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Reactor Division

INDEXED ABSTRACTS OF SELECTED REFERENCES
ON MOLTEN-SALT REACTOR TECHNOLOGY

D. W. Cardwell and P. N. Haubenreich

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
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UNION CARBIDE CORPORATION
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U. S. ATOMIC ENERGY COMMISSION

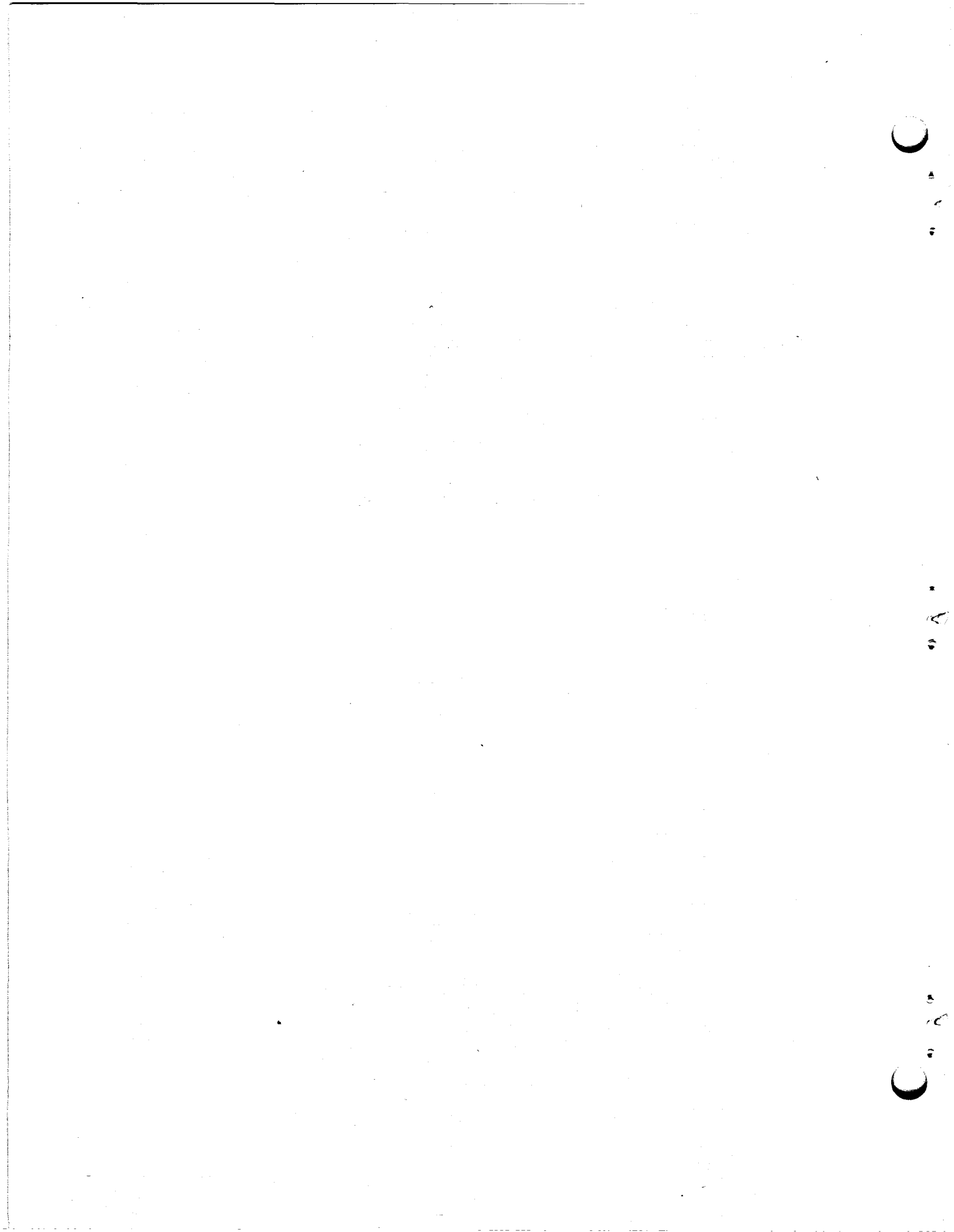


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The authors express their appreciation to Ruth Hofstra, John A. Carpenter, and A. F. Joseph of the ORNL Mathematics Division for providing technical guidance and programming required for file formatting, computer entry and automated printout of the MSR publication abstracts contained in this report. Appreciation is also expressed to Annabel Legg who prepared all text material for computer entry on our magnetic tape typewriter/converter and to various staff members of the Molten Salt Reactor Program who developed the abstracts and assigned keywords.

INDEXED ABSTRACTS OF SELECTED REFERENCES
ON MOLTEN-SALT REACTOR TECHNOLOGY

D. W. Cardwell and P. N. Haubenreich

ABSTRACT

Abstracts are given for 321 reports and articles which provide an introduction to MSR technology and describe major developments since 1960. Three indexes are provided: by keyword, by author, and by subject category.



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INTRODUCTION

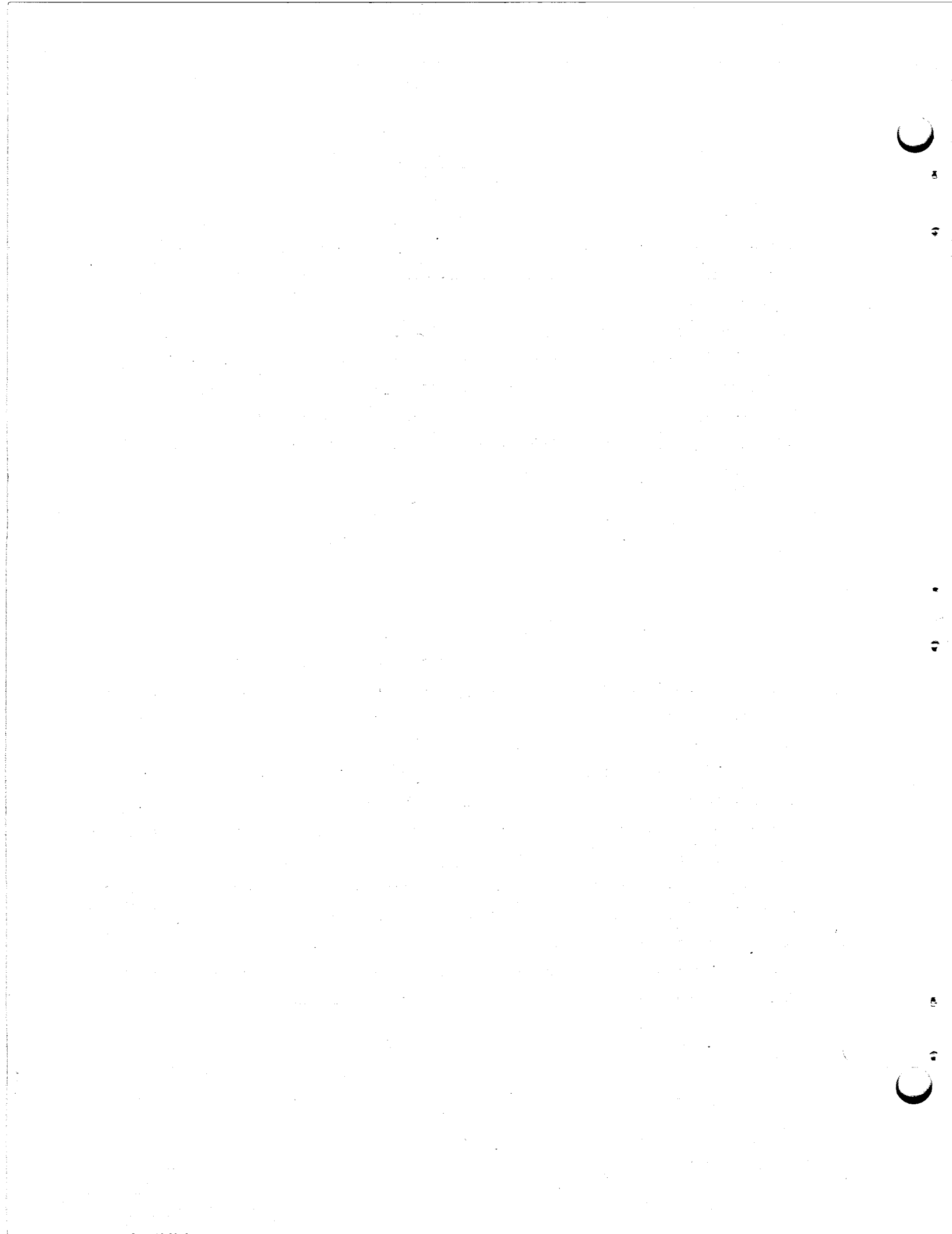
This document contains abstracts of 321 selected reports and papers which collectively provide a good, basic introduction to molten-salt reactor technology and describe major developments in the field since the initiation of the MSRE in 1960. As an aid in locating specific information, three indexes are provided: by keyword, by author, and by subject category.

The abstracts and indexes, prepared and printed by a computer, were taken from the file of the Molten-Salt Reactor Information System (MSRIS). This is a growing file in the IBM-360 computer at ORNL, which can be searched in various ways from remote terminals. A report is now being prepared to describe MSRIS and how to use it.

LIST OF ABSTRACTS

In the pages which follow, abstracts are listed in the alphabetical order of their primary subject categories. Each appears only once, even though its subject may extend into several other categories. Therefore to find all abstracts having information on a particular subject, it is necessary to use the category index.

Each entry in this list consists of the abstract itself plus certain other information about the reference. The first line is an identification number, assigned when the reference was added to the MSRIS file. The three letters in this number identify the primary subject category. If the material in the document extends significantly into another category, this is shown in the last line of the entry. Authors, title, and originating organization are listed on separate lines, then the document identification, date of publication and numbers of pages, figures, and references are given. Following the abstract is a list of keywords, with the most significant marked by asterisks.



Category A
Molten-Salt Reactor Programs

AAX670003

Briggs RB

SUMMARY OF THE OBJECTIVES, THE DESIGN, AND A PROGRAM OF DEVELOPMENT
OF MOLTEN-SALT BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1851 (June 1967), 84 p, 20 fig, 13 ref.

Molten-salt thermal breeder reactors are characterized by low specific inventory, moderate breeding gain with low fuel cycle cost, and high efficiency for converting heat into electricity. Studies indicate they should be able to produce electricity in 1000-MW (e) stations at a cost that is as low or lower than projected for advanced converter reactors or fast breeder power stations. The fuel utilization characteristics compare favorably with those of fast breeders. The present status of the breeder technology is being demonstrated in successful operation of the MSRE. A two-region Molten-Salt Breeder Experiment to demonstrate all the basic technology for full-scale breeders is proposed as the next step in the development. Design and construction of the MSBE would be accompanied by a program of fuels, materials, fuel reprocessing, and engineering development. Development, construction, and startup of the breeder reactor is estimated to take about eight years and to cost about \$125 million.

*development + *MSRP + *plans + *reviews + fuel cycle costs + MSBE + MSBR + natural resources + performance + power costs + technology

AAX670004

Carter WL + Whatley ME

FUEL AND BLANKET PROCESSING DEVELOPMENT FOR MOLTEN SALT
BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1852 (June 1967) 52 p, 10 fig, 13 ref.

This document describes the fuel and blanket processes for the MSBR, giving the current status of the technology and outlining the needed development. It is concluded that the principal needs are to develop the vacuum distillation and protactinium removal operations, which have been demonstrated in the laboratory but not on an engineering scale. A program to develop continuous fluoride volatility, liquid-phase reduction-reconstitution, improved xenon control, and special instrumentation should also be a major developmental effort. An estimate of manpower and cost for developing MSBR fuel and fertile processes indicates that it will require 288 man-years of effort over a 6-year period at a total cost of about \$18,000,000.

*development + *MSBR + *processing + blanket + costs + distillation + fuels + protactinium

AAX670005

Grimes WF

CHEMICAL RESEARCH AND DEVELOPMENT FOR MOLTEN-SALT BREEDER

Accession Number AAX670003 to AAX670005

Category A
 Molten-Salt Reactor Programs

AAX67C005 *Continued*

REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1853 (June 1967), 140 p, 26 fig, 69 ref.

Results of chemical research and development for molten salt reactors are summarized. These results indicate that LiF-BeF₂-UF₄ mixtures are feasible fuels for thermal breeder reactors. Such mixtures show satisfactory phase behavior, they are compatible with Hastelloy N and moderator graphite, and they appear to resist radiation and tolerate fission product accumulation. Mixtures of LiF-BeF₂-ThF₄ similarly appear suitable as blankets for such machines. Several possible secondary coolant mixtures are available; NaF-NaBF₄ systems seem, at present, to be the most likely possibility. Gaps in the technology are presented along with the accomplishments, and an attempt is made to define the information (and the research and development program) needed before an MSR can be operated with confidence.

*chemistry + *development + *MSRP + *research + *reviews + compatibility + fission products + fluorides + fluoroborates + molten salts + plans + two-fluid reactor

OTHER CATEGORIES: CXX

AAX670006

McCoy HE + Weir JR

MATERIALS DEVELOPMENT FOR MOLTEN-SALT BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1854 (June 1967), 88 p, 28 fig, 63 ref.

The materials development program is described for a two-region MSR with a uranium-bearing fluoride fuel salt, a thorium-bearing fluoride blanket salt, and a lower melting fluoride coolant salt. The primary structural materials are graphite and modified Hastelloy N. Individual fuel cells will be graphite tubes, which must withstand 10(23rd) neutrons/cm² and have very low permeability to gases and molten salts. Available graphites and their properties are described in detail. A program for obtaining and evaluating improved graphites is proposed. A program is described in detail for developing modified Hastelloy N, which will be used in all parts of the system except the core. Brazing alloys and a reasonable joint design have been developed for a joint between the graphite tubes and the modified Hastelloy N. Needed inspection techniques are considered. (This report is one of a set of 9 on development programs required for an MSR.)

modified Hastelloy N + graphite + *development + *materials + inspection + *MSRP + brazing + compatibility + mechanical properties + costs + reviews + *MSRP + *plans

OTHER CATEGORIES: EDX + FCX

AAX67C007

Category A
Molten-Salt Reactor Programs

AAX67C007 *Continued*

Scott D + Grindell AG

COMPONENTS AND SYSTEMS DEVELOPMENT FOR MOLTEN SALT BREEDER
REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1855 (June 1967), 56 p, 5 fig, 5 tab, 19 ref.

Studied thermal Molten-Salt Breeder Reactors to identify important design and development problems. The purpose was to organize these problems into a program which would produce components for use in a Molten-Salt Breeder Experiment. The reference-design concept is a two-region two-fluid system with the fuel salt separated from the blanket salt by graphite tubes. The energy produced in the reactor fluid is transferred to a secondary coolant-salt circuit, which couples the reactor to a supercritical steam cycle. The specific development problems to be studied include the reactor core and heat exchanger hydraulics, pumps for the three salt systems, heat transfer in the heat exchangers and boiler-superheater, mechanical valves for salt-flow control, control rod and drive, pressure relief in coolant system, cell insulation and heaters, and the cover-gas.

*components + *development + *MSBR + *MSRE + *plans +
*reviews + control-rod drives + control rods + cores +
heaters + heat exchangers + hydraulics +
pumps + steam systems + two-fluid reactor + valves +
thermal insulation

OTHER CATEGORIES: HXX

AAX67C008

Tallackson JR + Moore RL + Ditto SJ

INSTRUMENTATION AND CONTROLS DEVELOPMENT FOR MOLTEN-SALT
BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1856 (May 1967), 36 p, 2 ref.

Instrumentation used in the MSRE is a good basis for development of the instrumentation for large molten-salt breeder reactors. The development would involve primarily the testing and improvement of existing instrument components and systems. New or much improved devices are required for measuring flows and pressures of molten salts in the fuel and blanket circulating systems. No problems are foreseen that should delay the design or construction of a breeder reactor experiment. An estimate of costs of developing MSR instruments is given.

*development + *instrumentation + *MSBR + *systems +
components + control + flow measurement + MSRE + MSFP +
plans + measurement + radiation measurement +
temperature measurement + weigh cell

OTHER CATEGORIES: JXX

AAX67C009

Category A
Molten-Salt Reactor Programs

AAX670009 *Continued*

Perry AM

PHYSICS PROGRAM FOR MOLTEN-SALT BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1857 (June 1967) 40 p, 4 fig, 11 ref.

The sources of possible error in estimates of breeding performance of a Molten-Salt Breeder Reactor are discussed. Uncertainties in cross sections may contribute an uncertainty of about plus or minus 0.026 in breeding ratio. Other sources of error may arise from assumptions regarding behavior of fission products, or from inadequacies in methods of computation. A reactor physics development program is outlined which should provide a sound basis for design of a reactor experiment. The program includes theoretical investigation of system dynamic characteristics, evaluation of alternate core designs, development of computational methods, cross-section evaluation, development of computer codes and experimental physics. Program manpower requirements and costs are estimated. (This report is one of a set of nine on development programs required for an MSBR.)

MSBR + *breeding performance + *nuclear analysis +
*cross sections + computer codes + rare earths +
fission products + *MSRP + dynamic characteristics +
neutron yield + costs + *plans + stability + *design data +
calculations + methods

OTHER CATEGORIES: BXX

AAX670010

Kasten PR

SAFETY PROGRAM FOR MOLTEN-SALT BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1858 (June 1967) 42 p, 6 fig, 3 ref.

Investigations required in determining the safety characteristics of MSBR power plants are outlined, and the safety features of the major plant systems are described. Reactivity additions which need detailed study include those associated with net fuel addition to the core region, graphite behavior, changes in fluid flow conditions, and control rod movement. Reactivity coefficients which require evaluation include those associated with temperature, voids, pressure, fuel concentration, and graphite concentration. The integrity of plant containment under reactivity incident conditions and, also under circumstances where reactivity itself is not involved, needs to be evaluated. Stability analysis of the reactor plant is required. Physical behavior of materials and of equipment under MSBR conditions, as they relate to reactor safety, need to be determined experimentally. To delineate and resolve the basic safety problems associated with MSBR systems, about \$1.3 million is required over

Category A
Molten-Salt Reactor Programs

AAX67C010 *Continued*

a period of about eight years, with most of the effort (\$0.9 million) occurring during the first four years. (This report is one of a set of nine on development programs required for an MSBR.)

*MSRP + *safety + *analysis + *plans + reactivity + MSER + accidents + costs + containment + stability + dynamic characteristics + off-gas systems + processing
OTHER CATEGORIES: EGX

AAX67C011

Blumberg F

MAINTENANCE DEVELOPMENT FOR MOLTEN-SALT BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1859 (June 1967), 18 p, 1 fig, 6 ref.

The maintenance system of the proposed molten-salt breeder reactors will be based upon the technology in use and experience gained from the Molten-Salt Reactor Experiment. The unit replacement scheme, long-handled tools, movable maintenance shields, and the means for handling contaminated equipment will be similar for many operations. The techniques must be improved and extended and new techniques must be developed for maintaining some of the larger, more radioactive components of the breeder reactors. Remote welding is needed for major component replacement. Methods must be available for replacing the core and for the repair of heat exchanger. Finally, a general development and design surveillance program will be required. These programs are described and their cost is estimated. (This report is one of a set of 9 on development programs required for an MSBR.)

*maintenance + *MSBR + *plans + development + MSRE + remote welding
OTHER CATEGORIES: KEE

ABX580001

MacPherson HG

MOLTEN-SALT REACTORS

Oak Ridge National Laboratory, Tenn.

Part II of Fluid-Fuel Reactors, Addison-Wesley (1958), pp. 563-697.

The early history and 1958 development status of molten-salt reactors is covered in 7 chapters of this book, prepared for the second Geneva Conference. Chapter topics include chemistry, materials, nuclear aspects, heat-transfer equipment, the Aircraft Reactor Experiment, and a conceptual design of a power reactor. The concept presented has a core and blanket, with no moderator other than the LiF-BeF₂ carrier salt.

*ARE + *development + *MSRP + *reviews + *technology + chemistry + corrosion + Hastelloy N + inconels +

Accession Number AAX6700 10 to AEX580001

Category A
Molten-Salt Reactor Programs

ABX5800C1 *Continued*
molten salts

ABX64C004

Briggs RB

MOLTEN-SALT POWER REACTORS AND THE ROLE OF THE MSRE IN THEIR
DEVELOPMENT (PART OF MSRP SEMIANN PROG REPT 7/31/64)

Oak Ridge National Laboratory, Tenn.

ORNL-3708 (Nov. 1964), pp 3-21, 7 fig, 8 ref.

ORNL studies show the molten-salt reactor to be the most promising thermal-neutron thorium-U233 breeder concept. In this paper, a compact 500-MWe two-fluid breeder with graphite tubes separating fuel and fertile salts is described and its processing and economics are discussed. The MSRE was authorized in 1960 to investigate chemistry, materials, engineering and operation of the MSR concept. Success with the MSRE should lead to construction of a converter reactor that could be modified to become a breeder.

*MSRP + *two-fluid reactor + breeding performance +
design + development + economics + MSBR + MSRE + plans +
reviews

ABX670049

MacPherson HG

MOLTEN-SALT REACTOR SHOWS MOST PROMISE TO CONSERVE NUCLEAR
FUELS

Oak Ridge National Laboratory, Tenn.

Power Engineering 71, 1 and 2 (Jan and Feb 1967), 7 p,
6 fig, 6 ref.

The MSBR promises to combine simplified fuel recycle and stable fuel in a high-performance thermal breeder having low power costs. The present concept of an MSRE has fuel and fertile salts separated by graphite in a 14-ft reactor vessel. MSRE experience has shown molten salt reactors to be practical. A 50-MWe two-fluid breeder is suggested as the next step.

*breeding performance + *economics + *MSBR +
*natural resources + conceptual design + experience + MSRE +
plans + reviews

ABX680035

MacPherson HG

MOLTEN-SALT REACTORS

Oak Ridge National Laboratory, Tenn.

Proc. Intl. Conf. on Constructive Uses of Atomic Energy,
Washington, Nov. 1968, pp. 111-121, 7 fig, 4 ref.

Experiments on feasibility of molten salts as reactor fuels started in 1947 in the aircraft reactor program. The concept now features molten fluoride salt containing UF4 and ThF4 circulated through a graphite core. Advantages

Category A
Molten-Salt Reactor Programs

ABX680035 *Continued*

of low-pressure, high-temperature, fluid fuel promote safety and economy. Research and development have concentrated on materials, compatibility, components and the MSRE. Recent advances include improved materials and simplified processing. Conceptual design studies of one-fluid molten-salt breeder reactors indicate good breeding performance and low power costs.

*MSRP + *reviews + breeding performance + costs + development + MSBR + safety + technology

ABX690007

Haubenreich FN + Rosenthal MW

MOLTEN-SALT REACTORS

Oak Ridge National Laboratory, Tenn.

Science Journal 5 (6) (June 1969), 6 p, 5 fig, 4 ref.

Breeder reactors are needed to keep power costs down as uranium prices rise. Development emphasis is on fast breeders, which promise high gain. Thermal breeders must have fast processing to remove protactinium and poisons to achieve moderate gain, but fissile material investments can be low. The fluid-fuel molten-salt reactor with on-site processing promises low fuel cycle cost and acceptable doubling times. MSF development dates back to 1948 and includes successful operation of the MSRE at 650 deg C for over three years. The molten-salt breeder concept is now a graphite core with circulating salt containing both uranium and thorium, processed by reductive extraction into bismuth.

*breeding performance + *economics + *electrical power + *MSBR + *natural resources + *reviews + experience + fuel cycle costs + MSRE + MSRF + processing

ABX690056

Rosenthal MW + Robertson RC + Bettis ES

MOLTEN-SALT BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

Nucl. Engrg. Int. Vol. 14, No. 156 (May 1969), pp. 42C-425, 5 fig.

This article explains how molten-salt reactors offer low-cost power now and in the future because of good breeding performance and inherent advantages of molten-salt fuel. Brief descriptions are given of MSBR materials, core design, components, and processing scheme. After discussing MSR maintenance, safety, and costs, the authors conclude with an outline of work required to develop a large commercial MSBR.

*MSRP + *reviews + breeding performance + components + costs + development + MSBR + safety + technology

ABX700054

Category A
Molten-Salt Reactor Programs

ABX700054 *Continued*

Rosenthal MW + Kasten PR + Briggs RB

MOLTEN-SALT REACTORS -- HISTORY, STATUS, AND POTENTIAL

Oak Ridge National Laboratory, Tenn.

Nucl. Appl. Tech. 8, 107 (Feb. 1970), 11 p, 3 fig, 18 ref.

Molten-salt breeder reactors being developed at ORNL promise safe, low-cost power while extending resources of fissionable material. MSR technology, developing since 1947, was adequate for successful construction and operation of the MSRE which showed that circulating molten fuel is practical, that fluoride salts are stable under reactor conditions, and that corrosion is very low. The simple fuel processing necessary for a converter was demonstrated in the MSRE. Processing methods being developed should permit MSR's in which UF₄ and ThF₄ are combined in a single salt flowing through a graphite moderator to operate as economical breeders. Initial startup can be with U-235, U-233, or Pu-239. Construction costs should be about the same as light-water reactors and fuel costs should be much lower. Achievement of economic MSBR's requires development and construction of several MSR plants of increasing size.

*MSRP + *reviews + ARE + breeding performance +
capital costs + design + development + fuel cycle costs +
MSBR + materials + processing + safety + technology

ABX700055

Shaw M + Landis JW + Laney RV + Rosenthal MW + Layman WH

U. S. SURVEY: REACTOR DEVELOPMENT PROGRAM

United States Atomic Energy Commission

Nucl. Eng. Int. Vol. 15, No. 173 (Nov. 1970), pp. 899-904,

4 fig.

In the U.S.A. there was proliferation of reactor concepts in the 1950's eliminations in the 1960's; development efforts are now concentrated on 6 concepts: Light Water, Liquid Metal-cooled Fast Breeder, Light Water Breeder, Molten-Salt Breeder, High-Temperature Gas, and Gas-Cooled Fast Breeder. This article covers the development status of each. The molten-salt reactor program, since the conclusion of the MSRE, includes: design studies, reactor systems and equipment development, chemical processing, materials, and chemistry.

*AEC + *development + *electrical power + *reactors +
*reviews + foreign

ABX710020

Grenon M + Geist JJ

LES REACTEURS A SELS FONCUS

Euratom

Energie Nucleaire, Vol. 13, No. 2 (Mar.-Apr. 1971)

pp. 86-93, 10 fig.

Accession Number ABX700054 to AEX710020

Category A
Molten-Salt Reactor Programs

ABX710020 *Continued*

This article (in French) appears in a series on chemical sciences. The authors, formerly involved in the Euratom-USAEC exchange on molten-salt reactors, introduce the MSR as a potential breeder worthy of multinational consideration. They describe the concept, early development, recent progress, problems, advantages and possible future development. (An English translation, ORNL-tr-2508, is available from ORNL.)

*development + *economics + *MSBR + *MSEF +
breeding performance + foreign + reviews

ACA65C004

Haubenreich PN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRE PROG. REPT.
2/28/65)

Oak Ridge National Laboratory, Tenn.

ORNL-3812 (June 1965), pp. 5-60, 17 fig, 29 ref.

Construction of the salt systems and closely associated ancillary systems was completed and full-time pre-nuclear testing began in September. After leak-testing, purging and heating of the salt systems, flush salt and coolant salt were loaded. Transfers and circulation followed. Testing showed the need for modification of radiator doors, freeze-valve air supplies and controls, thermal shield water piping and some cooling air control valves. Krypton-85 was injected into the fuel system to test removal mechanisms.

*construction + *experience + *MSRE + *startup + *testing +
drying + freeze valves + krypton + loading + molten salts +
thermal insulation

OTHER CATEGORIES: MXX + KAB

ACA650010

Haubenreich PN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRE PROG. REPT.
8/31/65)

Oak Ridge National Laboratory, Tenn.

ORNL-3872 (Dec. 1965), pp. 7-78, 34 fig, 40 ref.

Pre-nuclear testing with flush salt was completed in March after 1000 hours of salt circulation. In preparation for low-power nuclear operation, nuclear instruments, the fuel sampler-enricher and one layer of the reactor cell roof blocks were installed and reactor operators received additional training. Fuel carrier salt containing depleted uranium was loaded and circulated for 10 days in May before additions of enriched U-235 began, first into the drain tanks, then through the pump bowl. Criticality was reached on June 1 at very near the predicted loading. Subsequent small additions of U-235 permitted calibration of the control rods and measurement of reactivity coefficients and

Accession Number ABX710020 to ACA650010

Category A
Molten-Salt Reactor Programs

ACA650010 *Continued*

provided enough reactivity to operate for several months at power. Zero-power measurements and dynamics tests were completed in July and final preparations for high-power operation were started.

*criticality + *experience + *MSRE + *operation +
*startup + control rods + dynamics tests + loading +
measurement + molten salts + reactivity + operators +
training + testing

OTHER CATEGORIES: MXX + KAB

ACA660008

Haubenreich PN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRP PROGR. REPT.
2/28/66)

Oak Ridge National Laboratory, Tenn.

ORNL-3936 (June 1966), pp. 7-92, 41 fig, 43 ref.

Preparations for high-power operation were completed. These included modifying coolant line archer sleeves, replacing radiator doors, inspecting fuel pump internals, measuring salt piping stresses, heat treating the reactor vessel, sealing and testing secondary containment, installing new core specimens, improving insulation on the radiator enclosure, and further training of operators. Nuclear operation resumed in December and tests at powers up to 1 MW verified predicted dynamic behavior. The power ascension was interrupted at 1 MW when valves and filters in the fuel off-gas system plugged. Investigation revealed radiation-polymerized decomposition products of oil that had leaked into the fuel pump bowl.

*experience + *MSRE + *operation + analysis + containment +
control rods + dynamics tests + heat treatments +
off-gas systems + piping + pumps + remote maintenance +
stability + startup + stress + testing

OTHER CATEGORIES: MXX + KAB + KBA

ACA660014

Haubenreich PN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRP PROGR. REPT.
8/31/66)

Oak Ridge National Laboratory, Tenn.

ORNL-4037 (Jan. 1967), pp. 1-94, 24 fig, 35 ref.

Power ascension was resumed in April after a large, efficient filter assembly was installed to protect the fuel system pressure control valve from oil decomposition products. Full power of 7.5 MW (limited by heat removal capability) was reached in May. Tests at each stage verified predictions except that xenon stripping was more effective and heat transfer from the radiator was lower than expected. Restrictions at the fuel off-gas charcoal bed inlets developed but were cleared by backblowing.)

Category A
Molten-Salt Reactor Programs

ACA660014 *Continued*

Operation was interrupted briefly by electrical failures in a component cooling pump and the fuel sampler, a false indication of containment cell leakage, and failure of a drive coupling on a main blower. Hub and blades of a main blower shattered on July 17, forcing a shutdown. Flaws were found in the hubs of the other blower and the spare and procurement of new blowers was started. The delay was used to remove core specimens, alter the radiator door seals, install equipment to handle radiolytic gas from the thermal shield, repair leaky cell coolers, and remove the off-gas particle trap for examination. Flush salt got into some gas lines when the fuel pump was accidentally overfilled, and this was melted out by temporary heaters.

*experience + *MSRE + *maintenance + *operation +
analysis + blowers + components + cracks + failures +
filters + fission products + heat transfer +
off-gas systems + radiolysis + remote maintenance +
samplers + startup

OTHER CATEGORIES: MXX + KAB + KBA

ACA670016

Haubenreich PN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRP PROGR. REPT.

2/28/67)

Oak Ridge National Laboratory, Tenn.

ORNL-4119 (July 1967), pp. 1-94, 36 fig, 42 ref.

Replacement blowers were received in October and operation was resumed after a 12-week shutdown. A restriction which appeared in the off-gas line at the fuel pump bowl was temporarily relieved by heating, but had to be cleared mechanically in November. After a successful 30-day run at full power, the reactor was shut down in January to inspect the blowers and to replace air line disconnects in the reactor cell whose leakage had interfered with measurement of containment cell leakage. At the same time the fuel off-gas filter assembly was replaced with two parallel particle traps of improved design. (In the first particle traps, expansion of some parts tended to throttle the flow upon heating by fresh fission products.) A comprehensive reactivity balance (including automatic computation at frequent intervals by the on-line computer) became operational and unexplained reactivity changes from the beginning of operation were shown to be only 0.05%. Full-power operation was resumed and continued through February.

*experience + *MSRE + *operation + analysis + blowers +
off-gas systems + reactivity + remote maintenance +
disconnects

OTHER CATEGORIES: MXX + KAE + KBA

ACA670023

Accession Number ACA660014 to ACA670023

Category A
Molten-Salt Reactor Programs

ACA67C023 *Continued*

Haubenreich FN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRE SEMIANN PFCG
REPT 8/31/67)

Oak Ridge National Laboratory, Tenn.

ORNL-4191 (Dec. 1967), pp 13-62, 33 fig, 25 ref.

Run 11 continued for 102 days, over 90% at full power, before a scheduled shutdown May 10. Makeup U-235, added at power for the first time, mixed in 2 minutes. The new offgas particle trap worked well, but the charcoal bed inlets occasionally plugged. A main blower bearing had to be replaced during the run. During the May-June shutdown, core specimens were replaced. A remote gamma spectrometer was tested and used to scan the primary heat exchanger. Minor maintenance was also done and annual tests were completed. The next run was 42 days at full power, with emphasis on beryllium additions and fuel sampling. Shutdown came after the fuel sampler cable tangled and was severed. Tools were developed and the latch was retrieved, but not the capsule. Operations analysis included long-term reactivity effects, thermocouple drift, and salt heat transfer. In preparation for replacement of the uranium in the fuel with U-233, neutronic characteristics with this fissile material were calculated.

*experience + *maintenance + *MSRE + *operation + analysis + bearings + components + gamma spectrometry + heat transfer + off-gas systems + performance + reactivity + remote maintenance + reactivity + temperature measurement + uranium-233 + samplers

OTHER CATEGORIES: MXX + KAB + MDX + MEC

ACA68C012

Haubenreich FN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRE SEMIANN PFCG
REPT 2/29/68)

Oak Ridge National Laboratory, Tenn.

ORNL-4254 (Aug. 1968), 49 p, 35 fig, 32 ref.

Early in the period the fuel sampler was reinstalled and full-power operation resumed. After the startup was interrupted to repair a component cooling pump, there was 6 months without a fuel drain. Fuel circulation was stopped 2 days in November for work on the sampler and during a xenon experiment at low power a bearing was replaced on a main blower. Otherwise no equipment problem interfered with operation. Operation at various fuel levels, temperatures and pressures showed effects on xenon stripping, neutronic noise, and gas in the access nozzle. Reactivity balances showed slight drift (0.1%) over the long run. An offgas sampler was installed downstream of the charcoal beds. The

Accession Number ACA670023 to ACA680012

Category A
Molten-Salt Reactor Programs

ACA68C012 *Ccontinued*

freeze flange thermal cycle test, stopped after 103 cycles, was resumed. Analyses of system dynamics with the proposed U-233 fuel predicted safe and stable operation.

*experience + *maintenance + *MSRE + *operation + analysis + bearings + components + dynamic characteristics + noise analysis + off-gas systems + reactivity + uranium-233 + xenon + samplers + freeze flanges
OTHER CATEGORIES: MXX + KAE + MDX + MEC

ACA68C019

Haubenreich PN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRE SEMIANN PFCG REPT 8/31/68)

Oak Ridge National Laboratory, Tenn.

ORNL-4344 (Feb. 1969), pp. 1-52, 28 fig, 43 ref.

A 6-month run, ending in March, concluded operation with U-235 after 900n equivalent full-power hours. After shutdown, gamma-spectrometric measurements were made on the fuel system, core specimens were replaced, the fuel offgas line was cleared and two heaters from the primary heat exchanger were repaired. All 15 rod-scam relays were replaced and 3 of the new relays failed. A capsule dropped in the fuel sampler could not be retrieved, but did not prevent fuel sampling. The on-site processing equipment was readied for removal of the uranium from the salt. After testing, the sulfur dioxide reaction system for disposal of excess fluorine was abandoned in favor of reaction with a caustic solution. In August the flush salt and the fuel salt were fluorinated, efficiently recovering the U as the hexafluoride. Corrosion products were precipitated and filtered in the final step before U-233 loading. Theoretical analyses of U-233 operation, including refined calculation of delayed neutron effects, showed that the system would be quite stable. After 268 test thermal cycles of the prototype freeze flange, a crack was found at the alignment stub.

*experience + *fluorination + *maintenance + *MSRE + *operation + analysis + dynamic characteristics + freeze flanges + samplers + uranium-233 + off-gas systems
OTHER CATEGORIES: MXX + KAE + MDX + MEC + LHX

ACA69C021

Haubenreich FN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRE SEMIANN PFCG REPT 2/28/69)

Oak Ridge National Laboratory, Tenn.

ORNL-4396 (Aug. 1969) pp. 1-47, 32 fig, 40 ref.

The MSRE began nuclear operation with U-233 in September and was brought to full power in January. Criticality was attained by adding 33 kg of U, as the UF₄-LiF eutectic, to the carrier salt from which the original U-235 had been

Category A
Molten-Salt Reactor Programs

ACA690021 *Continued*

stripped. Startup tests included measurements of rod worth and reactivity coefficients, dynamics tests, and noise analysis. When beryllium metal was added to adjust the reducing power of the salt, the entrained cover gas increased from less than 0.1 vol % to 0.6 vol %, apparently due to slight changes in the physical properties of the salt. During the power ascension, small perturbations in nuclear power were seen. Analysis indicated they were due to gas in the core, and they did not occur when gas entrainment was reduced by slowing the fuel pump. Before the power ascension, the fuel offgas line was cleared of a restriction. Shortly afterward a loose gear in the fuel sampler forced a 3-week shutdown, during which time a control-rod drive was serviced. Thermal cycle testing of the prototype freeze flange continued and test-stand operation of a fuel pump with a deeper bowl (Mark-2) began.

*experience + *maintenance + *MSRE + operation + analysis + control rods + dynamics tests + freeze flanges + off-gas systems + reactivity + uranium-233 + noise analysis
OTHER CATEGORIES: MXX + KAB + MDX + MEC

ACA690028

Haubenreich PN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRP SEMIANN PROG REPORT
8/31/69)

Oak Ridge National Laboratory, Tenn.

ORNL-4449 (Feb. 1970), pp. 1-38, 19 fig, 37 ref.

High-power operation with U-233, which began in January, continued until a scheduled shutdown on June 1. There were few equipment problems other than restrictions in the offgas lines, and the reactor was critical 95% of the time from January to June. Fuel samples were taken periodically to measure U-233 capture-to-fission ratio and to study fuel chemistry. Tests continued on the behavior of cover gas and xenon in the fuel system at various circulation rates. Continuous indicators of reactor pressure and neutron noise levels were installed and used. During the shutdown, a new core specimen array was installed, a stiff control rod was replaced, rod drives were repaired, and the offgas lines were cleared. A remote gamma-ray spectrometer was used to measure fission-product distributions with the salt drained and during the startup. Annual containment tests concluded the 10-week shutdown. Operation resumed with experiments comparing argon and helium as cover gases. Component development work included extension of the prototype freeze flange thermal cycle test through 400 cycles, and operation of the Mark-2 fuel pump with a high salt level to reduce entrainment.

*experience + *maintenance + *MSRE + *operation + analysis + control rods + cover gas + freeze flanges + gamma spectrometry + noise analysis + off-gas systems + uranium-233 + xenon
OTHER CATEGORIES: MXX + KAB + MDX + MEC

ACA700021

Category A
Molten-Salt Reactor Programs

ACA700021 *Continued*
Haubenreich PN
MOLTEN-SALT REACTOR EXPERIMENT (PART 1, MSRP SEMIANN PROG REPT
2/28/70)

Oak Ridge National Laboratory, Tenn.

ORNL-4548 (Aug. 1970), pp. 1-40, 14 fig, 49 ref.

Nuclear operation of the MSRE was concluded on Dec. 12, 1969. Principal activities during the final runs were studies of xenon stripping and tritium distribution, and sampling to determine fission product behavior. A remote gamma-ray spectrometer was also used during both operation and shutdown to observe fission product distributions. After the final shutdown the reactor was placed in standby, awaiting later examination. A small leak near a freeze valve during the shutdown released some fission products into the containment cell. Refined analyses of reactivity calculations and long-term behavior confirmed the good agreement. The prototype freeze flange undergoing thermal cycle testing was inspected after 470 cycles, then was run on to 540 cycles before the test was discontinued.

*experience + *MSRE + *operation + analysis + freeze flanges + gamma spectrometry + leaks + noise analysis + reactivity + tritium + xenon

OTHER CATEGORIES: MXX + KAB + MDX

ACA700035
Haubenreich PN
MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRE SEMIANN PROG
REPT 8/31/70)

Oak Ridge National Laboratory, Tenn.

ORNL-4622 (Jan. 1971), pp. 1-6, 3 fig, 12 ref.

The MSRE remained shut down, awaiting postoperation examination. Procedures and tools were prepared. Specimens were cut from the coolant system piping. Analysis of data taken with the remote gamma spectrometer during the final runs suggested that 'noble-metal' fission products quickly migrate, as extremely small particles, to metal surfaces or salt-gas interfaces. Existing data on radiolytic fluorine evolution from frozen salt indicate that evolution from the MSRE fuel in storage is easily prevented. MSRE component development ended with termination of pump endurance tests.

MSRE + analysis + experience + radiolysis + fluorine + examinations

OTHER CATEGORIES: MXX + CFX

ACA710028
Haubenreich PN
MOLTEN-SALT REACTOR EXPERIMENT (PART I, MSRP SEMIANN PROG
REPT 2/28/71)

Oak Ridge National Laboratory, Tenn.

ORNL-4676 (Aug. 1971), pp. 1-20, 16 fig, 16 ref.

Accession Number ACA700021 to ACA710028

Category A
Molten-Salt Reactor Programs

ACA710028 *Continued*

Portions of the fuel system were removed for examination as planned. Control rods, rod thimbles and one moderator bar were taken out and the interior of the reactor vessel was viewed. The fuel sampler cage was cut out and the pump bowl viewed. Portions of 6 heat exchanger tubes were removed through a hole cut in the shell. The salt leak was found at a freeze valve and the section was cut out. Conditions in the reactor were generally very good. A test showed the coolant flowmeter had been reading high and the reactor heat balance should have been 7.6 MW at full power.

*examinations + *MSRE + cores + cutting tools + experience + flow measurement + heat exchangers + heat balance + pumps + remote maintenance

OTHER CATEGORIES: MFX

ACB660009

(Staff Report)

MSBR DESIGN STUDIES (CHAP 6, MSRP SEMIANN PROG REPT 2/28/66)

Oak Ridge National Laboratory, Tenn.

ORNL-3936 (June 1966) pp 172-192, 7 fig, 4 ref.

A reference design concept is described for a 1000-MWe two-fluid MSBR with fuel and coolant salts separated by graphite tubes in a 14-ft reactor vessel. Flowsheet, layouts of the radioactive systems, and processing by fluorine volatility and distillation are presented. Also reported are calculated nuclear performance and costs.

*conceptual design + *MSBR + *two-fluid reactor + breeding performance + costs + flowsheets + layout + processing

ACB660015

Briggs RB

MOLTEN-SALT BREEDER REACTOR STUDIES (PART 3 MSRP SEMIANN PROG REPT 8-31-66)

Oak Ridge National Laboratory, Tenn.

ORNL-4037 (Jan. 1967), pp. 207-237, 10 fig. 5 ref.

Design study work for the two-region, two-fluid 1000 MW(e) MSBR included adoption of a modular concept, using four small reactors to facilitate maintenance, and revision of the primary heat exchangers to use bent tubes rather than bellows. Nuclear performances with and without Pa removal are compared. Steam system efficiencies and costs for 700 deg F feedwater are compared to a 580 deg F system. Performance data for other reactor types are presented, including a lead-cooled MSBR, and euthermal breeder, and a converter with the fertile and fissile materials in a single stream. Salt processing for the fuel and blanket salts is described. The two systems are similar, the salt being fluorinated, the off-gas sorbed, and the uranium tetrafluoride recovered by cold-trapping. A vacuum still

Category A
Molten-Salt Reactor Programs

ACB660015 *Continued*

separates the rare earths from the remaining salt. Concepts for continuous stills and fluorination units are described. Liquid-metal extraction and reductive precipitation are suggested as alternative methods.

*MSBR + *progress report + *conceptual design + *processing + *heat exchangers + *steam systems + *protactinium + *performance + breeding performance + electrical power + thermal power + heat transfer + fuel cycle costs + flowsheets + thermodynamics + design criteria + *modular design + *lead cooling + *two-fluid reactor + *steam cycle

OTHER CATEGORIES: IAC

ACB67C017

Briggs RB

MOLTEN-SALT BREEDER REACTOR DESIGN STUDIES (PART 3 MSBR SEMIANN PROG REPT, 2-28-67)

Oak Ridge National Laboratory, Tenn.

ORNL-4119 (July 1967), pp. 174-214, 21 fig, 6 ref.

Design study of the two-region, two-fluid 1000 MW(e) MSBR continued. The 250 MW(e) reactor module has a vessel 12 ft diam with 4-in. diam graphite balls between the core elements and the reflector. The revised primary heat exchangers have the long-shaft salt circulating pumps located above the units. The effect on reactivity of fissile concentration and fuel-volume fraction on the neutron flux distribution is explored. An off-gas system flowsheet is presented and the required gas injection and removal system discussed. The effect of xenon removal on the poison fraction was calculated. Processing of the salts in a continuous fluorinator with salt-protected walls may be adequate protection against corrosion. The relative volatility of the rare earths was investigated and the equations for buildup of non-volatiles on vaporizing surfaces are presented.

*MSBR + *progress report + *conceptual design + *processing + *pumps + *heat exchangers + *reactor vessel + *replacement + *void fractions + *volume fractions + performance + graphite + blanket + fluorination + corrosion protection + volatility + xenon + off-gas systems + *modular design + *two-fluid reactor

OTHER CATEGORIES: IAC

ACB67C024

Briggs RB

MSBR DESIGN AND DEVELOPMENT (PART 2 MSBR SEMIANN PROG REPT 8-31-67)

Oak Ridge National Laboratory, Tenn.

ORNL-4191 (Dec 1967), pp 63-101, 23 fig, 6 ref.

Design study of the two-fluid, two-region, 1000 MW(e) MSBR

Accession Number ACB660015 to ACB670024

Category A
Molten-Salt Reactor Programs

ACB670024 *Continued*

using four 250 MW(e) reactor modules involved new cell layouts to accommodate stresses in piping and pedestal-mounted equipment due to thermal expansions. The reactor cell wall construction was studied in more detail. The core graphite was rearranged to better accommodate dimensional changes due to neutron irradiation. More detailed drawings and data on the fuel and blanket-salt heat exchangers are presented. Reactor performance was evaluated in terms of the average core power density, optimized mainly on the basis of yield, and the fuel-cycle cost estimated. The useful life of the graphite as a function of the neutron flux is estimated from the Duncunrey Fast Reactor data. Flux-flattening is discussed and the temperature coefficients of reactivity calculated. The xenon-135 poisoning problem in the blanket and coolant-salt pumps are outlined, particularly with regard to the molten-salt journal bearing.

*MSBR + *progress report + *conceptual design + *pumps + *heat exchangers + *xenon + *graphite + *stress + thermal effects + bearings + breeding performance + design data + expansion + fuel cycle costs + mass transfer + neutron flux + neutron fluence + parametric studies + shrinkage + void fractions + volume fractions + development + radiation damage + *modular design + *two-fluid reactor
OTHER CATEGORIES: IAC

ACB680013

Briggs RB

MSBR DESIGN AND DEVELOPMENT (PART 2 MSRP SEMIANN PROG REPT
2-29-68)

Oak Ridge National Laboratory, Tenn.

CRNL-4254 (Aug. 1968), pp. 51-87, 22 fig, 10 ref.

A single fluid concept was adopted for the two-region 2000-MW(e) MSBR study reference design because it eliminated the graphite-to-metal joints in the two-fluid concept and because means for chemical processing of a single salt now appeared to be available. Flow diagrams and new plant layouts for the single-fluid system are presented. Drawings and design data for the single reactor vessel, the core graphite elements, and the salt drain tank are included. Tabulated data of reactor physics calculations indicate almost as good a performance as for the two-fluid reactor. The effect of use of coated graphite on the two-fluid reactor xenon-135 poison fraction is reported. A conceptual design is shown for a single-fluid MSBR fuel-salt pump, which does not require a salt-lubricated bearing as in the two-fluid concept. A salt-bearing experimental test loop and program are described, however. Remote maintenance problems in an MSBR plant are discussed. Preliminary

Accession Number ACB670024 to ACB680013

Category A
Molten-Salt Reactor Programs

ACB680013 *Continued*

results of analog computer studies of the dynamics of the two-fluid MSER are presented.

*MSBR + *progress report + *conceptual design +
*single-fluid reactors + performance + layout + joints +
flowsheets + data + reactor vessel + cores + graphite +
fuel cycle + neutron physics + coatings + xenon + pumps +
piping + instrumentation + maintenance +
OTHER CATEGORIES: IAD

ACB680020

Briggs RB

MSBR DESIGN AND DEVELOPMENT (PART 2 MSBR SEMIANN EBCG FEPT
8-31-68)

Oak Ridge National Laboratory, Tenn.

ORNL-4344 (Feb 1969) pp 53-108, 32 fig, 15 tables, 19 ref.

The single-fluid 1000 MW(e) reference plant uses a confinement building to permit replacement of the reactor core as an assembly. As shown on new flowsheet and layout drawings, the revised reactor has graphite spheres in the blanket and graphite control rods at the center. Details of a revised primary heat exchanger are presented. Neutron physics calculations for the revised concept were improved. Preliminary calculations for a 100-200 MW(t) MSER are reported. The MSBR Xe-135 poison as function of bubble stripping and graphite sealing was calculated and concepts for a bubble generator and gas separator described. The MSBR pumping requirements and first operation of the sodium fluoroborate test loop are discussed, as were the requirements for a steam generator test facility. Analyses of the dynamic response of the MSBR system (and the steam generator) indicates general feasibility. Neutron decay after shutdown was calculated. Resistance thermometers possibly can be used in the MSBR. Test equipment for measuring heat transfer properties of the salt are described and data given for thermal conductivity and heat capacity. Test equipment and first data on simulated mass transfer of xenon to bubbles are covered.

*MSBR + *progress report + *conceptual design + *reactors +
*heat exchangers + *pumps + *MSBE + *steam generators +
*control + *temperature measurement + *physical properties +
*gas injection + *gas separation + *performance +
*heat transfer + bubbles + mass transfer +
thermal conductivity + specific heat + structures +
maintenance + graphite + control rods + neutron physics +
xenon + fluoroborates + cores + reactor vessel +
single-fluid reactors + test facilities + void fractions +
primary salt + materials testing + instrumentation + piping +
spheres + containment
OTHER CATEGORIES: IAD

ACB690022

Accession Number ACB680013 to ACB690022

Category A
 Molten-Salt Reactor Programs

ACB690022 *Continued*

Briggs RB

MSBR DESIGN AND DEVELOPMENT (PART 2 MSRP SEMIANN PRCG REPT
2-28-69)

Oak Ridge National Laboratory, Tenn.

ORNL-4396 (Aug. 1969) pp 49-128, 57 fig, 26 tables, 50 ref.

Design studies of the single-fluid 1000 MW (e) MSBR continued with emphasis on the reactor core and vessel design, flow and temperature distributions, fission-product distribution in the systems, krypton and xenon purging, and the off-gas system heating loads. The diameter of the reactor cell was increased and the cell wall construction studied in more detail. Changes in the central core dimensions resulted in increased graphite life. Reactor afterheat sources, temperature distributions in graphite core and reflector and in reactor vessel are plotted. Development work includes methods for bubble generation and gas separation in fuel-salt system. Distribution of noble metal fission products is tabulated. Improved values were obtained for thermal conductivity of the fuel salt and an experimental loop to confirm heat transfer relationships has furnished preliminary data. Operation of the sodium fluoroborate test loop is described. A successful remotely-operated orbital welder for piping is reported. The controls system studies continued. Preliminary drawings and descriptions of the MSBR are included.

*MSBR + *progress report + *conceptual design + *reactors +
 *MSBR + *heat exchangers + *pumps + *steam generators +
 *physical properties + *control + *gas injection +
 *gas separation + *performance + *heat transfer + *cells +
 *test facilities + thermal insulation + bubbles +
 mass transfer + thermal conductivity + structures +
 welding + maintenance + cores + reactor vessel +
 noble metals + fission products + neutron physics + xenon +
 krypton + fluoroborates + decay + heat + graphite +
 *steam systems

OTHER CATEGORIES: IAD

ACB690029

Briggs RB

MSBR DESIGN AND DEVELOPMENT (PART 2 MSRP SEMIANN PROG REPT
8-31-69)

Oak Ridge National Laboratory, Tenn.

ORNL-4449 (Feb. 1970) pp 39-95, 41 fig, 12 tables, 38 ref.

Conceptual study of a single-fluid 1000 MW (e) reference design MSBR is essentially complete. Principal design data are tabulated. The plant layout was revised to include a domed confinement building which provides missile protection and acts as containment during maintenance. A waste storage cell is also provided. Seismic disturbances were considered in the design. Layout drawings are shown for all building

Accession Number ACB690022 to ACB690029

Category A
Molten-Salt Reactor Programs

ACB69C029 *Continued*

levels. The primary drain tank was revised to use a lithium-beryllium fluoride salt-to-water-to-air cooling system. Nuclear calculations were refined to include effect of plant size and to consider alternate reactor designs. Gamma and neutron heating was calculated for the reference design geometry and also for an MSBE with spherical vessel. The industrial program to develop a steam generator is discussed. Results of operation of the sodium fluoroborate test are reported. The requirements for the MSBE salt pump test stand are covered. Results of heat transfer and salt physical property studies are reported in some detail. The mass transfer test facility is completed and experimental work started.

*MSBR + *progress report + *conceptual design + *MSFE + *pumps + *steam generators + *drain tanks + *physical properties + *heat transfer + *test facilities + *performance + *containment + *cells + mass transfer + structures + welding + maintenance + *control + neutron physics + gas injection + gas separation + single-fluid reactors + primary salt + thermal conductivity + capture + absorption + earthquakes + dynamic characteristics + radiation heating + layout + flowsheets + data + waste disposal

OTHER CATEGORIES: IAD

ACB700022

Briggs RB

MSBR DESIGN AND DEVELOPMENT (PART 2 MSRP SEMIANN PROG REPT 2-28-70)

Oak Ridge National Laboratory, Tenn.

ORNL-4548 (Aug. 1970) pp 41-92, 25 fig, 15 tables, 32 ref.

Studies of the reference design for 1000 MW(e) single-fluid MSER were completed and the first draft of a report circulated. Principal design data are presented. Studies are being made of first-generation types of molten-salt reactors that would have poorer performance but would require less development, including a large MSRE type and a spherical reactor with graphite ball bed. A primary heat exchanger with bayonet tubes is compared to the reference design exchanger. The tritium distribution in a 1000 MW(e) MSBE was estimated and the effectiveness of various methods of reducing the amounts reaching the steam system were calculated. The nuclear physics calculations were refined, including estimates of the control rod worth. The steam generator development program is discussed and further tests from the sodium fluoroborate test loop reported. The pump test stand is described and the remotely-operated orbital welder for piping discussed in some detail. Simulation studies of dynamic response of MSBR controls systems are presented. Development work was continued on

Accession Number ACB69C029 to ACB700022

Category A
Molten-Salt Reactor Programs

ACB700022 *Continued*

gas bubble generation and separation from the fuel salt. Better values for the thermal conductivity of the salt and for heat transfer relationships were obtained from the experimental results.

*MSBR + *progress report + *conceptual design + *MSRE + *pumps + *steam generators + *converters + *physical properties + *heat transfer + *test facilities + *performance + *control + welding + mass transfer + thermal conductivity + maintenance + neutron physics + gas injection + gas separation + dynamic characteristics + graphite + spheres + *tritium + development + components

OTHER CATEGORIES: IAD

ACB700036

Briggs RB

MSBR DESIGN AND DEVELOPMENT (PART 2 MSRE SEMIANN PROG REPT 8-31-70)

Oak Ridge National Laboratory, Tenn.

ORNL-4622 (Jan. 1971) pp. 7-59, 43 fig, 11 tables, 35 ref.

With completion of the report draft on the single-fluid MSBR, ORNL directed major attention to MSRE technical problems but some studies continued on a demonstration plant and plans progressed for an industrial study of a large MSER station. Flowsheets and layout drawings are shown for a 300 MW(e) demonstration plant with low enough power density for the graphite to not require replacement. The primary heat exchangers are mounted horizontally to permit maintenance from the side, and detailed afterheat studies on an empty exchanger are reported. The drain tank uses a natural convection NaK cooling system. Nuclear physics studies continued on cores of low power density and for Th concentrations in the 10-18 mole % range with fuel cycle costs and yields tabulated. Batch processing was also considered. Capture cross section ratios for alpha for U-235 were determined experimentally. Bubble generator and gas separator testing is reported. Plans progressed for industrial study of a steam generator. The sodium fluoroborate loop testing included water injection with inconclusive results. Remote welder development emphasized consistently good welds without direct observation or manual adjustment. Partial load steady-state behavior of MSER was studied. Heat transfer tests and investigation of thermophysical properties continued. Data on transfer coefficients to helium bubbles are reported.

*MSBR + *MSBE + *progress report + *conceptual design + *physical properties + *gas separation + *heat transfer + *test facilities + *industrial studies + *performance + *neutron physics + *welding + mass transfer + reactors + heat exchangers + drain tanks + structures + layout + maintenance + control + decay + fission products +

Accession Number ACB700022 to ACB700036

Category A
Molten-Salt Reactor Programs

ACB700036 *Continued*
graphite + pumps + containment + earthquakes +
fluoroborates + bubbles + gas separation + components +
development
OTHER CATEGORIES: IAC

ACB710029
Briggs RB
MSBR DESIGN AND DEVELOPMENT (PART II, MSRP SEMIANN ERG
REPT 2/28/71)

Oak Ridge National Laboratory, Tenn.

CRNL-4676 (Aug. 1971), pp 21-72, 27 fig, 52 ref.

Conceptual design of a 1000-MW MSBR was completed and design studies of a large, 300-MWe demonstration reactor were started. Flowsheets, layouts and component design data for this reactor are presented. Conceptual design of a high-power-density, 150-MWth, molten-salt breeder experiment (MSBE) also was pursued to define development requirements. Development efforts focussed on coolant system technology and the removal and handling of gaseous fission products from the fuel. Plans progressed for industrial studies of steam generators and a 1000-MW MSBF plant.

*conceptual design + *development + *MSBE + *MSBF +
analysis + converters + coolants + design data +
flowsheets + gas separation + industrial studies +
progress report + steam generators + tritium
OTHER CATEGORIES: HEX + IAD + IAE

ACC650006
Lindauer RB
FUEL PROCESSING (PART 7 MSRP PROG REPT 2/28/65)
Oak Ridge National Laboratory, Tenn.
CRNL-3812 (June 1965), pp. 169-171, 2 fig.

The design, procurement and construction of the MSRE fuel processing system were essentially completed except for the salt sampler and the uranium absorption equipment. An electrolytic hygrometer is being tested for in-line monitoring of the removal of oxide from molten salt by treatment with hydrogen and hydrogen fluoride. Initial results are encouraging, but they indicate that hydrogen fluoride will have to be completely removed from the gas that is bypassed to the analyzer. Study of methods for the removal of volatilized chromium fluoride from the offgas stream during fluorination of molten salt has begun. Some data have been obtained for the sorption of chromium trifluoride on sodium fluoride pellets at 400 deg C.

*MSRE + *processing + *construction + absorption +
corrosion products + design + hydrogen compounds + oxides +
sodium fluoride + uranium
OTHER CATEGORIES: LHX + MBX

ACC650012

Category A
Molten-Salt Reactor Programs

ACC650012 *Continued*

Lindauer RB

FUEL PROCESSING (PART 7 MSRP PROG REPT 8/31/65)

Oak Ridge National Laboratory, Tenn.

ORNL-3872 (December 1965), p 152, 3 ref.

Construction of the MSRE fuel-processing system was completed, the system was tested, and the flush salt was processed for oxide removal. Operation of the plant was generally satisfactory, and about 115 ppm of oxide was removed from the salt in reducing the concentration to about 50 ppm.

*MSRE + *oxides + *processing + construction + operation + plant

OTHER CATEGORIES: LHX + MCD

ACC660010

(Staff Report)

MOLTEN-SALT REACTOR PROCESSING STUDIES (Part 7 MSRE Progr.

Rept 2/28/66)

Oak Ridge National Laboratory, Tenn.

ORNL-3936 (June 1966) pp. 193-211, 10 fig, 6 ref.

A close-coupled facility for processing the fuel and fertile streams will be an integral part of an MSRE system. Fuel will be processed on a 40-day cycle. Uranium will be fluorinated from the carrier salt which will then be recovered from fission products by distillation. Relative volatilities between lithium and rare earths have been measured to be 0.001 to 0.04 at 900 to 1050 deg C. Uranium hexafluoride will be absorbed in fuel salt containing uranium tetrafluoride and then reduced with hydrogen. Fluorinator corrosion can probably be eliminated by a layer of frozen salt on the wall. Experimental work with a small countercurrent continuous fluorinator gave recoveries of 90 to 96% of the uranium. Volatile chromium fluorides can be trapped with negligible uranium losses on sodium fluoride beds. A preliminary design study of the above facility has illuminated problems among which is handling high-heat-generating materials. The fixed capital cost for the conceptual plant was \$5.3 million; the salt inventory cost was \$0.196 million, and the direct operating cost was \$787,790 per year.

*MSBR + *processing + corrosion protection + costs + design + distillation + fluorination + lithium + rare earths + sodium fluoride + uranium + volatility + two-fluid reactor

OTHER CATEGORIES: LJX

ACC660016

(Staff Report)

MOLTEN-SALT REACTOR PROCESSING STUDIES (PART 9 MSRP PROG

REPT 8/31/66)

Category A
Molten-Salt Reactor Programs

ACC660016 *Continued*

Oak Ridge National Laboratory, Tenn.

ORNL-4037 (January 1967), pp. 227-237, 4 fig, 2 ref.

The MSBR processing plant would use cycle times of 40 days for the fuel salt and 20 days for the fertile salt. Using a recirculating equilibrium still relative volatilities have been obtained which are a factor of 50 lower than using a cold-finger technique. Uranium recoveries exceeding 99% have been attained with continuous fluorinators only 48 in. high. Corrosion protection by means of a frozen wall is being studied. Studies continued on alternative processing methods to replace vacuum distillation. Tests were made with the reduction-coprecipitation process using beryllium and with the liquid-metal extraction process using solutions of lithium in bismuth.

*MSBR + *processing + beryllium + bismuth +
distillation + flowsheets + fluorination +
lithium + reductive extraction process + volatility +
corrosion protection

OTHER CATEGORIES: LJX

ACC670018

(Staff Report)

MOLTEN-SALT REACTOR PROCESSING STUDIES (PART 10 MSRP PROG

REPT 2/28/67)

Oak Ridge National Laboratory, Tenn.

ORNL-4119 (July, 1967), pp. 204-213, 5 fig, 2 ref.

Studies on the fluorination-distillation flowsheet for MSBR processing continued. Fluorination studies with nonprotected systems using 1-in.-diam towers have demonstrated steady state recoveries up to 99.9% of the uranium with fluorine utilization of 15%. Studies on column protection involve the construction of a 5-in.-diam. nickel tower with provision to generate heat fluxes to create a frozen wall of salt. Relative volatilities measured at 1000 deg C and 0.5 mm mercury pressure were 3×10^{-3} rd, 3×10^{-4} th, 6×10^{-4} th and 2×10^{-4} th for cerium, lanthanum, neodymium and samarium trifluorides with respect to lithium fluoride. Equipment is being fabricated for the distillation of 48 liters of MSRE fuel salt after removal of the uranium by fluorination.

*MSBR + *processing + distillation + fluorination +
fuels + MSRE + rare earths + uranium + volatility

ACC670025

(Staff Report)

MOLTEN-SALT PROCESSING AND PREPARATION (PART 6 MSRE PROG

REPT 8/31/67)

Oak Ridge National Laboratory, Tenn.

ORNL-4191 (December 1967), pp 239-253, 6 fig, 10 ref.

Most of the effort in this period was on the distillation

Accession Number ACC660016 to ACC670025

Category A
Molten-Salt Reactor Programs

ACC670025 *Continued*

step in the fluorination-distillation flowsheet. Relative volatilities of rare earth fluorides were measured using both an equilibrium still and the transpiration method. Data from the two methods is in good agreement. Equipment for demonstration of vacuum distillation using MSRE fuel salt is being installed in a test facility for non-radioactive experiments before operation with MSRE salt.

A computer code has been prepared to provide information on fission product heat generation. An alternative process to distillation, reductive extraction of rare earths using lithium reductant in bismuth is being studied. Modifications are being made to the MSRE fuel processing facility to permit uranium recovery by fluorination after only 35 days decay. Design and equipment fabrication is in progress for preparing 40 kg of uranium-233 as the uranium-lithium fluoride eutectic for replacement of the present MSRE uranium fuel.

*MSBR + *processing + *distillation + volatility + rare earths + reductive extraction process + MSRE + uranium-233 + fluorination + lithium + bismuth
OTHER CATEGORIES: LCA

ACC680014

(Staff Report)

MOLTEN SALT PROCESSING AND PREPARATION (PART 6 MSRP PROG
REPT 2/29/68)

Oak Ridge National Laboratory, Tenn.

ORNL-4254 (August 1968) pp. 241-277, 18 fig, 18 ref.

Distribution coefficients were measured for uranium, thorium and rare earths between molten fluoride salts and lithium-bismuth solutions. Calculations were made for the isolation of protactinium from a single-fluid MSBR. Studies are underway on protecting a continuous fluorinator from corrosion by freezing a layer of salt on the vessel wall. Relative volatility measurements were made for uranium, rubidium, caesium and zirconium fluorides with respect to lithium fluoride. Four non-radioactive test runs were made with fuel carrier salt in the distillation unit to be used with the MSRE fuel salt. Reductive extraction processes for protactinium removal were evaluated. Small scale fluorination tests were made with simulated MSRE fuel salt. Preparation of the uranium-233 fuel concentrate for the MSRE is underway. Decay heat from fission products and protactinium has been calculated for a 2000-Mw single-region MSR.

*MSBR + *processing + distillation + distribution + fluorination + MSRE + rare earths + reductive extraction process + protactinium + thorium + uranium + uranium-233 + volatility

ACC680021

Accession Number ACC670025 to ACC680021

Category A
Molten-Salt Reactor Programs

ACC680021 *Continued*
(Staff Report)
MOLTEN SALT PROCESSING AND PREPARATION (PART 6 MSRE PFCG
REPT 8/31/68)

Oak Ridge National Laboratory, Tenn.

ORNL-4344 (February 1969), pp. 291-326, 22 fig, 13 ref.

Measurement of distribution coefficients for the reductive extraction process continued. The solubility of protactinium and thorium in bismuth was determined. Simulated molten salt-liquid bismuth contactor studies were started with mercury and water. Equipment is being installed for semicontinuous experiments on reductive extraction. A series of experiments was concluded which demonstrated the feasibility of a frozen salt wall for corrosion protection during fluorination. Equipment is being installed at the MSRE for demonstration of fuel salt distillation. Relative volatility measurements were made with thorium fluoride. Preparation of the uranium-233 fuel concentrate for the MSRE was completed using the two-step process. Development of this process is described. Final laboratory tests were made on several steps in the process for recovery of uranium from the MSRE (described in Part 1 of this report).

*MSBR + *processing + corrosion protection + distillation +
distribution + fluorination + protactinium +
reductive extraction process + MSRE + thorium + uranium +
uranium-233 + volatility

ACC690023
(Staff Report)
MOLTEN-SALT PROCESSING AND PREPARATION (PART 6 MSRE PROG
REPT 2/28/69)

Oak Ridge National Laboratory, Tenn.

ORNL-4396 (August 1969) pp. 270-299, 25 fig, 33 ref.

The proposed reductive extraction processing flowsheet for a single-fluid MSBR is described. Protactinium and rare-earth removal is included. A computer code has been developed to perform the necessary material balance calculations. The measurement of distribution coefficients for the reductive extraction process continued. The mutual solubilities of nickel and thorium in bismuth were determined. Experiments were carried out using quartz electrolytic cells. Preliminary testing of equipment for semi-continuous experiments on reductive extraction has begun. Data is reported on mercury-water tests in columns to simulate molten-salt-liquid-bismuth. Cold testing of the distillation unit at the MSRE was completed prior to distilling a portion of the MSRE fuel salt.

*MSBR + *processing + bismuth + computer codes +
distillation + distribution + electrolysis + flowsheets +
protactinium + reductive extraction process + MSRE +
solubility + thorium

Accession Number ACC680021 to ACC690023

Category A
Molten-Salt Reactor Programs

ACC690023 *Continued*
OTHER CATEGORIES: LKX

ACC690030
(Staff Report)
MOLTEN-SALT PROCESSING AND PREPARATION (PART 6 MSRP PRG
REPT 8/31/69)

Oak Ridge National Laboratory, Tenn.

ORNL-4449 (Feb. 1970) pp. 214-246, 27 fig. 26 ref.

Measurement of distribution coefficients in molten salt-metal systems continued and data is presented for transuranium elements, rare earths and thorium. The solubility of plutonium fluoride was measured in lithium-beryllium fluoride salt. Flowsheet analyses were made of protactinium isolation, rare-earth removal, thorium stripping, fission product concentrations and heat generation rates. Four runs were made with the semi-continuous system for contacting bismuth with molten salt. Electrolytic cell and salt-metal contactor development continued. Axial mixing was studied in both packed and bubble columns. About 11 liters of the MSRP fuel carrier salt was distilled. Design studies were carried out on: (1) heat transfer through the frozen salt walls of an electrolytic cell, (2) a continuous salt purification system and (3) plutonium capsules for refueling the MSRE.

*MSRP + *processing + bismuth + columns + design + distillation + distribution + electrolysis + flowsheets + heat transfer + MSRE + plutonium + protactinium + reductive extraction process + solubility + rare earths + thorium

ACC700023
(Staff Report)
MOLTEN-SALT PROCESSING AND PREPARATION (PART 6 MSRP PRG
REPT 2/28/70)

Oak Ridge National Laboratory, Tenn.

ORNL-4548 (August 1970) pp. 277-332, 42 fig, 33 ref.

A new processing flowsheet for a single-fluid MSRP is described. Electrolytic cells are eliminated by the use of a metal transport process for removing rare earths and fluorination followed by reductive extraction for protactinium isolation. Distribution of thorium and rare earths between lithium chloride and bismuth is being studied in support of the metal transfer process. Four more runs were made with the semicontinuous system for contacting bismuth with molten salt. Equipment is being prepared for a demonstration of the metal transport process. Contactor and electrolytic cell development is continuing. Data from the distillation of MSRP fuel carrier salt is presented. Material and energy balance calculations and calculations on the effect of chemical processing on

Category A
Molten-Salt Reactor Programs

ACC700023 *Continued*

nuclear performance were made for the MSBR processing plant. Installation of continuous salt purification equipment is in progress. Specially designed capsules were loaded with plutonium fluoride and added to the MSRE fuel salt.

*MSBR + *processing + bismuth + columns + design + distillation + distribution + electrolysis + flowsheets + metal transfer process + MSRE + plutonium + rare earths + reductive extraction process

OTHER CATEGORIES: LKX

ACC700037

(Staff Report)

MOLTEN-SALT PROCESSING AND PREPARATION (PART 5, MSRP SEMIANN PROG REPT 8/31/70)

Oak Ridge National Laboratory, Tenn.

ORNL-4622 (Jan. 1971), pp 199-224, 20 fig, 27 ref.

Calculations were made for a flowsheet using fluorination-reductive extraction for Pa isolation and metal-transfer for rare-earth removal. Calculations were also made on removal of uranium by oxide precipitation. More data were obtained on distribution of rare earths and thorium between bismuth solutions and molten salts. Engineering development included operation of the flow-through reductive-extraction facility, tests on performance of packed columns with two liquids differing widely in density, demonstration of the metal-transfer process, and experiments with electrolytic cells. Tests of a continuous salt purification system were started.

*development + *experiment + *MSBR + *processing + bismuth + chlorides + columns + data + distribution + electrolysis + flowsheets + fluorination + metal transfer process + oxide precipitation process + reductive extraction process

ACD650007

(Staff Report)

RADIATION CHEMISTRY (CHAP 5, MSRP SEMIANN PROG REPT 2/28/65)

Oak Ridge National Laboratory, Tenn.

ORNL-3812 (June 1965), pp 87-120, 25 fig, 5 ref.

In-pile capsule tests in the MTR were completed and post-irradiation examinations at CRNL were practically finished. Early tests had showed effects of fluorine evolution. Later tests, which included gas connections and external heating during reactor shutdown, proved that the fuel was stable, with no fluorine evolution, under operating conditions. (Radiolysis of cold salt had produced the gaseous fluorine.) Examination of salt, graphite and INOR-8 from the capsules showed no radiation-induced incompatibility.

*examinations + *in-pile tests + *radiolysis +

Accession Number ACC700023 to ACD650007

Category A
Molten-Salt Reactor Programs

ACD650007 *Continued*
capsules + compatibility + fluorine + graphite +
Hastelloy N + molten salts + progress report

ACD650011

(Staff Report)

CHEMISTRY (CHAP 6, MSRP SEMIANN PROG REPT 8/31/65)

Oak Ridge National Laboratory, Tenn.

CRNL-3872 (Dec. 1965), pp 111-151, 16 fig, 40 ref.

Analyses of MSRE salts during precritical testing, U-235 loading, and zero-power experiments showed that purity was maintained and corrosion was very low. Vapor pressures, HF solubility, and iodine removal in LiF-Bef₂ systems were determined. Phase relations in the NaF-NaBF₄ system and viscosity of NaBF₄ were determined. (This system is suggested as an inexpensive, lower-melting breeder coolant.) Preparations were made for studying Pa oxide precipitation. Efforts continued to improve analytical methods for MSRE salts and cover gas.

*analytical chemistry + *chemistry + *experience + *MSFE +
*progress report + data + experiment + fluorides +
fluoroborates + iodine + molten salts + oxides +
precipitation + protactinium

OTHER CATEGORIES: MCD + CXX + DXX

ACD650013

(Staff Report)

RADIATION CHEMISTRY (CHAP 5, MSRP SEMIANN PROG REPT 8/31/65)

Oak Ridge National Laboratory, Tenn.

CRNL-3872 (Dec. 1965), pp 106-110, 2 fig, 2 ref.

Design and development progressed on an in-pile molten-salt experiment to go in the ORR. It consists of a compact thermal-circulation loop of INOR-8 including a 2-inch graphite core and 85 cc of fuel salt.

description + development + in-pile tests

ACD660011

(Staff Report)

CHEMISTRY (CHAP 5, MSRP SEMIANN PROG REPT 2/28/66)

Oak Ridge National Laboratory, Tenn.

CRNL-3936 (June 1966), pp 122-171, 25 fig, 32 ref.

Improved analytical methods applied to MSRE fuel samples showed no anomalies and excellent purity. Flugging material in the offgas line proved to be oil decomposition products. Studies of physical chemistry of molten fluoride and the chemistry of Pa and fission product extraction continued. The latter included distillation, reductive extraction into liquid metals and oxide precipitation. Fabrication progressed on a molten-salt loop to go in the ORR.

*analytical chemistry + *chemistry + *experience + *MSFE +

Category A
Molten-Salt Reactor Programs

ACD660011 *Continued*

data + distillation + in-pile tests + liquid metals +
molten salts + off-gas systems + oxides + precipitation +
protactinium + rare earths + reduction

OTHER CATEGORIES: MCD + DXX + ICA + IDA + CXX

ACD660017

(Staff Report)

CHEMISTRY (CHAF 7, MSRP SEMIANN PROG REPT 8/31/66)

Oak Ridge National Laboratory, Tenn.

ORNL-4037 (Aug. 1966), pp 134-200, 24 fig, 42 ref.

The table of contents is as follows. Behavior of Fuel and Coolant Salts in MSRE. Physical Chemistry of Fluoride Melts; Viscosity and Density of Molten Beryllium Fluoride; Transpiration Studies in Support of the Vacuum Distillation Process; Estimated Thermophysical Properties of MSRE Coolant Salt. Separation in Molten Fluorides: Extraction of Rare Earths from Molten Fluorides into Molten Metals; Removal of Rare Earths from Molten Fluorides by Simultaneous Precipitation with UF₃; Removal of Protactinium from Molten Fluorides by Oxide Precipitation; Extraction of Protactinium from Molten Fluorides into Molten Metals; Protactinium Studies in the High-Alpha Molten-Salt Laboratory. Radiation Chemistry: Xenon Diffusion and Possible Formation of Cesium Carbide in an MSRE; Fission Product Behavior in the MSRE. Development and Evaluation of Analytical Methods for Molten-Salt Reactors: Determination of Oxide in Radioactive MSRE Samples; Spectrophotometric Studies of Molten-Salt Reactor Fuels; Voltammetric and Chronopotentiometric Studies of Uranium in Molten LiF-BeF₂-ZrF₄; In-Line test Facility; Analysis of Helium Blanket Gas. Development and Evaluation of Equipment and Procedures for Analyzing Radioactive MSRE Salt Samples: Samples Analyses; Quality-Control Program.

*analytical chemistry + *beryllium fluoride + *capsules +
*experiment + *fission products + *graphite +
*hydrocarbons + *MSRE + *noble metals + *oxides +
*physical properties + *rare earths + *xenon + actinides +
analysis + behavior + boron trifluoride + carbides + cells +
circulation + compatibility + concentration + cores +
corrosion + corrosion products + cover gas + decay +
density + deposition + diagrams + dissolving + distillation +
electrical properties + entrainment + equilibrium +
examinations + fission + fluorides + fluoroborates +
gamma radiation + gamma spectrometry + gases + Hastelloy N +
inert gases + lithium fluoride + molten salts +
oxide precipitation process + phase equilibria +
progress report + protactinium + protactinium fluorides +
rare gases + research + sampling + specific heat +
uranium fluorides + vapor pressure + viscosity

ACD670019

Accession Number ACD660011 to ACD670019

Category A
Molten-Salt Reactor Programs

ACD67C019 *Continued*

(Staff Report)

CHEMISTRY (CHAP. 7, MSRP SEMIANN. PROGR. REPT. 2/28/67)

Oak Ridge National Laboratory, Tenn.

ORNL-4119, (July 1967), pp. 118-166, 13 fig, 39 ref.

The following topics are included in the table of contents: Chemistry of the MSRE, Fuel salt composition and purity, MSRE fuel circuit corrosion, Extent of UF₄ reduction during MSRE Fuel preparation, Adjustment of the UF₃ concentration in the MSRE fuel salt, Fission product behavior in the MSRE, Long-term surveillance specimens, Uranium analyses of graphite specimens, Fuel salt samples, Effect of operating conditions. Effect of beryllium additions, Pump bowl volatilization and plating tests, Uranium on pump bowl metal specimens, Freeze valve capsule experiments, Special pump bowl tests, General discussion of fission product behavior, Physical chemistry of fluoride melts, The oxide chemistry of LiF-B₂F₆-ZrF₄ mixtures, Solubilities of SmF₃ and NdF₃ in Molten LiF-B₂F₆ (66-34 mole %), Possible MSBR blanket-salt mixtures, Separations in molten fluorides, Removal of rare earths from molten fluorides by precipitation on solid UF₃, Extraction of protactinium from molten fluorides into molten metals, Extraction of rare earths from molten fluorides into molten metals, Protactinium studies in the high-alpha molten-salt laboratory, Preliminary study of the system LiF-ThF₄-PaF₄, Development and Evaluation of analytical methods for molten-salt reactors, Determinations of oxide in MSRE salts, Determination of U(3+)/U(4+) ratios in radioactive fuel by a hydrogen reduction method, EMF measurements on the Nickel-Nickel(II) couple in Molten fluorides, Studies of the anodic uranium wave in molten LiF-B₂F₆-ZrF₄, Spectrophotometric studies of molten-salt reactor fuels, Analytical chemistry analyses of radioactive MSRE fuels, Sample analyses, Quality control program.

*analysis + *analytical chemistry + *cells + *chemistry +
 *fuels + *graphite + *noble metals + *oxides +
 *protactinium + *protactinium fluorides +
 *reductive extraction process + *sampling +
 *surveillance + beryllium fluoride + bismuth + capsules +
 chemical properties + chemical reactions + compatibility +
 concentration + corrosion + corrosion products +
 electrolysis + gamma spectrometry + hot cells +
 in-pile tests + liquid metals + lithium fluoride +
 materials + materials testing + metals + molybdenum +
 MSRE + nickel + nickel alloys + progress report +
 rare earths + secondary salts + testing +
 thorium fluorides + uranium fluorides + zirconium fluoride
 OTHER CATEGORIES: CXX + LXX + MCD

ACD67C020

Accession Number ACD67C019 to ACD67C020

Category A
 Molten-Salt Reactor Programs

ACD67C020 *Continued*
 (Staff Report)
 CONVECTION LOOP IN ORR (CHAP. 8, MSRP SEMIANN. PROGR.
 REPT. 2/28/67)

Oak Ridge National Laboratory, Tenn.

ORNL-4119 (July 1967), pp. 167-173, 0 fig, 4 ref.

Irradiation of the first molten salt convection loop experiment in the Oak Ridge Research Reactor was terminated Aug. 8, 1966, after development of 1.1×10^{16} fissions/cc (0.27% U-235 burnup) in the $\text{LiF-FeF}_2\text{-ZrF}_4\text{-UF}_4$ (65.16-28.57-4.90-1.36 mole %) fuel. Average fuel power densities of up to 105 w/cc were attained in the core, which was made of MSRE Grade graphite. The table of contents for the report on these experiments contains the following topics: Objectives and description, First loop experiment, In-pile irradiation assembly, Operations, Chemical analysis of salt, Corrosion, Fission products, Nuclear heat and neutron flux, Hot-cell examination of components, Evaluation of system performance, Heaters, Coolers, Temperature control, Sampling and addition, Salt circulation, Second in-pile irradiation assembly, Operation.

*experiment + *fission products + *gamma radiation +
 *graphite + *Hastelloy N + *leaks + noble metals +
 *thermal convection + actinides + analysis +
 beryllium fluoride + circulation + compatibility + cores +
 corrosion + decay + dismantling + examinations + fission +
 fluorides + molten salts + gamma spectrometry + hot cells +
 lithium fluoride + materials testing + MSRE +
 progress report + radiation damage + research + sampling +
 stress + stress rupture + uranium fluorides + uranium-235

ACD67C026

Grimes WR

CHEMISTRY (PART 3, MSRP SEMIANN. PROGR. REPT. 8/31/67)

Oak Ridge National Laboratory, Tenn.

ORNL-4191 (Dec. 1967), pp. 102-175, 39 fig, 66 ref.

Sampling of the MSRE fuel and coolant salts is described and the analyses are interpreted. Results from examinations of metal and graphite surveillance specimen from the core and of specimen exposed to pump bowl gases are presented. The fact that metallic fission products appear in the cover gas prompted a study of the chemistry and volatilization behavior of the little-known intermediate valence fluorides of molybdenum. Oxide fluoride equilibria in fuel systems was studied in a research for separation processes. Phase behavior, decomposition pressure and corrosiveness of fluoroborate coolants is described. Recovery of protactinium and removal of fission products by reductive extraction is discussed. Developmental studies in analytical chemistry directed primarily to improvements in analyses of

Accession Number ACD67C02C to ACD67C026

Category A
Molten-Salt Reactor Programs

ACD67C026 *Continued*

radioactive samples of fuel for oxide and uranium trifluoride and for impurities in helium offgas from the MSRE.

*analytical chemistry + *boron trifluoride + *coolants + *corrosion + *fission products + *fluoroborates + *molten salts + *graphite + *oxides + *rare earths + *reductive extraction process + analysis + behavior + beryllium fluoride + bismuth + blanket + capsules + cells + chemical reactions + chemistry + chromium + compatibility + cores + corrosion products + cover gas + decomposition + distribution + equilibrium + equipment + experiment + fissile materials + gas analysis + gases + helium + hot cells + hydrocarbons + impurities + inert gases + inventories + irradiation + liquidus + materials + materials testing + mists + molybdenum + MSRE + off-gas systems + oxide precipitation process + phase equilibria + physical properties + protactinium + protactinium fluorides + radiolysis + rare gases + reduction + sampling + solubility + solidus + spectrophotometry + stability + surveillance + testing + thorium + thorium fluorides + uranium + uranium fluorides + vapor pressure

OTHER CATEGORIES: CXX + LXX + MCD

ACD67C027

Bohlmann EG

IRRADIATION EXPERIMENTS (PART 4, MSRE SEMIANN. REPORT. FEPT. 3/31/67)

Oak Ridge National Laboratory, Tenn.

ORNL-4191 (Dec. 1967) pp. 176-195, 10 fig, 7 ref.

A second thermal convection in-pile loop containing fission fuel was terminated when a crack developed in the core outlet pipe. The crack was caused by radiation embrittlement of the Hastelloy N and stresses encountered during a reactor setback. Sufficient operating time had, however, been achieved to produce fission product concentration levels equivalent to equilibrium in a breeder; therefore an exhaustive evaluation of the experiment is presented.

*fission products + *gamma radiation + *graphite + *Hastelloy N + *in-pile tests + *nickel metals + *thermal convection + actinides + analysis + analytical chemistry + beryllium fluoride + circulation + compatibility + cores + corrosion + corrosion products + decay + dismantling + embrittlement + examinations + experiment + fission + fluorides + molten salts + gamma spectrometry + hot cells + leaks + lithium fluoride + materials testing + molybdenum + MSRE + progress report + radiation damage + rare earths + research + sampling + stress + stress rupture + uranium fluorides

ACD680015

Accession Number ACD67C026 to ACD680015

Category A
Molten-Salt Reactor Programs

ACD680015 *Continued*

Grimes WR

CHEMISTRY (PART 3, MSRP SEMIANN. PROGR. REPT. 2/29/68)

Oak Ridge National Laboratory, Tenn.

ORNL-4254 (Aug. 1968), pp. 88-173, 55 fig, 102 ref.

The chemistry of the MSRE is discussed from the standpoint of fuel composition and purity, corrosion chemistry, and isotopic composition of the uranium in the fuel. Fission product behavior in the fuel and in the cover gas is described. Results on fission products found on samples of graphite and metal from the core are given. Other topics are "Proton Reaction Analysis for Lithium and Fluorine in MSR Graphite" and "Surface Phenomena in Molten Salts". Items pertaining to the physical chemistry of molten salts are the thermodynamics of LiF-BeF₂ melts from EMF measurements, and electrical properties of melts. The chemistry of silica in LiF-BeF₂ melts is presented. A Molten Salt Chemistry Information Center is described. Synthesis and properties of molybdenum and niobium fluorides is discussed. Reprocessing of fuel by reductive extraction into molten bismuth is described, with special emphasis on protactinium recovery. The behavior of BiF₃ and fluoroborate mixtures is examined from the standpoint of phase relations, non-ideality, thermodynamics, corrosion and compatibility.

actinides + beryllium + beryllium fluoride + bubbles +
 capsules + cells + concentration + coolants + corrosion +
 cover gas + decomposition + distribution +
 electrical properties + electrical conductivity +
 electrolysis + equilibrium + examinations + experiment +
 fissile materials + fuel preparation + gamma spectrometry +
 Hastelloy N + heat transfer + hot cells +
 hydrofluorination + hydrogen + in-pile tests +
 interfacial tension + irradiation + liquidus +
 lithium fluoride + materials + mists + MSRE +
 off-gas systems + oxide precipitation process + oxides +
 oxidation + phase equilibria + progress report +
 protactinium + protactinium fluorides + rare gases +
 reaction rates + reduction + research + sampling +
 sodium fluoride + solubility + specific heat +
 test facilities + testing + thermal properties + thorium +
 thorium fluorides + uranium + uranium fluorides +
 vapor pressure + viscosity + volatility +
 zirconium fluoride + *bismuth +
 *boron trifluoride + *chemical properties +
 *chemical reactions + *chemistry + *compatibility +
 *corrosion products + *fluorides + *fluoroborates +
 *molten salts + *graphite + *noble metals +
 *physical properties + *processing + *rare earths +
 *reductive extraction process
 OTHER CATEGORIES: CXX + MCD

ACD680016

Accession Number ACD680015 to ACD680016

Category A
Molten-Salt Reactor Programs

ACD68C016 *Continued*

Bohlmann EG

IRRADIATION EXPERIMENTS (PART 4, MSRE SEMIANN. PRCG. REPT.
2/29/68)

Oak Ridge National Laboratory, Tenn.

ORNL-4254 (Aug. 1968) pp. 174-182, 2 fig, 8 ref.

The isotope balance on a second in-pile convection loop containing fissioning fuel is given. Fission product behavior is described. In this loop the fuel wetted the graphite, presumably because of trace amounts of moisture present in the helium used in loading, sampling and draining. Accordingly a study was made of the effect of moisture on the wetting of graphite by MSRE carrier salt. Also presented is a design for a third in-pile molten salt convection loop.

actinides + cells + compatibility + cover gas +
experiment + fissile materials + fluorides + fuels +
gas analysis + gases + inert gases + laboratory equipment +
materials + noble metals + MSRE + sampling + *chemistry +
*examinations + *fission products + *molten salts +
*graphite + *in-pile tests + *thermal convection

ACD68C022

Grimes WB

CHEMISTRY (PART 3, MSRE PROGR. REPT. 8/31/68)

Oak Ridge National Laboratory, Tenn.

ORNL-4344 (Aug. 1968) pp. 109-199, 57 Fig, 134 Ref.

MSRE chemistry topics include Feasibility of Fueling with PuF_3 , Burnup, High Temperature Fuel-Graphite Compatibility, and Examination of a Corroded Sample Capsule. Fission Product Behavior is discussed in connection with Specimens from the Core, Analyses for Li and F, and Surface Phenomena in Molten Salts. Items under Physical Chemistry of Molten Salts include Molybdenum Fluoride Chemistry, Alkali Fluoroborates, Physical Properties of ThF_4 -containing Melts, Electrochemical Studies, Spectroscopy, Oxide Chemistry, and the Chemistry of Silica in LiF-BaF_2 . Fuel Reprocessing was studied in experiments on the Reductive Extraction of Pa and of Rare Earths into Bismuth. Analytical Studies included oxide determination, $\text{U}(3+)$ and total reducing power, $\text{U}(5+)$ in $\text{LiF-BaF}_2\text{-ZrF}_4$, $\text{Ni}(0)/\text{Ni}(+)$ couple, $\text{Cr}(2+)$, Hot Cell Spectrophotometer, Spectra of $\text{U}(5+)$ and $\text{U}(6+)$, a Gas Chromatograph for the Off-gas system, Hydrocarbons in MSRE Helium, Gamma Spectroscopy, Fission Product Penetration in Graphite, U-235 Analyses, and Determination of U.

*analytical chemistry + *bismuth + *chemical properties +
*chemical reactions + *chemistry + *compatibility +
*corrosion products + *fission products + *fluorides +
*fluoroborates + *molten salts + *graphite +
*physical properties + *processing +

Accession Number ACD680016 to ACD680022

Category A
Molten-Salt Reactor Programs

ACD68C022 *Continued*

*reductive extraction process + *solubility +
*surveillance + beryllium + beryllium fluoride + bubbles +
burnup + capsules + carbides + cells + chromium +
concentration + coolants + corrosion + cover gas +
density + distribution + electrical properties +
electrical conductivity + electrolysis + equilibrium +
examinations + expansion + experiment + fissile materials +
fuel preparation + gamma spectrometry + Hastelloy N +
heat transfer + hot cells + hydrofluorination + hydrogen +
hydrogen compounds + interfacial tension + liquidus +
lithium fluoride + materials + mists + MSRE +
noble metals + off-gas systems +
oxide precipitation process + oxides + oxidation +
phase equilibria + plutonium fluorides + precipitation +
progress report + protactinium + protactinium fluorides +
rare earths + rare gases + reaction rates + reduction +
research + sampling + sodium fluoride + solidus +
specific heat + spectrophotometry + test facilities +
testing + thermal conductivity + thermal properties +
thorium + thorium fluorides + uranium + uranium fluorides +
vapor pressure + viscosity + volatility +
zirconium fluoride

OTHER CATEGORIES: CXX + EXX + MCD

ACD68C023

Bohlmann EG

IRRADIATION EXPERIMENTS (PART 4, MSRE PROGR. REPT. 8/31/68)

Oak Ridge National Laboratory, Tenn.

ORNL-4344 (Aug. 1968), pp. 200-210, 1 fig, 8 ref.

Examinations of the graphite from an CHF convection loop showed the salt had wetted the graphite, contrary to previous experiences in very dry inert gases. Subsequent laboratory studies show that extremely minute concentrations of water (approximately 1 ppm) promote wetting at points of three phase contact of salt, graphite and gas. A second Hastelloy-N capsule containing NaBF₄-NaF (92-8 mole %) was irradiated for 1460 hr at 600 deg C in three successive spent HFIR fuel elements; no deleterious effects were observed. Fluorine due to the delayed neutrons by B-10F₃ + n to LiF + alpha + F₂ was deemed to be tolerably low. The jumper section of the MSRE off-gas line, 2 ft downstream from the pump bowl, was recovered for examination. All internal surfaces were covered with a thin, sootlike film, and no other deposits were found. A group of fission products, largely "noble metals" (Mo, Ru, Ag, Te) were present in quantities several hundred times the amounts expected from the inventory of salt present in the deposit; this substantiated earlier observations that metals could be transferred in the off-gas.

Accession Number ACD680022 to ACD680023

Category A
Molten-Salt Reactor Programs

ACD680023 *Continued*

*boron trifluoride + *fission products + *fluoroborates +
*in-pile tests + *noble metals + *off-gas systems +
*wetting + behavior + capsules + chemical reactions +
chemistry + compatibility + coolants + corrosion +
cover gas + delayed neutrons + molten salts +
gamma radiation + gamma sources + gases + hydrocarbons +
inert gases + inventories + materials + materials testing +
mists + MSRE + radiation damage + radiolysis + rare gases +
sodium fluoride + testing + uranium + uranium fluorides
OTHER CATEGORIES: MCD

ACD690024

Grimes WR

CHEMISTRY (PART 3. MSRP PROGR. REPT. 2/28/69)

Oak Ridge National Laboratory, Tenn.

ORNL-4396 (Feb. 1969), pp. 129-196, 32 fig, 122 ref.

MSRE chemistry topics include the uranium material balance, corrosion, adjustment of U(3+)/Sigma U, and foaming behavior. Fission product disposition in the MSRE is described under Examination of Graphite from the Core, Distribution of Fission Products, Fission Product Inventory, Off-gas Analyses, and Material Recovered from the off-gas line. Also the formation of aerosols from the MSRE was studied extensively in a hot cell; additionally tracer level studies were also made. The chemistry of the fluorides of Nb, Mo and Ru was studied by mass spectroscopy. Under Physical Chemistry of Molten Salts are 15 items dealing with such topics as CeF_3 (a standard for FuF_3) solubility, Zone Melting, Phase Relations, Solubility of Th(c) in $LiF-ThF_4$, Densities, Crystal Structure, Spectroscopy in a Diamond-Windowed Cell, Distribution of U(4+) between fuel and (U-Th) $_2$ Solid Solution, Reference Electrodes, Concentration Cells, Electrical Conductance. Chemistry in support of fuel reprocessing deals with reductive extraction of Zr, U, Pa, rare earths, and Th. Analytical Chemistry is represented by Determination of Oxide and Oxidation State, Eaf, Voltammetric, and Spectrographic Studies, Wetting Behavior, Contaminants in Blanket Gas from $NaEF_4$ tests, and the Determination of Bi in MSRP Salts.

*analytical chemistry + *bismuth + *chemical properties +
*chemical reactions + *chemistry + *corrosion products +
*examinations + *fission products + *fluorides +
*fluoroborates + *molten salts + *graphite + *liquidus +
*materials + *noble metals + *phase equilibria +
*processing + *reduction + *reductive extraction process +
*solidus + *solubility + *surveillance +
actinides + beryllium fluoride +
beryllium oxide + blanket + boron trifluoride + bubbles +
carbon + capsules + cells + compatibility + concentration +

Accession Number ACD680023 to ACD690024

Category A
Molten-Salt Reactor Programs

ACD690024 *Continued*

coolants + cover gas + density + deposition + diagrams +
distribution + electrical conductivity + electrolysis +
equilibrium + experiment + foaming + freezing +
fuel preparation + fuels + gamma spectrometry +
gas analysis + gases + hydrofluorination + hydrogen +
hydrogen compounds + inert gases + in-pile tests +
interfacial tension + ions + laboratory equipment +
lithium fluoride + melting + metal transfer process +
mists + MSRE + off-gas systems +
oxide precipitation process + oxides + oxidation +
physical properties + plutonium fluorides +
potassium fluorides + precipitation + protactinium +
protactinium fluorides + potassium fluorides +
precipitation + protactinium fluorides +
rare earths + rare gases + reaction rates + sampling +
secondary salts + sodium fluoride + spectrophotometry +
surface tension + thorium + thorium fluorides +
uranium + uranium fluorides + volatility + zirconium +
zirconium fluoride

OTHER CATEGORIES: CXX + DXX + MCD

ACD690025

Bohlmann EG

IRRADIATION EXPERIMENTS (PART 4, MSRP SEMIANN PROG REPT
2/28/69)

Oak Ridge National Laboratory, Tenn.

ORNL-4396 (Aug. 1969), p 210.

The program of in-pile molten-salt loops was suspended.
Laboratory experiments showed that the wetting of graphite
that was seen in the ORF molten-salt loop was probably
caused by traces of moisture in the gas used to transfer
salt.

experiment + graphite + inert gases + in-pile tests +
molten salts + progress report + wetting

ACD690031

Grimes WR

CHEMISTRY (PART 3, MSRP PROGR. REPT. 8/31/69)

Oak Ridge National Laboratory, Tenn.

ORNL-4449 (Aug. 1969), pp. 96-163, 29 fig, 108 ref.

MSRE Chemistry topics include the composition of the fuel,
plutonium material balance, and gas behavior. Fission
product behavior is deduced from surveillance specimens from
laboratory studies of metal fission product chemistry. A
measurement of the surface tension of the fuel in the
reactor is presented. Chemical and physical properties of
alkali fluoroborates are given under 10 topic headings.
Topics relating to the Physical Chemistry of Molten Salts
include phase relations, heterogeneous equilibria,
liquidus temperature, solubility of thorium, U(3+)/U(4+)

Accession Number ACD690024 to ACD690031

Category A
Molten-Salt Reactor Programs

ACD690031 *Continued*

ratio, spectrum of UF₃, concentration cells, electrical conductance, viscosity, and density. Items of interest in connection with fuel reprocessing are the reductive extraction of Pa, rare earths and thorium, and also the separation of zirconium as a platinide. In connection with analytical chemistry, there is oxide determination and removal, U(3+)/U(4+) determination, electroanalytical studies, spectral studies, hot cell spectrophotometry, and bismuth determination.

*analytical chemistry + *chemical properties +
 *chemical reactions + *chemistry + *coolants +
 *fission products + *fluorides + *fluoroborates +
 *molten salts + *graphite + *physical properties +
 *surveillance + beryllium +
 beryllium fluoride + bismuth + boron trifluoride +
 bubbles + cells + concentration + corrosion products +
 cover gas + density + distribution +
 electrical conductivity + equilibrium + examinations +
 fuel preparation + gamma spectrometry + helium +
 hot cells + hydrofluorination + hydrogen + impurities +
 inert gases + interfacial tension + kinetic equations +
 liquids + lithium fluoride + materials + melting +
 metal transfer process + noble metals +
 oxide precipitation process + oxides + phase equilibria +
 plutonium fluorides + potassium fluorides + precipitation +
 protactinium + protactinium fluorides + rare earths +
 rare gases + reaction rates + reduction +
 reductive extraction process + research + sampling +
 sodium fluoride + solidus + solubility +
 spectrophotometry + surface tension + testing +
 thorium fluorides + uranium fluorides + viscosity +
 void fractions + zirconium
 OTHER CATEGORIES: CXX + LXX + MCD

ACD700024

Grimes WR

CHEMISTRY (PART 3, MSRP PROGR. REPT. 2/28/70)

Oak Ridge National Laboratory, Tenn.

ORNL-4548 (Feb. 1970), pp. 93-187, 50 fig, 153 ref.

MSRE chemistry topics discussed are corrosion, appearance of Nb-95 in the fuel salt, isotopic composition of U and Pu, and surface tension and wetting behavior. Fission product behavior was demonstrated by samples from the core and from the pump bowl. Laboratory studies of the metals that are fission products are presented. Fourteen topics are discussed under Physical Chemistry of Molten Salts and six under Properties of the Alkali Fluoroborates. These include the oxide chemistry of Pu in molten fluorides, and the solubility of the corrosion product, Na₃CrF₆, in fluoroborate melts. Basic chemistry work in support of

Category A
Molten-Salt Reactor Programs

ACD70C024 *Continued*

fuel reprocessing included distribution of Ce, Eu, and Sr between bismuth and LiCl, and 7 other topics related to reductive extraction. The following analytical chemistry topics are presented: Determination of Oxide in MSRE Salt, Determination of U(3+)/Sigma U(4+) Ratios, Spectral Studies, Tritium in the Effluent Gases of the MSRE, Reference Electrodes in Molten Fluorides, Removal of Oxide from NaBF₄, Volatile AlCl₃ Complexes.

*chemical properties + *chemical reactions + *chemistry + concentration + *coolants + *corrosion + *fission products + *fluorides + *fluoroborates + *molten salts + *graphite + *materials + *oxides + *physical properties + *progress report + *rare earths + *surveillance + *tritium + analytical chemistry + barium + bismuth + compatibility + cesium + corrosion products + cover gas + distribution + electrical conductivity + equilibrium + fuels + hydrogen + ions + lithium chloride + lithium fluoride + measurement + liquidus + metal transfer process + MSRE + noble metals + oxide precipitation process + phase equilibria + plutonium + plutonium fluorides + reaction rates + reductive extraction process + research + sampling + secondary salts + sodium fluoride + solidus + solubility + specific heat + spectrophotometry + surface tension + technology + testing + thorium fluorides + uranium fluorides + uranium-232 + uranium-233 + uranium-235
OTHER CATEGORIES: CXX + DXX + MCD

ACD70C038

Grimes WR

CHEMISTRY (PART 3, MSRP SEMIANN PRG REPT 8/31/70)

Oak Ridge National Laboratory, Tenn.

ORNL-4622 (Aug. 1970), pp 60-118, 35 fig, 143 ref.

Fission product behavior in the MSRE is analyzed in terms of age of the products, time of exposure for short exposures, surface roughness, flow conditions, and the comparison of deposition on graphite with that on metal. A possible mechanism for "smokes" of metallic fission products is advanced. The chemistry of molybdenum and niobium fluorides is treated. Various properties of alkali fluoroborates, including tritium retention were measured. Phase relations, Pu solubility, oxide chemistry, entropies and conductances were investigated. Fission product separation studies were expanded to include chemistry of molten chlorides. Analytical methods being studied include electrochemistry, studies of NaBF₄, coolant salt, and in-line analyses.

*analysis + *analytical chemistry + *bismuth + *boron trifluoride + *chemical properties + *chemical reactions + *chemistry + *coolants + *deposition +

Accession Number ACD700024 to ACD700038

Category A
Molten-Salt Reactor Programs

ACD700038 *Continued*

*electrical conductivity + *fission products + *fluorides +
 *fluoroborates + *graphite + molten salts + *noble metals +
 *oxides + *processing + *physical properties + *rare earths +
 *reductive extraction process + *solubility +
 *surveillance + *thermodynamics + actinides +
 beryllium + beryllium fluoride + capsules + cells +
 compatibility + concentration + corrosion + diagrams +
 distribution + electrical properties + entrainment +
 equilibrium + examinations + experiment + fissile materials +
 fuel preparation + gamma spectrometry + gases + Hastelloy N +
 hydrogen compounds + inert gases + inventories +
 irradiation + liquidus + lithium fluoride + materials +
 measurement + mists + molybdenum + MSRE + off-gas systems +
 oxide precipitation process + oxidation + phase equilibria +
 primary salt + progress report + protactinium +
 protactinium fluorides + rare earths + reaction rates +
 reduction + sampling + sodium fluoride + spectrophotometry +
 test facilities + testing + thermal properties + thorium +
 thorium fluorides + uranium + uranium fluorides +
 vapor pressure + volatility + zirconium fluoride

ACE650008

(Staff Report)

METALLURGY (Chap. 4 MSRP Prog. Rept. 2/28/65)

Oak Ridge National Laboratory, Tenn.

ORNL-3812 (June 1965), pp. 63-86, 13 fig, 12 ref.

Reaction of Hastelloy N with impure nitrogen apparently gave a protective oxide film. Plugs were welded to seal the MSRE heat exchanger after four tubes had been removed. Welds in Hastelloy N showed reasonably good creep and tensile properties. In a poorly weldable heat silicon and aluminum concentrated in a grain-boundary eutectic. Brazing alloys were sought for joints between various refractory metals and graphite or Hastelloy N. Density, gas evolution, and porosity were measured on MSRE graphite. Oxidation accelerated by thermal cycling led to a crack in a Hastelloy N bayonet tube in a development test. Plans for Hastelloy N creep measurements in the ORR and the MSRE surveillance rig are described.

drain tanks + scaling + cracks + tungsten + tantalum +
 alloys + compatibility + corrosion + creep + development +
 ductility + embrittlement + equipment + examinations +
 failures + microstructure + brazing + graphite +
 Hastelloy N + heat treatments + impurities + in-pile tests +
 inspection + irradiation + joints + materials +
 materials testing + mechanical properties + metallography +
 metallurgy + metals + molybdenum + nitrogen +
 progress report + radiation damage + rupture +
 stress rupture + surveillance + testing + welding
 OTHER CATEGORIES: EAX + FEX + GEX

ACE650014

Category A
Molten-Salt Reactor Programs

ACE650014 *Continued*

(Staff Report)

METALLURGY (Chap. 4 MSRP Prog Rept. 8/31/65)

Oak Ridge National Laboratory, Tenn.

ORNL-3872 (Dec. 1965), pp 81-105, 20 fig. 11 ref.

Molten fluorides were circulated in loops of Hastelloy N and type 304 stainless steel, and molten lead in several steels and Nb-1% Zr. The MSRE surveillance rig is described. Irradiation creep of graphite is discussed for advanced reactors. Creep of Hastelloy N is reported as affected by thermal and mechanical treatments, prior irradiation, and simultaneous irradiation. Welding air-melted but not vacuum-melted Hastelloy N impaired strength and ductility. Autoradiography and microprobe analysis showed intermetallic precipitation in Hastelloy N.

precipitation + modified Hastelloy N + alloys + compatibility + corrosion + creep + development + ductility + embrittlement + examinations + failures + fluorides + molten salts + microstructure + graphite + Hastelloy N + heat treatments + impurities + in-pile tests + inspection + iron alloys + irradiation + joints + lead + liquid metals + loop + mass transfer + materials + materials testing + mechanical properties + metallography + metallurgy + metals + progress report + radiation damage + rupture + stainless steels + stress rupture + surveillance + testing + thermal convection + welding

OTHER CATEGORIES: ECX + FBX + GXX

ACE660012

(Staff Report)

METALLURGY (Chap. 4 MSRP Prog. Rept. 2/28/66)

Oak Ridge National Laboratory, Tenn.

ORNL-3936 (June 1966), pp. 95-121, 18 fig, 10 ref.

Molten lead was circulated in Nb-1% Zr and in a chromium-molybdenum steel; molten fluorides in Hastelloy N and type 304 stainless steel. MSRE surveillance specimens were examined after pre-nuclear operation. Irradiation darkened grain boundaries of Hastelloy N but did not induce reaction with nitrogen. Development of graphite-to-Hastelloy N brazed joints included pressure testing with molten fluorides. Procurement, characterization, and MSRE requirements of graphite are described. Irradiation decreased rupture life and creep ductility of Hastelloy N. Effects of pre- and post-irradiation heat treatments on tensile properties of Hastelloy N are shown. Attempts were made to improve adverse mechanical properties of Hastelloy N welds.

alloys + compatibility + corrosion + creep + density + development + ductility + electrical conductivity + embrittlement + equipment + examinations + expansion + failures + fluorides + molten salts + microstructure +

Accession Number ACE650014 to ACE660012

Category A
Molten-Salt Reactor Programs

ACE660012 *Continued*

brazing • graphite + Hastelloy N + heat treatments +
 impurities + in-pile tests + inspection + iron alloys +
 irradiation + joints + lead + liquid metals + loop +
 mass transfer + materials + materials testing +
 mechanical properties + metallography + metallurgy + metals +
 molybdenum + nitrogen + physical properties + procurement +
 progress report + radiation damage + rupture +
 specifications + stainless steels + stress rupture +
 surveillance + testing + thermal convection + welding +
 tensile properties

OTHER CATEGORIES: EXX + FXX + GXX

ACE660018

Adams GM

MSRP MATERIALS (Chap. 6 MSRP Prog. Rept 8/31/66)

Oak Ridge National Laboratory, Tenn.

ORNL-4037 (Jan. [1967], pp. 97-133, 33 fig. 20 ref.)

Mechanical damage from unequal expansion of graphite and Hastelloy N required revision and replacement of the MSRE surveillance rig. Aluminum alloy from a blower failure was removed from the MSRE radiator; simulated tests showed no damage of aluminum to Hastelloy N. Graphite studies included characterization of several grades, measurement of creep under irradiation at 700 and 1000 deg C, brazing to molybdenum and Hastelloy N, and molten-salt corrosion of brazed joints. Observations on Hastelloy N include good weldability with titanium modifications but poor with zirconium, postirradiation ductility improved by titanium additions but not tungsten or niobium, and extension of creep studies to 982 deg C. Molten fluorides were circulated in loops of Hastelloy N, type 304 stainless steel, and NE-1% Zr clad with type 446 stainless steel, and Croloy 9M. A Croloy 9M loop circulating lead plugged and was examined.

heat exchangers + lead + zirconium + tungsten + iron alloys +
 aluminum + alloys + brazing + corrosion protection +
 columns • compatibility + corrosion + creep + density +
 development + distillation • electrical conductivity +
 embrittlement + ductility + equipment + examinations +
 expansion + failures + fluorides + molten salts +
 fluoroborates + graphite + Hastelloy N + heat treatments +
 in-pile tests + inspection + irradiation + joints + loop +
 mass transfer + materials + materials testing •
 measurement + mechanical properties + metallography +
 metallurgy + metals + modified Hastelloy N + molybdenum +
 MSRE + physical properties + procurement + progress report +
 radiation damage + reliability + rupture + stainless steels +
 stress rupture + surveillance + testing +
 alloy composition + welding + liquid metals

OTHER CATEGORIES: EXX + FXX + GXX

ACE670021

Category A
Molten-Salt Reactor Programs

ACE670021 *Continued*

Cook WH + McCoy HE + Kennedy CR + Werner WJ + Litmar AP +
Canonic DA + Haseltine DM

MOLTEN-SALT REACTOR PROGRAM MATERIALS (Chap. 6 MSRP Prog.
Rept. 2/28/67)

Oak Ridge National Laboratory, Tenn.

ORNL-4119 (July 1967), pp. 95-117, 20 fig. 11 ref.

The first Hastelloy N surveillance specimens were removed from the MSRE, and tensile properties, particularly ductility, are shown as functions of temperature and strain rate. Hastelloy N was unaffected by contact with aluminum. MSBR graphite studies included requirements, studies of prospective grades, irradiation plans, and brazing to Hastelloy N. Loops are described to study corrosion of Hastelloy N, stainless-steel-clad niobium-1% zirconium, and type 304L stainless steel in molten salts.

aluminum + brazing + compatibility + corrosion + embrittlement + ductility + examinations + fluorides + molten salts + fluoroborates + graphite + Hastelloy N + heat treatments + in-pile tests + irradiation + loop + mechanical properties + metallography + MSRE + progress report + stainless steels + surveillance + testing

OTHER CATEGORIES: EDX + FBX + GAX

ACE670028

McCoy HE + Weir JR

MATERIALS DEVELOPMENT (Part 5 MSRP Prog. Rept. 8/31/67)

Oak Ridge National Laboratory, Tenn.

ORNL-4191 (Dec. 1967), pp 196-238, 37 fig, 24 ref.

Creep and microstructural results are given for Hastelloy N MSRE surveillance specimens; microstructures include weld metal and modified alloys. Other Hastelloy N studies include aging modified alloy, weldability of zirconium-modified alloys, precipitate morphology, residual welding stress, corrosion by molten salts, tellurium compatibility, titanium diffusion, and compatibility with graphite.

Graphite studies include procurement, physical and mechanical characterization, fabrication, coating with molybdenum, gas impregnation, and start of HFIR irradiation. Molten salt corrosion to type 304 L stainless steel, Croloy 9M, and a graphite-to-molybdenum brazed joint were also studied.

diffusion + iron alloys + zirconium + elasticity + alloys + brazing + carbides + coatings + compatibility + corrosion + creep + density + deposition + development + electrical conductivity + embrittlement + ductility + equipment + examinations + fabrication + failures + fluorides + molten salts + fluoroborates + graphite + Hastelloy N + heat treatments + impregnation + in-pile tests + inspection + irradiation + joints + loop + mass transfer + materials + materials testing +

Accession Number ACE670021 to ACE670028

Category A
Molten-Salt Reactor Programs

ACE670028 *Continued*
measurement + mechanical properties + metallography +
metals + modified Hastelloy N + molybdenum + MSRE +
physical properties + precipitation + procurement +
progress report + radiation damage + reliability +
rupture + sealing + stainless steels + stress rupture +
surveillance + testing + alloy composition + welding +
x-rays

OTHER CATEGORIES: EXX + FXX + GXX

ACE680017

McCoy HE + Weir JR

MATERIALS DEVELOPMENT (Part 5 MSRP Prog. Rept. 2/29/68)

Oak Ridge National Laboratory, Tenn.

ORNL-4254 (Aug. 1968), pp. 183-240, 43 fig, 33 ref.

Hastelloy N investigations include creep of MSRE
surveillance specimens, effect of strain rate on ductility
of irradiated and control specimens, creep of modified
alloy, electron microscopy of effects of silicon and
titanium on precipitate morphology, titanium diffusion,
weld stresses, weld development, oxidation, and corrosion
of standard and modified alloys by various molten salts.
Graphite studies include procurement, density,
microstructure, increase in porosity by oxidation, x-ray
diffraction, gas impregnation, sealing with molybdenum,
ultrasonic measurement of elastic properties, and brazing
to molybdenum and Hastelloy N. Also studied were
irradiation of brazing alloys, ultrasonic inspection of
brazed joints, and electrical resistivity of modified
Hastelloy N.

alloys + brazing + carbides + coatings + compatibility +
corrosion + creep + density + deposition + development +
electrical conductivity + embrittlement + ductility +
equipment + examinations + fabrication + failures +
fluorides + molten salts + fluoroborates + graphite +
Hastelloy N + heat treatments + impregnation +
in-pile tests + inspection + irradiation + joints + loss +
mass transfer + materials + materials testing + measurement +
mechanical properties + metallography + metallurgy + metals +
modified Hastelloy N + molybdenum + MSRE + oxidation +
physical properties + precipitation + procurement +
progress report + radiation damage + reliability +
remote welding + rupture + sealing + stainless steels +
stress rupture + surveillance + testing +
alloy composition + welding + x-rays + tensile properties

OTHER CATEGORIES: EXX + FXX + GXX

ACE680024

McCoy HE + Weir JR

MATERIALS DEVELOPMENT (Part 5 MSRP Prog. Rept. 8/31/68)

Oak Ridge National Laboratory, Tenn.

Category A
Molten-Salt Reactor Programs

ACE680024 *Continued*

ORNL-4344 (Feb. 1969), pp. 211-290, 62 fig, 32 ref.

Graphite and Hastelloy N surveillance specimens were removed from the MSRE and replaced; creep tests were run on removed Hastelloy N and two modified alloys. Efforts on several grades of graphite include procurement, density, resistivity, permeability, bend testing, x-ray diffraction, gas impregnation, sealing with molybdenum, HFIR irradiation, and small-angle x-ray scattering. Effects of titanium content and irradiation temperature on creep, effect of aging on tensile properties, weldability, and molten salt corrosion were studied on modified Hastelloy N. Standard Hastelloy N studies included resistance to salts and air, transition joints with graphite, mechanical properties of welds in irradiated specimens, and fluted tubing. Precipitate morphologies were studied in both. Other corrosion studies included Haynes alloy No. 25 in fluoroborates, stainless steel, and a chemical separation still. Bearing coatings were studied by x-ray diffraction.

alloys + bearings + brazing + carbides + cermets +
coatings + columns + compatibility + corrosion + creep +
density + deposition + development + distillation +
electrical conductivity + embrittlement + ductility +
cobalt + equipment + examinations + expansion +
fabrication + failures + fluorides + molten salts +
fluoroborates + graphite + hafnium + Hastelloy N +
heat treatments + impregnation + in-pile tests +
inspection + irradiation + joints + loop + mass transfer +
materials + materials testing + measurement +
mechanical properties + metallography + metallurgy + metals +
modified Hastelloy N + molybdenum + MSRE + oxidation +
physical properties + precipitation + procurement +
progress report + radiation damage + reliability +
remote welding + rupture + sealing + stainless steels +
stress rupture + surveillance + testing +
alloy composition + welding + x-rays + tensile properties
OTHER CATEGORIES: FXX + FXX + GXX

ACE690026

Eatherly WF + McCoy HE + Weir JR

MATERIALS DEVELOPMENT (PART 5, MSRE SEMIANN PROG REPT

2/28/69)

Oak Ridge National Laboratory, Tenn.

ORNL-4396 (Aug. 1969), pp. 211-268, 54 fig, 44 ref.

Magnetic particles from the MSRE pump bowl were quite diverse as shown by microprobe analysis and metallography. Graphite topics include radiation damage fundamentals, binder chemistry, hot pressing, physical characterization, thermal conductivity, x-ray diffraction, electron microscopy, gas impregnation, and irradiation in HFIR. Creep of modified Hastelloy N was studied as affected by

Accession Number ACE680024 to ACE690026

Category A
Molten-Salt Reactor Programs

ACE690026 *Continued*

titanium and carbon contents, aging, and irradiation at various temperatures. Electron microscopy identified the carbide precipitates formed in Hastelloy N and various modifications. Corrosion in various molten salts and in air and welding stresses were measured for Hastelloy N. Molten fluoride corrosion was compared for Hastelloy N, the modified alloy, and stainless steel. Other studies include brazing Hastelloy N to graphite, brazing molybdenum, corrosion of TZM by fluorides, and thermal cycling of bearing materials.

alloy composition + bearings + bismuth + compatibility + coolants + corrosion + creep + defects + development + ductility + embrittlement + examinations + expansion + fabrication + failures + ferroalloys + fluorides + fluoroborates + forming + microstructure + graphite + heat treatments + impregnation + impurities + Hastelloy N + in-pile tests + iron alloys + irradiation + mass transfer + materials + mechanical properties + metallography + metallurgy + metals + modified Hastelloy N + molten salts + MSRE + molybdenum + physical properties + processing + procurement + progress report + radiation damage + rupture + sealing + stainless steels + stress + surveillance + test facilities + testing + welding + brazing + x-rays

ACE690032

Eatherly WF + McCoy HE + Weir JR

MATERIAL DEVELOPMENT (PART 5, MSRP SEMIANN PROG REPT E/31/69)
Oak Ridge National Laboratory, Tenn.

ORNL-4449 (Feb. 1970), pp 165-213, 48 fig, 22 ref.

Preliminary examination of graphite and Hastelloy N surveillance specimens from the MSRE and tensile results on the Hastelloy N are reported. Graphite studies include x-ray diffraction determination of anisotropy, electron microscopy, gas impregnation, irradiation effects to high fluences, and interpretation of radiation damage. For titanium-modified Hastelloy N, creep was studied as affected by titanium content and aging, and tensile properties were related statistically to several variables. Combinations among Ti, Hf, Nb, and Y additions improved postirradiation ductility; titanium plus hafnium was best. Commercial heats are often inferior to laboratory-melted Hastelloy N modifications. Electron microscopy traced the improvements to formation of MC-type carbides. Other studies include compatibility of Hastelloy N with coolant salts and other fluids, depositing tungsten, brazing molybdenum, and remote welding.

bismuth + coatings + compatibility + coolants + corrosion + creep + defects + density + deposition + development + ductility + embrittlement + equipment + examinations +

Accession Number ACE690026 to ACE690032

Category A
Molten-Salt Reactor Programs

ACE690032 *Continued*
 expansion + fabrication + failures + fluorides +
 fluoroborates + forming + microstructure +
 graphite + hafnium + heat treatments +
 impregnation + impurities + Hastelloy N + in-pile tests +
 irradiation + mass transfer + materials +
 mechanical properties + metallography + metallurgy + metals +
 modified Hastelloy N + molten salts + MSRE + molybdenum +
 physical properties + procurement + progress report +
 radiation damage + reliability + remote maintenance +
 research + rupture + sealing + steam generators + stress +
 surveillance + test facilities + testing + tungsten +
 brazing + x-rays

ACE700025

Weir JR + McCoy HE

MATERIALS DEVELOPMENT (PART 5, MSRP SEMIANN PEGG FEET
 2/28/70)

Oak Ridge National Laboratory, Tenn.

ORNL-4548 (Aug. 1970), pp 188-276, 83 fig, 45 ref.

Microstructural changes and creep properties were studied on Hastelloy N surveillance specimens from the MSRE. Graphite studies include electron damage; density, resistivity, and permeability of promising grades; hot and isostatic pressing; thermal conductivity measurement; x-ray diffraction; electron microscopy; gas impregnation; irradiation; and lifetime calculation. Effects of aging, irradiation, and composition on mechanical properties, electron microscopy, weldability, and corrosion by various molten salts were studied for modified Hastelloy N. Back extrusion, welding, and brazing of molybdenum, compatibility of alloys with bismuth, coating with tungsten, and oxidation of steels were studied in support of fuel reprocessing. Also reported are progress in remote welding, failure analysis of loop components, and compatibility testing of bearings.

alloy composition + bearings + bismuth + coatings +
 compatibility + coolants + corrosion + creep + defects +
 deposition + development + embrittlement + examinations +
 expansion + fabrication + failures + ferroalloys +
 fluorides + fluoroborates + forming + microstructure +
 graphite + hafnium + heat treatments +
 impregnation + impurities + Hastelloy N + in-pile tests +
 iron alloys + irradiation + mass transfer + materials +
 mechanical properties + metallography + metallurgy +
 metals + modified Hastelloy N + molten salts + MSRE +
 molybdenum + physical properties + procurement +
 progress report + pyrocarbon + radiation damage +
 remote maintenance + rupture + sealing + stainless steels +
 steam generators + stress + surveillance + test facilities +
 testing + tungsten + welding + brazing

ACE700039

Category A
Molten-Salt Reactor Programs

ACE700039 *Continued*

Weir JR

MATERIALS DEVELOPMENT (PART 4 MSRP EBCGE. FEET., 8/31/70)

Oak Ridge National Laboratory, Tenn.

ORNL-4622 (Jan. 1971) pp. 119-198, 75 fig, 51 ref.

Tubing and thermocouple wells from the MSRE coolant circuit showed very little corrosion. Graphite development included procurement of new grades, determination of density, resistivity, anisotropy, and microstructure, characterization of pitch, building thermal conductivity apparatus, pore sealing, and irradiation effects on density, porosity, and pore seals. Investigations of Hastelloy N modifications include effects of titanium content and aging on hardness, tensile properties, creep, and postirradiation creep, postirradiation creep of alloys containing various combinations of Ti, Nb, and Fe, weldability, creep of commercially melted alloys, and microstructure. Loop studies of corrosion included type 304 stainless steel, Hastelloy N and several modified alloys exposed to fuel, blanket, and coolant salts. The electron microprobe was used to study the corrosion of Hastelloy N by power plant steam, and study was started on a duplex steam-generator tube made of Incoloy 800 and Nickel 280. Development of processing equipment included back-extrusion, welding, and brazing of molybdenum components, compatibility of Mo, TZM, Nb, Ni-1% Zr, Ta, T-111, graphite, Fe-5% Mo, and brazed joints in Mo with molten bismuth, and deposition of coatings by reduction of MoF₆ and WF₆ vapors and MoF₆ dissolved in molten fluorides.

brazing + creep + compatibility + contactors + density + deposition + ductility + electrical conductivity + examinations + fabrication + fluorides + fluoroborates + bismuth + graphite + hardness + Hastelloy N + heat exchangers + heat treatments + impregnation + iron + irradiation + loop + molybdenum + modified Hastelloy N + nickel + procurement + progress report + stainless steels + steam generators + tantalum + thermal conductivity + welding + alloy composition + microstructure + molten salts + niobium • tensile properties

OTHER CATEGORIES: EXX + FCX + GXX

ACX640008

(Staff Report)

MOLTEN-SALT REACTOR PROGRAM SEMI-ANNUAL PROGRESS REPORT FOR
PERIOD ENDING JANUARY 31, 1964

Oak Ridge National Laboratory, Tenn.

ORNL-3626 (July 1964), 166 p.

Status and progress are reported. Contents are abstracted and filed in 5 parts: MSRE, Metallurgy, Radiation Chemistry, Chemistry, and Fuel Processing.

Accession Number ACE700039 to ACX640008

Category A
 Molten-Salt Reactor Programs

ACX640008 *Continued*
 MSRP + progress report

ACX640014
 (Staff Report)
 MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT
 FOR PERIOD ENDING JULY 31, 1964
 Oak Ridge National Laboratory, Tenn.
 ORNL-3708 (Nov. 1964), 395 p.

This report is a review in depth rather than a report of 6 months' progress. It is a collection of papers given at an information meeting at ORNL, August 18-19, 1964, near the end of MSRE construction. The papers cover the background and report the status of the technology of molten-salt thermal-breeder reactors as of mid-1964. Separate abstracts are filed for each of 16 papers.

MSRP + progress report + reviews + technology
 OTHER CATEGORIES: AEX

ACX650003
 (Staff Report)
 MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT FOR
 PERIOD ENDING FEBRUARY 28, 1965
 Oak Ridge National Laboratory, Tenn.
 ORNL-3812 (June, 1965), 176 p.

Status and progress are reported. Contents are abstracted and filed in 5 parts: MSRE, Metallurgy, Radiation Chemistry, Chemistry, and Fuel Processing.

MSRP • progress report

ACX650009
 (Staff Report)
 MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT FOR
 PERIOD ENDING AUGUST 31, 1965
 Oak Ridge National Laboratory, Tenn.
 ORNL-3872 (Dec. 1965), 156 p.

Status and progress are reported. Contents are abstracted and filed in 5 parts: MSRE, Metallurgy, Radiation Chemistry, Chemistry, and Fuel Processing.

MSRP + progress report

ACX660007
 (Staff Report)
 MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT FOR
 PERIOD ENDING FEBRUARY 28, 1966
 Oak Ridge National Laboratory, Tenn.
 ORNL-3936 (June, 1966), 216 p.

Status and progress are reported. Contents are abstracted and filed in 5 parts: MSRE, Metallurgy, Chemistry, MSRE Design Studies, and MSR Processing Studies.

MSRP + progress report

ACX660013

Category A
Molten-Salt Reactor Programs

ACX66C013 *Continued*

(Staff Report)

MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT FOR PERIOD
ENDING AUGUST 31, 1966

Oak Ridge National Laboratory, Tenn.

ORNL-4037 (Jan. 1967), 242 p.

Status and progress are reported. Contents are abstracted
and filed in 5 parts: MSRE, MSRP Materials, Chemistry, MSBR
Design Studies, and MSR Processing Studies

MSRP + progress report

ACX67C015

(Staff Report)

MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT FOR PERIOD
ENDING FEBRUARY 28, 1967

Oak Ridge National Laboratory, Tenn.

ORNL-4119 (July 1967), 219 p.

Status and progress are reported. Contents are abstracted
and filed in 6 parts: MSRE, Materials, Chemistry, In-Pile
Loops, MSBR Design Studies, and MSR Processing Studies

MSRP + progress report

ACX670022

(Staff Report)

MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT FOR
PERIOD ENDING AUGUST 31, 1967

Oak Ridge National Laboratory, Tenn.

ORNL-4191 (Dec. 1967), 260 p.

Status and progress are reported in six parts with
these titles: Molten-Salt Reactor Experiment, MSER
Design and Development, Chemistry, Molten-Salt Irradiation
Experiments, Materials Development, and Molten-Salt
Processing and Preparation. Separate abstracts are
filed for each part.

MSRP + progress report

ACX68C011

(Staff Report)

MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT
FOR PERIOD ENDING FEBRUARY 29, 1968

Oak Ridge National Laboratory, Tenn.

ORNL-4254 (Aug. 1968), 282 p.

Status and progress are reported in six parts with these
titles: Molten-Salt Reactor Experiment, MSBR Design
and Development, Chemistry, Molten-Salt Irradiation
Experiments, Materials Development, and Molten-Salt
Processing and Preparation. Separate abstracts are
filed for each part.

MSRP + progress report

ACX680018

(Staff Report)

MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT FOR

Category A
Molten-Salt Reactor Programs

ACX680018 *Continued*

PERIOD ENDING AUGUST 31, 1968

Oak Ridge National Laboratory, Tenn.

ORNL-4344 (Feb. 1969), 333 p.

Status and progress are reported in six parts with these titles: Molten-Salt Reactor Experiment, MSBR Design and Development, Chemistry, Molten-Salt Irradiation Experiments, Materials Development, and Molten-Salt Processing and Preparation. Separate abstracts are filed for each part.

MSRP + progress report

ACX690020

(Staff Report)

MOLTEN-SALT REACTOR PROGRAM SEMI-ANNUAL PROGRESS REPORT FOR
PERIOD ENDING FEBRUARY 28, 1969

Oak Ridge National Laboratory, Tenn.

ORNL-4396 (Aug. 1969), 307 p.

Status and progress are reported in six parts with these titles: Molten-Salt Reactor Experiment, MSBR Design and Development, Chemistry, Molten-Salt Irradiation Experiments, Materials Development, and Molten-Salt Processing and Preparation. Separate abstracts are filed for each part.

MSRP + progress report

ACX690027

(Staff Report)

MOLTEN-SALT REACTOR PROGRAM SEMI-ANNUAL PROGRESS REPORT FOR
PERIOD ENDING AUGUST 31, 1969

Oak Ridge National Laboratory, Tenn.

ORNL-4449 (Feb. 1970), 252 p.

Status and progress are reported in six parts with these titles: Molten-Salt Reactor Experiment, MSBR Design and Development, Chemistry, Molten-Salt Irradiation Experiments, Materials Development, and Molten-Salt Processing and Preparation. Separate abstracts are filed for each part except Irradiation Experiments, where there was no activity.

MSRP + progress report

ACX700018

(Staff Report)

MOLTEN-SALT REACTOR PROGRAM SEMI-ANNUAL PROGRESS REPORT FOR
PERIOD ENDING FEBRUARY 28, 1970

Oak Ridge National Laboratory, Tenn.

ORNL-4548 (Aug. 1970), 338 p.

Status and progress are reported in six parts with these titles: Molten-Salt Reactor Experiment, MSBR Design and Development, Chemistry, Molten-Salt Irradiation Experiments, Materials Development, and Molten-Salt

Category A
Molten-Salt Reactor Programs

ACX700018 *Continued*

Processing and Preparation. Separate abstracts are filed for each part except Irradiation Experiments, where there was no activity.

MSRP + progress report

ACX700034

(Staff Report)

MOELTEN-SALT REACTOR PROGRAM SEMI-ANNUAL PROGRESS REPORT

Oak Ridge National Laboratory, Tenn.

ORNL-4622 (Jan. 1971), 230 p.

Status and progress are reported in five parts with these titles: Molten-Salt Reactor Experiment, MSER Design and Development, Chemistry, Materials Development, and Molten-Salt Processing and Preparation. Separate abstracts are filed for each part.

MSRP + progress report

ACX710027

(Staff Report)

MOLTEN-SALT REACTOR PROGRAM SEMI-ANNUAL PROGRESS REPORT FOR PERIOD ENDING FEBRUARY 28, 1971

Oak Ridge National Laboratory, Tenn.

ORNL-4676 (Aug. 1971), 277 p.

Status and progress are reported in five parts with these titles: Molten-Salt Reactor Experiment, MSER Design and Development, Chemistry, Materials Development, and Molten-Salt Processing and Preparation. Separate abstracts are filed for each part.

MSRP + progress report

ADX640021

Kasten PR

THE MOSEL REACTOR CONCEPT

Kernforschungsanlage Julich, Germany

Proc. 3rd Int. Conf. in Peaceful Uses of Atomic Energy, Geneva, (Aug. 31 - Sept. 9 1964), Vol. 6, pp. 363-369, 1 fig. 23 ref.

The Molten Salt Epithermal (MOSEL) reactor concept features a core fluid of $UF_4-NaF-BeF_2$ and a blanket fluid of $ThF_4-NaF-BeF_4$ separated by nickel alloy tubes. Processing is by fluorine volatility. Parasitic absorption of neutrons (predominantly epithermal) is moderate and breeding ratios between 1.08 - 1.22 appear attainable. The core concentration is about 300 g fissile/liter and fissile inventory cost is a major item. Nevertheless power costs appear reasonable.

*breeding performance + *conceptual design + *molten salts + *reactors + costs + foreign + fluorination + Hastelloy N + inconels + inventories + neutron spectra + nickel +

Accession Number ACX700018 to ADX640021

Category A
Molten-Salt Reactor Programs

ADX640021 *Continued*
sodium fluoride
OTHER CATEGORIES: IAF

ADX67C046

Gat U

COOLING CONCEPTS FOR A COMPACT MOSEL (MOLTEN SALT) REACTOR
Kernforschungsanlage Julich, Germany
Nucl. Engrg. and Design 5 (1967), pp. 113-122, 10 fig,
23 ref.

This review of engineering possibilities of the MSEL reactor considers cooling by direct contact of salt and molten lead, either internally or externally to the core. Internal cooling reduces the fuel inventory and fuel cycle costs but concentrates engineering problems in the core zone.

*conceptual design + *lead cooling + *reactors + cores +
foreign + heat transfer + inventories

OTHER CATEGORIES: IAF

ADX690063

Jensen RJ + Swanson E

UTILITY APPLICATION OF MOLTEN-SALT BREEDER REACTOR

Northern States Power Co, Minneapolis + Black and Veatch,
Kansas City

Proc. American Power Conf. 31 (1969), pp. 222-230, 5 fig,
7 ref.

The MSBR concept, supported by the successful MSFE, is under study by utilities as a contender with fast breeders for long-range power. Its low breeding ratio is offset by low fissile inventory and it can use U-233, U-235, or Pu as initial fuel. On-site processing is required. A schedule for 1976 startup of a 300- to 500-Mwe prototype is presented.

*electrical power + *industry + *MSBR + *prototypes +
*utilities + architect-engineering + breeding performance +
converters + economics + natural resources

Category B
Reactor Analysis

BAX680006

Perry AM

INFLUENCE OF NEUTRON DATA IN THE DESIGN OF OTHER TYPES OF
POWER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2157 (March 1968), 23 p, 5 fig, 10 ref.

This report was presented at the Second Conference on Neutron Cross Sections and Technology, Washington, March 4-7, 1968. The effects of cross-section uncertainties on estimates of breeding performance and of power cost for a molten-salt breeder reactor are shown to be small. Uncertainty in breeding ratio due to cross-section uncertainties is less than plus or minus 0.02, and the uncertainty in power costs is less than plus or minus 0.3 mills/kwhr (e). Similarly small effects are shown for the high-temperature gas-cooled reactor. The need for further refinements in nuclear data is related primarily to the calculation of temperature coefficients of reactivity.

*breeding performance + *cross sections + *design data +
*errors + MSBR + nuclear analysis + power costs + reactivity +
reactors + thorium + uranium-233

OTHER CATEGORIES: BFX

BAX700008

Kasten PR + Craven CW + Wright RC

CROSS-SECTION AND NUCLEAR-CONSTANT DATA FOR HEAVY METAL
NUCLIDES (FUELS)

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2851 (Rev.) (Apr. 1970) 21 p, 18 fig, 0 ref.

Cross sections and nuclear constants of fissile and fertile materials and of higher isotopes are summarized in graphical form, based on ENDF/B data. The resulting figures permit visual appreciation of nuclear data in present use, and relative comparison of data for the different fissile, fertile, and higher-isotope materials. Nuclides considered are Th-232, U-238, Pa-233, Np-239, U-233, U-235, Pu-239, Pu-241, U-234, U-236, Pu-240, and Pu-242. The revision differs from the original report only in the quality of reproduction of graphs.

*neutron physics + *cross sections + *neutron yield +
fission + absorption + *fissile materials + thorium +
uranium + plutonium + neptunium + *data + isotopes +
capture + *fertile materials

BBX670012

Carlsmith RS + Bennett LL + Edison GE + Gift FH + Thomas WE +
Welfare FG

REVIEW OF MOLTEN SALT REACTOR PHYSICS CALCULATIONS

Oak Ridge National Laboratory, Tenn.

Accession Number BAX680006 to BBX670012

Category B
Reactor Analysis

BBX670012 *Continued*

ORNL-TM-1946 (Aug. 1967) 59 p, 13 fig, 41 ref.

A set of calculations was made to check the reactivity and breeding ratio of the reference design of a two-region, two-fluid MSR. The review covered cross section selection, fission product treatment, multigroup cell calculations, two-dimensional reactor criticality calculations, equilibrium depletion calculations and startup depletion calculations. Insofar as possible, the cross sections and calculations methods were made independent of those used previously. The reference composition gave a $k(\text{eff})$ of 0.95. This discrepancy was traced to use of a low value for thorium resonance integral in previous calculations. When the reactor was made critical by the addition of 14% more U-233, the breeding ratio was 1.062 compared with 1.054 in the previous calculations. Reoptimization of the composition would probably decrease this difference in breeding ratio.

*calculations + MSR + *two-fluid reactor +
*breeding performance + reactivity + burnup +
criticality + cross sections + fission products +
computer codes + nuclear analysis + *design data +
*errors + *reviews + thorium

OTHER CATEGORIES: BFX

BFX680009

Perry AM + Smith OL + Kerr HT
NEW DEVELOPMENTS IN MSR PHYSICS

Oak Ridge National Laboratory, Tenn.

Summary of paper presented at ANS Winter Meeting,

Washington, D. C., Nov. 10-15, 1968, ANS Transactions 11,

(2) 619, 2 p.

Developments which permit separation of protactinium and fission products from MSR fuel and from each other raise the possibility of a single-fluid MSR. The fluid which contains uranium and thorium is made to function as both fuel and fertile material by adjusting the degree of neutron moderation that occurs in various regions of a zoned core. Zoned cores have higher yields and lower fuel inventories than uniform cores. Power costs can be reduced (.1 to 0.3 mills/kwh(e) below those for comparable uniform cores. Temperature coefficients of reactivity in zoned-core single-fluid reactors are such that dynamic characteristics are expected to be acceptable. (Reprints of this paper are not available but similar, and more recent, data are presented in Nucl. Appl. and Tech. 8, 208 (Feb. 1970). See BFX700016.)

*MSR + *single-fluid reactors + *breeding performance +
reactivity + dynamic characteristics + specific inventory +
nuclear analysis + neutron physics + blanket + moderators +
fuel cycle costs

BFX700016

Category B
Reactor Analysis

BFX70C016 *Continued*

Perry AM + Bauman HF

REACTOR PHYSICS AND FUEL CYCLE ANALYSES

Oak Ridge National Laboratory, Tenn.

Nucl. Appl. Tech. 8, (2) 208 (Feb. 1970), 12 p, 12 fig,
5 ref.

General nuclear characteristics, breeding performance and fuel-cycle costs are discussed for a reference design, single-fluid MSBR operating on a thorium -- uranium-233 fuel cycle with full chemical processing. This design has a breeding ratio near 1.06 specific fissile inventory of 1.5 kg/Mw(e), fuel doubling time of 20 yr, and a fuel cycle cost near 0.7 mil/kwh(e). Either enriched uranium or plutonium may be used as a startup fuel. If chemical processing for Pa isolation and rare-earth removal is omitted, the design has a conversion ratio of 0.8 to 0.9. The fuel cycle cost penalty for operation as a converter is around 0.1 mil/kwh(e)

*MSBR + *single-fluid reactors + *fuel cycle costs +
*breeding performance + nuclear analysis + thorium +
uranium-233 + plutonium + processing + specific inventory +
rare earths + noble metals + fission products +
rare gases + protactinium

BFX70C056

Carlsmith FS + Lane JA

POWER REACTORS FOR THE FUTURE -- AN EVALUATION

Oak Ridge National Laboratory, Tenn.

Proc. American Power Conf. Vol. 32 (1970) pp. 98-104,
4 fig, 16 ref.

A review by the AEC of the U.S. civilian nuclear power program consisted of two phases: determination of the characteristics of various reactor types and simulation of optimal growth patterns using these reactors. This paper reports on the first phase including uranium, thorium and plutonium usage, separative work requirements, and capital, operating, fuel, and total power costs for 70 fueling variations in 7 reactor concepts. The figures reported are those that were available in 1967. An MSER with 1.1 kg/MWe specific inventory, 1.07 breeding ratio and capital costs around those for light-water reactors had a total power cost slightly lower than any other reactor.

*breeding performance + *power costs + *reactors + AEC +
capital costs + economics + fuel cycle costs +
optimizations + natural resources

EGX67C045

(Staff Report)

SAFETY STUDIES FOR MSBR (PART 5 NUCL SAFETY ERG ANN ERG

REPT 12/31/67)

Oak Ridge National Laboratory, Tenn.

Accession Number BFX700016 to EGX670045

Category B
Reactor Analysis

BGX670045 *Continued*

ORNL-4228 (April 1968), pp. 287-307, 8 fig, 8 ref.

These studies, the first reported as part of the Safety Program, are aimed at information needed for safety criteria for MSER's. Indications are that MSER systems have favorable inherent safety and stability characteristics, that fission-product behavior strongly influences emergency cooling requirements, and that MSER materials are compatible.

*analysis + *design criteria + *MSER + *safety +
afterheat • compatibility + deposition +
fission products + molten salts + reactivity + stability +
dynamic characteristics

Category C
Reactor Chemistry

CAX68C032

Thoma RE

CHEMICAL FEASIBILITY OF FUELING MOLTEN SALT REACTORS WITH
PuF₃

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2256 (June 1968), 37 p, 5 fig, 20 ref.

The feasibility of starting molten salt reactors with PuF₃ was evaluated with respect to chemical compatibility within fuel systems and to removal of plutonium from the fuel by chemical reprocessing after Pu-239 burncut. Compatibility within reactor containment systems is moderately well-assured but requires confirmation of PuF₃ solubility and oxide tolerance before tests can be made using the MSRE. Although separation of plutonium and protactinium in the chemical reprocessing plant, as would be desirable in a large breeder reactor, has not yet been demonstrated, conceptual designs of processes for effecting such separations are available for development.

*chemistry + *compatibility + *dissolving + *fuels +
*plutonium fluorides + *primary salt + *processing +
*separations + actinides + beryllium fluoride +
carriers • chemical properties + concentration +
corrosion + fluorides + graphite + Hastelloy N +
lithium fluoride + oxides + physical properties +
reactors + replacement + stability + thorium fluorides +
uranium fluorides

OTHER CATEGORIES: LDA

CAX690052

Fredricksen JA + Gilpatrick IC + Barton CJ

SOLUBILITY OF CERIUM TRIFLUORIDE IN MOLTEN MIXTURES OF
LiF, BeF₂, AND ThF₄

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2335 (Jan. 1969), 25 p, 9 fig, 7 ref.

The Solubility of CeF₃ was determined at various temperatures in six mixtures of LiF, BeF₂, ThF₄ of the type that may be used to fuel a molten salt breeder reactor. Comparison of earlier data on the solubility of PuF₃ and CeF₃ in fluoride solvents makes it possible to predict that the solubility of PuF₃ in single-region fuel compositions at reactor operating temperatures will be more than adequate. The solubility data as a function of solvent composition were best correlated by a model that assumes BeF₂ to be complexed as the BeF₄⁽²⁻⁾ ion and ThF₄ as the ThF₅⁽¹⁻⁾ ion.

*beryllium fluoride + *dissolving + *fluorides +
*liquidus + *lithium fluoride + *phase equilibria +
*plutonium fluorides + *rare earths + *solubility +
*thorium fluorides + actinides + chemistry + compatibility +
fissile materials + fuels + mixtures + MSBE • MSBR +
replacement + solidus

CAX690053

Category C
Reactor Chemistry

CAX690053 *Continued*

McDuffie HF + McCoy HE + Robertson EC + Scott D +
Thoma RE

ASSESSMENT OF MOLTEN SALTS AS INTERMEDIATE COOLANTS FOR
LMFERS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2696 (Sept. 1969), 29 p, 7 fig, 23 ref.

Several molten salts were considered as intermediate coolants for LMFBR's. Included were fluorides, chloride, carbonate, nitrate-nitrite and fluoroborate salts. Chemical reactions that could occur between sodium and fluoroborates lead to the conclusion that carbonates might be a better choice for LMFBR's. Use of carbonates avoids the safety considerations and related costs that arise from the reactions of sodium with water if steam generator fails and with air if a coolant pipe ruptures. In the absence of these safety considerations, sodium is clearly superior to the molten salts as an intermediate coolant for LMFBR's because the lower thermal conductivity and higher viscosity of the salts would result in higher equipment costs.

*carbonates + *chemical properties + *chemical reactions +
*chlorides + *coolants + *fluorides + *fluoroborates +
*liquid metals + *LMFBR + *NaK + *physical properties +
*secondary salts + accidents + afterheat +
applications + behavior + boron trifluoride + compatibility +
concentration + containment + corrosion + decomposition +
density + economics + emergency cooling + failures +
heat exchangers + heat transfer + leakage + leaks +
liquidus + lithium chloride + lithium fluoride + mixtures +
phase equilibria + potassium fluorides + safety +
sodium fluoride + solidus + specific heat + stability +
steam generators + thermal conductivity + viscosity
OTHER CATEGORIES: CCX + CEX

CAX690061

Thoma RE + Ricci JE

FRACTIONAL CRYSTALLIZATION REACTIONS IN THE
SYSTEM LiF-BeF₂-ThF₄

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2596 (July 1969), 33 p, 16 fig, 9 ref.

Equilibrium and non-equilibrium crystallization reactions in the system LiF-BeF₂-ThF₄ are analyzed in relation to their potential application to molten salt reactor fuel reprocessing. Heterogeneous equilibria in the temperature range from the liquidus at 590 deg C to the solidus at 350 deg C are described quantitatively and in detail by means of ten typical isothermal sections and by three temperature-composition sections. The implications of metastable fractionation in this temperature interval are discussed as a possible feed control step in reductive

Category C
Reactor Chemistry

CAX69C061 *Continued*

extraction reprocessing of molten salt breeder reactor fuels.

crystallization + data + fluorides + freezing + measurement + processing + separations

CAX710023

Mailen JC + Smith FJ + Ferris LE

SOLUBILITY OF PLUTONIUM TRIFLUORIDE IN MOLTEN 2 LITHIUM FLUORIDE-BERYLLIUM FLUORIDE

Oak Ridge National Laboratory, Tenn.

J. Chem. and Eng. Data, 12 (Jan. 1971), 2 p, 1 fig, 7 ref.

The solubility of plutonium trifluoride in molten 2 lithium fluoride-beryllium fluoride was determined over the temperature range of 550-660 deg C. The results can be expressed by the least-squares equation: $\log S(\text{mole } \% \text{ plutonium trifluoride}) = 3.2305 - 3096/T(\text{deg K})$. The solid phase present at equilibrium was probably pure plutonium trifluoride.

*molten salts + actinides + fluorides +

*plutonium fluorides + *solubility + MSRE

CCX680033

Kohn HW

BUBBLES, DECES, AND ENTRAINMENT IN MOLTEN SALTS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2373 (Dec. 1968), 21 p, 5 fig, 42 ref.

The author describes production of droplets from splashes and bursting bubbles and reports experiments with molten salts which showed that jet drops could preferentially remove a surface film. He concludes that this phenomenon could contribute to removal of metallic fission products from the fuel salt in the MSRE pump bowl.

*bubbles + *entrainment + *fission products + *molten salts +

*gas injection + *gas separation + *interfacial tension +

*mists + *MSRE + *noble metals + *surface tension +

beryllium fluoride + chemistry + circulation + cover gas +

experiment + fissile materials + fluorides + foaming +

fuels + gases + inert gases + lithium fluoride +

molybdenum + off-gas systems + pumps + sprays +

viscosity + void fractions

OTHER CATEGORIES: CFX + CJX

CCX680038

Cantor S

PHYSICAL PROPERTIES OF MSR FUEL, COOLANT, AND FLUSH SALTS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2316 (Aug. 1968), 49 p, 2 fig, 49 ref.

Experimental values or estimates are given for properties of seven salts of interest for MSBR's. Properties include viscosity, thermal conductivity, electrical

Category C
Reactor Chemistry

CCX68C038 *Continued*

conductivity, phase transition behavior, specific heat, heat of fusion, density, expansivity, compressibility, vapor pressure, surface tension, and gas solubilities.

*data + *fluorides + *fluoroborates + *physical properties + density + solubility + specific heat + surface tension + thermal conductivity + viscosity

CDX670035

Malinauskas AP + Rutherford JI + Evans (III) RE
GAS TRANSPORT IN MSRE MODERATOR GRAPHITE. I. REVIEW OF
THEORY AND COUNTERDIFFUSION EXPERIMENTS

Oak Ridge National Laboratory, Tenn.

ORNL-4148 (Sept. 1967), 39 p, 7 fig, 6 ref.

The authors develop equations describing gas transport in porous media. The experimental findings are limited but significant. Under MSRE conditions it appears quite justifiable to ignore normal diffusion effects in gas transport computations so that all the caseous-diffusion information necessary to correlate fission-product-migration data can be gained through simple permeability measurements. The more complex interdiffusion experiments are not required. Thus a complete flow-property survey of all MSRE moderator materials can be performed with a minimum expenditure of time and effort.

*diffusion + *fission products + *gas flow + *graphite + *MSRE + *xenon + analysis + behavior + concentration + deposition + design criteria + distribution + films + fluids + inert gases + materials + measurement + moderators + physical properties + rare earths + testing
OTHER CATEGORIES: FEX

CEX640018

Blankenship FF

EFFECTS OF RADIATION ON THE COMPATIBILITY OF MSRE MATERIALS
(PART OF MSRE SEMIANN PROG REPT 7/31/64)

Oak Ridge National Laboratory, Tenn.

CRNL-3708 (Nov. 1964), pp 252-287, 16 fig, 4 ref.

Capsules containing fuel salt, graphite, INCF-8 and molybdenum were irradiated in the MTR and later examined at CRNL. Enhanced attack and other anomalous effects appeared to be due to fluorine that was produced by radiolysis of frozen salt at low temperature. Much of this article is concerned with investigation of this phenomenon. Typical radiolytic yield of fluorine from frozen fuel salt was 0.02 molecules per 100 eV absorbed energy. Internal recombination was sufficient to prevent any evolution of gaseous fluorine at temperatures above about 80 deg C.

capsules + compatibility + experiment + fluorine + graphite + in-pile tests + irradiation + materials + molten salts + molybdenum + MSRE + radiolysis +

Accession Number CCX680038 to CEX640018

Category C
Reactor Chemistry

CEX640018 *Continued*
recombination

CLX700010

Haubenreich PN

FLUORINE PRODUCTION AND RECOMBINATION IN FROZEN MSR SALTS
AFTER REACTOR OPERATION

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3144 (Sept. 1970) 36 p, 9 fig, 12 ref.

Exposure of capsules of MSR fuel salts in the MTR between 1961 and 1964 showed that when the salt was chilled below about 80 deg C, F₂ was produced by radiolysis at a rate of 0.02 molecules/100 ev. Other experiments confirmed the radiolysis of frozen salt and provided data on the effect of temperature on recombination. The data on yield and recombination have recently been reviewed and used in answering questions involved in storing and disposing of irradiated salt from the MSRE and future molten-salt reactors. The energy source in the MSRE salt is low enough that no fluorine evolution is expected for over a year after heating to induce recombination. Salt from a high-power MSR can be stored in bare cans with no fluorine evolution if the surroundings are kept at about 200 deg F.

*fluorine + *molten salts + *radiolysis + *storage +
*waste disposal + afterheat + analysis + experiment +
heat transfer + MSRE + primary salt + reaction rates +
recombination

CXX640020

Grimes WR

CHEMICAL BASIS FOR MOLTEN-SALT REACTORS (PART OF MSRP
SEMIANN PROG REPT 7/31/64)

Oak Ridge National Laboratory, Tenn.

ORNL-3708 (Nov. 1964), pp. 214-251, 29 ref.

Requirements of high-temperature fluid-fuel reactors are best met by mixtures of fluorides including UF₄ and ThF₄, which have low vapor pressure, good heat transfer properties, little parasitic absorption of neutrons, and immunity to radiation damage. The selection and characteristics of MSRE fuel and coolant salts are discussed.

chemistry + coolants + fluorides + fuels + molten salts +
MSRE + MSRP + phase equilibria + physical properties +
reviews

CXX700049

Grimes WR

MOLTEN-SALT REACTOR CHEMISTRY

Oak Ridge National Laboratory, Tenn.

Nucl. Appl. Tech. 8, 137 (Feb. 1970) 19 p, 8 fig, 58 ref.

Category C
Reactor Chemistry

CXX700049 *Continued*

Considerations leading to the choice of MSR fuel composition are discussed under the headings: Phase Behavior Among Fluorides, Oxide Fluoride Phase Behavior, MSRE and MSBR Fuel Compositions and Choice of Coolant. Physical properties of fuels and coolant are tabulated. In connection with the chemical compatibility of MSRE materials, topics included are: Thermodynamic Data for Molten Fluorides, Oxidation (Corrosion) of Metal and Compatibility of Graphite with Fluorides. Chemical behavior in the MSRE is discussed in terms of Behavior of the Fuel Components, the Corrosion Products and the Fission Products. Separations chemistry is treated in terms of Separation of Protactinium and of Fission Products by several methods including Reduction. While much research and development remain to be accomplished, it is demonstrated that there is no fundamental chemical difficulty with design and operation of a single-fluid molten salt breeder system.

*behavior + *bismuth + *chemical properties +
 *chemical reactions + *chemistry + *compatibility +
 *coolants + *corrosion + *corrosion products +
 *equilibrium + *fission products + *fluorides +
 *fluoroborates + *fuels + *graphite + *MSRE +
 *noble metals + *oxide precipitation process + *oxides +
 *protactinium fluorides + *separations + actinides +
 beryllium fluoride + beryllium oxide + boron trifluoride +
 chromium + components + concentration + density +
 deposition + dissolving + experiment + fissile materials +
 gases + Hastelloy N + liquid metals + liquidus +
 lithium fluoride + mass transfer + melting + mists + MSRE +
 MSBR + phase equilibria + physical properties +
 primary salt + protactinium + radiation damage +
 rare earths + rare gases + reaction rates + fuel cycle +
 sampling + single-fluid reactors + solidus + solubility +
 thermal conductivity + thorium fluorides +
 uranium fluorides + uranium-235 + vapor pressure +
 viscosity + zirconium fluoride

Category E
Graphite

EBX69C039

Greenstreet WL + Smith JE + Yahr GT + Valachovic RS
THE MECHANICAL BEHAVIOR OF ARTIFICIAL GRAPHITES AS
PORTRAYED BY UNIAXIAL TESTS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2727 (Dec. 1969), 46 p, 27 fig, 5 ref.

Tensile and compressive stress-strain curves were measured and combined with previous measurements to show behavior of several specimens of reactor-grade graphite, principally AGOT, under several conditions of cyclic loading and unloading. Hysteresis was considerable but diminished on successive cycles, becoming very small after several cycles.

graphite + compressive properties + fatigue +
tensile properties + testing

EBX700041

Yahr GT

DETERMINATION OF RELATIVE THERMAL RUPTURE RESISTANCES OF
GRAPHITES

Oak Ridge National Laboratory, Tenn.

ORNL-4467 (Jan. 1970), 47 p, 18 fig. 42 ref.

Polycrystalline graphite has remarkable resistance to thermal-stress-induced fracture. Nevertheless, selection of a particular grade of graphite for certain applications must include consideration of this property. Currently the type and grade of graphite are often selected on the basis of elastic analyses, since thermal shock tests are too expensive for screening devices. This report describes a test rapid and economical enough for screening candidate materials. Thin disks of graphite are heated at the center with an inert-gas shielded-arc nonconsumable electrode welder, each at a different, but constant, power level. The minimum power input to the welder that will consistently cause the graphite to fracture is determined. The graphite that requires the highest power level to produce a fracture is the one most resistant to thermal shock. This test ranked 21 grades, or types, of graphite. An appendix contains mechanical and thermal properties of the specimens, obtained from the literature for determination of figures of merit ratings of thermal shock resistance. None of the standard figures of merit gave reliable predictions.

*graphite + elasticity + experiment + physical properties +
*rupture + *thermal effects + *testing +
thermal properties + thermal shock

EBX700042

Dergunov NN + Barbancov VN + Kurakin VM + Zaitsev GG +
Strokov VI + Abrakhimov U

SHORT-TERM STRENGTH, CREEP, AND DUCTILITY OF GRAPHITE AT

Category E
Graphite

EBX70C042 *Continued*
300 TO 3500 DEG K

Not given.

LA-4462-TR (Sept. 1970), 5 p, 4 fig, 10 ref.

A facility and procedure for studying the tensile, impact, and compressive strengths, creep and ductility of graphite at temperatures from 300 to 3500 deg K are described. Experimental data on the mechanical characteristics and variations of the creep rate and ductility of three grades of graphite differing in grain size in this temperature are presented. The tensile and compressive properties depended in a complex fashion on grain size, temperature, and orientation; the coarse-grained material was weakest. At about 3000 deg K, the tensile behavior changed from brittle to ductile and the temperature dependence of creep increased greatly. This document is a translation of Paper No. 5 in the Transactions of the Fifth All-Union Scientific-Technical Conference, Kiev, Ukrainian SSR, December 1967.

creep + ductility + *graphite + experiment + *equipment + microstructure + tensile properties + compressive properties

EBX70C043

Fontana A + Winand R

STUDY OF THE WETTABILITY OF GRAPHITE BY DIFFERENT MELTEN
SODIUM-FLUORIDE-ZIRCONIUM TETRAFLUORIDE-ZIRCONIUM DIOXIDE
MIXTURES IN THE PRESENCE OF VARIOUS GASEOUS ATMOSPHERES

Universite Libre de Bruxelles, Belgium

J. Nucl. Mater. 35, 87 (Apr. 1970), 5 p, 4 fig, 5 ref.

The wettability of graphite at 1050 deg C by molten sodium fluoride containing up to 25% zirconium fluoride and 3.75% zirconium oxide was studied by observing the contact angle of drops of the molten salt under a hot-stage microscope in the presence of different gases. The wettability of graphite by these mixtures became less as the ZrF₄ content was increased and the ZrO₂ content reduced. The wettability of graphite by zirconia-free mixtures increased when an argon atmosphere was replaced by CO₂, while for mixtures containing ZrO₂ the change of atmosphere had no effect. CF₄ and CO did not appear to have any significant influence on the wettability of graphite by the mixtures examined. Polished graphite surfaces were much less well wetted by NaF-ZrF₄-ZrO₂ mixtures than machined graphite surfaces. (auth)

wetting + graphite + oxides + molten salts

ECX710011

Chang SJ + Carpenter JA + Alton DW

VISCOELASTIC ANALYSIS OF IRRADIATED GRAPHITE WITH VARIABLE
CREEP COEFFICIENT

Oak Ridge National Laboratory, Tenn.

Category E
Graphite

ECX710011 *Continued*

ORNL-TM-3242 (May 1971), 31 p, 5 fig, 3 ref.

This report is an addendum to ORNL-TM-2407 concerning a method of stress analysis for irradiated graphite which may be used for MSBR core design. To provide a refined analysis, the present method includes the effect of a variable creep coefficient which is caused by the nonuniform temperature distribution. To facilitate a simple formulation, it is assumed that the temperature dependence of the elastic response of the material is approximated to be inversely proportional to the creep rate. It is shown that the problem reduces to the solution of several associated (fictitious) elastic problems which have a common elastic modulus inversely proportional to the creep rate of the irradiated graphite. Numerical examples in the previous report were recalculated based on the present theory. It shows, for large dose values, an improvement to the previous method. A computer program is written for the purpose and can include the previous solution as a special case.

stress + analysis + graphite + radiation damage +
elasticity + MSBR + creep

EDX640016

Cook WH

MSRE GRAPHITE (IN MSRP PROGR. REPT., 7/31/64)

Oak Ridge National Laboratory, Tenn.

ORNL-3708 (Nov. 1964), pp. 373-389, 10 fig, 18 ref.

The graphite purchased for the MSRE, grade CGE, is a new nuclear graphite that is basically an extruded petroleum coke bonded with coal tar pitch heated to 2800 deg C. Low permeation is obtained through a series of impregnations and heat treatments. The final heat treatment was at 2800 deg C or higher. Experimental equipment and processes were used on a commercial scale for the first time. The graphite was produced as a 2-1/2-in.-square x 72-in.-long bars, which were machined to the required shapes. The average bulk density is 1.86 grams per cubic centimeter. Its matrix is not permeated by molten salts under conditions more severe than those expected in the MSRE. It exceeded all the requirements specified for the MSRE except that it had longitudinal cracks. Tests indicated that the cracks should not have any significant adverse effect on the operation of the MSRE. The shrinkage of the graphite at 350 to 475 deg C under exposures between 0.60×10^{20} to 1.40×10^{20} neutrons/sq-cm (E greater than 2.9 MeV) indicated that this should not create any important adverse effects on the operation of the MSRE.

*graphite + *MSRE + cracks + density + elasticity +
examinations + fabrication + inspection + intrusion +
irradiation + physical properties + progress report +

Accession Number ECX710011 to EDX640016

Category E
Graphite

EDX640016 *Continued*

procurement + reviews + specific heat + specifications +
thermal conductivity + heat treatments + tensile properties +
flexural properties + compressive properties + microstructure
OTHER CATEGORIES: ACE

EDX68C031

Kasten PR + Bettis ES + Cook WH + Eatherly WF +
Holmes DK + Kedl RJ + Kennedy CR + Kirslis SS +
McCoy HE + Perry AM + Robertson RC + Scott D +
Strehlow RA

GRAPHITE BEHAVIOR AND ITS EFFECTS ON MSBR PERFORMANCE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2136 (Feb. 1968), 97 p, 22 fig, 43 ref.

Graphite behavior under MSBR conditions is reviewed and its influence on MSBR performance estimated. The deposition of fission products on graphite does not appear to be large (10 to 35 % of the noble-metal fission products based on MSRE experience). Taking into account graphite replacement every two years, fission product deposition reduces the MSBR breeding ratio by about 0.002. Also, it appears that xenon poisoning can be kept at a 0.5% fraction poisoning level by using pyrolytic carbon as a pore impregnant to seal the surface and/or by efficient gas stripping of the fuel salt fluid by injection and removal of helium bubbles. Published in slightly abbreviated form in Nucl. Eng. Design 9, 157-95 (1969).

+MSBR + *graphite + *reviews + irradiation +
mechanical properties + creep + development +
breeding performance + xenon

EDX690051

Kasten PR + Bettis ES + Cook WH + Eatherly WF +
Holmes DK + Kedl RJ + Kennedy CR + Kirslis SS +
McCoy HE + Perry AM + Robertson RC + Scott D +
Strehlow RA

GRAPHITE BEHAVIOR AND ITS EFFECTS ON MSBR PERFORMANCE

Oak Ridge National Laboratory, Tenn.

Nucl. Eng. Design 9, 157 (Feb. 1969), 39 p, 18 fig, 40 ref.

Graphite behavior under MSBR conditions is reviewed and its influence on MSBR performance estimated. The deposition of fission products on graphite does not appear to be large (10 to 35% of the noble-metal fission products based on MSRE experience). Taking into account graphite replacement every two years, fission product deposition reduces the MSBR breeding ratio by about 0.002. Also, it appears that xenon poisoning can be kept at a 0.5% fraction poisoning level by using pyrolytic carbon as a pore impregnant to seal the surface and/or by efficient gas stripping of the fuel salt fluid by injection and removal of helium bubbles. Published with somewhat more detail as ORNL-TM-2316.

Category E
Graphite

EDX690051 *Continued*

*MSBR + *graphite + *reviews + irradiation +
mechanical properties + creep + development +
breeding performance + xenon

EXX700048

Engle GB + Frice RJ + Bokros JC + White JL

RESEARCH ON GRAPHITE -- THREE-YEAR SUMMARY REPORT

May 15, 1967, through May 14, 1970

Gulf General Atomic, San Diego, Calif.

GA-9975 (June 1, 1970), 111 p, 36 fig, 47 ref.

A detailed summary is given of work at GGA related to the formation, properties, and irradiation performance of graphite; other forms of carbon were studied to complement work on graphite. The seven main topics are Morphology of the Carbonaceous Mesophase Formed in the Pyrolysis of Coal-Tar Pitch, Catalytic Graphitization of Pyrolytic Carbons, Petroleum Cokes, and Graphites, Characterization of Graphites, Model Materials, Annealing of Irradiated Pyrolytic Carbons, Highly Oriented Graphites, and Artificial Commercial Graphites. Reports giving more detail are listed.

carbon + coke + creep + density + expansion + fabrication +
graphite + irradiation + physical properties + pyrocarbon +
thermal conductivity + x-rays + lattice + heat treatments +
microstructure

Category F
Hastelloy N and Related Alloys

FAX620004

DeVan JH + Evans (III) RB

CORROSION BEHAVIOR OF REACTOR MATERIALS IN FLUORIDE SALT MIXTURES

Oak Ridge National Laboratory, Tenn.

ORNL-TM-328 (Sept. 1962), 35 p, 10 fig, 12 ref.

The report discusses (1) corrosion experiments dealing with fluoride salts in support of the MSRE, and (2) analytical methods employed to interpret corrosion and mass-transfer behavior. The products of corrosion are soluble in the molten salt; accordingly passivation is precluded and corrosion depends directly on the thermodynamic driving force of the corrosion reactions. Compatibility, therefore, demands salt constituents that are not appreciably reduced by useful structural alloys and container materials whose components are near thermodynamic equilibrium with the salt medium. Utilizing information gained in corrosion testing of commercial alloys and in fundamental interpretations of the corrosion process, ORNL developed a high-strength nickel-base alloy containing 17% Mo, 7% Cr, and 5% Fe. Several long-term corrosion loops and in-pile capsule tests completed with this alloy demonstrate the excellent corrosion resistance to fluoride salt mixtures at high temperatures. Thermodynamic methods are presented for predicting corrosion rates. Radiotracer studies confirmed the corrosion model. Also published as pp. 557-579 in Corrosion of Reactor Materials, IAEA, Vienna, 1962, Vol. II.

alloys + compatibility + *corrosion + *development +
fluorides + *Hastelloy N + loop + nickel + fuels +
thermodynamics + molten salts

OTHER CATEGORIES: FBD

FAX620005

DeVan JH + Evans (III) RB

CORROSION BEHAVIOR OF REACTOR MATERIALS IN FLUORIDE SALT MIXTURES

Oak Ridge National Laboratory, Tenn.

Corrosion of Reactor Materials, Vol. II (Proc. Conf.

June 4-8, 1962) IAEA, Vienna, p. 557, 23 p, 10 fig, 12 ref.

This paper covers the same material as ORNL-TM-328 (FAX620004).

alloys + compatibility + *corrosion + *development +
fluorides + *Hastelloy N + loop + nickel + fuels +
thermodynamics + *molten salts

OTHER CATEGORIES: FBD

FAX690035

DeVan JH

EFFECT OF ALLOYING ADDITIONS ON CORROSION BEHAVIOR OF NICKEL-MOLYBDENUM ALLOYS IN FUSED FLUORINE MIXTURES

Accession Number FAX620004 to FAX690035

Category F
Hastelloy N and Related Alloys

FAX690035 *Continued*

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2021 Vol. 1 (May 1969), 45 p, 13 fig, 16 ref.

Corrosion properties of nickel-molybdenum alloys with various solid-solution strengthening additions were tested in thermal convection loops, which circulated salt mixtures between 815 and 650 deg C. The alloys contained 17 to 20% Mo and various percentages of Cr, Al, Ti, V, Fe, Nb, and W. Loops of individual alloys were exposed to the salt mixture NaF-LiF-KF-UF₄ (11.2-45.3-41.0-2.5 mole %) for 500 and 1000 hr. The corrosion susceptibility of alloying additions increased in this order: Fe, Nb, V, Cr, W, Ti, and Al. Metallographic examinations showed relatively light attack for all alloys except those containing combined aluminum and titanium or aluminum and chromium. A nickel-base alloy containing 17% Mo, 7% Cr, and 5% Fe, designated Hastelloy N, had the best combination of strength and corrosion resistance among the compositions tested.

alloys + compatibility + corrosion + development +
fluorides + fuels + molten salts + Hastelloy N + loop +
nickel alloys + alloy composition
OTHER CATEGORIES: FEB

FAX690045

DeVan JH

EFFECT OF ALLOYING ADDITIONS ON CORROSION BEHAVIOR OF
NICKEL-MOLYBDENUM ALLOYS IN FUSED FLUORIDE MIXTURES

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2021 Vol. 1 (May 1969), 45 p, 13 fig, 16 ref.

Corrosion properties of nickel-molybdenum alloys with various solid-solution strengthening additions were tested in thermal convection loops, which circulated salt mixtures between 815 and 650 deg C. The alloys contained 17 to 20% Mo and various percentages of Cr, Al, Ti, V, Fe, Nb, and W. Loops of individual alloys were exposed to the salt mixture NaF-LiF-KF-UF₄ (11.2-45.3-41.0-2.5 mole %) for 500 and 1000 hr. The corrosion susceptibility of alloying additions increased in this order: Fe, Nb, V, Cr, W, Ti, and Al. Metallographic examinations showed relatively light attack for all alloys except those containing combined aluminum and titanium or aluminum and chromium. A nickel-base alloy containing 17% Mo, 7% Cr, and 5% Fe, designated Hastelloy N, had the best combination of strength and corrosion resistance among the compositions tested.

compatibility + corrosion + development + fluorides +
fuels + Hastelloy N + loop
OTHER CATEGORIES: FEB

FBA660020

Category F
Hastelloy N and Related Alloys

FBA660020 *Continued*

McCoy HE

STUDIES OF THE CARBON DISTRIBUTION IN HASTELLOY N

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1353 (Feb. 1966), 24 p, 14 fig, 6 ref.

A small heat of Hastelloy N was prepared in which a portion of the carbon atoms were tagged as carbon-14. The response to heat treatment was studied to determine whether the changes in mechanical properties could be correlated with the observed changes in the carbon distribution. Although marked segregation resulted, the changes in mechanical properties did not appear to be related. A second objective was to determine whether the relatively large precipitate particles in this alloy were carbides. These precipitates, in both their stringer (low-temperature) and lamellar (high-temperature) forms, were found to be as low in carbon as the matrix or lower. It is hypothesized that the other alloying elements reduce the solubility of molybdenum in nickel so that the precipitate is basically nickel-molybdenum intermetallic compounds.

*heat treatments + welding + *metallography + *Hastelloy N + carbon + mechanical properties + *microstructure

FBA680029

Gehlbach RE + McCoy HE

PHASE INSTABILITY IN HASTELLOY N

Oak Ridge National Laboratory, Tenn.

Int. Sym. on Structural Stability in Superalloys,

Seven Springs, Pa., Sept. 4-6, 1968, Vol. II, pp 346-366.

Available from Dr. John Radavich, AIME High-Temperature Alloys Committee, Micromet Laboratories, West Lafayette, Ind., 21 p, 14 fig, 0 ref.

Though Hastelloy N is basically a solid-solution alloy, thermomechanical treatments change its mechanical properties and microstructure. Identifying and characterizing precipitates involved microscopy, extraction replication, x-ray diffraction, and electron probe microanalysis. Chemical analysis with a microprobe attachment for the electron microscope and electron diffraction were employed to identify individual particles, agglomerates, and grain-boundary films on extraction replicas without interference from the matrix. The microstructure is characterized by stringers of massive primary precipitates of the Ni₃Mo₃C type. Exposure between 500 and 1000 deg C precipitates particles of the Ni₂Mo₄C type in the grain boundaries. In air-melted heats with 0.6% Si, the precipitates are enriched in silicon and are not dissolved at high temperatures but melt and transform to a noncarbide phase. In vacuum-melted heats with low silicon contents, carbides go into solid solution. The only precipitates that form in

Category F
Hastelloy N and Related Alloys

FBA680029 *Continued*

air-melted alloys at as high as 1180 deg C are the Ni₃(Mo, Cr)₃(C, Si) and Ni₂(Mo, Cr)₄(C, Si) types. The amount and behavior are highly silicon dependent; this impurity stabilizes the particles. The delta-NiMo intermetallic is probably responsible for the increased embrittlement at high annealing temperatures.

heat treatments + Hastelloy N + precipitation + microstructure

FBB65C018

Donnelly FG

TUBE PLUGGING IN THE MOLTEN-SALT REACTOR EXPERIMENT

PRIMARY HEAT EXCHANGER

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1023 (Feb. 1965) 11 p, 6 fig, 4 ref.

To reduce the pressure drop through the shell side of the MSRE primary heat exchanger, it was decided to remove four of the outer U-tubes. This required sealing the eight tube stubs produced. A plug design and seal welding procedure were developed to assure a high-integrity seal between the molten fuel salt on the shell side and the coolant salt on the tube side of the heat exchanger. The plugs had a slight interference fit (0.0000 to 0.0002 in.) with the tubes and were machined for edge-welding. The plug material was Hastelloy N, as was the entire heat exchanger. The tube end was manually welded to the plug with a gas tungsten-arc torch. The conditions were adjusted to provide weld metal penetration equivalent to at least the thickness of the tube wall. Visual, dye-penetrant, and radiographic examinations of the welds gave every indication that high-integrity welds had been made that would successfully isolate the fuel salt from the coolant salt during the planned operation of the heat exchanger.

fabrication + Hastelloy N + heat exchangers + MSRE + welding

OTHER CATEGORIES: HCX

FBB66C021

Gilliland FG + Venard JT

ELEVATED TEMPERATURE MECHANICAL PROPERTIES OF WELDS IN A

Ni-Mo-Cr-Fe ALLOY

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1341 (Jan. 1966), 35 p, 18 fig, 6 ref.

The contents of this TM appear in Welding J. (N. Y.) 45, 103-s-110-s (1966) AC FBB660022.

*Hastelloy N + *welding + heat treatments + creep + ductility + tensile properties

OTHER CATEGORIES: FEC

FBB66C022

Gilliland FG + Venard JT

ELEVATED TEMPERATURE MECHANICAL PROPERTIES OF WELDS IN A

Category F
Hastelloy N and Related Alloys

FBB66022 *Continued*

Ni-Mo-Cr-Fe ALLOY

Oak Ridge National Laboratory, Tenn.

Welding J. (N. Y.) 45, 103-s (Mar. 1966), 8 p, 18 fig, 6 ref.

Tensile tests on transverse weld samples of Hastelloy N in the as-welded and annealed conditions show a good combination of strength and ductility from 70 to 1800 deg F. Tensile properties of these compare favorably with those of the base metal. Stress relieving at 1600 deg F for 2 hr lowered the tensile yield strength. Creep-rupture tests at 2200, 1300, and 1500 deg F showed significant improvement in strength and ductility at 1300 deg F from stress relief in hydrogen. In creep-rupture behavior, both as-welded and stress-relieved specimens were as good as the base metal. The nil-ductility temperature, as determined by simulated heat-affected zone thermal cycle tests was 2300 deg F. Reasonable recovery of mechanical properties followed a simulated welding cycle with a 2300 deg F maximum temperature. The contents of this paper also appear as CRNL-TM-1341, AC-FEE660022.

*Hastelloy N + *welding + heat treatments + creep + ductility + tensile properties

OTHER CATEGORIES: FBC

FBB690040

McCoy HE + Canonico DA

PREIRRADIATION AND POSTIRRADIATION MECHANICAL PROPERTIES
OF HASTELLOY N WELDS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2483 (Mar. 1969), 43 p, 20 fig, 16 ref.

Welds were made by the TIG process in several heats of Hastelloy N. The mechanical properties of transverse weld samples and the base metal were compared in tensile tests over the range of 75 to 1600 deg F and in creep tests at 1200 deg F. The as-fabricated welds exhibited lower fracture strains than the base metal under all test conditions, but the properties of the welds were improved markedly by post-weld heat treatments. The postirradiation tensile and creep properties of the welds and base metal at elevated temperatures were about the same, although the properties were widely different before irradiation. Mechanical properties of all specimens tested are tabulated in an Appendix. The same material without the appendix is published in Welding J. (N. Y.) 48, 203-s-211-s (1969), AC-FBB690041.

*Hastelloy N + *welding + creep + ductility + *irradiation + tensile properties

OTHER CATEGORIES: FEE

FBB690041

McCoy HE + Canonico DA

Accession Number FBB660022 to FEB690041

Category F
Hastelloy N and Related Alloys

FBB690041 *Continued*
PREIRRADIATION AND POSTIRRADIATION MECHANICAL PROPERTIES
OF HASTELLOY N WELDS

Oak Ridge National Laboratory, Tenn.

Welding J. (N.Y.) 48, 203-s (May 1969), 9 p, 18 fig, 16 ref.

This paper presents the same material as CFNL-TM-2483 (FBB690040) except that it does not include the appendix which lists the properties of all specimens.

*Hastelloy N + *welding + creep + ductility + *irradiation + tensile properties

OTHER CATEGORIES: FEE

FBB700028

McCoy HE + Gunkel RW + Slaughter GM

TENSILE PROPERTIES OF HASTELLOY N WELDED AFTER IRRADIATION

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2858 (Apr. 1970), 24 p, 7 fig, 8 ref.

Fusion welds affecting 75% of the cross section were made in small tensile samples 0.125 in. in diameter) of Hastelloy N irradiated to thermal fluences up to 9.4×10^{20} neutrons/sq-cm. All of the unirradiated samples and 67% of the irradiated samples were satisfactorily welded by a specialized technique. Surface contamination is suspected to cause the unsuccessful welds in the irradiated samples. The welded irradiated samples generally had as good tensile properties at 25 and 650 deg C as the irradiated base metal. The weld metal decreased appreciably at 650 deg C and made a significant contribution to the overall fracture strain. The fracture location in the irradiated samples tested at 650 deg C shifted from the weld metal to the base metal following the post-weld anneal of 8 hr at 870 deg C. Porosity near the fusion line of the irradiated samples probably resulted from helium bubbles, but this did not seem to affect the location of the fracture.

welding + irradiation + Hastelloy N • ductility + tensile properties

FBB700031

Cepolina AG + Cancicco DA

THE MEASUREMENT OF RESIDUAL STRESSES

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3113 (Oct. 1970) 32 p, 12 fig, 16 ref.

A modification of the Sachs 'boring-out' method for determining residual stresses permits determination of the distribution of stresses and their levels over extremely short increments of distances. This technique was used for measuring residual stresses in gas tungsten-arc welds made in Hastelloy N. Circular welds 6 in. in diameter were simultaneously deposited on both flat faces of a 1/2-in.-thick plate, 12 in. in diameter. The maximum tangential residual stress was found to be about 50,000 psi and was

Category F
Hastelloy N and Related Alloys

FBB700031 *Continued*

not particularly affected by either the shielding gas or heat input. Stress relieving at 1600 deg F for 4.5 hr proved to be the optimum heat treatment and reduced the tangential residual stress to about 5000 psi. Lowering of the maximum residual stress to about 10,000 psi was achieved at 1400 deg F after 6 hr; however, lower temperatures even for times as long as 100 hr only reduced the maximum residual stress by about 25%.

Hastelloy N + stress + heat treatments + testing

FBC590001

Carlson RG

FATIGUE STUDIES OF INOR-8

Battelle Memorial Institute, Columbus, Ohio

BMI-1354 (June 1959), 16 p, 8 fig, 1 ref.

The temperature and frequency dependence of fatigue properties of Hastelloy N were determined by rotating-beam fatigue tests. Stress-lifetime data were obtained for temperatures of 1100, 1300, and 1500 F, and cyclic frequencies of 100, 600, and 3000 cpm. The fatigue strength decreased with increasing temperature. No appreciable frequency effect was found up to 1300 F. At 1500 F, the fatigue strengths of specimens tested 600 and 3000 cpm were equal, while the fatigue strength at 100 cpm was substantially lower. A critical frequency is associated with each temperature, above which frequency has no effect, but below which fatigue strength decreases with decreasing frequency. Fatigue strength was higher for finer grained material.

Hastelloy N + microstructure + fatigue

FBC610001

Swindeman FW

THE MECHANICAL PROPERTIES OF INOR-8

Oak Ridge National Laboratory, Tenn.

ORNL-2780 (Jan. 1961), 76 p, 45 fig, 20 ref.

Tensile, creep, and relaxation tests were performed on INOR-8. (This is the alloy later called Hastelloy N.) The mechanical properties are summarized and discussed in relation to the composition, microstructure, and environment. The results indicate that the minimum strength properties of INOR-8 are sufficient to permit the use of workable design stresses up to 1300 deg F, although certain areas are pointed out where additional information is desirable.

Hastelloy N + creep + ductility + heat treatments + tensile properties + alloy composition

FBC640017

McCoy HE

Category F
Hastelloy N and Related Alloys

FBC640017 *Continued*

INFLUENCE OF SEVERAL METALLURGICAL VARIABLES ON THE
TENSILE PROPERTIES OF HASTELLOY N

Oak Ridge National Laboratory, Tenn.

ORNL-3661 (Aug. 1964), 63 p, 35 fig, 10 ref.

The tensile properties of Hastelloy N were measured after various heat treatments. One vacuum-melted and four air-melted heats were studied. The vacuum-melted material exhibited good ductility after all heat treatments. Annealing the air-melted material to temperatures in excess of 2150 deg F significantly reduced the minimum fracture strain. Holding at about 1600 deg F for an extended period recovered the fracture ductility. Aging in the 1100 to 1200 deg F range material that had been previously annealed at 2150 deg F significantly reduced the ductility. These changes in ductility occurred with very small changes in tensile strength. These effects can be explained in terms of the formation of a brittle grain boundary layer along which a crack can propagate easily at elevated temperatures. Interrupting the continuity of this layer by overaging or cold working recovers good fracture ductility. The formation of this layer is associated with the presence of trace alloying elements.

Hastelloy N + heat treatments + ductility + metallography +
tensile properties

FBC650017

Venard JT

TENSILE AND CREEP PROPERTIES OF INCR-8 FOR THE MCLTEN-SALT
REACTOR EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1017 (Feb. 1965), 22 p, 19 fig, 6 ref.

Tensile and creep-rupture testing has been carried out on three heats of Hastelloy N selected from those used for the MSRE. The primary aim was to collect strength information representative of the construction material and to compare the data on these commercial heats with that from early experimental heats. The data reported are ultimate tensile strength, 0.2% off-set yield strength, elongation, and reduction in area from room temperature to 982 deg C (1800 deg F). Creep-rupture behavior was investigated at 493, 704, and 816 deg C (1100, 1300, and 1500 deg F). In general, the commercial MSRE construction material shows greater strength and ductility than did earlier heats. Additional confidence in the MSRE design strength values is thus in order.

*Hastelloy N + creep + ductility + MSRE + tensile properties +
alloy composition

FBD690036

Koger JW + Litran AP

Category F
Hastelloy N and Related Alloys

FBD69C036 *Continued*

COMPATIBILITY OF HASTELLOY N AND CROLOY 9M WITH
NaBF₄-NaF-KBF₄ (90-4-6 mole %) FLUOROBORATE SALT
Oak Ridge National Laboratory, Tenn.

ORNL-TM-2490 (Apr. 1969), 41 p, 20 fig, 15 ref.

The compatibility of relatively impure (greater than 3000 ppm impurities) NaBF₄-NaF-KBF₄ (90-4-6 mole %) tested with Hastelloy N and Croloy 9M was tested in natural circulation loops at a maximum temperature of 605 deg C with a temperature difference of 145 deg C. The Croloy 9M-loop was completely plugged after 1440 hr and the Hastelloy N loop was three-quarters plugged after 8760 hr (one year). All major alloying elements mass transferred as the result of nonselective attack by virtue of the initial oxygen and water contamination of the salt. Saturation concentrations of 700 ppm Fe and 470 ppm Cr were determined for the fluoroborate salt at 460 deg C. Initially, soluble metal fluoride compounds formed in the hot leg. The reverse reaction in the cold leg causes the metal to deposit and to diffuse into the cold leg. This continues until an equilibrium concentration of one or more metal fluorides is reached in the salt at the cold-leg temperature and these compounds start depositing or the equilibrium constant of the reaction changes so much with temperature that the pure metal is deposited.

*iron alloys + *Hastelloy N + compatibility + *corrosion +
*fluoroborates + *impurities + mass transfer + solubility +
coolants + thermal convection + molten salts

OTHER CATEGORIES: GEX

FBE650015

Martin WR + Weir JR

EFFECT OF ELEVATED TEMPERATURE IRRADIATION ON THE STRENGTH AND
DUCTILITY OF THE NICKEL-BASE ALLOY HASTELLOY N

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1005 (Feb. 1965), 17 p, 7 fig, 17 ref.

The contents of this TM are the same as an article with the same title in the April, 1965 Nuclear Applications, which see.

*Hastelloy N + ductility + *tensile properties + irradiation

OTHER CATEGORIES: FBC

FBE650016

Martin WR + Weir JR

EFFECT OF ELEVATED TEMPERATURE IRRADIATION ON THE STRENGTH AND
DUCTILITY OF THE NICKEL-BASE ALLOY HASTELLOY N

Oak Ridge National Laboratory, Tenn.

Nucl. Appl. 1: 160 (1965) 8 p, 6 fig. 20 ref.

The tensile properties of Hastelloy N have been determined after irradiation at 600 deg C to 7×10^{20} n/sq-cm (E greater than 1 MeV) and 9×10^{20} n/sq-cm (thermal). The

Accession Number FBD69C036 to FBE650016

Category F
Hastelloy N and Related Alloys

FBE650016 *Continued*

strength and ductility were determined for the range 20 to 900 deg C. The stress-strain relationship is not affected by irradiation at 700 deg C. Ductility, as measured by the true uniform and fracture strains, is reduced for deformation temperatures of 500 deg C and above. The loss in ductility results in a reduction in the true tensile strength, especially in intergranular failure, such as low strain rates and elevated temperature. Post irradiation annealing does not improve ductility. These data are compatible with helium generation from the (n, alpha) reaction of boron as the cause of low ductility. (Also published as ORNL-TM-1005.)

*Hastelloy N + ductility + *tensile properties + irradiation
OTHER CATEGORIES: FBC

FBE660019

Martin WR + Weir JR

POSTIRRADIATION CREEP AND STRESS RUPTURE OF HASTELLOY N

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1515 (June 1966) 31 p, 12 fig, 15 ref.

The contents of this TM are the same as an article with the same title in the March 1967 Nuclear Applications, which see.

*Hastelloy N + *creep + ductility + irradiation + microstructure

OTHER CATEGORIES: FBC

FBE670029

Martin WR + Weir JR

POSTIRRADIATION CREEP AND STRESS RUPTURE OF HASTELLOY N

Oak Ridge National Laboratory, Tenn.

Nucl. Appl. 3, 167 (Mar. 1967), 11 p, 10 fig, 17 ref.

The creep ductilities of irradiated Hastelloy N at 650 deg C have been determined at several neutron exposures. Elevated-temperature irradiation embrittlement greatly reduces the stress-rupture strength as measured in postirradiation uniaxial stress tests. The reduction in ductility to values as low as 0.4% is due to an irradiation effect related to the process of intergranular fracture. Intergranular cracks, once formed, propagate with greater ease in the irradiated alloy as compared with a sample exposed to a lesser radiation exposure. (Also published as ORNL-TM-1515.)

*Hastelloy N + *creep + ductility + irradiation + microstructure

OTHER CATEGORIES: FBC

FBE670030

McCoy HE + Weir JR

IN- AND EX-REACTOR STRESS-RUPTURE PROPERTIES OF HASTELLOY N TUBING

Oak Ridge National Laboratory, Tenn.

Category F
Hastelloy N and Related Alloys

FBE670030 *Continued*

ORNL-TM-1906 (Sept. 1966), 27 p, 14 fig, 28 ref.

The stress-rupture properties of two heats of Hastelloy N tubing have been determined at 760 deg C in the irradiated and unirradiated conditions. Irradiation reduced the rupture life and the rupture strain but produced no detectable effects on the creep rate. Small variations in behavior of tubular specimens tested during irradiation and small rod specimens tested after irradiation are explained on the basis of differences in stress states and sizes of test sections. The effects of irradiation are rationalized on the basis of the behavior of helium which is formed in the metal as a result of the reaction of boron-10 with thermal neutrons. (Also published with some condensation in Nuclear Applications (MSRIS Accession FEE680025).

Hastelloy N + creep + ductility + irradiation

OTHER CATEGORIES: FBC

FBE670031

McCoy HE

AN EVALUATION OF THE MOLTEN SALT REACTOR EXPERIMENT

HASTELLOY N SURVEILLANCE SPECIMENS -- FIRST GROUP

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1997 (Nov. 1967), 57 p, 35 fig, 12 ref.

Gives test results on the effect of various variables (temperature, strain rate, prestrain, etc.) on the tensile ductility of irradiated and unirradiated Hastelloy N. Specimens removed from the MSRE after 7823 MWh had been at 645 deg C for 4800 and accumulated 1.3×10^{20} neutrons/sq-cm (thermal). The high-temperature ductility was reduced similarly to that observed for the same materials in the ORR in helium. No corrosion was observed, but a 1 to 2 mil carbon-rich layer was noted where specimens touched graphite

*MSRE + *surveillance + *Hastelloy N + *compatibility +
fluorides + creep + corrosion + irradiation +
tensile properties + tensile properties

FBE680025

McCoy HE + Weir JR

STRESS-RUPTURE PROPERTIES OF IRRADIATED AND UNIRRADIATED HASTELLOY N
TUBES

Oak Ridge National Laboratory, Tenn.

Nucl. Appl. 4, 96 (Feb. 1968), 9 p, 6 fig, 24 ref.

The stress-rupture properties of two heats of Hastelloy N tubing have been determined at 760 deg C in the irradiated and unirradiated reaction of boron-10 with thermal neutrons.

(Reported in more detail in ORNL-TM-1906.)

Hastelloy N + creep + ductility + irradiation

OTHER CATEGORIES: FEC

FBE680026

McCoy HE

EFFECTS OF IRRADIATION ON THE MECHANICAL PROPERTIES OF TWO VACUUM-

Category F
Hastelloy N and Related Alloys

FBE680026 *Continued*

MELTED HEATS OF HASTELLOY N

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2043 (Jan. 1968), 43 p, 24 fig, 18 ref.

The mechanical behavior of two vacuum-melted heats of Hastelloy N was tested at 650 and 760 deg C. The material was subjected to several thermal-mechanical treatments and then irradiated at 650 and 760 deg C to a thermal dose of 2.3×10^{20} (20th) neutrons/sq-cm. The results are compared with those for unirradiated specimens that were given a similar thermal treatment. The various thermal-mechanical treatments had some relatively small effects on the tensile properties of unirradiated material, but the creep properties were very similar. The primary effects of irradiation were reductions in the creep-rupture life and the rupture ductility in both creep and tensile tests. These observations are explained on the basis of helium production in the metal by the boron-10 (n, alpha) transmutation.

*Hastelloy N + creep + ductility + heat treatments + tensile properties + irradiation

OTHER CATEGORIES: FBC

FBE690034

McCoy HE

AN EVALUATION OF THE MOLTEN-SALT REACTOR EXPERIMENT

HASTELLOY N SURVEILLANCE SPECIMENS -- SECOND GEORGE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2359 (Feb. 1969), 69 p, 45 fig, 22 ref.

Two rods of standard Hastelloy N from the surveillance position outside the core vessel were exposed to nitrogen plus 2 to 5% oxygen for 11,000 hr. The alloy was compatible with this environment, showing only superficial oxidation and no nitriding. These samples were exposed to a thermal fluence of 1.3×10^{19} (19th) neutrons/sq-cm, and both tensile and creep tests showed significant changes in mechanical properties, particularly the strain at fracture. These changes are in good agreement with those for material irradiated in helium in the CRR. Two rods of modified Hastelloy N containing small additions of titanium and zirconium from the core with a thermal fluence of 4.1×10^{20} (20th) neutrons/sq-cm, showed slightly improved postirradiation mechanical properties and acceptable corrosion resistance.

*MSRE + *surveillance + *Hastelloy N + *modified Hastelloy N + *compatibility + nitrogen + oxygen + fluorides + creep + rupture + corrosion + irradiation + irradiation + tensile properties + microstructure

OTHER CATEGORIES: FCE

FBE690044

McCoy HE

Category F
Hastelloy N and Related Alloys

FBE690044 *Continued*
VARIATION OF THE MECHANICAL PROPERTIES OF IRRADIATED
HASTELLOY N WITH STRAIN RATE

Oak Ridge National Laboratory, Tenn.

J. Nucl. Mater. 31, 67 (May 1969), 19 p, 12 fig, 44 ref.

The postirradiation mechanical properties of several heats of Hastelloy N, both vacuum- and air-melted, have been measured after exposure to thermal fluences of 2 to 6×10^{20} neutrons/sq-cm. At strain rates normally encountered in tensile tests, the fracture strain is quite sensitive to strain rate in the range of 500 to 650 deg C. At 650 deg C a minimum fracture strain was observed at a strain rate of approximately 0.1% hour; the strain increased rapidly with increasing strain rate and increased gradually with decreasing strain rate. Although the fracture strains at high strain rates differed significantly for test temperatures of 650 and 760 deg C, the strains were the same under creep conditions. A titanium-modified alloy showed improved resistance to irradiation damage. Qualitative explanations are given for each of these observations.

irradiation + Hastelloy N + modified Hastelloy N +
ductility + creep + tensile properties + alloy composition
OTHER CATEGORIES: FBC

FBE700027

McCoy HE

AN EVALUATION OF THE MOLTEN-SALT REACTOR EXPERIMENT HASTELLOY N
SURVEILLANCE SPECIMENS -- THIRD GROUP

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2647 (Jan. 1970), 88 p, 56 fig, 8 ref.

We examined the third group of hastelloy N surveillance samples from the MSRE. Standard Hastelloy N was exposed in the core to a thermal of 9.4×10^{20} neutrons/sq-cm over 15,289 hr at 650 deg C and outside the reactor vessel to 2.6×10^{19} neutrons/sq-cm over 20,789 hr at 650 deg C. The former samples were exposed to the fuel salt and the latter to nitrogen plus 2 to 5% oxygen. The material seemed quite compatible with both environments. Postirradiation tests showed that the fracture strain was reduced at 25 deg C and above 500 deg C. The reduction at 25 deg C is likely due to carbide precipitation and that above 500 deg C is due to helium from boron-10(n, alpha). Accumulated results allow us to follow changes in fracture strain with thermal fluence from 1.3×10^{19} to 9.4×10^{20} neutrons/sq-cm. Two heats of modified Hastelloy N were irradiated in the core to a thermal fluence of 5.3×10^{20} neutrons/sq-cm over 9789 hr at 650 deg C. The postirradiation properties were better than those of standard Hastelloy N.

*MSRE + *surveillance + *Hastelloy N + *modified Hastelloy N +
*compatibility + nitrogen + oxygen + fluorides + creep +
corrosion + irradiation + tensile properties + microstructure

Accession Number FBE690044 to FBE700027

Category F
Hastelloy N and Related Alloys

FBE700027 *Continued*
OTHER CATEGORIES: FCE

FBE710017

McCoy HE + Gehlbach RE

INFLUENCE OF IRRADIATION TEMPERATURE ON THE CREEP-RUPTURE
PROPERTIES OF HASTELLOY N

Oak Ridge National Laboratory, Tenn.

Nucl. Technol. 11, 45 (May 1971), 16 p, 17 fig, 15 ref.

The variation of the postirradiation creep-rupture properties with irradiation temperature has been determined for air- and vacuum-melted Hastelloy N. The air-melted material was high in silicon and formed a stable carbide of the M₆C type. The properties of this material were not dependent upon the irradiation temperature over the range studied. The vacuum-melted alloys formed a M₂C-type carbide whose size and morphology depended markedly upon the irradiation temperature. When the carbides were finely dispersed by irradiation at about 650 deg C, the postirradiation properties were equivalent to those of the air-melted material. Irradiation at about 760 deg C resulted in coarser dispersions of the M₂C carbide and inferior postirradiation properties.

creep + ductility + Hastelloy N + irradiation +
thermal effects + alloy composition + microstructure +
carbides

OTHER CATEGORIES: FCE

FBE710018

McCoy HE

AN EVALUATION OF THE MOLTEN-SALT REACTOR EXPERIMENT
HASTELLOY N SURVEILLANCE SPECIMENS -- FOURTH GROUP

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3036 (March 1971), 91 p, 67 fig, 14 ref.

Two heats of standard Hastelloy N were removed from the core of the MSRE after 22,533 hr at 650 deg C, a thermal fluence of 1.5×10^{21} neutrons per square centimeter, and a fast fluence (greater than 50 keV) of 1.1×10^{21} neutrons per square centimeter. The mechanical properties have systematically deteriorated with increasing fluence. However, the change in properties is due to the helium produced by the B-10(n, alpha) Li-7 transmutation and can be reduced by changes in chemical composition. Some heats with modified composition have been exposed to the core of the MSRE and show improved resistance to irradiation. The corrosion of the Hastelloy N has been largely due to the selective removal of chromium. The rates of removal are much as predicted from the measured diffusion rate of chromium. Other superficial structure modifications have been observed, but they likely result from carbide precipitation along slip bands that were formed during

Category F
Hastelloy N and Related Alloys

FBE710018 *Continued*
machining.

modified Hastelloy N + MSRE + surveillance + corrosion +
creep + ductility + fluorides + Hastelloy N + irradiation +
alloy composition + molten salts + microstructure +
tensile properties

OTHER CATEGORIES: FEC + FBC + FCE

FBX640015

Taboada A

METALLURGICAL DEVELOPMENTS (IN MSRE PROGR. REPT., 7/31/64)

Oak Ridge National Laboratory, Tenn.

ORNL-3708 (Nov. 1964), pp. 330-372, 27 fig, 8 ref.

Metallurgical developments in support of the MSRE show that Hastelloy N is satisfactory. Physical, tensile, creep, and fatigue properties are given. Compatibility is excellent with circulating molten fluorides, graphite in molten fluorides, and air to 1800 deg F. Fabrication is described, including a complex heat exchanger. Annealing is needed after cold work, and attention to composition is required to ensure weldability. Irradiation causes some loss of tensile strength and ductility. Inconel-clad gadolinia-alumina control rod elements were fabricated.

*Hastelloy N + brazing + ceramics + compatibility +
control rods + corrosion + creep + density + ductility +
electrical conductivity + elasticity + erosion +
expansion + fabrication + fluorides + graphite +
heat exchangers + heat treatments + inconels + irradiation +
joints + loop + machining + melting + casting +
metallography + MSRE + oxidation + physical properties +
procurement + progress report + rare earths + reviews +
specific heat + specifications + surveillance + testing +
thermal conductivity + welding + microstructure +
tensile properties + fatigue + molten salts

OTHER CATEGORIES: ACE

FCC690048

Sessions CE + Lundy TS

DIFFUSION OF TITANIUM IN MODIFIED HASTELLOY N

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2392 (Jan. 1969), 24 p, 7 fig, 13 ref.

Diffusion coefficients of titanium-44 in titanium-modified Hastelloy N were determined over the range 800 to 1250 deg C by serial sectioning by latheing or grinding and counting by gamma-spectroscopy. The data were fitted to $D = (15.3 \text{ plus or minus } 2.2) \exp(-73,000 \text{ plus or minus } 3300/RT) \text{ sq-cm/sec}$. Results were used to predict the maximum loss by diffusion of titanium from the alloy in a typical molten-salt breeder reactor at 700 deg C. Expected increases of titanium in the molten salt are no more than 5 to 10 ppm for two years of operation, based on

Accession Number FBE710018 to FCC690048

Category F
Hastelloy N and Related Alloys

FCC690048 *Continued*

a simplified diffusion model. Also published as J. Nucl. Mater. 31, 316-22 (1969).

diffusion + titanium + modified Hastelloy N

FCC690049

Sessions CE + Lundy TS

DIFFUSION OF TITANIUM IN MODIFIED HASTELLOY N

Oak Ridge National Laboratory, Tenn.

J. Nucl. Mater. 31, 316 (July 1969), 7 p, 5 fig, 13 ref.

Diffusion coefficients of titanium-44 in titanium-modified Hastelloy N were determined over the range of 800 to 1250 deg C by serial sectioning by latheing or grinding and counting by gamma-spectroscopy. The data were fitted to $D = (15.3 \text{ plus or minus } 2.2) \exp(-73,000 \text{ plus or minus } 3300/RT) \text{ sq-cm/sec}$. Results were used to predict the maximum loss by diffusion of titanium from the alloy in a typical molten-salt breeder reactor at 700 deg C. Expected increases of titanium in the molten salt are no more than 5 to 10 ppm for two years of operation, based on a simplified diffusion model. Also published as CNL-TM-2392.

diffusion + titanium + modified Hastelloy N

FCC700040

Sessions CE

INFLUENCE OF TITANIUM ON THE HIGH-TEMPERATURE DEFORMATION

AND FRACTURE BEHAVIOR OF SOME NICKEL BASED ALLOYS (THESIS)

Oak Ridge National Laboratory, Tenn.

ORNL-4561 (July 1970), 189 p, 44 fig, 68 ref.

Adding 0.5% titanium to nickel-12% molybdenum-7% chromium-0.06% carbon decreased the creep rate, increased stress-rupture life and ductility. Increasing carbon from 0.003 to 0.3% increased rupture life and decreased creep rate four orders of magnitude and increased ductility threefold for various stresses. Fracture at 650 deg C changed from intergranular at low carbon to mixed trans- and intergranular at high carbon for similar heat treatments. Increasing titanium up to 1.2% favored formation of an MC-type carbide during aging at 650 and 760 deg C rather than the M₂C that forms at lower titanium. A heavy distribution of MC at grain boundaries resulted in superior ductility in both creep and tensile tests, presumably by reducing grain-boundary shearing and limiting growth of cracks. When alloys with 1.2% titanium were solution annealed at 1260 deg C and aged, MC carbides precipitated on dislocations, causing growth of stacking faults, which increased strength but impaired ductility. Small titanium additions improved ductility of pure nickel at 600 deg C.

*modified Hastelloy N + *creep + nickel + *ductility +

*heat treatments + carbides + precipitation + microstructure +

Category F
Hastelloy N and Related Allcys

FCC700040 *Continued*
*alloy composition
OTHER CATEGORIES: FCA

FCC700044

Sessions CE

INFLUENCE OF TITANIUM ON THE HIGH-TEMPERATURE DEFORMATION
AND FRACTURE BEHAVIOR OF SOME NICKEL-BASED ALLOYS

Oak Ridge National Laboratory, Tenn.

Scripta Met. 4, 795 (Oct. 1970), 4 p, 1 ref.

This publication is an abstract of report ORNL-4561 (MSRIS
Accession FCC700040).

*modified Hastelloy N + *creep + nickel + *ductility +
*heat treatments + carbides + precipitation +
microstructure + *alloy composition

FCC710010

Sessions CE + Stansbury EE

THERMAL STABILITY OF TITANIUM-MODIFIED HASTELLOY N AT
650 and 760 DEG C

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3321 (July 1971), 43 p, 17 fig, 19 ref.

The influence of small titanium additions on the thermal stability of Ni-12% Mo-7% Cr-0.07% C was investigated. The mechanical properties at 650 deg C (tensile tests at 0.002/min strain rate and creep tests at 40,000 psi stress) were measured for four heats of this alloy with titanium contents from 0.15 to 1.2%. Solution annealing temperatures were 1177 or 1260 deg C, and subsequent precipitation heat treatments were conducted at 650 and 760 deg C. Titanium increases the stability of a complex MC-type carbide. At low titanium levels the MC carbide is stable at 650 deg C but is unstable at 760 deg C, where an M₂C-type carbide is precipitated, resulting in inferior properties. For the higher titanium concentrations the MC carbide is stable on aging at 760 deg C and results in excellent properties after a solution anneal at 1177 deg C. However, high-titanium alloys are significantly less ductile if they are solution annealed at 1260 deg C and aged at either 650 or 760 deg C. The heat with the lowest carbon content (0.04% C) was most resistant to property changes on aging up to 10,000 hr at both 650 and 760 deg C.

*modified Hastelloy N + *development + *alloy composition +
*heat treatments + aging + creep + ductility +
microstructure + tensile properties + carbides +
thermal effects

FCD710016

Evans (III) EE + Koger JW + DeVan JH

CORROSION IN POLYTHEMAL LOOP SYSTEMS II. A SOLID-STATE
DIFFUSION MECHANISM WITH AND WITHOUT LIQUID FILM EFFECTS

Accession Number FCC700040 to FCD710016

Category F
Hastelloy N and Related Alloys

FCD710016 *Continued*

Oak Ridge National Laboratory, Tenn.

ORNL-4575, Vol. II (June 1971) 74 p, 16 fig, 49 ref.

The corrosion of alloys exposed to nonisothermal circulating liquids is important in systems with liquid coolants or coolant-fuel combinations. Mathematical descriptions were developed to explain transport of constituents of alloys. This report specializes to cases in which solid-state diffusion in the alloy dominates the corrosion. Equations are derived for both transient and steady-state cases; transients are negligible. Applicability is demonstrated by comparison of predicted values with experimental results for two systems. The first involves hot-to-cold-zone transfer of nickel in Inconel 600 pumped loops circulating sodium. Actual corrosion is much higher than predicted; this suggests that the true corrosion reaction overrides a slow diffusion process. The second system is transfer of chromium in Hastelloy N thermal convection loops with molten salt. Three examples are considered; (1) corrosion at all points, transfer to salt only; (2) hot-to-cold-zone transfer; and (3) cold-to-hot-zone transfer. Early Cr-51 tracer experiments (example 1) suggest that solid-state diffusion applies to certain molten-salt systems.

*corrosion + *models + *diffusion + analysis +
computer codes + coolant loops + fluorides + Hastelloy N +
inconels + liquid metals + loop + mass transfer +
mathematics + sodium + models

OTHER CATEGORIES: GCX

FCE690043

McCoy HE + Weir JR

DEVELOPMENT OF A TITANIUM-MODIFIED HASTELLOY N WITH
IMPROVED RESISTANCE TO RADIATION DAMAGE

Oak Ridge National Laboratory, Tenn.

Irradiation Effects in Structural Alloys for Thermal and
Fast Reactors, ASTM STP 457, Am. Soc. for Testing and
Materials, 1969, p. 290, 22 p, 11 fig, 19 ref.

The effects of neutron irradiation on the high-temperature mechanical properties of Hastelloy N are generally that the creep-rupture life and ductility are reduced. The ductility is a strong function of the strain rate and shows a minimum at a minimum creep rate of about 0.1 %/hour. The resistance to radiation damage can be enhanced greatly by adding titanium. The postirradiation creep-rupture ductility and strength rise sharply as the titanium content is increased above 0.3%. Postirradiation creep-rupture tests at 650 C on specimens irradiated to a thermal fluence of 5×10^{20} neutrons/sq-cm indicate that a ductility minimum still exists as a function of strain rate. However, the minimum strain is 3 to 5% as compared with

Accession Number FCD710016 to FCE690043

Category F
Hastelloy N and Related Alloys

FCE690043 *Continued*

0.5% for the standard alloy. In-reactor creep-rupture tests indicated the same improved properties.
irradiation + ductility + creep + Hastelloy N + modified Hastelloy N + alloy composition

FCE710004

McCoy HE

INFLUENCE OF TITANIUM, ZIRCONIUM, AND HAFNIUM ADDITIONS ON THE RESISTANCE OF MODIFIED HASTELLOY N TO IRRADIATION DAMAGE AT HIGH TEMPERATURE - PHASE I

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3064 (Jan. 1971), 146 p, 117 fig, 12 ref.

The influence of small additions of Ti, Zr, and Hf on the mechanical properties of a modified Hastelloy N with the nominal composition Ni-12% Mo-7% Cr-0.2% Mn-0.5% C is described. Test results are from numerous, small, laboratory melts and several 100-lb melts from commercial vendors. Additions of Ti, Zr, and Hf improved the properties of the alloy both unirradiated and after irradiation. Irradiation temperature had a marked effect upon the properties of all alloys investigated. Generally, good properties were observed when the irradiation temperature was 650 deg C or less and poor when the temperature was 700 deg C or higher. We attributed this large effect of irradiation temperature to coarsening of the carbide structure at the higher temperature.

modified Hastelloy N + irradiation + microstructure + creep + ductility + tensile properties + alloy composition

OTHER CATEGORIES: FCC

FCX690033

McCoy HE + Weir JR + Beatty RI + Cook WH + Kennedy CR +

Litman AP + Gehlbach RE + Sessions CE + Koger JW

MATERIALS FOR MOLTEN-SALT REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2511 (May 1969), 43 p, 13 fig, 33 ref.

The contents of this TM are the same as an article in the Feb. 1970 Nuclear Applications, which see.
MSRE + *graphite + Hastelloy N + *modified Hastelloy N + alloy composition + mechanical properties + sealing + fluoroborates + corrosion + compatibility + iron alloys + *reviews + progress report

OTHER CATEGORIES: EEX + ECX

FCX700026

McCoy HE + Beatty RL + Cook WH + Gehlbach RE + Kennedy CR +

Koger JW + Litman AP + Sessions CE + Weir JF

NEW DEVELOPMENTS IN MATERIALS FOR MOLTEN-SALT REACTORS

Oak Ridge National Laboratory, Tenn.

Nucl. Appl. Tech. 8, 156 (Feb. 1970), 14 p, 13 fig, 34 ref.

Category F
Hastelloy N and Related Alloys

FCX700026 *Continued*

Operating experience with the Molten-Salt Reactor Experiment (MSRE) has demonstrated the excellent compatibility of the graphite-Hastelloy N-fluoride salt system at 650 deg C. Several improvements in materials are needed for a molten-salt breeder reactor with a basic plant life of 30 years; specifically, (1) Hastelloy N with improved resistance to embrittlement by thermal neutrons, (2) graphite with better dimensional stability in a fast neutron flux, (3) graphite that is sealed to obtain very low surface permeability, and (4) a secondary coolant that is inexpensive and has a melting point of about 400 deg C. A brief description is given of work in progress to satisfy each of these requirements. Significant improvements are being made in each area. (This paper was also published as ORNL-TM-2511.)

MSRE + graphite + Hastelloy N + modified Hastelloy N + alloy composition + mechanical properties + sealing + molten salts + fluoroborates + corrosion + compatibility + iron alloys + reviews + progress report
OTHER CATEGORIES: EDX + FCX

FXX690047

McCoy HE

THE INOR-8 STORY

Oak Ridge National Laboratory, Tenn.

Review (Oak Ridge National Laboratory) 3, 35 (Fall 1969)

15 p, 9 fig.

Semitechnical language reviews the development of Hastelloy N, and describes the current program to improve irradiation stability. Studies of compatibility with molten fluorides, oxidation resistance, strength and fabricability led to the basic nickel-base alloy containing 15-18% Mo, 6-8% Cr, 5% Fe, 1% Mn, and 1% Si. More recent studies of microstructure and mechanical properties as influenced by irradiation show the need for reducing silicon and molybdenum and adding small amounts of titanium, hafnium, and niobium.

reviews + Hastelloy N + modified Hastelloy N + development + alloy composition

OTHER CATEGORIES: FCX

Category G
Materials Other than Hastelloy N and Graphite

GAX670033

Stiegler JO + Weir JR

EFFECTS OF IRRADIATION ON DUCTILITY

Oak Ridge National Laboratory, Tenn.

Chap. 11, p. 311 in Ductility, Papers Presented at a Seminar
of the American Society for Metals Oct 14-15, 1967, ASM,
Metals Park, Ohio, 1968, 32 p, 19 fig, 58 ref.

The mechanisms and effects of radiation damage to metals are presented, with emphasis on effects on tensile elongation. Displacement cascades from fast neutrons and transmutation effects, including the introduction of helium, are treated. Many examples show effects of several variables. Most results are shown for type 304 stainless steel, including titanium-modified material. Some results are given for Hastelloy N, molybdenum, and tungsten. Electron micrographs show bubbles and other damage. Also published as ORNL-TM-2019, AC-GAF680028.

*ductility + *stainless steels + Hastelloy N + molybdenum + tungsten + *irradiation + tensile properties + microstructure
OTHER CATEGORIES: FBE

GAX680028

Stiegler JO + Weir JR

EFFECTS OF IRRADIATION ON DUCTILITY

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2019 (Jan. 1968), 55 p, 19 fig, 58 ref.

The mechanisms and effects of radiation damage to metals are presented, with emphasis on effects on tensile elongation. Displacement cascades from fast neutrons and transmutation effects, including the introduction of helium, are treated. Many examples show effects of several variables. Most results are shown for type 304 stainless steel, including titanium-modified material. Some results are given for Hastelloy N, molybdenum, and tungsten. Electron micrographs show bubbles and other damage. Also published as pp. 311-342 in Ductility, Papers presented at a Seminar of the American Society for Metals Oct 14-15, 1967, ASM, Metals Park, Ohio, 1968, AC-GAF670033.

*ductility + *stainless steels + Hastelloy N + molybdenum + tungsten + *irradiation + tensile properties + microstructure
OTHER CATEGORIES: FBE

GAX700045

Koger JW + Litran AP

CATASTROPHIC CORROSION OF TYPE 304 STAINLESS STEEL IN A

SYSTEM CIRCULATING FUSED SODIUM FLUOROBORATE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2741 (Jan. 1970), 22 p, 5 fig, 12 ref.

A type 304 stainless steel liquid level probe contacted sodium fluoroborate containing 8 mole % sodium fluoride in an Inconel 600 pump loop at constant temperatures in the range 540 to 690 deg C for 192 hr. The probe exhibited

Accession Number GAX670033 to GAX700045

Category G
Materials Other than Hastelloy N and Graphite

GAX700045 *Continued*

heavy attack, evidence by severe leaching of chromium, iron, manganese, and silicon from the alloy. Equivalent uniform attack was about 4 mils/day. Corrosion of the stainless steel, which is inferior to nickel-base alloys in fused fluorides, became catastrophic in this system due to dissimilar-metal effects.

inconels + stainless steels + corrosion + fluoroborates + liquid level measurement + loop + molten salts

GCX610002

Adamsen GM + Crouse RS + Manly WD

INTERIM REPORT ON CORROSION BY ZIRCONIUM-BASE FLUORIDES

Oak Ridge National Laboratory, Tenn.

ORNL-2338 (Jan. 1961), 60 p, 34 fig, 3 ref.

The mixture NaF-ZrF₄-UF₄ (50-46-4 mole %), was circulated in thermal convection loops for 500 to 5000 hr at a hot-leg temperature of 1500 deg F. In Inconel-600 loops, subsurface voids were formed by selective leaching of chromium. After 500 hr of operation the voids were found to depths of about 10 mils, and the depth increased about 4 mils/1000 hr. The effects of time, hot-leg temperature, temperature drop, fluoride purity, loop size and shape, and inhibitors on the depth of corrosion were studied. The attack was reduced when a portion of the uranium was trivalent. A few tests were carried out in loops constructed from nickel, stainless steels, iron, Hastelloy B, molybdenum, and niobium. A limited amount of work was done on Inconel loops circulating alkali-metal-base mixtures (NaF, LiF, KF, UF₄) with portions of the uranium in the trivalent state. Reduced attacks were found.

*corrosion + *fluorides + *inconels + iron + metallography + molybdenum + thermal convection + loop + nickel + stainless steels + *molten salts + niobium

GCX680030

McCoy HE + McElroy DL

ELECTRICAL RESISTIVITY ANOMALY IN NICKEL-BASE ALLOYS

Oak Ridge National Laboratory, Tenn.

Trans. ASM (Am. Soc. Metals) 61, 730 (Dec. 1968), 12 p, 16 fig, 17 ref.

The electrical resistivity of eight nickel-base alloys containing iron, chromium, and molybdenum was measured to 1000 C. Alloys with more than 50 wt % Ni showed a rapid increase in resistivity between 400 and 600 C and a decreasing resistivity from about 600 to 1000 C. For these alloys the resistivity below 600 C can be changed by annealing and by cold working. The resistivity of alloys with less than 50 wt % Ni increased with temperature with a slope decrease between 400 and 600 C. The effects of annealing and cold working were relatively minor for these

Category G
Materials Other than Hastelloy N and Graphite

GCX68C030 *Continued*

alloys. The resistivity variations do not uniquely depend on any one alloying constituent although there is a weak correlation with the total nickel content. Electron microscope results indicate that these changes may be associated with short-range order.

electrical conductivity + nickel + inconels + Hastelloy N + heat treatments

GDX69C042

Koger JW + Litman AP

COMPATIBILITY OF MOLYBDENUM-BASE ALLOY TZM WITH LITHIUM

FLUORIDE-BERYLLIUM FLUORIDE-THORIUM FLUORIDE-URANIUM (IV)

FLUORIDE (68-20-11.7-0.3 mole %) AT 1100 deg C

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2724 (Dec. 1969), 16 p, 2 fig, 9 ref.

The TZM alloy (Mo-0.5% Ti-0.08% Zr-0.02% C) showed very little attack by the fused salt (LiF-BeF₂-ThF₄-UF₄, 68-20-11.7-0.3 mole %) at 1100 deg C for 1011 hr. Corrosion manifested itself as leaching of titanium and possibly zirconium from the alloy. The TZM alloy exposed to the salt partially recrystallized, while that exposed to the vapor did not. This recrystallization was attributed to the removal of titanium and zirconium. On the basis of this single test the magnitude and mechanism of corrosion indicate no serious problems for long-term use of TZM in the vacuum distillation processing scheme for the Molten Salt Breeder Reactor. However, the strength properties of the TZM alloy would approach those of unalloyed molybdenum as salt exposure increased; this is not considered a problem now.

compatibility + corrosion + molybdenum + processing + distillation + equipment + capsules + MSBR + molten salts

GDX710025

Nicholson EL

CONCEPTUAL DESIGN AND DEVELOPMENT PROGRAM FOR THE

MOLYBDENUM REDUCTIVE EXTRACTION EQUIPMENT TEST STAND

Oak Ridge National Laboratory, Tenn.

ORNL-CF-71-7-2 (July 1971), 45 p, 5 fig, 24 ref.

Reductive extraction reprocessing of molten-salt breeder reactor fuel requires that the fuel salt be contacted with molten bismuth containing lithium and thorium metals (as reducing agents) in order to remove protactinium and rare earths from the fuel salt. Bismuth is extremely corrosive to the usual materials of construction for molten salt systems, but molybdenum appears to have adequate corrosion resistance. To date, difficulties in fabrication of molybdenum have ruled against its use for vessels for engineering-scale experiments but development work in progress indicates that equipment for reductive extraction

Category G
Materials Other than Hastelloy N and Graphite

GDX710025 *Continued*

reprocessing can now be fabricated from this material. A small packed column, representative of a typical equipment unit in reductive extraction reprocessing, will be built of molybdenum and operated for metallurgical and chemical engineering evaluation in a versatile test stand in which this and future molybdenum components may be tested. This report describes the conceptual designs of the test stand and molybdenum equipment and discusses the fabrication and process development work that will be required before the equipment can be designed and built. A brief summary of the state of the art of molybdenum metallurgy is also included.

*conceptual design + *reductive extraction process +
*extraction columns + *molybdenum + *fabrication +
*bismuth + molten salts + *test facilities + *SBF +
development + plans + materials

OTHER CATEGORIES: LDB

GFX660023

Tolson GM + Taboada A

A STUDY OF LEAD AND LEAD-SALT CORROSION IN THERMAL-
CONVECTION LOOPS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1437 (Apr. 1966) 19 p, 10 fig, 5 ref.

Thermal-convection loop tests of several structural alloys were operated using circulating molten lead. Screening tests included carbon steel between 900 and 1100 deg F, type 410 stainless steel between 910 and 1210 deg F, Croloy 2-1/4 under both conditions, and niobium 1% zirconium between 1000 and 1400 deg F. Two loops contained surge tanks in which fluoride salts, Nb-1% Zr alloy, and graphite were placed in contact with the lead to determine the compatibility of these materials in a direct-cooled lead system. All of the steel loops tended to plug in the cold regions because dendritic crystals of iron and chromium formed. The hot-leg attack consisted of general surface removal, with a few large pits extending to a greater depth. The Nb-1% Zr alloy showed no measurable attack; however, niobium crystals were found in the cold leg of a loop that operated 5000 hr.

compatibility + *thermal convection + *corrosion + *iron +
linings + *lead + liquid metals + *mass transfer +
secondary systems + *stainless steels + coolants

GGX670034

Tolson GM + Taboada A

MSRE CONTROL ELEMENTS: MANUFACTURE, INSPECTION, DRAWINGS,
AND SPECIFICATIONS

Oak Ridge National Laboratory, Tenn.

ORNL-4123 (July 1967) 53 p, 8 fig, 7 ref.

The control elements for the Molten Salt Reactor are

Accession Number GDX710025 to GGX670034

Category G
Materials Other than Hastelloy N and Graphite

GGX67C034 *Continued*

Gd2O3-Al2O3 bushings canned in Inconel. The report includes material selection and development of fabrication methods. The can was made from fully inspected Inconel closed by four TIG welds. The Gd2O3-Al2O3 bushings were made by conventional pressing and sintering methods after a special prereaction step was used. The bushings were given thermal shock tests, weighed, dimensionally inspected, and given a final visual inspection for chips or cracks. As-built drawings, specifications, and manufacturing procedures are included. By methods described in this report, 160 MSRE control rod elements were manufactured.

welding + specifications + rare earths + MSRE +
nickel alloys + inconels + *fabrication + *control rods +
ceramics

GXX68C039

Metzger GE

SURVEY OF STRUCTURAL MATERIALS FOR THE MOLTEN SALT
EXPERIMENTAL (MOSEL) REACTOR

Wright-Patterson Air Force Base, Ohio.

Nucl. Eng. and Design, Vol. 7, No. 1, (Jan. 1968).

Survey of metal-base structural materials for use in molten lead and fluoride salts at temperatures between 500 and 1000 deg C. The mechanical properties, fabrication and corrosion properties are considered with respect to the Molten Salt Experimental MOSEL Reactor Concept.

*alloy composition + *converters + lead cooling +
corrosion + fabrication + mechanical properties +
molten salts + niobium + tantalum

Category H
Reactor Component Development

HAX700050

Kedl RJ

FLUID DYNAMIC STUDIES OF THE MOLTEN-SALT REACTOR EXPERIMENT
CORE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3229 (Nov. 19, 1970), 33 p, 16 fig, 10 ref.

In the MSRE reactor vessel, fluid fuel was circulated at 1200 gpm down through an annular region and up through 1140 passages in the graphite core. The core design was based on preliminary tests in a one-fifth scale model, followed by detailed measurements with water solutions in a full-scale mockup of the reactor vessel and internals. This report describes the models, the testing, and the data from which velocity, pressure drop and flow patterns are deduced. It also describes how the measurements were extrapolated to molten salt at 1200 deg F in the actual reactor. The few observations possible in the reactor were consistent with the predicted behavior.

cores + design + development + flow measurement +
fluid flow + MSRE + reactor vessel + models

HBX620006

Smith PG

WATER TEST DEVELOPMENT OF THE FUEL PUMP FOR THE MSRE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-79 (March 1962), 47 p, 19 fig, 6 ref.

A vertical-shaft, sump-type centrifugal pump with overhung impeller, of conventional hydraulic design was specified for circulating molten salts in the MSRE. This report describes water tests of a prototype, including hydraulics and the performance of a spray device for stripping gas from the circulating liquid.

*development + *MSRE + *prototypes + *pumps + components +
design + hydraulics + testing

OTHER CATEGORIES: MAB

HBX670042

Smith PG

EXPERIENCE WITH HIGH-TEMPERATURE CENTRIFUGAL PUMPS IN
NUCLEAR REACTORS AND THEIR APPLICATION TO MOLTEN-SALT
THERMAL BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1993 (Sept. 1967) 44 p, 12 fig, 8 tab, 24 ref.

Design features, development problems, and operating experience were compiled for liquid-metal- and molten-salt-circulating pumps used in various nuclear reactors and test facilities. The compilation was made to determine problem areas and select combinations of features for the pumps required by each of the three molten-salt systems. The short-shaft pump is favored for the coolant-salt system because of reliability, the long

Category H
Reactor Component Development

HBX670042 *Continued*

shaft for fuel and blanket salt systems because it provides greater thermal and radiation protection to the drive motor.
*design + *development + *liquid metals + *molten salts +
*pumps + *reviews + MSBR + two-fluid reactor

HBX69C058

Grindell AG • McGlothlan CK

CONCEPTUAL SYSTEM DESIGN DESCRIPTION OF THE SALT PUMP TEST
STAND FOR THE MOLTEN SALT BREEDER EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2043 (Aug. 1969) 53 p, 7 fig, 9 tab.

A stand is required to test the salt pumps for the Molten Salt Breeder Experiment (MSBE). It will be designed to accommodate pumps having capacities ranging from 3000 to 7000 gpm and operating with salt of specific gravities to 3.5 at discharge pressures to 400 psig and temperatures to 1300 deg F normally and to 1400 deg F for short times. Both the drive-motor electrical supply and the heat removal system for the loop will be designed for 1500 hp.

Preventive measures to protect personnel and equipment from the hazardous effects of a salt leak will be taken.

*description + *MSBE + *pumps + *test facilities + design +
development + molten salts + plans + testing

HBX690059

Wilson LV + Grindell AG

PRELIMINARY SYSTEMS DESIGN DESCRIPTION (TITLE I DESIGN) OF
THE SALT PUMP TEST STAND FOR THE MOLTEN SALT BREEDER
EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2780 (Dec. 1969), 100 p, figs, tabs.

The preliminary system design description and the Title I design calculations of the test stand are presented.

Descriptions, functions, and design requirements for components and subsystems are provided. The principles of operation of the test stand, the safety precautions, and the maintenance philosophy are discussed. The Quality-Assurance Program Plan is being prepared.

*description + *MSBE + *pumps + *test facilities +
design + development + molten salts + plans +
quality assurance

HBX70C012

Smith PG

DEVELOPMENT OF FUEL- AND COOLANT-SALT CENTRIFUGAL PUMPS FOR
THE MSRE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2987 (Oct 1970), 50 p, 18 fig, 15 ref.

The two salt pumps in the MSRE are vertical-shaft sump pumps with overhung impeller and oil-lubricated bearings.

Accession Number HBX67C042 to HBX70C012

Category H
Reactor Component Development

HBX70C012 *Continued*

The fuel pump delivers 1200 gpm and the coolant pump 800 gpm of salt at 1000 - 1200 deg F. The fuel pump is designed for remote replacement of the motor or entire rotary element and includes in the pump tank a spray device for removing xenon from the circulating fuel. A replacement fuel pump with larger tank and longer shaft was developed but never installed. This report describes the development, testing with molten salts and performance in the MSRE.

*development + *MSRE + *pumps + components + design + experience + hydraulics + maintenance + molten salts + testing

OTHER CATEGORIES: MAB

HCX68C037

Kedl EJ + McGlothlan CK

TUBE VIBRATION IN MSRE PRIMARY HEAT EXCHANGER

Cak Ridge National Laboratory, Tenn.

ORNL-TM-2098 (Jan. 1968) 43 p, 8 fig, 4 tab, 16 ref.

The primary heat exchanger for the Molten Salt Reactor Experiment was completed in 1963. Preoperational tests with water revealed excessive tube vibration and high fluid pressure drop on the shell side of the exchanger. Modifications were made to correct these deficiencies. From January 1965 through November 1967 the heat exchanger has operated for about 14,000 hrs in molten salt without indications of leakage or change in performance.

*design + *development + *heat exchangers + *MSRE + experience + hydraulics + testing + vibration

OTHER CATEGORIES: MAB

HCX710022

Bettis CE + Crowley WK + Nelms HA + Fickel TW +

Siman-Tov M + Stoddart WC

COMPUTER PROGRAMS FOR MSBR HEAT EXCHANGERS

Cak Ridge National Laboratory, Tenn.

ORNL-TM-2815 (April 1971), 158 p, 7 fig, 22 ref.

Three programs were developed to make design calculations for the heat exchangers for molten-salt reactors. The programs are: for the primary heat exchangers, PRIME1; for the reheaters, RETEX; and for the steam generator-superheaters, SUPEX. Each type of exchanger is described, the basic equations used in each analysis are given, and the logic used in each program is discussed briefly in this report. The programs developed were used in designing the four 556 MW primary exchangers, eight 36.6 MW reheaters, and sixteen 121 MW steam generator superheaters. All are basically baffled shell and tube exchangers; the steam generator superheater is a U-tube, U-shell exchanger. Flow diagrams, lists of input required and output received, complete program listings, and the nomenclature for the

Category H
Reactor Component Development

HCX710022 *Continued*

programs as well as example computer input and output for the exchangers described are appended.

*computer codes + *heat exchangers + *MSBR + primary salt + secondary salts + steam generators + steam cycle + stress + thermal properties + analysis + fluid flow + conceptual design + design data + expansion + single-fluid reactors + fluoroborates + vibration + Hastelloy N

OTHER CATEGORIES: HDX

HFX620007

Richardson M

DEVELOPMENT OF FREEZE VALVE FOR USE IN MSRE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-128 (Feb. 1962), 24 p, 8 fig, 2 ref.

Early in the MSRE development program three types of devices were tested for blocking flow in small salt lines by freezing a plug in a restricted section. After 100 test cycles, one design was chosen for further development and testing.

development + freeze valves + MSRE + prototypes + testing

OTHER CATEGORIES: MAB

HIX660026

Hitch BF + Ross RG + McDuffie HF

TESTS OF VARIOUS PARTICLE FILTERS FOR REMOVAL OF OIL MISTS AND HYDROCARBON VAPOR

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1623 (Sept. 1966) 27 p, 9 fig, 3 tab.

Various filter and adsorbent materials were examined for possible use in the removal of oil mists and hydrocarbon vapors. A controlled flow of oil was injected into a heated nickel reaction vessel to cause vaporization and some cracking of the oil. Helium flowing through the reaction vessel carried the oil mist and hydrocarbon vapor through a filter system. Filter effectiveness was determined by the use of a combination of felted metal filters and ceramic fibers in a configuration proposed for use in the MSRE. Granulated charcoal removed hydrocarbon vapors (C-6 and above) in a manner consistent with the established adsorption isotherms for this material.

*development + *filters + *off-gas systems + adsorption + charcoal + components + filtration + hydrocarbons + materials + mists + MSRE + testing

HXX640019

Scott D

COMPONENT DEVELOPMENT IN SUPPORT OF MSRE (PART OF MSRE SEMIANN PROG REPT 7/31/64)

Accession Number HCX710022 to HXX640019

Category H
Reactor Component Development

HXX640019 *Continued*

Oak Ridge National Laboratory, Tenn.

ORNL-3708 (Nov. 1964), pp 167-190, 24 fig.

Development and operation of prototype units for evaluation of performance and maintainability are described in this paper. Included are the core hydraulic mockup, the heat exchanger hydraulic tests, electric heaters, freeze flanges, freeze valves, control rods and drives, and the fuel sampler-enricher.

components + development + maintenance + MSRE +
prototypes + testing

OTHER CATEGORIES: MAD

Category I
Reactor Design

IAA650024

Alexander LG + Carter WL + Craven CW + Janney EB +
Van Winkle R

MOLTEN-SALT CONVERTER REACTOR -- DESIGN STUDY AND CCST

ESTIMATES FOR A 1000-MWE STATION

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1060 (Sept. 1965), 348 p, 45 fig, 112 ref.

In 1961-1962 a study was made of a molten-salt converter reactor based on technology to be demonstrated in the MSBR. The conceptual design is a one-fluid reactor with cylindrical graphite moderator elements in a 20-ft reactor vessel. Ten cu ft of salt (of 2500 cu ft) is removed daily for recovery and recycle of uranium in a central plant serving many such reactors. With highly enriched U-235 feed the equilibrium conversion ratio is 0.90. Estimated power costs encourage continued effort on molten-salt reactors.

*conceptual design + *converters + *costs +
breeding performance + design data + economics +
fluorination + MSRP + plans

IAA66C030

Kasten FB + Bettis ES + Fauman HF + Carter WL +

McDonald WB + Robertson RC + Westsik JH

SUMMARY OF MOLTEN-SALT BREEDER REACTOR DESIGN STUDIES

Oak Ridge National Laboratory, Tenn.

CONF-66-524 (Proc. 2nd Int. Thorium Fuel Cycle Symposium,
Gatlinburg, May 3-6, 1966), pp. 41-63, 7 fig, 4 ref.

This paper discusses molten-salt reactor technology and presents a conceptual design, breeding performance, and cost estimates for a two-region, two-fluid MSBR with graphite tubes in the core. (A more detailed presentation is in CRNL-3996, IAA660025.)

*conceptual design + *MSBR + *two-fluid reactor +
breeding performance + capital costs + economics +
flowsheets + processing + protactinium + thorium

IAB67C043

Briggs RB

EFFECTS OF IRRADIATION ON THE SERVICE LIFE OF THE

MOLTEN-SALT REACTOR EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ANS Trans. 10(1), (June 1967), pp. 166-167.

Thermal neutron irradiation adversely affects the high-temperature stress-rupture life of the Hastelloy that was used in the MSRE. An allowance was made in the design for damaging effects of irradiation, but the much better understanding of the effects, obtained during the years the reactor was being built, indicated that the allowance might not be sufficient and that the service life of the reactor should be reevaluated. Concluded that the reactor vessel

Accession Number IAA650024 to IAB670C43

Category I
Reactor Design

IAB670043 *Continued*

would have a minimum service life of 20,000 hr.
*analysis + *Hastelloy N + *MSRE + *operation +
*radiation damage + *stress rupture + design + limits +
plans + reactor vessel
OTHER CATEGORIES: MAA

IAC660024

Kasten PR + Bettis ES + Fauman HF + Carter WL +
McDonald WB + Robertson RC + Westsik JH
SUMMARY OF MOLTEN-SALT BREEDER REACTOR DESIGN STUDIES
Oak Ridge National Laboratory, Tenn.

ORNL-TM-1467 (March 1966), 31 p, 7 fig, 11 tables, 4 ref.

A preliminary report on the conceptual design studies of a two-fluid two-region molten-salt thermal-breeder reactor power station of 1000 MW(e) capacity. A much more detailed report on the same studies was subsequently published as ORNL-3996, MSRI accession IAC660025, which see.

*MSBR + *conceptual design + *performance + *power costs +
reactors + containment + structures + molten salts +
processing + neutron physics + *two-fluid reactor

IAC660025

Kasten PR + Bettis ES + Robertson RC
DESIGN STUDIES OF 1000-MW(e) MOLTEN-SALT BREEDER REACTORS
Oak Ridge National Laboratory, Tenn.

ORNL-3996 (Aug. 1966), 150 p, 43 fig, 52 tables, 30 ref.

Design and evaluation studies were made of a two-region molten-salt thermal-breeder reactor which uses fuel and blanket salts separated by the walls of graphite tubing, which acts as the moderator. A coolant salt transports the heat from the primary heat exchangers to steam generators and reheaters. The reference design fuel salt is LiF-B₂F₆-UF₄ (68.3-31.2-0.5 mole %), the blanket salt is LiF-ThF₄-B₂F₆ (71.0-27.0-2.0 mole %), and the coolant salt is NaF-NaBF₄ (61.1-38.9 mole %). On-site fuel recycle processing was assumed, with fluoride volatility and vacuum distillation employed for the fuel salt and direct protactinium-removal processing used for the blanket salt. Estimated power cost is about 2.7 mills/kWhr, the specific inventory about 0.7 kg/MW(e), the fuel doubling time about 13 yrs and the estimated fuel-cycle cost is 0.35 mills/kWhr. General flowsheets and conceptual designs for the reactor, primary heat exchangers, salt circulating pumps and steam generators are presented. Cost and performance estimates are also given. Several alternate designs are briefly described: (a) a modified primary heat exchanger design; (b) a system using 580 deg F rather than 700 deg F feedwater; (c) a modular concept using four small reactors rather than one large reactor; (d) an MSBR operating without Pa removal in the chemical processing plant; (e) a concept

Category I
Reactor Design

IAC660025 *Continued*

in which the fuel salt is cooled by direct contact with circulating molten lead; (f) a single-stream core-breeder with direct Pa removal; (g) a lead-cooled reactor without graphite moderator operating in the intermediate-to-fast range (10 to 20 kev), and (h), a graphite-moderated single-region, single-fluid converter reactor.

*two-fluid reactor + *MSER + *conceptual design + *performance + *power costs + *capital equipment + *neutron physics + reactors + pumps + heat exchangers + steam generators + steam systems + off-gas systems + fuels + blanket + coolants + physical properties + breeding performance + fuel cycle costs + protactinium + lead + converters + cooling

IAC700047

Robertson RC + Eriggs RB + Smith OL + Bettis ES
TWO-FLUID MOLTEN-SALT BREEDER REACTOR DESIGN STUDY (STATUS AS OF JANUARY 1, 1968).

Oak Ridge National Laboratory, Tenn.

ORNL-4528 (Aug. 1970), 80 p, 44 fig, 30 tables, 45 ref.

The January 1, 1968 status of the conceptual design study of a 1000 MW(e) MSBR power station employing separate fuel and blanket salts in the reactor is reported. The requirements for Hastelloy N, the graphite moderator and reflector, and for the fissile, fertile and heat-transport salts are discussed and the properties of the available materials are tabulated. The selected fuel salt is LiF-BeF₂-UF₄ (68.5-31.3-0.2 mole %), the blanket salt is LiF-ThF₄-FeF₂ (11-27-2 mole %) and the coolant salt is NaBF₄-NaF (92-8 mole %). (The lithium is separated Li-7.) Conceptual designs are presented for the reactors, pumps, primary heat exchangers, drain tanks and steam-generating equipment. Flowsheets are given for the main systems, but the off-gas, afterheat removal, fuel-processing, and steam-power systems are described only in sufficient detail to indicate feasibility and to estimate costs. The reference design, with a power density of 20 kW/liter and an estimated graphite life of 8 years, has a breeding ratio of 1.06, a specific power of 1.77 MW(t)/kg, and a fuel yield of 4.07%/year. The dimensions of the four reactor vessels are about 14 ft D x 20 ft high. The estimated construction cost of the power station is about \$141/kW (1968 prices) and, based on 14.7% fixed charges and 80% plant factor, the estimated power production cost is about 4 mills/kWh. The estimated fuel-cycle cost is 0.7 mills/kWh.

*two-fluid reactor + *MSBR + *conceptual design + *performance + *power costs + *capital equipment + *neutron physics + reactors + control rods + drain tanks + heat exchangers + structures + steam generators + pumps +

Accession Number IAC660025 to IAC700047

Category I
Reactor Design

IAC70C047 *Continued*

steam systems + off-gas systems + containment + molten salts +
coolants + physical properties + graphite + neutron flux +
breeding performance + fission products + noble metals +
fuel cycle costs + afterheat

IAC700051

Bettis ES + Robertson RC

THE DESIGN AND PERFORMANCE FEATURES OF A SINGLE-FLUID
MOLTEN-SALT BREEDER REACTOR

Oak Ridge National Laboratory, Tenn.

Nucl. Appl. Tech. 8, 190 (Feb. 1970), 18 p, 9 fig, 5 ref.

A conceptual design has been made of a single-fluid 1000 MW(e) MSBR power station. The reactor vessel is 22 ft in diam x 20 ft high, of Hastelloy N, with graphite moderator and reflector. The fuel is U-233 carried in a LiF-B₂F₆-ThF₄ mixture which is molten above 930 deg F. With continuous chemical processing to isolate protactinium and remove fission products, conversion of thorium to U-233 exceeds fissile burnup. The estimated fuel yield is 3.3% per year. The estimated construction cost of the station is comparable to PWR total construction costs. The power production cost, including fuel-cycle and graphite replacement costs, with private utility financing, is estimated to be less than that for present-day light-water reactors, largely due to the low fuel-cycle cost and high plant thermal efficiency. After some engineering development, such a plant appears feasible and practical. (Companion papers in the same issue discuss the status of material development, fuel processing, and potential of the MSBR concept.)

*MSBR + *conceptual design + *performance + *power costs +
*capital equipment + *materials + *processing +
*fuel cycle costs + plant + reactors +
heat exchangers + pumps + off-gas systems +
steam systems + maintenance

IAC71C013

Robertson RC (editor)

CONCEPTUAL DESIGN STUDY OF A SINGLE-FLUID MOLTEN-SALT
BREEDER REACTOR

Oak Ridge National Laboratory, Tenn.

ORNL-4541 (Feb. 1971), 192 p, 92 fig, 62 tables, 129 ref.

Conceptual design of a 1000-MW(e) molten-salt thermal breeder reactor power station indicates that such a plant is technically feasible and economically attractive. The plant operates on the Th - U-233 cycle, using a fuel salt of the composition LiF-B₂F₆-ThF₄-UF₄ (71.7-16.0-12.0-0.3 mole %). The salt is pumped through a 22-ft diam x 20 ft high graphite-moderated and reflected reactor vessel and then through primary heat exchangers where it is cooled.

Category I
Reactor Design

IAC710013 *Continued*

from 1300 deg F to 1050 deg F. The core graphite is replaced by remote maintenance procedures at 4-yr intervals. The chief material of construction for the salt systems is Hastelloy N improved by additives to increase the resistance to irradiation damage. Tritium, Xe and Kr are sparged from the circulating fuel salt by helium bubbles. An off-gas system removes the fission-products for storage and decay and recycles the helium. A 1-gpm side stream of fuel salt is continuously processed to remove Pa-233, recover the bred U-233, and to adjust the fissile content. Heat is transported from the four primary heat exchangers by a circulating coolant salt, NaF4-NaF (92-8 mole %), to steam generators and reheaters supplying a 3500 psia 1000 deg F/1000 deg F steam turbine. The specific inventory of the plant is 1.5 kg fissile/MW(e), the breeding ratio is 1.06 and the annual yield is about 3.3%. The net thermal efficiency is 44%, and the estimated capital cost is about the same as for a light-water nuclear power station. The fuel-cycle cost is about 0.8 mills/kWhr. Flowsheets and conceptual designs of the major components are presented. Cost and performance estimates are tabulated. The principal design uncertainties are in areas of tritium confinement, fuel-salt processing, graphite and Hastelloy N behavior under irradiation, suitability of coolant salt, maintenance procedures, and the behavior of fission-product particulates.

*MSBR + *conceptual design + *performance + *power costs + *capital equipment + coolants + physical properties + graphite + Hastelloy N + reactors + cores + control rods + drain tanks + heat exchangers + structures + pumps + steam generators + steam systems + off-gas systems + containment + neutron flux + processing + breeding performance + fission products + noble metals + fuel cycle costs + afterheat + bubbles + gas separation + helium + maintenance + control + instrumentation + freeze valves + sites + heat generation

IAC710014

Robertson RC

ESTIMATED COST OF ADDING A THIRD SALT-CIRCULATING SYSTEM FOR CONTROLLING TRITIUM MIGRATION IN THE 1000-MW(e) MSBR

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3428 (July 1971), 26 p, 2 fig, 1 ref.

Controlling tritium migration to the steam system of the 1000-MW(e) reference design MSBR power station by interposing a KNO₃-NaNO₂-NaNO₃ salt-circulating system to chemically trap the tritium would add about \$13 million to the total of \$206 million now estimated as the cost of the reference plant if Hastelloy N is used to contain the

Accession Number IAC710013 to IAC710014

Category I
Reactor Design

IAC710014 *Continued*

LiF-BeF₂ salt employed to transport heat from the fuel salt to the nitrate-nitrite salt, and about \$10 million if Incoloy could be used. The major expenses associated with the modification are the costs of the additional heat exchangers (\$9 million), the additional pumps (\$5 million), and the LiF-BeF₂ inventory (\$4.8 million). Some of the expense is offset by elimination of some equipment from the feedwater system (\$2 million), through use of less expensive materials in the steam generators and reheaters (about \$2 million), and through an improved thermal efficiency of the plant (worth about \$1 million). In addition to acting as an effective tritium trap the third circulating system would simplify startup and operation of the MSBR. A simplified flowsheet for the modified plant, a cell layout showing location of the new equipment, physical properties of the fluids, design data and cost estimates for the new and modified equipment are presented.

*MSBR + *tritium + *capital costs + conceptual design + loop + coolants + heat exchangers + pumps + power costs + fuel cycle costs + steam systems

IAD700052

Bettis ES + Bauman HF

MOLTEN-SALT CONVERTER REACTORS

Oak Ridge National Laboratory, Tenn.

Power Engrg. Vol. 74 No. 8, 42 (Aug 1970) 3 p, 2 fig.

Development of rapid fuel-salt processing and longer-lived graphite is needed before molten-salt breeder reactor power stations are built. The molten-salt converter reactor, however, is generally within present technology; the graphite would last the lifetime of the plant and the occasional fuel processing would involve only the well-proven fluoride volatility process. The breeding ratio would be about 0.84, but the fuel-cycle cost would be only about 0.8 mills/kwh and the construction costs are expected to be attractively low. The reactor design receiving the most study is the type used successfully in the MSRE: a vessel filled with reflector and moderator graphite having salt flow passages formed by grooves in the faces of the pieces. A pebble-bed type of converter reactor has a structure that easily accommodates dimensional changes in the graphite and the shapes are economical to manufacture, but the salt-to-graphite ratio in the core is higher than desired. Thermal reactors operating on the Th - U-233 cycle are more efficient than those using the U-238 - Pu-239 cycle, but Pu can be used efficiently for startup and makeup fuel. Construction of a molten-salt converter reactor would lead to low-cost power in the near term, provide a market for the Pu produced in light-water reactors, and give impetus to

Accession Number IAC710014 to IAD700052

Category I
Reactor Design

IAD700052 *Continued*

development of the molten-salt breeder reactor which will be needed to assure low cost power in the future.

*conceptual design + *performance + MSEF + MSFE +
*fuel cycle costs + *materials + plutonium + *converters

IAE70C059

McWherter JF

MOLTEN SALT BREEDER EXPERIMENT DESIGN BASES

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3177 (Nov. 1970), 54 p, 15 fig, 12 ref.

The design bases for the MSBE are based on information from the MSRE and the reference plant design of a 1000 MW(e) single-fluid MSBR. Calculations indicate that a 150 MW (thermal) reactor is a reasonable size that meets the project objectives for the MSBE. The primary salt for the MSBE contains both the fissile (U-233) and the fertile (Th) material. The heat generated in the primary system is transferred by a secondary salt loop to the steam generators. Provisions are made in the MSBE core to permit exposure of removable graphite samples at conditions similar to those expected in the MSBR. The pumps and heat exchangers in the MSBE are similar to those proposed for the MSBR.

conceptual design + design + design criteria +
design data + graphite + irradiation + materials testing +
MSBE + MSEF + reactors + test facilities

IAF67C047

Taube M + Mielcarski M + Poturaj-Gutniak S + Kowalew A

NEW BOILING SALT FAST BREEDER REACTOR CONCEPTS

Inst. of Nuclear Research, Warsaw, Poland

Nucl. Engrg. and Design 5 (1967), pp. 109-112, 1 fig, 30 ref.

Use of molten chlorides in homogeneous-core fast breeder reactors is envisaged. In the SAWA reactor concept the core is filled with a molten mixture of NaCl-AlCl₃-UCl₃-PuCl₃. Heat is removed by boiling in the core, producing AlCl₃ vapor. The WARS concept uses UCl₃ and PuCl₃ in a mixture of NaCl and KCl, with boiling mercury removing the heat.

*boiling + *cores + *conceptual design + *reactors +
breeding performance + chlorides + fast neutrons +
foreign + mercury + molten salts

IAF670048

Taube M + Kowalew A + Poturaj-Gutniak S + Mielcarski M

KONZEPTION DER SALZFLÜSSIGKEITREAKTOREN SAWA UND WARS

Inst. of Nuclear Research, Warsaw, Poland

Kernenergie 10 (1967), pp. 184-186, 12 ref.

Fast breeder reactors with homogeneous cores of molten chlorides are not impossible: two concepts have been

Category I
Reactor Design

IAF670048 *Continued*

envisioned. In the SARA reactor concept the core is filled with a molten mixture of NaCl-AlCl₃-UCl₃-PuCl₃. Heat is removed by boiling in the core, producing AlCl₃ vapor. The WARS concept uses UCl₃ and PuCl₃ in a mixture of NaCl and KCl, with boiling mercury removing the heat. (This article, in German, is very similar to one in English: IAF670047.)

*boiling + *cores + *conceptual design + *reactors +
breeding performance + chlorides + fast neutrons +
foreign + mercury + molten salts

IAF690014

Perry AM

A HIGH-YIELD MOLTEN-SALT BURST REACTOR

Oak Ridge National Laboratory, Tenn.

Proceedings of the National Topical Meeting on Fast Burst

Reactors, Albuquerque, N. Mex., Jan. 28-30, 1969,

Conf-690102, 387, 15 p, 11 fig, 3 ref.

A pulsed molten-salt reactor appears capable of producing neutron fluences of 10¹⁶ nvt in neutron-irradiation specimens in single bursts with widths of less than 1 msec.

A reactor design is presented which achieves these goals, using as fuel lithium-uranium fluoride (73-27 mole %) eutectic salt. Neutronic, mechanical, and hydraulic analyses of the reactor are discussed.

*molten salts + *reactors + neutron spectra +
nuclear analysis + neutron fluence + *materials testing +
*irradiation + fuels + excursions + neutron sources +
neutron physics + description + *conceptual design +
experiment + fast neutrons

IBA710005

Tallackson JR

THERMAL RADIATION TRANSFER OF AFTER HEAT IN MSBR HEAT
EXCHANGERS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3145 (March 1971), 108 p, 43 fig, 28 ref.

About 40 percent of the noble-metal fission products are expected to deposit on metal surfaces in the fuel loop of an MSR, predominantly in the heat exchangers. The normal means of afterheat removal is continued circulation of the salts, but the design must permit afterheat removal from the heat exchangers entirely by radiative heat transfer without compromising containment. Steady-state temperature profiles in 5 'reference design' heat exchangers ranging in size from 94 MW(t) to 565 MW(t) were computed. The transients following a drain were estimated for the largest and smallest heat exchangers. The maximum temperatures, occurring 3 to 4 hrs after shutdown, were estimated to be about 2100 deg F and 1800 deg F in the 565-MW and 94-MW

Category I
Reactor Design

IBA710005 *Continued*

units respectively. The calculated temperatures are believed to be conservatively high. Elimination of one of the two outer shells from the 'reference design' exchangers would reduce steady-state temperatures by 200 deg F to 300 deg F. It is concluded that MSER heat exchangers with ratings of 500 - 600 MW(t) can be designed to accommodate safely this worst case afterheat situation.

accidents + afterheat + cooling + design +
radiation heating + heat exchangers + heat transfer + MSBE +
MSBR + noble metals

OTHER CATEGORIES: HCX

IBB67C039

(Staff Report)

DESIGN STUDY OF A HEAT-EXCHANGE SYSTEM FOR ONE MSER
CONCEPT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1545 (Sept. 1967), 201 p, 9 fig, 12 tab, 40 ref.

A system is described which uses five types of heat exchangers to transfer the heat generated in the reactor core of one concept of a 1000 MW(e) MSER to the supercritical steam needed to drive a turbine for the generation of electrical power. The two major design approaches reported here are for flow circuits in which heat is transferred from the molten core fuel and fertile blanket salts to the molten coolant salt and then to the supercritical fluid. The Case-A system involves relatively high fuel- and blanket-salt pressures in the reactor core. These pressures are reduced in the Case-B system by reversal of the flows of the fuel and blanket salts through the reactor core and the respective pumps and exchangers, while the operating pressures of the coolant-salt system are raised above those in the Case-A system. The criteria used, assumptions made, relationships employed, and the results obtained in the design for each of the five types of exchangers used in these cases are reported. The resulting design for the Case-B heat-exchange system and the exchangers appears to be the most workable one.

*design + *heat exchangers + *steam generators +
*two-fluid reactor + blanket + components + computer codes +
molten salts + MSBR + steam cycle

OTHER CATEGORIES: IEC

IBB71C015

Fraas AF

A NEW APPROACH TO THE DESIGN OF STEAM GENERATORS FOR
MOLTEN SALT REACTOR POWER PLANTS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2953 (June 1971), 69 p, 25 fig, 20 ref.

A new type of steam generator has been devised to meet the special requirements of high-temperature liquid-metal and

Category I
Reactor Design

IBB710015 *Continued*

molten-salt reactor systems. The basic design concept is such that boiling heat transfer instabilities and their attendant severe thermal stresses are avoided even for a temperature difference of as much as 1000 deg F between the feedwater and the high-temperature liquid, thus giving good control characteristics even under startup conditions. This is accomplished by employing a vertical reentry tube geometry with the feedwater entering the bottom of the inner small diameter tube (approximately 1/4 in. diam) through which it flows upward until evaporated to dryness. The slightly superheated steam emerging from the top of the small central tube then flows back downward through the annulus between the central tube and the outer tube. A portion of the heat transferred from the high-temperature liquid to the superheated steam in the annulus is in turn transferred to the water boiling in the central tube. Design studies indicate that this type of boiler not only avoids thermal stress and salt freezing problems but it also gives a relatively compact and inexpensive construction. Further, it appears to make possible a simple plant control system with exceptionally good plant response to changes in load demand.

*steam generators + *design + steam systems + control

IBD680036

Peebles FN

REMOVAL OF XENON-135 FROM CIRCULATING FUEL SALT OF THE MSBR
BY MASS TRANSFER TO HELIUM BUBBLES

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2245 (July 1968), 33 p, 8 fig, 2 tab, 21 ref.

Removal of dissolved xenon-135 by mass transfer to helium bubbles offers an attractive means of controlling the xenon-135 poison level in molten salt breeder reactors. To provide necessary engineering information for evaluation of the proposed method, the existing data on rates of mass transfer to gas bubbles were reviewed. Rather extensive literature references point to reliable equations for prediction of mass transfer rates to single bubbles rising in stationary liquids under the two extreme cases of a rigid bubble interface and of a perfectly mobile bubble interface. In general, experimental data are available which support these predictions. No reliable criterion for predicting the transition from one type behavior to another is available.

*bubbles + *mass transfer + *xenon + analysis + data +
design + models + MSER + reviews

OTHER CATEGORIES: HEX

Category J
Instrumentation and Controls

JAA710009

Chang SI

A SYSTEMATIC PROCEDURE FOR DETERMINING SYSTEM PARAMETERS BY
PERFORMANCE INDEX MINIMIZATION (THESIS)

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3311 (May 1971), 97 p, 8 fig, 19 ref.

A method was developed for determining optimum control system parameters by performance index minimization. The controller parameters appearing in a mathematical model of the system are optimized by adjusting them in such a manner as to minimize certain integral measures of the difference between the actual and desired output response. The minimization is performed according to an optimization scheme known as the steepest descent procedure by a computer code written for the IBM-360. The techniques were used to solve several problems in order to demonstrate the validity and practicality of the methods. The problems included control parameter optimization for a nineteenth order model of the Molten Salt Breeder Reactor. The method easily found the optimum controller parameters for this system in only 5 minutes of computer time.

control + design + MSBR + computer codes

JAB69C018

Sides WH

MSBR CONTROL STUDIES

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2489 (June, 1969) 43 p, 16 fig, 7 ref.

A preliminary study was made of the dynamics and control of a 1000 Mw(e), single-fluid MSBR by an analog computer simulation. An abbreviated, lumped-parameter model was used. The control system included a steam temperature controller and a simplified version of the MSRE reactor temperature control system. The results of the study indicate a need for a variable speed, secondary-salt pump for close control of the steam temperature. During severe transients, considerable care must be taken in designing the control system if freezing or overheating of the salts is to be avoided.

*MSBR + *control + *dynamic characteristics + *simulation +
analog systems + analysis + behavior + computers +
excursions + instrumentation + plant + stability + systems
OTHER CATEGORIES: BCX

JAB70C017

Sides WH

CONTROL STUDIES OF A 1000-Mw(e) MSBR

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2927 (May, 1970) 46 p, 16 fig, 7 ref.

Preliminary studies of the dynamics and control of
a 1000-Mw(e), single-fluid MSBR were continued. Previous

Accession Number JAA710009 to JAB700017

Category J
Instrumentation and Controls

JAB700017 *Continued*

studies were reported in CRNL-TM-2489, MSBE Control Studies, W. H. Sides, Jr. An analog simulation of an expanded lumped-parameter model was used. Steam temperature control was accomplished by applying the load demand signal directly to the reactor outlet temperature controller as well as to the steam generators.

*MSBR + *control + *dynamic characteristics + *simulation + analog systems + analysis + behavior + computers + excursions + instrumentation + plant + stability + systems
OTHER CATEGORIES: BCX

JAB710008

Sides WH

MSBR CONTROL STUDIES: ANALOG SIMULATION PROGRAM

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3102 (May 1971), 29 p, 8 fig, 6 ref.

An analog computer simulation of the proposed 1000-MW MSBR was devised and preliminary studies were made of dynamics and control. This report describes the reactor plant model and the computer simulation. The analog simulation of the plant consisted of a lumped-parameter heat transfer model for the core, primary heat exchanger, and steam generator; a two-delayed-neutron-group model of the circulating-fuel nuclear kinetics with temperature reactivity feedbacks; and the external control system. So that the model would have the maximum dynamic range, the system differential equations were not linearized, and as a result the model was severely limited spatially to minimize the number of equations. In addition, the pressure in the water side of the steam generator, as well as in the rest of the plant, and the physical properties of the salts and water were taken to be time invariant. The temperature of the feedwater to the steam generators was also held constant. Results and conclusions are given in CRNL-TM-2927 (MSRIS accession JAB700017).

MSBR + control + simulation + dynamic characteristics + analog systems + computers
OTHER CATEGORIES: JBX

JCX690019

Clark FH + Burke OW

DYNAMIC ANALYSIS OF A SALT SUPERCRITICAL WATER HEAT EXCHANGER AND THROTTLE USED WITH MSBE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2405 (Jan. 1969), 42 p, 24 fig, 6 tables, 3 ref.

A linearized, coarse space mesh model of a salt-supercritical water heat exchanger and the downstream throttle was simulated on analog computers. The effects on certain output quantities of changes in certain input quantities were noted. The output quantities were

Category J
Instrumentation and Controls

JCX690019 *Continued*

heat-exchanger water outlet temperature and pressure, salt outlet temperature, and enthalpy output. The input quantities were heat-exchanger water inlet temperature and pressure, salt inlet temperature, salt velocity, and throttle setting. Changes were studied only around design steady state.

*heat exchangers + analysis + dynamic characteristics + computers + steam systems + feedback + MSBR + analog systems + mathematics + stability + models + coolants + heat transfer + secondary salts + secondary systems + simulation

OTHER CATEGORIES: JAE

JDX670037

Russell JA + Knowles DG

DESCRIPTION OF FACILITY RADIATION AND CONTAMINATION SYSTEMS
INSTALLED IN THE MOLTEN-SALT REACTOR EXPERIMENT BLDG. 7503
Oak Ridge National Laboratory, Tenn.

ORNL-TM-1127 (Rev. 1) (August 1967), 23 p. 7 fig.

A radiation monitoring system continuously and automatically determines the conditions in the entire facility and records this information at a central control panel. When preset values are exceeded, audible and visual alarms inside and outside of the building area are actuated. Beta-Gamma constant air monitors sound caution alarms at 100 counts/min and high level alarms at 400 counts/min. The gamma monitors sound bells at 7.5 mF/hr. Building evacuation alarms are actuated from coincidence modules. This report describes the system and briefly discusses operating experience. An earlier version of this document was issued in May 1965.

*environment + *instrumentation + *MSRE +
*radiation measurement + contamination + monitors +
sampling + stack + off-gas systems

OTHER CATEGORIES: MAC

JDX690060

Bauman, CD

FISSION-PRODUCT MONITORING IN HIGH-TEMPERATURE GAS-COOLED
REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2791 (Dec. 1969), 33 p, 5 fig, 1 tab, 41 ref.

The report proposes the development of an instrumentation system capable of identifying and measuring the accumulation of fission products in high-temperature gas-cooled reactor (HTGR) coolant loop circuits and loop components. Discussed is the applicability of ionization chambers, beta and gamma spectrometers, charged-wire precipitators, Cerenkov detectors, filters, diffusion tubes, thermal gradient tubes, deposition tubes, and impactors as plateout monitors. It is recommended that the deposition-tube and gamma-spectrometer systems be further developed as component

Category J
Instrumentation and Controls

JDX69C060 *Continued*

plateout monitors and tested first in a high-temperature gas loop and then in reactor service.

*development + *fission products + *instrumentation +
*off-gas systems + components + coolant loops + HTGF +
plans + testing + monitors

OTHER CATEGORIES: IBD

JEX650020

Engel JR

APPLICATION OF AN ON-LINE DIGITAL COMPUTER TO A REACTOR
EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ANS Trans. 8(2), (1965), pp. 585-586.

The MSRE uses an on-line digital computer for acquiring and analyzing data. 273 analog process signals are scanned every 5 sec. These are stored on magnetic tape every 10 min. Alarms are taken from 171 signals. A number of calculations using current reactor data are performed periodically and on demand. Summaries of the calculations are recorded by typewriters. The Bunker-Ramc Model-340 computer can supply data for processing by other computers.

*data acquisition systems + *MSRE + computers +
data processing + instrumentation + operation + plans

OTHER CATEGORIES: MAC

JFX660027

Moore RL

CLOSED-CIRCUIT TELEVISION VIEWING IN MAINTENANCE OF
RADIOACTIVE SYSTEMS AT CRNL

Oak Ridge National Laboratory, Tenn.

ANS Trans. 9(2), (1966), pp. 530-531.

Discusses the development of television camera systems for viewing remote maintenance operations at reactors. (A later, more detailed reference on this subject is JAC67C036.)

description + equipment + MSRE + remote maintenance +
viewing devices

OTHER CATEGORIES: KEA + MEE

JFX67C036

Moore RL

CLOSED-CIRCUIT TELEVISION VIEWING IN MAINTENANCE OF
RADIOACTIVE SYSTEMS AT ORNL

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2032 (Nov. 1967), 13 p, 7 fig.

This report discusses factors affecting the use of closed-circuit television in radioactive systems, then describes equipment used for closed-circuit television viewing at the Homogeneous Reactor Test and at the MSRE. The results of a radiation test of a miniature,

Category J
Instrumentation and Controls

JFX67C036 *Continued*

radiation-resistant television camera are also presented.
*instrumentation + *MSRE + *remote maintenance +
viewing devices + equipment + glass + maintenance +
manipulators + optics + in-pile tests
OTHER CATEGORIES: KEA

Category K
Operation and Maintenance

KBB690006

Holz PP

FEASIBILITY STUDY OF REMOTE CUTTING AND WELDING FOR NUCLEAR
PLANT MAINTENANCE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2712 (Nov. 1969), 53 p, 13 fig, 21 ref.

Remote cutting and welding are potentially valuable in the maintenance of radioactive portions of reactors, particularly if the system is designed to exploit these techniques. ORNL has started to adapt an orbital cut-and-weld system (developed for the Air Force) to permit completely remote work applications. This report describes factors involved in radioactive system maintenance, summarizes some previous work on remote maintenance development, and explains how the automated orbital cutting and welding machinery system may overcome problems that have been encountered in nuclear repair work. Progress of the ORNL study is summarized, including descriptions of the prototype equipment and the results of machining and welding tests. The report describes additional requirements for development to provide fully remote operations and controls and proposes a long range program for development of a complete system for radioactive system equipment replacement by cutting and welding.

*maintenance + *reactors + *remote welding + cutting tools +
development

Category L
Fuel Preparation and Processing

LAX690010

Chandler JM + Bolt SE

PREPARATION OF ENRICHING SALT (LITHIUM-7 URANIUM-233
TETRAFLUORIDE) FOR REFUELING THE MOLTEN SALT REACTOR

Oak Ridge National Laboratory, Tenn

ORNL-4371 (March 1969), 73 p, 24 fig, 6 ref.

The Molten Salt Reactor Experiment has been refueled with an enriching salt concentrate. Its preparation in a shielded cell of the Thorium-Uranium Recycle Facility at ORNL was required because of the high uranium-232 content (222 ppm) of the uranium-233. A two-step process was used in which the uranium oxide was reduced to uranium dioxide by treatment with hydrogen and converted to uranium tetrafluoride by hydrofluorination. Lithium fluoride was added and the eutectic was formed by fusing the components. The eutectic was purified by treatment with hydrogen, which reduced the corrosion products to metal and subsequently allowed their removal by filtration. The quality of the product was well within the requirements established for the MSRE. The fuel concentrate, containing 39 kg of uranium (91.4% uranium-233), was packaged in nine containers of various sizes (0.5 to 7 kg of uranium) for addition to the reactor fuel drain tank and in 45 enrichment capsules, each containing 96 g of uranium, for addition to the bowl of the fuel circulating pump. The fuel was shipped in shielded carriers to the MSRE to accommodate the reactor enrichment schedule.

*fuel preparation + *uranium-233 + corrosion products +
filtration + fluorides + hydrofluorination + hydrogen +
lithium fluoride + MSRE + uranium-232

OTHER CATEGORIES: MCD

LAX700013

Chandler JM + Bolt SE

URANIUM-233-BEARING SALT PREPARATION FOR THE MSRE

Oak Ridge National Laboratory, Tenn.

Nuclear Applications and Technology, Dec. 1970, 16 p, 5 fig,
2 ref.

The MSRE has been refueled with an enriching salt concentrate, lithium-7 fluoride - uranium-233 tetrafluoride (73-27 mole %). Sixty-three kilograms of this was prepared in a shielded cell in the Thorium-Uranium Recycle Facility at Oak Ridge National Laboratory. The preparation process involved reducing uranium trioxide to uranium dioxide by treatment with hydrogen, converting the uranium dioxide to uranium tetrafluoride by hydrofluorination, and fusing the uranium tetrafluoride with lithium fluoride. Its preparation in a shielded cell was required because of the high uranium-232 content (222 ppm) of the uranium. The product salt, containing 39 kg of uranium (91.4% uranium-233) was low in oxide content (50 ppm) and the

Accession Number LAX690010 to LAX700013

Category L
Fuel Preparation and Processing

IAX700013 *Continued*

concentration of the corrosion products, chromium, iron, and nickel, was minimal at less than 0.5% total. (Abstractor's note: This work was reported in CRNL-4371. See LGX6900100.)

*fuel preparation + *MSRE + *uranium-233 +
corrosion products + hydrofluorination + lithium fluoride +
reduction + uranium-232

OTHER CATEGORIES: MCD

IAX710019

Shaffer JH

PREPARATION AND HANDLING OF SALT MIXTURES FOR THE
MOLTEN-SALT REACTOR EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ORNL-4616 (Jan. 1971), 41 p, 30 fig, 36 ref.

A molten mixture of LiF, BeF₂, ZrF₄, and UF₄ served as the circulating fuel for the Molten-Salt Reactor Experiment. Its secondary coolant for transferring heat to an air-cooled radiator was a molten mixture of LiF and BeF₂. A third mixture that was chemically identical to the coolant mixture was used in place of the fuel for pre-nuclear operations and subsequently to flush the reactor core after a fuel drain. Approximately 26,000 lb of these fused fluoride mixtures were prepared from component fluoride salts and loaded into the reactor facility by ORNL's Reactor Chemistry Division. Techniques for handling molten fluorides and their production process for attaining high chemical purity were developed and applied simultaneously with the development of the molten-salt nuclear reactor concept. The plans and operations which were part of the fueling of the MSRE are described.

*molten salts + *procurement + *production +
hydrofluorination + MSRE + loading + beryllium fluoride +
lithium fluoride + uranium fluorides + zirconium fluoride

LBX680027

Mailen JC + Cathers GI

FLUORINATION OF FALLING DROPLETS OF MOLTEN FLUORIDE SALT AS
A MEANS OF RECOVERING URANIUM AND PLUTONIUM

Oak Ridge National Laboratory, Tenn.

ORNL-4224 (November 1968) 21 p, 7 fig, 10 ref.

A fluorination method in which molten-fluoride droplets fall countercurrently through fluorine was devised for the recovery of uranium and plutonium from molten fluoride salts. Advantages over methods in which fluorine is bubbled through the salt are: (1) higher removal rates for both uranium and plutonium, (2) lower corrosion rates since the molten salt does not contact the vessel wall, (3) possibility of continuous operation, and (4) minimal corrosion-product contamination of the fluorinated salt.

Category L
Fuel Preparation and Processing

IBX680027 *Continued*

Experimental equipment was developed, and small-scale fluorinations were made using several salt solutions. From this data it was calculated that, by using a 5-ft-long fluorination column at 650 deg C, 99.9% of the uranium can be removed from 100-micron-diameter droplets of MSBR fuel. With an 11-ft-long tower at 640 deg C, 99% of the plutonium can be removed from 100-micron-diameter droplets of salt. In similar experiments using salt droplets containing protactinium-231, no protactinium was fluorinated, even at temperatures as high as 613 deg C.

*columns + *fluorination + corrosion + corrosion products + molten salts + MSBR + plutonium + protactinium + uranium

ICA670014

McNeese LE

CONSIDERATIONS OF LOW PRESSURE DISTILLATION AND ITS
APPLICATION TO PROCESSING OF MOLTEN-SALT REEFER
REACTOR FUELS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1730 (March 1967), 46 p, 12 fig, 10 ref.

Distillation at low pressure was examined as a method for removing rare earth fluorides from the fuel stream of an MSBR. It was concluded that distillation allows adequate rare earth fluoride removal with the simultaneous recovery of more than 99.5% of the fuel salt.

Characteristics of equilibrium and molecular distillation were noted and expressions for the relative volatility of rare earth fluorides were derived for these types of distillation. Expressions for the separation potential of several modes of distillation were derived and reported rare earth fluoride relative volatilities were shown to allow a great deal of latitude in still design and operational mode.

It was concluded that a single contact stage such as a well mixed liquid pool provides adequate rare earth fluoride removal and that rectification is not required. The buildup of rare earth fluorides at the vaporization surface was shown to seriously reduce the effectiveness of a distillation system. Liquid circulation was shown to be an effective means for preventing this buildup.

*distillation + *molten salts + fuels + MSBR + rare earths + volatility

ICA680008

Hightower JR + McNeese LE

MEASUREMENT OF THE RELATIVE VOLATILITIES OF FLUORIDES OF
CERIUM, LANTHANUM, PRAESODYMIUM, NEODYMIUM, SAMARIUM,
EUROPIUM, BARIUM, STRONTIUM, YTRIUM, AND ZIRCONIUM IN
MIXTURES OF LITHIUM-FLUORIDE AND BERYLLIUM-FLUORIDE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2058 (Jan. 1968, 43 p, 8 fig, 9 ref.)

One step in processing the fuel stream of a molten-

Category L
Fuel Preparation and Processing

ICA680008 *Continued*

salt breeder reactor is removal of rare earth fission product fluorides from the lithium fluoride-beryllium fluoride carrier salt by low pressure distillation. For designing the distillation system we have measured relative volatilities of the fluorides of cerium, lanthanum, praeodymium, neodymium, samarium, europium, barium, strontium, yttrium, and zirconium with respect to lithium fluoride, the major component. The measurements were made using a recirculating equilibrium still operated at 1000 deg C and at pressure from 0.5 to 1.5 mm mercury. Errors from several sources were estimated and shown to be small.

*distillation + *rare earths + beryllium fluoride + equilibrium + fission products + lithium fluoride + measurement + MSBR + processing + volatility

ICA690037

Smith FJ + Ferris LM + Thompson CT

LIQUID-VAPOR EQUILIBRIA IN LITHIUM FLUORIDE-BERYLLIUM

FLUORIDE AND LITHIUM FLUORIDE-BERYLLIUM FLUORIDE-THORIUM
FLUORIDE SYSTEMS

Oak Ridge National Laboratory, Tenn.

ORNL-4415 (June, 1969) 18 p, 4 fig, 21 ref.

Liquid-vapor equilibrium data for several lithium fluoride-beryllium fluoride and lithium fluoride-beryllium fluoride-thorium fluoride systems were obtained by the transpiration method over the temperature range of 900 to 1050 deg C. Relative volatilities, effective activity coefficients, and apparent partial pressures are tabulated for the major components, as well as for solutes such as uranium tetrafluoride, zirconium tetrafluoride, caesium fluoride, rubidium fluoride, and some rare-earth fluorides. The values are in reasonable agreement with those reported in the literature. Results of this study show that distillation may not be feasible as a primary separations method in the processing of single-fluid MSBR fuels.

*equilibrium + *volatility + beryllium fluoride + distillation + lithium fluoride + MSBR + rare earths + separations + thorium fluorides

ICB680007

Carter WL + Lindauer RB + McNeese IE

DESIGN OF AN ENGINEERING-SCALE, VACUUM DISTILLATION

EXPERIMENT FOR MOLLEN-SALT REACTOR FUEL

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2213 (Nov. 1968), 133 p, 43 fig, 14 ref.

Experimental equipment has been designed for an engineering-scale demonstration of vacuum distillation of molten-salt reactor fuel. The distillation is carried out at about 1000 deg C and 1-mm mercury to separate carrier

Category L
Fuel Preparation and Processing

ICB680007 *Continued*

salt from less volatile fission products, primarily the rare earths. The experiment is designed for either continuous salt feeding or for batchwise operation. Sampling of the distillate furnishes data on the separation between salt and fission products as a function of still bottoms concentration. The equipment consists of a 48-liter feed tank, a 12-liter still, a 10-in. diam x 51-in. condenser, a 48-liter receiver, plus associated temperature, pressure and level control instrumentation. All vessels and parts contacted by molten salt are made of Hastelloy N. The unit is heated by shell-type, ceramic heaters. About 90% of the experimental program will be devoted to nonradioactive operation using mixtures of lithium, beryllium, zirconium, and selected rare earth fluorides. The experiment will be concluded by distilling a 48-liter batch of uranium-free spent fuel from the MSRE.

*distillation + *molten salts + beryllium fluoride + Hastelloy N + lithium fluoride + MSRE + rare earths + zirconium fluoride

ICB710007

Hightower JR + McNeese LE

LOW-PRESSURE DISTILLATION OF MOLTEN FLUORIDE MIXTURES:

NONRADIOACTIVE TESTS FOR THE MSRE DISTILLATION EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ORNL-4434 (January 1971), 52 p, 16 fig, 10 ref.

Equipment built to demonstrate the low-pressure distillation of a 48-liter batch of irradiated fuel salt from the MSRE consisted of a feed tank, a 12-liter, one-stage still reservoir, a condenser, and a condensate receiver. The equipment was tested in 1968 by processing six 48-liter batches of nonradioactive $\text{LiF-EeF}_2\text{-ZrF}_4\text{-NdF}_3$ (65-30-5-0.3 mole %) at 1000 deg C. A distillation rate of 1.5 cu-ft of salt per day per square root of vaporization surface was achieved in the nonradioactive tests. Evidences of concentration polarization and/or entrainment were noted in some runs but not in others. Automatic operation was easily maintained in each run, although certain deficiencies in the liquid-level measuring devices were noted. Condensation of volatile salt components in the vacuum lines and metal deposition in the feed line to the still pot are problems needing further attention. These results showed that use of distillation in MSRE fuel salt processing is feasible and that the test equipment was satisfactory for use with radioactive material from the MSRE.

distillation + entrainment + fluorides + processing + molten salts + rare earths + liquid level measurement + beryllium fluoride + lithium fluoride + zirconium fluoride +

Accession Number ICB680007 to ICB710007

Category L
Fuel Preparation and Processing

ICB710007 *Continued*
MSRE + separations + testing

ICC710024

Hightower JR + McNeese LE + Hannaford EA + Cochran FD
LOW PRESSURE DISTILLATION OF A PORTION OF THE FUEL CARRIER
SALT FROM THE MOLTEN SALT REACTOR EXPERIMENT
Oak Ridge National Laboratory, Tenn.

ORNL-4577 (August 1971), 56 p, 17 fig, 10 ref.

High-temperature low-pressure distillation of irradiated MSRE fuel carrier salt was demonstrated. Twelve liters of this salt was distilled in 23 hr with still pot temperatures of 900-980 deg C and condenser pressures of 0.1-0.8 torr. Eleven condensate samples taken during the run were analyzed for Li, Be, Zr, Cs-137, Zr-95, Ce-144, Em-147, Eu-155, Y-91, Sr-90, and Sr-89. Effective relative volatilities, with respect to LiF, for Be and Zr agreed with values measured previously. Effective relative volatilities for the slightly volatile materials Ce, Y, and Sr were much higher than previously measured values. The high values are believed to be the result of sample contamination, although concentration polarization may have also been a contributor. The effective relative volatility for Cs-137 was only 20%, or less, or previously measured value; no explanation of this discrepancy is available. Although the effective relative volatilities for the lanthanides were higher than anticipated, the values observed would still allow adequate recovery of LiF-7 from waste salt streams by distillation.

*distillation + *experience + *molten salts + *MSRE +
beryllium fluoride + cesium + entrainment + fluorides +
processing + rare earths + liquid level measurement +
lithium fluoride + zirconium fluoride + separations +
operation + volatility

OTHER CATEGORIES: LIX

IDA690012

Ferris LM

SOME ASPECTS OF THE THERMODYNAMICS OF THE EXTRACTION OF
URANIUM, THORIUM, AND RARE EARTHS FROM MOLTEN LITHIUM
FLUORIDE-BERYLLIUM FLUORIDE INTO LIQUID LITHIUM-BISMUTH
SOLUTIONS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2486 (March 1969), 17 p, 18 ref.

Expressions for the equilibrium distribution of uranium, thorium, lanthanum, and other solutes between lithium fluoride-beryllium fluoride solutions and lithium-bismuth solutions at 600 to 700 deg C were calculated, using thermodynamic data from the literature. The results obtained experimentally for uranium were in reasonably good agreement with the calculated values. However,

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Fuel Preparation and Processing

LDA690012 *Continued*

the results for thorium and lanthanum reflect the high degree of uncertainty that exists in the available thermodynamic data for these solutes. It is concluded, therefore, that an accurate measure of the relative extractability of the various solutes can be obtained only by experimental means.

beryllium fluoride + bismuth + distribution + equilibrium + lithium + lithium fluoride + rare earths + thorium + uranium + *reductive extraction process + *design data

OTHER CATEGORIES: CBX

IDA690013

Mailen JC + Ferris LM + Mcgueira ED

ESTIMATE OF THE SOLUBILITY OF PROTACTINIUM IN LIQUID BISMUTH

Oak Ridge National Laboratory, Tenn.

Inorg. Nucl. Chem. Letters, 5 (1969) pp 869-872, 8 ref.

The solubility of protactinium in bismuth was determined to be about 1200 ppm at 500 deg C. The method used was the extraction of protactinium from lithium-beryllium fluoride into bismuth at 600 deg C under conditions such that the concentrations of other metals in bismuth was very low and then cooling to 500 deg C and sampling.

solubility + protactinium + bismuth + molten salts

IDA690038

Dahlke O + Gans W + Knacke O + Muller F

DISSOCIATION PRESSURE OF BISMUTH IN THE SYSTEM:

BISMUTH-THORIUM

Technischen Hochschule, Aachen and Kernforschungsanlage, Julich, Germany

Z. Metallk., 60(5) (1969) pp. 464-468, 5 fig, 8 ref.

ORNL-tr-2218

The dissociation pressure of bismuth in the system bismuth-thorium was determined using the effusion method due to Knudsen between 600 and 1400 deg C. Isothermal dissociation curves and x-ray diagrams corroborated the compounds thorium dibismuthide and trithorium tetrabismuthide which do not form solid solutions. The supposed compound dithorium bismuthide was not found using measurements of the partial pressure.

*bismuth + thorium + reductive extraction process

OTHER CATEGORIES: CEX

LDA700014

Schilling CE + Ferris LM

THE SOLUBILITY OF THORIUM IN LIQUID BISMUTH

Oak Ridge National Laboratory, Tenn.

Journal of the Less-Common Metals, 20 (1970) pp. 150-159,

1 fig, 12 ref.

The solubility of thorium in bismuth was determined over the

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Fuel Preparation and Processing

LDA70C014 *Continued*

temperature range 450 - 900 degrees C, using a filtration technique. Data obtained using mild steel equipment can be represented by the expression: $\log S (\text{ppm Th}) = 7.7085 - 3852/T(\text{deg K})$. Several data points were obtained using molybdenum apparatus. These values are in good agreement with those determined in mild steel.

*solubility + thorium + bismuth + filtration + *data

LDA700015

Smith FJ + Ferris LM

MUTUAL INTERACTIONS OF THORIUM, NICKEL AND BISMUTH IN
Th-Ni-Bi SOLUTIONS

Oak Ridge National Laboratory, Tenn.

J. Inorg. Nucl. Chem. 32 (1970) pp 2863-2868, 2 fig, 8 ref.

Thorium and nickel, dissolved in liquid bismuth, were found to interact with each other and with the solvent to form a solid ternary compound of the apparent composition thorium-nickel-2 bismuth. This interaction was studied at 50 deg intervals, over the temperature range of 550-700 deg C. At each temperature studied, the joint solubility of thorium and nickel could be expressed as a mole fraction product. $S = \text{mole fraction thorium} \times \text{mole fraction nickel}$. The variation of the solubility with temperature can be expressed as $\log S = 1.115 - 6397/T(\text{deg K})$. The standard free energy of formation of ThNi_2Bi was estimated to be -51 kcal/mole at 650 deg C.

*solubility + thorium + nickel + bismuth + *data

OTHER CATEGORIES: CBX

LDA700046

Ferris LM + Mailen JC + Lawrence JJ + Smith FJ +
Noqueira ED

EQUILIBRIUM DISTRIBUTION OF ACTINIDE AND LANTHANIDE ELEMENTS
BETWEEN MOLTEN FLUORIDE SALTS AND LIQUID BISMUTH SOLUTIONS

Oak Ridge National Laboratory, Tenn.

J. Inorganic Nucl. Chem, 32, (1970) pp. 2019-2035, 9 fig,
21 ref.

The equilibrium distributions of several actinide and lanthanide elements between liquid bismuth solutions and a variety of lithium fluoride-containing molten fluoride salts were determined between 500 and 700 deg C. At each temperature, the distribution coefficients (mole fraction in bismuth phase divided by mole fraction in salt) obeyed the relationship $D = (EnLi)_K(1st)$ in which r is the valence of the element in the salt phase. Over the range of conditions investigated, thorium, protactinium and zirconium existed as tetravalent species in the salt; uranium, neptunium, plutonium, americium, curium, californium, lanthanum, and neodymium were trivalent; and europium was divalent. The distribution behavior of each element was affected by salt

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Fuel Preparation and Processing

LDA70C046 *Continued*

composition and temperature. Values of log K (1st) increased regularly with decreasing temperature. With lithium fluoride-beryllium fluoride-thorium fluoride (72-16-12 mole %) as the salt phase, the values of log K (1st) at 600 deg C for the trivalent actinide elements varied systematically with atomic number, and passed through a minimum near Z = 96 (curium).

*distribution + actinides + beryllium fluoride + bismuth + lithium fluoride + protactinium fluorides + rare earths + thorium fluorides + zirconium fluoride

LGX650002

McNeese LE + Scott CD

RECONSTITUTION OF MSR FUEL BY REDUCING URANIUM

HEXAFLUORIDE GAS TO URANIUM TETRAFLUORIDE IN A MOLTEN SALT

Oak Ridge National Laboratory, Tenn

ORNL-TM-1051 (March 11, 1965), 15 p, 5 fig, 7 ref

The direct reduction of uranium hexafluoride to uranium tetrafluoride in a molten salt is proposed as a step in the purification of fuel salt from a molten salt reactor. This step would replace the conventional method of reduction in which uranium hexafluoride is reduced to uranium tetrafluoride power in a hydrogen-fluorine flame. Reduction of the uranium hexafluoride in a molten salt will result in a shorter and more direct process for fuel salt purification. The reduction is to be effected in two steps which consist of absorption of uranium hexafluoride into a molten salt containing uranium tetrafluoride and of reduction of the resulting intermediate fluorides to uranium tetrafluoride with hydrogen. Experimental data on the absorption step are presented and information concerning the reduction of intermediate fluorides is considered.

*hydrogen + *reduction + absorption + chemical reactions + fluorine + molten salts + hydrogen + uranium fluorides

LHX690011

Lindauer BE + McGlothlan CK

DESIGN, CONSTRUCTION, AND TESTING OF A LARGE MOLTEN-SALT FILTER

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2478 (March 1969), 35 p, 9 fig, 12 ref.

The Molten Salt Reactor Experiment uses mixtures of fluoride salts as fuel. Routine on-site processing of these molten salts results in formation of corrosion products. This report describes development, design, construction, installation, and testing of a large salt filter to remove these corrosion products. The filter is designed to remove approximately 15 kilograms of corrosion products from 9000 kilograms of flush and fuel salt at a temperature

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LHX690011 *Continued*
of 1200 deg F.

*filters + *MSFE + construction + corrosion products +
design + development + fluorides + testing

LIX650023

Lindauer RB

MSRE DESIGN AND OPERATIONS REPORT, PART VII, FUEL HANDLING
AND PROCESSING PLANT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-907 (May 1965), 96 p, 20 fig, 8 ref.

The on-site plant is designed to remove oxides from the salts by treatment with H₂-HF mixtures and uranium as UF₆ by treatment with fluorine. The report includes plant description, safety analyses, and procedures. (A revision covering changes in equipment and plans, issued in 1967, is filed as LIA670013.)

*MSRE + *processing + equipment + flowsheets +
fluorination + hydrogen + hydrogen compounds + operation +
oxides + plans + plant + safety

OTHER CATEGORIES: MAA

LIX67C013

Lindauer FB

MSRE DESIGN AND OPERATIONS REPORT, PART VII, FUEL HANDLING
AND PROCESSING PLANT

Oak Ridge National Laboratory, Tenn

ORNL-TM-907 Revised (Dec. 1967), 65 p, 19 fig, 7 ref.

Flowsheets and equipment for the MSRE Fuel Processing Plant are described. The plant is designed to remove oxides from the flush and fuel salts by treatment with hydrogen-hydrogen fluoride gas mixtures and to recover uranium by fluorination. Consequences of the maximum credible accident are described in addition to the expected radiation levels during processing.

*MSRE + *processing + accidents + equipment + flowsheets +
fluorination + hydrogen + hydrogen compounds + operation +
plant + safety + uranium

OTHER CATEGORIES: MAA

LIX69C008

Lindauer FB

PROCESSING OF THE MSRE FLUSH AND FUEL SALTS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2578 (Aug. 1969), 75 p, 25 fig, 12 ref.

The MSRE Fuel Processing Plant, shakedown tests of equipment and procedures, and the uranium recovery operation are described. The MSRE flush and fuel salt batches were fluorinated to recover 6.5 and 2.16 kg of uranium, respectively. Known losses during processing were less than 0.1%. Gross beta and gamma decontamination

Category I
Fuel Preparation and Processing

LIX69C008 *Continued*

factors of 1.2×10 (9th) and 8.6×10 (8th) were obtained. Corrosion averaged about 0.1 mil/hr. The corrosion product fluorides were reduced and filtered to provide a carrier salt having a lower concentration of metallic contaminants than the original carrier salt.

*MSRE + *processing + corrosion + decontamination + filtration + fluorination + losses + operation + plant + uranium

OTHER CATEGORIES: MCD

LIX690009

Chandler JM + Lindauer RB

PREPARATION AND PROCESSING OF MSRE FUEL

Oak Ridge National Laboratory, Tenn.

CONF-69081 (August 1969), Symposium on Reprocessing of Nuclear Fuels, Nuclear Metallurgy Volume 15, Ames, Iowa, August 25-27, 1969, pp. 97-120, 8 fig, 4 ref.

The MSRE has been refueled with an enriching salt concentrate, lithium-7 fluoride-uranium-233 tetrafluoride (73-27 mole %), which was prepared in a shielded cell in the Thorium-Uranium Recycle Facility at ORNL. The preparation process involved reducing uranium trioxide (uranium-232 content, 222 ppm) to uranium dioxide by treatment with hydrogen, converting the dioxide to tetrafluoride by hydrofluorination, and fusing the tetrafluoride with lithium-7 fluoride. The original MSRE fuel salt, which contained 220 kg of uranium (35% uranium-235), was fluorinated to volatilize the uranium as the hexafluoride which was absorbed on sodium fluoride. The uranium was decontaminated from fission products by a factor of almost ten to the ninth. Fluorine utilization averaged 39%. Corrosion products were removed from the barren carrier salt by reduction and filtration. Corrosion rates for surfaces exposed to fluorine during fluorination averaged 0.1 mil/hr for 47 hours.

*fuel preparation + *MSRE + *processing + decontamination + filtration + hydrofluorination + hydrogen + lithium fluoride + reduction + sodium fluoride + uranium fluorides

OTHER CATEGORIES: LAX + MCD

LJX660006

Scott CD + Carter WL

PRELIMINARY DESIGN STUDY OF A CONTINUOUS FLUORINATION-VACUUM-DISTILLATION SYSTEM FOR REGENERATING FUEL AND FERTILE STREAMS IN A MSRE

Oak Ridge National Laboratory, Tenn.

ORNL-3791 (Jan. 1966), 123 p, 31 fig, 39 ref.

A preliminary design and engineering evaluation is made of a conceptual plant for treating the fuel and fertile streams of a 1000 MWE MSBR. The requirements are to recover unburned fuel and fused salts from the fuel

Category L
Fuel Preparation and Processing

IJX660006 *Continued*

stream and the bred uranium and fused salts from the fertile stream. Decontamination must be sufficient for attractive breeding. Plant capacity is 15 cu ft/day of fuel salt and 105 cu ft/day of fertile salt. The fuel stream is purified by fluorination and distillation. Volatile fission product fluorides are removed by sorption on sodium fluoride. The fertile stream is fluorinated to remove bred uranium sufficiently fast to keep a low concentration in the blanket. The chief conclusions of this study are that this process can be engineered with a normal amount of development work and that integration of the processing and reactor facilities is of primary importance in lowering the processing cost. The cost of the processing plant contributes about 0.2 mil/kwh to the fuel cycle cost.

*design + *MSBR + *processing + blanket + costs +
decontamination + distillation + fluorination +
fuel cycle costs + sodium fluoride + uranium +
*two-fluid reactor

IJX660032

Whatley ME + Carter WL + Lindauer RB + McNeese LE +
Scott CD + Hightower JR

ENGINEERING DEVELOPMENT OF ON-SITE PROCESSING FOR
MOLTEN-SALT BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

CONF-66-524 (Proc. 2nd Int. Thorium Fuel Cycle Symposium,
Gatlinburg, May 3-6, 1966), pp. 653-669, 9 fig, 4 ref.

A processing scheme is described for a two-fluid MSBR. The fuel stream is fluorinated to recover uranium and distilled to recover lithium. The blanket salt (LiF-ThF₄) is fluorinated to recover bred U-233.

*conceptual design + *distillation + *fluorination +
*processing + costs + flowsheets + molten salts + MSBR +
thorium

IJX670032

Carter WL + Whatley ME

FUEL AND BLANKET PROCESSING DEVELOPMENT FOR MOLTEN SALT
BREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1852 (June 1967) 52 p, 10 fig, 13 ref.

This document describes the fuel and blanket processes for the two-fluid MSBR, giving the 1967 status of the technology and outlining the needed development. It is concluded that the principal needs are to develop the vacuum distillation and protactinium removal operations, which have been demonstrated in the laboratory but not on an engineering scale. A program to develop continuous fluoride volatility, liquid-phase reduction-reconstitution, improved xenon control, and special instrumentation should

Category L
Fuel Preparation and Processing

LJX670032 *Continued*

also be a major developmental effort. An estimate of manpower and cost for developing MSBR fuel and fertile processes indicates that it will require 288 manyears of effort over a 6-year period at a total cost of about \$18,000,000.

*development + *MSBR + *processing + blanket + costs + distillation + fuels + protactinium + two-fluid reactor

LKX620003

Carter WL + Milford RP + Stockdale WG

DESIGN STUDIES AND COST ESTIMATES OF TWO FLUORIDE VOLATILITY PLANTS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-522 (Oct. 10, 1962), 81 p, 25 fig, 16 ref

Studies are made for 1.2 cu ft/day and 12 cu ft/day plants processing fuel from a 1000-MWE one-region fused salt converter reactor. Two conditions were considered for the smaller plant: (1) retention of the waste salt for protactinium-233 decay and recovery by a second fluorination and (2) discard of all protactinium-233 as waste after the first fluorination. The larger plant was considered only for the case of protactinium decay and recovery. With protactinium recovery, the capital cost is \$25,750,000 for the larger plant and \$12,556,000 for the smaller plant. Operating cost is \$2,241,000 for the larger plant and \$1,103,000 for the smaller plant. With protactinium discard the capital cost is \$10,188,000 for the smaller plant. Processing consists of fluorination, absorption on sodium fluoride, condensation in cold traps, reduction in a hydrogen-fluorine flame, dissolution in makeup salt and recycle to the reactor.

*capital costs + *converters + *operating costs + *processing + absorption + design + molten salts + protactinium + reactors + sodium fluoride + volatility

OTHER CATEGORIES: LJX

LKX700030

McNeese LE

ENGINEERING DEVELOPMENT STUDIES FOR MOLTEN-SALT BREEDER REACTOR PROCESSING NO. 1

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3053 (Nov. 1970), 85 p, 9 fig, 24 ref.

Several operations associated with MSBR processing are under study. This report describes: (1) a recently completed facility for semi-continuous engineering experiments on reductive extraction, (2) experiments related to the development of electrolytic cells for use with molten salt and bismuth, (3) consideration of selective crystallization of thorium bismuthide from bismuth-thorium-rare earth solutions as a means for separating thorium from the rare

Category I
Fuel Preparation and Processing

LKX70C030 *Continued*

earths, and (4) a computer code that calculates the nuclear, chemical, and physical processes occurring in the fuel stream of an MSBR. This work was carried out in the Chemical Technology Division during the period October through December 1968.

*development + *processing + *MSBR + bismuth + computer codes + contactors + electrolysis + rare earths + reductive extraction process + thorium

LKX710001

McNeese LE

ENGINEERING DEVELOPMENT STUDIES FOR MOLTEN-SALT BREEDER REACTOR PROCESSING No. 2

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3137 (Jan. 1971) 102 p, 50 fig, 5 ref.

Several operations associated with MSBR processing are under study. This report describes (1) a proposed reductive-extraction flowsheet for a single fluid MSBR, (2) material-balance calculations that show the effects of the removal times for zirconium, alkali metals and alkaline earths, europium, and protactinium on reactor performance and that indicate the magnitudes of the heat generation and mass flows associated with the reactor off-gas, (3) calculated results showing the steady-state performance of a protactinium isolation system, (4) an evaluation of the use of the protactinium isolation system to limit the uranium concentration in the blanket of a single-fluid MSBR, (5) calculations to predict the steady-state performance of a rare-earth removal system based on reductive extraction, (6) preliminary testing of the semicontinuous reductive-extraction facility, (7) experiments related to the development of electrolytic cells for use with molten salt and bismuth, and (8) installation of equipment at the MSRE for demonstrating low-pressure distillation of molten salt using irradiated MSRE fuel carrier salt. This work was carried out by the Chemical Technology Division during the period January - March 1969.

*development + *processing + *MSBR + bismuth + electrolysis + heat generation + MSRE + protactinium + rare earths + reductive extraction process + zirconium

IXX660031

Blankenship FF

CHEMICAL SEPARATIONS IN MOLTEN FLUORIDES

Oak Ridge National Laboratory, Tenn.

CONF-660524 (Proc. 2nd Int. Thorium Fuel Cycle Symposium, Gatlinburg, May 3-6, 1966), pp. 647-652, 8 ref.

There are several favorable reactions in molten fluoride systems that would permit rapid removal of fission products. Xenon and krypton are only slightly soluble and transfer into any gas phase. Iodine can be removed by sparging with

Category I
Fuel Preparation and Processing

LXX66C031 *Continued*

a mixture of H₂ and HF. Rare earths can be separated by distillation, by reduction to insoluble intermetallic beryllides in the salt, or by reduction and extraction into molten Li-Bi alloys. Protactinium can be removed by oxide precipitation.

*chemistry + *fission products + *molten salts +
*separations + beryllium + chemical reactions +
distillation + experiment + iodine + oxides + processing +
protactinium + rare earths + reductive extraction process

LXX70C029

Whatley ME + McNeese LE + Carter WL + Ferris IM +
Nicholsen EL

ENGINEERING DEVELOPMENT OF THE MSBR FUEL RECYCLE

Oak Ridge National Laboratory, Tenn.

Nucl. Appl. Tech. 8, 170 (Feb. 1970), 9 p, 5 fig, 16 ref.

The MSBR being developed at ORNL requires continuous processing of the fuel salt, lithium fluoride-beryllium fluoride-thorium fluoride (72-16-12 mole %) containing approximately 0.3 mole % uranium-233 tetrafluoride. The reactor and processing plant are planned as an integral system. The main functions of the processing plant will be to isolate protactinium-233 from the neutron flux and to remove the rare-earth fission products. The method being developed involves the selective chemical reduction of the various components into liquid bismuth solutions at approximately 600 deg C, utilizing multistage counter-current extraction. Protactinium, which is easily separated from uranium, thorium, and the rare earths, would be trapped in the salt phase in a storage tank located between two extraction contactors and allowed to decay to uranium-233. Rare earths would be similarly separated from thorium; however this operation will be more difficult because the separation factors are lower. Electrolytic cells would be used to introduce reductant into the bismuth phase at the cathode and to return extracted materials to the salt phase at the anode.

*development + *engineering + *MSBR + *processing +
bismuth + chemical reactions + extraction columns +
fission products + flowsheets + fuel cycle + molten salts +
MSBR + processing + separations +
reductive extraction process

OTHER CATEGORIES: LDX

LXX710021

McNeese LE

ENGINEERING DEVELOPMENT STUDIES FOR MSBR PROCESSING No. 3

Oak Ridge National Laboratory, Tenn,

ORNL-TM-3138 (May 1971), 97 p, 35 fig, 13 ref.

This report describes (1) calculated steady-state performance of a protactinium isolation system for

Accession Number LXX660031 to LXX710021

Category L
Fuel Preparation and Processing

LXX710021 *Continued*

reactors fueled with uranium or plutonium; (2) material-balance calculations showing the effect of fission product removal times on reactor performance; (3) experiments on reductive extraction in a mild-steel flow-through facility; (4) a simulation of a flow control system for the semicontinuous reductive extraction system; (5) development of electrolytic cells for use with molten salt and bismuth; (6) an analysis of the transfer of materials in electrolytic cells; (7) measurement of axial dispersion in packed columns using immiscible liquids having large density differences; (8) calculated heat generation rates and temperatures in a protactinium extraction column; and (9) low-pressure distillation of irradiated MSRE fuel carrier salt.

*distillation + *processing + *protactinium +
*reductive extraction process + bismuth + electrolysis +
extraction columns + fission products + heat generation +
molten salts + MSRE + plutonium + uranium

OTHER CATEGORIES: LDX + ICC

LXX710026

McNeese LE

ENGINEERING DEVELOPMENT STUDIES FOR MSBR PROCESSING NO. 4

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3139 (Aug. 1971), 124 p, 67 fig, 16 ref.

This report describes (1) experiments on the hydrodynamics of packed column operation, carried out in a mild-steel reductive extraction facility, (2) measurement of axial dispersion in packed columns in which immiscible fluids leaving large density differences are flowing countercurrently, (3) a simplified method for estimating the effect of axial dispersion on countercurrent column performance, (4) estimates of the effect of axial dispersion in packed column contactors used for MSBR processing, (5) measurements of axial dispersion coefficients in an open bubble column, (6) experiments related to the development of electrolytic cells for use with molten salt and bismuth, (7) the design and installation of the Flow Electrolytic Cell Facility, (8) the calibration of an orifice-head pct flowmeter for use with the Flow Electrolytic Cell Facility, (9) the development of an induction type of bismuth-salt interface detector, and (10) calculations regarding the removal of ThF₄ from molten-salt streams by reductive extraction.

*extraction columns + *processing + bismuth + dispersion +
electrolysis + flow measurement + liquid level measurement +
molten salts + reductive extraction process +
thorium fluorides

Category M
MSRE

MAC680034

Tallackson JR

NUCLEAR AND PROCESS INSTRUMENTATION -- PART IIA, MSRE
DESIGN AND OPERATIONS REPORT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-729 (Feb. 1968), 397 p, 180 fig, 102 ref.

The first part of this document gives a general description of the entire MSRE instrumentation and control system including control of auxiliary equipment and the instrument power system. Considerations which influenced the design are also discussed and the physical layout of the instrumentation system is described. The second part is a detailed description of the safety instrumentation and nuclear control systems. Included are neutron instruments, safety circuits, control-rod system, and the heat load control. Also described are the instruments and interconnections of the radiation and contamination monitoring system, process radiation monitors, and the data logger-computer system.

*design + *design criteria + *instrumentation + *MSRE +
*radiation measurement + *safety + computers + control +
control rods + control-rod drives +
data acquisition systems + electrical circuits +
health physics + monitors + off-gas systems

OTHER CATEGORIES: JBX

MAD690004

Gabbard CH

DESIGN AND CONSTRUCTION OF CORE IRRADIATION-SPECIMEN ARRAY
FOR MSRE RUNS 19 and 20

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2743 (Dec. 22, 1969) 23 p, 7 fig, 0 ref.

A new MSRE core specimen array was designed and fabricated to replace the type of metallurgical surveillance specimen array that was used in the MSRE through Run 18. The main purpose of the new array is to measure the capture-to-absorption ratio of uranium-233 and to determine the effects of salt velocity, turbulence, and surface finish on the deposition of fission products on graphite and on Hastelloy N. Two additional test specimens were included, one of pyrolytic graphite to determine if there is permeation of fuel salt or its constituents into the graphite and one of Hastelloy N to expose a series of electron microscope screens in a trapped gas pocket.

*MSRE + *irradiation + design + fabrication + *uranium-233 +
cross sections + fission products + adsorption + graphite +
*Hastelloy N + surveillance + pyrocarbon + intrusives

OTHER CATEGORIES: MBX

MAX650019

Category M
MSRE

MAX650019 *Continued*

Robertson RC

MSRE DESIGN AND OPERATIONS REPORT, PART I, DESCRIPTION OF
REACTOR DESIGN

Oak Ridge National Laboratory, Tenn.

ORNL-TM-728 (Jan. 1965), 567 p, 112 fig, 61 tables, 176 ref.

This report is one of a series describing the MSRE. The reactor and other major components and systems are fully described and detailed flowsheets show process data and instrumentation for both the main and auxiliary systems. The fuel salt was LiF-BeF₂-ZrF₄-UF₄ and the coolant salt used to transport heat from the primary heat exchanger to the air-cooled radiator (heat sink) was LiF-BeF₂.

All parts of the systems in contact with the salts were fabricated of standard Hastelloy N. Operating temperatures for the fuel salt were 1175 deg F to 1225 deg F, and for the coolant salt were 1025 deg F to 1100 deg F. A drain tank, cooled by boiling water in thimbles, was used to store the fuel salt. The site facilities, building services, containment cells and other structures, and the various reactor system electrical circuits, are also described. (Nuclear calculations, operational procedures, performance data, and maintenance aspects are given in companion reports ORNL-TM-729, 730, 731, 732, 733, 908, 909, 910, and 911.)

*MSRE + *design + reactors + heat exchangers + pumps +
off-gas systems + drain tanks + graphite + Hastelloy N +
processing + molten salts + experiment +
temperature measurement + thermal shield + absorbers +
electrical circuits + freeze valves +
liquid level measurement + sampling + flowsheets +
instrumentation + maintenance

OTHER CATEGORIES: IAB

MBX640003

McDonald WB

MSRE DESIGN AND CONSTRUCTION (PART OF MSRP SEMIANN PROG
REPORT 7/31/64)

Oak Ridge National Laboratory, Tenn.

ORNL-3708 (Nov. 1964), pp. 22-82, 44 fig.

This paper gives a brief description of the MSRE design and considerable detail (including 35 photographs) of component fabrication and installation, which was nearing completion at the time of writing in August, 1964.

*construction + *MSRE + components + description +
fabrication + progress report

OTHER CATEGORIES: MAB

MBX700002

Webster BH

QUALITY-ASSURANCE PRACTICES IN CONSTRUCTION AND MAINTENANCE OF THE

Category M
MSRE

MBX700002 *Continued*

MOLTEN-SALT REACTOR EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2999 (June 1970), 106 p, 4 fig.

The MSRE was built at ORNL to demonstrate the practicality of the molten-salt reactor concept. Site construction and installation of auxiliary systems were by outside contractors, while the primary reactor systems were installed by ORNL forces. Design, procurement, construction, and maintenance followed ASME codes, ORNL practices, and special procedures developed for the MSRE by the ORNL Reactor Division group primarily responsible for quality assurance. This report describes the program, the problems that were encountered, and the lessons that were learned. Four years of reliable operation of the MSRE proved the success of the quality-assurance program.

*MSRE + quality assurance + construction + inspection + maintenance + procedures + testing + welding

OTHER CATEGORIES: MEK

MCA660001

Guymon RH + Haubenreich PN + Engel JR

MSRE DESIGN AND OPERATIONS REPORT PART XI, TEST PROGRAM

Oak Ridge National Laboratory, Tenn.

ORNL-TM-911 (Nov. 1966) 84 p, 0 fig, 2 ref.

The test program for operation of the MSRE with U-235 fuel is divided into 4 major phases: precritical testing, initial critical measurements; low-power measurements, and reactor capability investigations. Within each phase individual tests are briefly described to define the objectives and general procedures to be followed. Internal Test Memos are identified which contain detailed procedures and check lists used in performing the tests.

*MSRE + *plans + *testing + *operation + primary system + secondary systems + containment + components + experiment + surveillance + procedures

OTHER CATEGORIES: KAE

MCA680004

Engel JR

MSRE DESIGN AND OPERATIONS REPORT, PART XI-A, TEST PROGRAM FOR URANIUM-233 OPERATION

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2304 (Sept. 1968), 18 p, 1 fig, 9 ref.

General plans for operating the MSRE with uranium-233 fuel are outlined. The equipment and procedure for loading highly radioactive uranium-233 are described. Experiment plans include measurement of the initial critical loading, control-rod calibration, evaluation of temperature, fuel concentration and power coefficients of reactivity, power calibration, control-system tests, and dynamics tests. Continued reactor operation will permit study

Category M
MSRE

MCA680004 *Continued*

of the reactivity behavior, noise analysis, measurement of the capture-to-fission ratio for uranium 233, and additional investigation of salt, uranium, fission-product, and materials behavior.

*MSRE + *plans + *uranium-233 + *operation + dynamic characteristics + fuels + stability + reactivity + *surveillance + *loading + criticality + *experiment + chemistry + materials + fission products + nuclear analysis

OTHER CATEGORIES: KAB

MCB650021

Smith AN

MSRE DESIGN AND OPERATIONS REPORT, PART IX, SAFETY PROCEDURES AND EMERGENCY PLANS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-909 (June 1965), 46 p, 16 fig, 3 ref.

Contains brief description of basic plan, emergency philosophy, organization and responsibilities, emergency procedures, description of possible local emergencies and plans of action, and background information.

MSRE + operation + plans + procedures + safety

MCB650022

Guymon RH

MSRE DESIGN AND OPERATIONS REPORT, PART VIII, OPERATING PROCEDURES (VOL. I)

Oak Ridge National Laboratory, Tenn.

ORNL-TM-908 (Vol. I), (Dec. 1965), 478 p.

Contains brief training material, description of auxiliary systems (electrical, air, water, ventilation, etc.), and complete startup check list for each system.

MSRE + operation + operators + procedures + startup

MCB660029

Guymon RH

MSRE DESIGN AND OPERATIONS REPORT. Part VIII, OPERATING PROCEDURES (VOL II)

Oak Ridge National Laboratory

ORNL-TM-908 (Vol II), (Jan. 1966), 539 p.

Contains procedures for reactor startup, fuel sampling, instrument calibration and heat balances, response to unusual operating conditions, routine observations, reactor shutdown, shutdown operations, maintenance. Includes checklists for various operations.

MSRE + operation + plans + procedures

MCB690054

Guymon RH + Haubenreich PN

OPERATING SAFETY LIMITS FOR THE MSRE

Category M
MSRE

MCB690054 *Continued*

Oak Ridge National Laboratory, Tenn.

ORNL-TM-733 (3rd revision) (July 1969), 11 p.

This document prescribes limits for parameters describing the operating conditions of the MSRE. It covers all items directly related to the health and safety of the public. Selected items affecting only the safety of the operators and the protection of the Experiment against disabling accidents are also included. Earlier editions were issued in April 1965, August 1965, and September 1966.

*MSRE + *operation + *safety limits + procedures + safety

OTHER CATEGORIES: KAB

MCB710012

Guymon RH

MSRE PROCEDURES FOR THE PERIOD BETWEEN EXAMINATION AND
ULTIMATE DISPOSAL (PHASE III OF DECOMMISSION PROGRAM)

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3253 (Feb. 1971), 41 p, 3 fig, 9 ref.

This document describes the condition of the MSRE and specifies procedures to be followed after the post-operation examinations and before the ultimate disposal of the fissile and radioactive material in the reactor. The fuel salt will be kept frozen in the sealed drain tanks, within secondary containment whose only opening is through filters to a stack. Surveillance will consist of remote monitoring and daily visits by X-10 plant personnel. Personnel access will be controlled by the security fence around the reactor building. The MSRE Procedures specify remedial actions for abnormal conditions. Also specified are procedures and responsibilities for maintenance, modifications, and removal of surplus equipment.

MSRE + procedures + storage + surveillance +
administration + containment + flowsheets + maintenance +
operation + plans + testing

MCC660005

Ball SJ

SIMULATORS FOR TRAINING MOLTEN-SALT REACTOR EXPERIMENT
OPERATORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1445 (Apr. 5, 1966), 25 p, 11 fig, 4 ref.

Two on-site reactor kinetics simulators were developed for training operators of the Molten-Salt Reactor Experiment (MSRE) in nuclear startup and power-level operating procedures. Both simulators were set up on general purpose, portable Electronic Associates, Inc., TR-10 analog computers and were connected to the reactor control and instrumentation system. The training program was successfully completed. Also, the reactor

Category M
MSRE

MCC660005 *Continued*

control and instrumentation system, the operating procedures, and the rod and radiator-docr drives were checked out. Some minor modifications were made to the system as a result of the experience with these simulators.

analog systems + *MSRE + *training + simulation +
*operators + startup + testing

MCC670044

Guymon RH

MSRE -- TRAINING, PREPARATION FOR OPERATION AND OPERATING
TECHNIQUES

Oak Ridge National Laboratory, Tenn.

Suppl. to ANS Trans (10), Conf. on Reactor Cp. Exp.
(July 1967), p. 35.

Three periods of training were given: Initial (system familiarity prior to checkout), Precritical (A simulator was hooked to actual controls and instrument readcuts), and Pre-power (more simulator work, operation of power systems). Operations procedures and practices are also discussed.

MSRE + operators + procedures + training

MCD680010

Haubenreich PM + Engel JR

RECENT EXPERIENCE WITH THE MOLTEN-SALT REACTOR EXPERIMENT

Oak Ridge National Laboratory, Tenn.

Summary of paper presented at ANS Winter Meeting,
Washington D. C. Nov. 10-15, 1968, ANS Transactions 11 (2)
619, 1 p.

Operating experience with U-235 fuel in the MSRE is described. From initial criticality in June 1, 1965 to March 1968, the reactor was critical 11,515 hr and generated 9005 equivalent full-power hours of energy at power levels to 8 Mw. The shutdown in March 1968 was to permit changeover to U-233 fuel. Operation with U-235 demonstrated system reliability and materials compatibility. Information was collected on fission-product behavior, nuclear characteristics, and reactivity behavior. Operating difficulties were associated with a very small oil leak in the fuel circulating pump, the fuel sampler, air-line disconnects, and secondary cooling blowers. Some remote-maintenance techniques were demonstrated. (A convenient summary of MSRE operating experience through June 1969 is published in Nucl. Appl. Tech. 8, 118 (Feb. 1970). See MCD700001.)

*MSRE + *operation + *experience + *uranium-235 +
reactivity + components + fission products + fuels +
rare gases + maintenance

OTHER CATEGORIES: KAE

MCD690017

Category M
MSRE

MCD69C017 *Continued*
Engel JR + Haubenreich PN
OPERATION OF THE MOLTEN SALT REACTOR EXPERIMENT WITH U-233
FUEL

Oak Ridge National Laboratory, Tenn.

Summary of paper presented at ANS Conference on Reactor
Operating Experience, San Juan, Puerto Rico, Oct. 1-3,
1969, ANS Transactions, 12 (Suppl), 10, 2 p, 3 ref.

The MSRE was operated for 3000 equivalent full-power hours
from September, 1968 through September, 1969 with U-233
fuel. Static and dynamic characteristics of the reactor were
essentially as predicted. Special fuel samples were taken
to evaluate U-233 integral cross-section ratios in an MSR
spectrum. Experiments were conducted to measure cover-gas
entrainment in the fuel loop and its effect on fission-
product stripping. Fission-product distribution was
studied with a remote gamma-ray spectrometer. Small amounts
of plutonium fluoride were added to the fuel. Fuel
chemistry and materials compatibility studies continued to
show good system behavior. (Preprints of the full paper
(25 p, 8 fig, 14 ref) are available from the authors;
similar material is presented in Nucl. Appl. Tech. 8, 118
(Feb. 1970). See MCD700001.)

*MSRE + *operation + *experience + fuels + *uranium-233 +
*plutonium + *nuclear analysis + dynamic characteristics +
reactivity + fission products + bubbles + cross sections +
corrosion + gamma spectrometry + inert gases + noble metals +
rare gases + control rods + pumps + off-gas systems + helium +
argon + cover gas + void fractions

OTHER CATEGORIES: KAB

MCD69C055

Haubenreich PN

MOLTEN-SALT REACTOR PROGRESS

Oak Ridge National Laboratory, Tenn.

Nucl. Engrg. Int. Vol. 14, No. 155 (April 1969),
pp. 325-329, 3 fig.

This article touches briefly on earlier MSR technology
development, then describes the MSRE and its operating
experience. (A more recent, lengthier paper along the
same lines is in the Feb. 1970 Nucl. Appl. Tech.)

*experience + *MSRE + *operation + design + MSRP +
startup

OTHER CATEGORIES: KAB

MCD69C062

Blumberg F + Dyer FF + Houtzeel A

MSRE USES REMOTE GAMMA SPECTROMETRY FOR FISSION PRODUCT
DEPOSITION STUDIES

Oak Ridge National Laboratory, Tenn.

ANS Trans. 12(2) (Dec. 1969), p. 842.

Category M
MSRE

MCD690062 *Continued*

Describes adaptation of gamma ray spectroscopy and remote handling to investigate fission product behavior in inaccessible portions of MSRE system. Enabled study of deposition of fission products on metal surfaces of the system where heat removal is of particular importance. Equipment consisted of GE(Li) diode detector placed at the exit of a one-eighth-inch by 12-in.-long aperture and a 400-channel analyzer.

*design + *gamma spectrometry + *MSRE + analysis +
deposition + experience + fission products + operation +
remote maintenance

OTHER CATEGORIES: JDX + MDE

MCD700001

Haubenreich PN + Engel JR

EXPERIENCE WITH THE MOLTEN-SALT REACTOR EXPERIMENT

Oak Ridge National Laboratory, Tenn.

Nucl. Appl. Tech. 8: 118 (1970) 19 p, 6 fig, 16 ref.

The MSRE is an 8-Mw(th) reactor in which molten fluoride salt at 1200 degrees F circulates through a core of graphite bars. Its purpose was to demonstrate the practicality of key features of molten-salt power reactors. Operation with U-235 in the fuel salt amounted to 9000 equivalent full-power hours between June 1965 and March 1968. At the end of a 15-month demonstration of reliability the reactor was shut down and the U-235 was stripped from the salt in on-site fluorination equipment. U-233 was added to the salt and operation was resumed in October 1968. Over 2500 EFPH has been produced with U-233 through July 1969. The MSRE has shown that salt handling in an operating reactor is quite practical, the salt chemistry is well-behaved, there is practically no corrosion, the nuclear characteristics are very close to predictions, and the system is dynamically stable. Containment of fission products has been excellent, component performance has been good, and maintenance of radioactive components has been accomplished safely and without unreasonable delay.

*experience + *MSRE + *maintenance + *operation + components +
description + fluorination + performance + reactivity +
reliability

OTHER CATEGORIES: KAB + MEC

MDA620001

Engel JR + Haubenreich PN

TEMPERATURES IN THE MSRE CORE DURING STEADY-STATE POWER
OPERATION

Oak Ridge National Laboratory, Tenn.

ORNL-TM-378 (Nov. 5, 1962) 58 p, 14 fig, 8 ref.

Accession Number MCD690062 to MDA620001

Category M
MSRE

MDA620001 *Continued*

Overall fuel and graphite temperature distributions were calculated for a detailed hydraulic and nuclear representation of the MSRE fueled with highly enriched uranium-235. These temperature distributions were importance and volume weighted to obtain nuclear and bulk mean temperatures for both materials. At the design power level of 10 Mw, with the reactor inlet and outlet temperatures at 1175 deg F and 1225 deg F, respectively, the nuclear mean fuel temperature is 1213 deg F. The bulk average temperature of the fuel in the reactor vessel (excluding the volute) is 1198 deg F. For the same conditions and with no fuel permeation, the graphite nuclear and bulk mean temperatures are 1257 deg F and 1226 deg F, respectively. Fuel permeation of 2% of the graphite volume raises these values to 1264 deg F and 1231 deg F, respectively. Power coefficients of reactivity are calculated under various assumptions of system temperature control.

*analysis + *MSRE + *cores + *reactivity + fluid flow + heat generation + neutron flux + nuclear analysis + *thermal effects + calculations

MDA620002

Prince BE + Engel JR

TEMPERATURE AND REACTIVITY COEFFICIENT AVERAGING IN THE MSRE
Oak Ridge National Laboratory, Tenn.

CRNL-TM-379 (Oct. 15, 1962) 26 p, 6 fig, 5 ref.

Use is made of the concept of 'nuclear average temperature' to relate the spatial temperature profiles in fuel and graphite attained during high power operation of the MSRE to the neutron multiplication constant. Based on two-group perturbation theory, temperature weighting functions for fuel and graphite are derived, from which the nuclear average temperatures may be calculated. Similarly, importance-averaged temperature coefficients of reactivity are defined. The values of the coefficients calculated for the MSRE were -4.4×10^{-5} deg F for the fuel and -7.3×10^{-5} for the graphite. These values refer to a reactor fueled with salt which does not contain thorium. They were about 5% larger than the values obtained from a one-region, homogeneous reactor model, thus reflecting the variation in the fuel volume fraction throughout the reactor and the effect of the control rod thimbles on the flux profiles.

*MSRE + *nuclear analysis + *reactivity + *thermal effects + models + neutron physics + calculations + methods

OTHER CATEGORIES: BEX

MDA630002

Haubenreich FN

INHERENT NEUTRON SOURCE IN CLEAN MSRE FUEL SALT

Category M
MSRF

MDA63C002 *Continued*

Oak Ridge National Laboratory, Tenn.

ORNL-TM-611 (Aug. 1963), 17 p, 6 ref.

Alpha particles from uranium interact with beryllium and fluorine to produce an inherent source of neutrons in the MSRE fuel salt. The spontaneous fission source is relatively insignificant. Calculations are described which predict an inherent source of $3 \text{ to } 5 \times 10^{15}$ neutrons/sec in the 25 cu ft of salt in the MSRE core.

calculations + MSRE + neutron sources + fuels + molten salts

MDA64C001

Engel JR + Haubenreich PN + Prince BE

MSRE NEUTRON SOURCE REQUIREMENTS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-935 (Sept. 1964), 37 p, 11 fig, 6 ref.

The alpha-n source inherent in the uranium-235 fuel salt meets all the safety requirements for a neutron source in the MSRE. Subcritical flux distributions were calculated to determine the combination of external source strength and detector sensitivity required for monitoring the reactivity. If more sensitive detectors than the servo-driven fission chambers are installed in the instrument shaft to monitor the filling operation, the calculations indicate that the required source strength can be reduced from 4×10^{17} n/sec to 7×10^{16} n/sec. An antimony-beryllium source with an initial strength of 4×10^{18} n/sec would still produce 7×10^{16} n/sec one year after installation. Abstractor's note: The external source ultimately selected for use in MSRE was americium-curium-beryllium.

*calculations + *neutron sources + *MSRE + safety + neutron flux + nuclear analysis

MDA640002

Engel JR + Prince BE

CRITICALITY FACTORS IN MSRE FUEL STORAGE AND DRAIN TANKS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-759 (Sept. 1964), 18 p, 3 fig, 3 ref.

Calculations indicate that there is no danger in the fuel storage or drain tanks with uranium-235 fuel salt of normal concentration even if the salt is frozen and cooled to 20 deg C and the tank is submerged in water, provided the uranium remains evenly dispersed in the salt. If segregation of the uranium occurs during freezing and all the uranium accumulates in a region near the center of a tank, criticality will occur at 20 deg C for concentrations factors of 4 or more. Criticality can be avoided by keeping the salt molten or by dividing the fuel charge among two or more tanks before it is allowed to freeze.

*calculations + *criticality + *freezing + *fuels +

Category M
MSRE

MDA640002 *Continued*
uranium-235 + *drain tanks + storage + density +
phase equilibria + *MSRE + nuclear analysis
OTHER CATEGORIES: BGX

MDA640006

Haubenreich PN + Engel JR + Prince BE + Claiborne HC
MSRE DESIGN AND OPERATIONS REPORT PART III -- NUCLEAR ANALYSIS
Oak Ridge National Laboratory, Tenn.

ORNL-TM-730 (Feb. 1964), 199 p, 63 fig, 52 ref.

Early calculations of effects of core size and fuel-to-graphite ratio had determined the core design. This report describes the calculated nuclear characteristics of the MSRE with 3 fuel compositions. One had thorium and highly enriched U-235; a second, highly enriched U-235 and no thorium; and a third, 35% enriched U-235 and no thorium, all in a carrier salt of lithium, beryllium and zirconium fluorides. Calculated quantities include critical loadings, fluxes, temperature distributions, temperature coefficients, delayed neutron effects, control rod worth, dynamics, and neutron sources.

*MSRE + *nuclear analysis + control rods + criticality +
delayed neutrons + heat generation + neutron flux +
reactivity + shielding

OTHER CATEGORIES: BBX + BCX

MDA640007

Beall SE + Haubenreich PN + Lindauer RE + Tallackson JF
MSRE DESIGN AND OPERATIONS REPORT, PART V -- REACTOR
SAFETY ANALYSIS REPORT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-732 (Aug. 1964), 300 p, 109 fig, 50 ref.

The MSRE is described with emphasis on component design, instrumentation and controls, site, layout, and containment. Plans and staff for startup and operation are outlined. Three different fuel compositions and a power level of 10 Mw were considered. Nuclear incidents that conceivably could cause damage are analyzed and it is concluded that none could breach the containment. The secondary containment design is shown to be adequate even if all the fuel were spilled in the cell.

*analysis + *MSRE + *safety + accidents + containment +
control + instrumentation

OTHER CATEGORIES: BGX + JEX

MDA650001

Ball SJ + Kerlin TW
STABILITY ANALYSIS OF THE MOLTEN-SALT REACTOR EXPERIMENT
Oak Ridge National Laboratory, Tenn.

ORNL-TM-1070 (Dec. 1965), 80 p, 23 fig, 20 ref.

A detailed analysis shows that the Molten-Salt Reactor Experiment is inherently stable with uranium-235 fuel.

Category M
MSRE

MDA650001 *Continued*

It has sluggish transient response at low power, but this creates no safety or operational problems. The study included analysis of the transient response, frequency response, and pole configuration. The effects of changes in the mathematical model for the system and in the characteristic parameters were studied. A systematic analysis was also made to find the set of parameters, within the estimated uncertainty range of the design values, that gives the least stable condition. The system was found to be inherently stable for this condition, as well as for the design condition. Comparisons are made with previous models which underestimated stability. Reasons are given to explain the increase in stability with increasing power level.

*MSRE + *stability + *dynamic characteristics + *uranium-235 + analysis + nuclear analysis + calculations + *models

OTHER CATEGORIES: BCX

MDA660003

Prince BE

PERIOD MEASUREMENTS ON THE MOLTEN SALT REACTOR EXPERIMENT
DURING FUEL CIRCULATION: THEORY AND EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1626 (Oct. 1966), 36 p, 8 fig, 10 ref.

A theory of period dependence on the fuel circulation is developed from the general space-dependent reactor kinetics equations. A procedure for computer evaluation of the resulting inhour-type equation is presented, together with numerical results relating the reactivity to the observed asymptotic period, both with the fuel circulating and with it stationary. The calculated reactivity difference between the time-independent flux conditions for the noncirculating and the circulating fuel states is in close agreement with the value inferred from the MSRE rod calibration experiments. Rod-hump period measurements made with the fuel circulating were converted to differential rod worth by use of this model. These results are compared with similar rod sensitivity measurements made with the fuel stationary. The rod sensitivities measured under these two conditions agree favorably, within the limits of precision of the period measurements. Due to the problem of maintaining adequate precision, however, the period-rod sensitivity measurements provide a less conclusive test of the theoretical model than the reactivity difference between the time-independent flux conditions.

*analysis + *circulation + *dynamics tests + *experiment + *MSRE + control rods + criticality + delayed neutrons + experience + models + reactivity + startup + uranium-235

OTHER CATEGORIES: BCX + MDC

MDA660004

Category M
MSRE

MDA66C004 *Continued*

Engel JR + Haubenreich PN + Ball SJ
ANALYSIS OF FILLING ACCIDENTS IN MSRE
Oak Ridge National Laboratory, Tenn

ORNL-TM-497 (Aug. 1966), 41 p, 14 fig, 2 ref.

Whenever the MSRE is shut down, the fuel salt is drained from the core. Then, during a normal startup, the graphite and the fuel are preheated and the control rods are positioned so that the reactor remains subcritical while it is being filled. Certain abnormal circumstances could result in criticality and a power excursion in the partially filled core. Various postulated incidents were surveyed and the worse case was analyzed in detail. This case involved selective freezing in the drain tanks to concentrate the uranium in the molten salt fraction. Physical restrictions on the fill rate and safety actions of control rods and gas control valves limited the calculated power and temperature excursions so that any damage to the reactor would be prevented. Abstractor's note: It was subsequently shown that the degree of uranium concentration required for a serious filling accident cannot be attained by partial freezing of the salt. Protective circuits and administrative procedures to prevent abnormal fills were retained.

*MSRE + *accidents + *analysis + *freezing + excursions +
*simulation + *nuclear analysis
OTHER CATEGORIES: BGX

MDA67C038

Haubenreich PN

SAFETY ASPECTS OF THE MSRE

Oak Ridge National Laboratory, Tenn.

Nucl. Safety Vol. 8 No. 3 (1967) pp. 226-235, 2 fig, 12 ref.

Fluid-fuel and solid-fuel reactors, although similar in ultimate containment requirements, differ in the kinds of accidents that can cause system damage. Some fluid fuels are susceptible to segregation, and filling accidents are more likely, but afterheat is more easily handled. The surviving fluid-fuel concept, the molten-salt reactor, is being developed into a thermal breeder. The MSRE, in which molten fluoride salts circulate at 1200 deg F, is a step in that development. Safety analyses and experience have shown the MSRE to be safe, and no serious problems are expected in designing safe molten-salt breeder reactors.

*accidents + *MSRE + *reviews + *safety + afterheat +
analysis + containment + experience
OTHER CATEGORIES: BGX

MDA670040

Kedl RJ + Hcutzeel A

Category M
MSRE

MDA670040 *Continued*

DEVELOPMENT OF A MODEL FOR COMPUTING XE-135 MIGRATION IN
THE MSRE

Oak Ridge National Laboratory, Tenn.

ORNL-4069 (June, 1967), 77 p, 22 fig, 3 tab, 21 ref.

The report deals primarily with developing a model for computing the migration of Xe-135 in the MSRE and with experiments conducted to establish the model. A preoperational experiment was run in the MSRE with Kr-85 tracer, and many of the gas-transport constants were inferred from the results. Equivalent transport constants for calculating the Xe-135 migration gave a poisoning of about 1.4% without circulating bubbles and well below 1% with bubbles. Preliminary measurements made on the critical reactor show xenon poisoning of 0.3 to 0.4%. Since physical measurements confirm that there are bubbles in the system, the conclusion is drawn that the computation model, the krypton experiment, and reactor operation agree.

*analysis + *mass transfer + *models + *MSRE + *xenon +
computer codes + diffusion + graphite + krypton

OTHER CATEGORIES: BFX + IBA

MDA670041

Kedl RJ

A MODEL FOR COMPUTING THE MIGRATION OF VERY SHORT LIVED
NOBLE GASES INTO MSRE GRAPHITE

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1810 (July 1967), 26 p, 4 fig, 1 tab, 6 ref.

A model describing the migration of very short-lived noble gases from the fuel salt to the graphite in the MSRE CORE was developed. From the migration rate, the model computes (with certain limitations) the daughter-product distribution in graphite as a function of reactor operational history. Noble-gas daughter-product concentrations (EA-140, CE-141, SR-89, and Y-91) were measured in graphite samples removed from the MSRE core after 7800 MWhr of power operation. Concentrations of these isotopes computed with this model compare favorably with measured values.

*analysis + *experience + *fission products + *graphite +
*mass transfer + *MSRE + computer codes + diffusion +
krypton + molten salts + xenon

OTHER CATEGORIES: IEA

MDA680003

Haubenreich FN + Engel JR + Gattard CH + Guymon RH +
Prince BE

MSRE DESIGN AND OPERATIONS REPORT PART V-A, SAFETY ANALYSIS
OF OPERATION WITH U-233

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2111 (Feb. 1968), 80 p, 24 fig, 36 ref.

This report presents data and analyses that support

Accession Number MDA670040 to MDA680003

Category M
MSRE

MDA680003 *Continued*

the conclusion that it is safe to load and operate the MSRE with uranium-233. It summarizes pertinent experience with the MSRE and new information on materials through December, 1967. Procedures for producing, handling and loading enriching salt are described and their safety assessed. The nuclear characteristics of the reactor with U-233 fuel are presented and the possibility of breach of the primary containment due to credible nuclear incidents is reexamined taking into account the different dynamics characteristics, the action of the safety system and the condition of the salt system after two years of operation.

*analysis + *MSRE + *safety + *uranium-233 + containment + dynamic characteristics + engineered safeguards + excursions + fuel preparation + Hastelloy N + loading + radiation damage + reactivity

OTHER CATEGORIES: BGX

MDA690001

Steffy RC

INHERENT NEUTRON SOURCE IN MSRE WITH CLEAN U-233 FUEL
Oak Ridge National Laboratory, Tenn.

ORNL-TM-2685 (Aug. 10, 1969) 22p, 2 fig, 11 ref.

After about three years of nuclear operation, the MSRE fuel, enriched U-235, was replaced with a U-233 fuel mixture. In this new mixture there are quantities of U-232, U-233, and U-234. Each of these, along with the U-232 decay chain, is a strong alpha emitter and interacts with fluorine, beryllium, and lithium to produce neutrons. This neutron source is time-dependent because of the buildup of U-232 daughters, and at the time of reaching criticality with the U-233 fuel, the neutron source in the MSRE core was about 4×10^{10} (8th) neutron/sec, primarily from the reactions $\text{Be}^9(\text{Alpha}, \text{N})\text{C}^{12}$ and $\text{F}^{19}(\text{Alpha}, \text{N})\text{Na}^{22}$. Alpha-N reactions with lithium will produce less than 3×10^{10} (6th) neutrons/sec. Spontaneous fission will produce less than 10^{10} (2rd) neutrons/sec.

*MSRE + *neutron sources + *fuels + *uranium-233 + *uranium-232 + decay + beryllium + fluorine + lithium + calculations

MDA690002

Steffy RC • Wood FJ

THEORETICAL DYNAMIC ANALYSIS OF THE MSRE WITH U-233 FUEL
Oak Ridge National Laboratory, Tenn.

ORNL-TM-2571 (July, 1969) 42 p, 17 fig, 11 ref.

A study undertaken to characterize the dynamics of the U-233 fueled MSRE prior to operation revealed that the system is inherently asymptotically stable at all

Category M
MSRE

MDA690002 *Continued*

power levels above zero. The motivation for these studies was the expected difference between the MSRE dynamic response with U-233 fuel and with U-235 fuel because of the smaller delayed-neutron fraction of U-233. An existing system model, previously verified for U-235 fuel, was modified for use in this work. The reactor system response to reactivity perturbations is rapid and nonoscillatory at high power, and it becomes sluggish and oscillatory at lower powers. These characteristics were determined by three methods:

(1) transient-response analyses, including a check of the validity of the linear model, (2) a frequency-response and sensitivity study, (3) stability analyses, both by inspection of the system eigenvalues and application of the recently developed, modified Mikhailov criterion.

*MSRE + *dynamic characteristics + *stability + *models + *uranium-233 + fuels + nuclear analysis + reactivity + delayed neutrons + computer codes + mixing + calculations + methods

OTHER CATEGORIES: BCX

MDA690005

Burke CW + Clark FH

ANALYSIS OF TRANSIENTS IN THE MSRE SYSTEM WITH URANIUM-233 FUEL

Oak Ridge National Laboratory, Tenn.

CRNL-4397 (June 1969), 47 p, 19 fig, 4 ref.

The uranium-233 fueled MSRE system was simulated on the CRNL analog computer. The simulated system was used to evaluate the existing MSRE control and safety systems when used on the uranium-233 fueled system. The pertinent results and conclusions were as follows: (1) The safety system will limit the 'startup accident' so that the peak power will be 100 kw. (2) A quantity of uranium-233 sufficient to cause a reactivity change of approximately -1% delta K/K when precipitated out of the fuel at some point in the system external to the core could be swept back into the core in a concentrated form without causing excessive core damage. (3) The existing controller will control the uranium-233 fueled system in a stable manner; however, an increased velocity feedback gain will be required.

analysis + dynamic characteristics + computers + feedback + MSRE + analog systems + stability + mathematics + models + simulation + control + reactivity + safety + startup + heat transfer + control rods + uranium-233 + accidents + control-rod drives + delayed neutrons + dynamics tests + excursions + kinetic equations

MDA700006

Accession Number MDA690002 to MDA700006

Category M
MSRE

MDA700006 *Continued*

Ulrich WC

AN EXTENDED HYDRAULIC MODEL OF THE MSRE CIRCULATING FUEL SYSTEM (THESIS)

Oak Ridge National Laboratory, Tenn.

ORNL-TM-3007 (June 1970) 53 p, 6 fig, 14 ref.

The hydraulic portion of a combined hydraulic-neutronic mathematical model for determining the effects of helium gas entrained in the circulating fuel salt of the MSRE on the neutron flux-to-pressure frequency response was extended to include effects due to the fuel pump and helium cover-gas system. By comparing the computed results with experimental data, it was concluded that pressure perturbations introduced by the fuel pump were the main source of the naturally occurring neutron flux fluctuations in the frequency range of one to a few cycles per second. It was also noted that the amplitude of the neutron flux-to-pressure frequency-response function was directly proportional to the pressure in the fuel-pump bowl; however, further work will be required before completely satisfactory results are obtained from the extended model.

Recommendations are proposed which should prove useful in future modeling of similar hydraulic systems.

*MSRE + *models + *hydraulics + *dynamic characteristics + *measurement + reactivity + computer codes + hydrodynamics + primary system + nuclear analysis + cover gas + calculations
OTHER CATEGORIES: MDC

MDA700007

Bell MJ

CALCULATED RADIOACTIVITY OF MSRE FUEL SALT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2970 (May 1970) 21 p, 0 fig, 8 ref, 12 tables.

Calculations have been made of the inventory and radioactivity of the fission products and transuranium isotopes present in the MSRE fuel salt. The calculations included operation with both U-235 and U-233 fuels, the effect of stripping of noble gases, and fluorination of the fuel salt after the period of U-235 operation. Results are presented which give the inventory and radioactivity of individual isotopes in the salt up to January 1, 1975. After storage for 5 years, the gamma-ray shielding required for shipping fuel is determined by thallium-208 and neutrons from uranium-232 daughters produce the controlling radiation dose through a lead shield.

*MSRE + *fission products + *isotopes + storage + fuels + *decay + uranium-235 + uranium-233 + uranium-232 + disposal + processing + shielding + *inventories + rare gases + radioactivity + calculations

MDA700032

Category M
MSRE

MDA700032 *Continued*

Robinson JC

ANALYTICAL DETERMINATION OF THE NEUTRON FLUX-TO-PRESSURE
FREQUENCY: APPLICATION TO THE MOLTEN-SALT REACTOR
EXPERIMENT

Oak Ridge National Laboratory, Tenn.

Nucl. Sci. and Eng. 42(3), 382-396 (December 1970), 15 p,
7 fig, 2 tables, 18 ref.

The neutron flux-to-pressure frequency response for a molten-salt-fueled reactor with a small amount of gas entrained in the molten salt was determined analytically. The one-dimensional conservation equations describing the flow of the compressible molten-salt gas mixture and the one-group neutron diffusion equations were written in the linearized perturbed form, and Laplace transformation in time was performed. The coupled set of equations describing the conservation of mass for the molten salt, conservation of mass for the gas, and conservation of momentum for the salt-gas mixture (the hydraulic equations) was solved by employing matrix exponential techniques. The remaining equations were solved by more conventional schemes. The matrix exponential technique was selected to obtain a solution for the hydraulic equations over the techniques normally employed (nodal or nodal) for stability studies in boiling water systems because the validity of the solution is independent of the frequency of interest, and the total number of simultaneous equations required to be solved for application of boundary conditions (closing the flow loop) is small. Results from the computed neutron flux-to-pressure frequency response for the molten-salt-fueled reactor under study show that the shape of the modulus of the frequency response for frequencies below 1 to 2 cycles/sec is independent of the void fraction (volume fraction occupied by the gas), and the magnitude of the modulus of the frequency response is proportional to the void fraction. Therefore, we conclude that the amount of void in the system can be inferred by comparing the analytical frequency response with an experimental frequency response. (This conclusion was verified and is reported in the following paper.)

MSRE + bubbles + dynamic characteristics + noise analysis +
void fractions

MDA710003

Kerlin TW + Fall SJ + Steffy RC

THEORETICAL DYNAMICS ANALYSIS OF THE MOLTEN-SALT REACTOR
EXPERIMENT

Oak Ridge National Laboratory, Tenn.

Nuclear Technology, Feb. 1971, 15 p, 24 fig, 12 ref.

The dynamic characteristics of the MSRE were calculated for operation with U-235 and U-233 fuels. The analysis

Category M
MSRE

MDA710003 *Continued*

included calculation of the transient response for reactivity perturbations, frequency response for reactivity perturbations, stability and sensitivity to parameter variations. The calculations showed that the system dynamic behavior is satisfactory for both fuel loadings.

MSRE + dynamic characteristics + feedback + kinetic equations + simulation + stability

MDB700003

Gabbard CH

REACTOR POWER MEASUREMENT AND HEAT TRANSFER PERFORMANCE IN THE MOLIEN SALT REACTOR EXPERIMENT

Oak Ridge National Laboratory, Tennessee

ORNL-TM-3C02 (May 1970) 32 p, 6 fig, 16 ref.

The operating power of the MSRE as determined by a heat balance on the fuel and coolant salt systems, was 8.0 MW. Changes in the isotopic composition of uranium and plutonium in the fuel salt indicated a power lower by about 7 - 10%. Attempts to resolve this discrepancy have been inconclusive. The coolant salt flow rate was found to be the only potential source of significant error in the heat balance. A calibration check of the instruments is planned. The heat-removal capabilities of the fuel-salt to coolant-salt heat exchanger and coolant-salt to air radiator were below the predictions of the original design calculations. In the case of the primary heat exchanger, the overestimate was due to the use of erroneous, estimated physical property data. In the case of the radiator, the overestimate in the design was only partially explained by the improper selection of an air 'film' temperature. There was no decrease in heat transfer capability of the two heat exchangers over more than 3 years of operation.

*MSRE + *heat balance + *heat transfer + *heat exchangers + molten salts + *performance + *operation + reactors + *experience + *flow measurement + isotopes + analysis + *thermal power + components + reviews + design + design data + physical properties + thermal properties + instrumentation + specific heat + heat balance

MDB700033

Robinson JC + Fry DN

EXPERIMENTAL NEUTRON FLUX-TO-PRESSURE FREQUENCY RESPONSE FOR THE MOLIEN-SALT REACTOR EXPERIMENT: DETERMINATION OF VOID FRACTION IN FUEL SALT

Oak Ridge National Laboratory, Tenn.

Nucl. Sci. and Eng. 42(3), 397-405 (Dec. 1970), 9 p, 10 fig, 1 table, 14 ref.

Small pressure perturbations were introduced into the primary fuel pump bowl of the MSRE operating at its nominal power of 8 MW(th). The experimental neutron flux-to-

Category M
MSRE

MDB700033 *Continued*

pressure frequency response was then obtained from a cross-power and auto-power spectral density analysis of the resulting signals from a neutron-sensitive ionization chamber and a pressure transducer. By comparing the frequency dependence of the experimental frequency response determined for the reactor operating at power with the frequency response determined from analysis of mathematical models, the selection of the more appropriate boundary condition set from a choice of two possible boundary condition sets was possible. Then the analytical frequency response was fitted by the least-squares method to the experimental frequency response to obtain the void fraction in the molten salt fuel. A void fraction of 0.61 plus or minus 0.04% was determined from the frequency response; this value compares favorably with a value of 0.6 plus or minus 0.1% determined by other techniques. Conclusions are that the analytical model leads to acceptable results for the neutron flux-to-pressure frequency response and that properly designed dynamic tests involving small reactivity perturbations (introduced by means other than rod motion) can be used to extract specific nuclear parameters for a nuclear system operation at power.

MSRE + bubbles + dynamic characteristics + dynamics tests + measurement + noise analysis + void fractions

OTHER CATEGORIES: MCD

MDB71002

Kerlin TW + Ball SJ + Steffy RC + Buckner MR

EXPERIENCES WITH DYNAMIC TESTING METHODS AT THE MOLTEN-SALT REACTOR EXPERIMENT

Oak Ridge National Laboratory, Tenn.

Nuclear Technology, Feb. 1971, 15 p, 19 fig, 13 ref.

A series of reactivity-to-power frequency response measurements was made on the Molten-Salt Reactor Experiment. This was done for U-233 and U-235 fuels, for a range of operating power levels, at several points in the system operating history, and for several different test procedures. A comparison of experimental results with prior theoretical predictions confirmed the validity of the theoretical predictions. The test program included measurements using the pseudorandom binary sequence, pseudorandom ternary sequence, n-sequence, and the multifrequency binary sequence.

MSRE + dynamic characteristics + dynamics tests + feedback + noise analysis + stability + experience

OTHER CATEGORIES: MCD

MDC66002

Kerlin TW + Ball SJ

EXPERIMENTAL DYNAMIC ANALYSIS OF THE MOLTEN-SALT REACTOR

Category M
MSRE

MDC66002 *Continued*
EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1647 (Oct. 13, 1966), 58 p, 29 fig, 29 ref.

Dynamics tests were performed on the uranium-235 fueled MSRE for the full range of operating power levels to determine the power-to-reactivity frequency response. Three types of input disturbances were used: the pseudorandom binary reactivity input, the pulse reactivity input, and the step reactivity input. The frequency response of the uncontrolled reactor system displayed resonant behavior in which the frequency of oscillation and the damping increased with increasing power level. Measured periods of natural oscillation ranged from thirty minutes at 75 KW to two minutes at 7.5 MW. These oscillations were lightly damped at low power, but strongly damped at higher power. The measured results generally were in good agreement with predictions. The main conclusion is that the system has no operational stability problems and that the dynamic characteristics are essentially as predicted.

*MSRE + *dynamics tests + *dynamic characteristics +
*experiment + measurement + *stability + testing +
reactivity + thermal power + procedures + analysis + methods
OTHER CATEGORIES: BCX

MDC670001

Engel JR + Prince BE

THE REACTIVITY BALANCE IN THE MSRE

Oak Ridge National Laboratory, Tennessee

ORNL-TM-1796 (Mar. 10, 1967) 54 p, 16 fig, 15 ref.

Experience with a reactivity balance calculation is described for approximately 1 year of MSRE power operation with uranium-235 fuel. Computations were performed every 5 minutes by an on-line digital computer. Results were used initially to evaluate xenon poisoning in the reactor and subsequently to monitor for anomalous reactivity changes. Sensitivity for detecting short-term changes in fuel composition is 10 times greater than chemical analysis. No significant long-term drift in reactivity is observed at zero power with no xenon present. A more detailed and comprehensive report of the theoretical base and the entire experience with uranium-235 fuel is presented in CFNL-4674, accession number MEC710006.

*MSRE + *reactivity + *experience + *xenon +
*nuclear analysis + operation + data processing + bubbles +
control rods + rare earths + models + fission products +
fuels + uranium-235
OTHER CATEGORIES: BBX

MDC670002

Accession Number MDC660002 to MIC670002

Category M
MSRE

MDC670002 *Continued*

Engel JR + Prince BE

THE REACTIVITY BALANCE IN THE MSRE

Oak Ridge National Laboratory, Tenn.

Abstract of paper presented at the Thirteenth Annual Meeting
of the American Nuclear Society, San Diego, Calif.,
June 11-15, 1967, American Nuclear Society Transactions
10(1), p 337

Experience with a reactivity balance calculation is
described for approximately 1 year of MSRE power operation
with uranium-235 fuel. This abstract (and preprint
of paper) is a synopsis of ORNL-TM-1796, same title,
accession number MDC670001.

*MSRE + *reactivity + *experience + *xenon +
*nuclear analysis + operation + data processing + bubbles +
control rods + rare earths + models + fission products +
fuels + uranium-235

MDC680002

Prince BE + Ball SJ + Engel JR + Haubenreich FN + Kerlin TW
ZERO-POWER PHYSICS EXPERIMENTS ON THE MOLTEN-SALT REACTOR
EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ORNL-4233 (Feb. 1968), 60 p, 24 fig, 25 ref.

This report describes the techniques and results of
a program of experiments designed to measure the important
neutronic characteristics of the MSRE, under conditions
of negligible nuclear heat generation. The program
includes the initial critical U-235 loading, the control-
rod calibration (period-differential worth and rod
drop-integral worth measurements), determinations of
the reactivity loss due to fuel circulation, the 'static'
reactivity coefficients of excess U-235 concentration and
isothermal core temperature, the fuel salt temperature
reactivity coefficient, the pressure effects on reactivity,
and a series of system dynamics tests (frequency response,
transient flow, and neutron flux noise measurements).
These measurements, carried out in June 1965, form much
of the experimental baseline for evaluation of the
nuclear operation at full power with U-235 fuel. The
report includes discussions of the comparisons of the
measurement results with the corresponding neutronic
characteristics calculated from theoretical models.

*analysis + *criticality + *dynamics tests + *experiment +
*MSRE + *reactivity + *uranium-235 + circulation +
control rods + delayed neutrons + experience + models +
startup

OTHER CATEGORIES: KAE + MCD

MDC680005

Fry DN + Kryter EC + Robinson JC

Accession Number MDC670002 to MDC680005

Category M
MSRE

MDC680005 *Continued*

MEASUREMENT OF HELIUM VOID FRACTION IN THE MSRE FUEL SALT
USING NEUTRON-NOISE ANALYSIS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2315 (Aug. 1968), 32 p, 10 fig, 17 ref.

Investigations were made at the MSRE during power operation with uranium-235 fuel to determine if the amount of helium gas in the fuel salt could be measured using neutron noise analysis. The neutron power spectral density (NPSD) was measured at different reactor operating conditions and compared with analytical model predictions of the NPSD for the same conditions. Results showed that the principal source of small neutron density fluctuation observed in the MSRE is helium bubbles circulating in the fuel salt. The measurements showed that NPSD in the frequency range from 0.5 to 2 cps varied as the square of helium void fraction as predicted by the model, and that the minimum void fraction was more nearly zero than the previously accepted value of 0.1%. It is concluded that changes in the circulating void fraction can be inferred with good sensitivity directly from neutron noise measurements, and, consequently, NPSD can complement and enhance the value of the MSRE reactivity balance calculations.

*analysis + *experience + *models + *MSRE + *noise analysis +
*reactivity + *void fractions + dynamic characteristics +
experiment + uranium-235

OTHER CATEGORIES: MCD

MDC690003

Robinson JC + Fry DN

DETERMINATION OF THE VOID FRACTION IN THE MSRE USING SMALL
INDUCED PRESSURE PERTURBATIONS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2318 (Feb. 1969), 58 p, 11 fig, 22 ref.

With the MSRE operating at 5 Mw, sawtooth pressure perturbations were introduced into the fuel-pump bowl to determine the amount of helium gas entrained in the circulating fuel. The pressure and neutron flux signals were simultaneously amplified and recorded on magnetic tape. Then the signals were analyzed using auto-power spectral density, cross-power spectral density, cross-correlation, and direct Fourier transform techniques to obtain the neutron-flux-to-pressure frequency-response function. An analytical model, developed previously to aid in the interpretation of the fluctuations of the neutron flux in an unperturbed system, was used to infer from the experimental data the amount of helium void (interpreted as a void fraction) entrained in the fuel salt. A description of the analytical

Category M
MSRE

MDC690003 *Continued*

model and its experimental verification are included in this report. The void fraction was determined to be between 0.023 and 0.045%. The uncertainty of this inference is attributed to assumptions made in the model. (Abstractor's note: This work was subsequently reported in Nucl. Sci. & Tech., see MDB700009.)

*analysis + *dynamics tests + *experiment + *models +
*MSRE + *reactivity + *void fractions +
dynamic characteristics + experience + uranium-235 +
pressure + theory

OTHER CATEGORIES: MCD

MDC690015

Fry DN + Kryter RC + Robinson JC

ANALYSIS OF NEUTRON NOISE IN A MOLTEN SALT REACTOR
OPERATING AT POWER

Oak Ridge National Laboratory, Tenn.

Summary of paper presented at ANS Annual Meeting, Seattle,
Wash., June 15-19, 1969, ANS Transactions 12(1), 299, 2 p,
1 fig, 5 ref.

Neutron flux noise in the MSRE was studied by Fourier analysis of an ionization chamber signal over the frequency range from 0.1 to 15 cycle/sec. Measurements at various operating conditions and cross correlations with other reactor signals showed marked changes with changes in the circulating void fraction and a high degree of correlation between neutron flux noise and pressure noise. It appears that cross correlation of neutron and pressure noise offers a non-perturbing method of determining the circulating void fraction with the reactor operating at power. (This work is described in detail in ORNL-TM-2315. See MIC680005.)

*MSRE + *noise analysis + *void fractions + measurement +
neutron flux + helium + cover gas

OTHER CATEGORIES: MCD

MDC690016

Robinson JC + Fry DN

THE FREQUENCY RESPONSE OF THE NEUTRON FLUX TO PRESSURE IN A
CIRCULATING FUEL REACTOR - ANALYTICAL AND EXPERIMENTAL

Oak Ridge National Laboratory, Tenn.

Summary of paper presented at ANS Annual Meeting, Seattle,
Wash., June 15-19, 1969, ANS Transactions 12 (1) 292, 2 p,
3 ref.

An analytical model was developed to compute the neutron-flux-to-pressure frequency response in a reactor fuelled with circulating molten salt. Since entrained cover gas makes the circulating fluid compressible, pressure perturbations induce reactivity, and hence neutron flux, perturbations. Analysis indicated that the neutron-flux-to-pressure frequency response is proportional to the

Category M
MSRE

MDC690016 *Continued*

circulating void fraction in the frequency range from 0.01 to 0.1 cycle/sec. Sawtooth pressure perturbations were imposed on the MSRE at full power and the frequency response was measured. Experimental data were best fitted with a circulating void fraction of 0.04%. (A detailed discussion of the model and experimental results is presented in ORNL-TM-2318. See MDC-690003.)

*MSRE + *dynamic characteristics + *bubbles + *models +
*nuclear analysis + experiment + measurement +
*void fractions + cover gas + helium + calculations

OTHER CATEGORIES: MCD

MDC700004

Steffy RC

FREQUENCY RESPONSE TESTING OF THE MOLTEN SALT REACTOR
EXPERIMENT (THESIS)

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2823 (Mar. 1970) 118 p, 27 fig, 31 ref.

Tests to determine the neutron flux-to-reactivity frequency response were performed on the MSRE with the reactor at various power levels between zero and full power and with the reactor fueled with a U-235 fuel mixture and a U-233 fuel mixture. Test patterns employed were pseudo-random binary sequences (PRBS) and pseudorandom ternary sequences (PRTS) of various sequence lengths and minimum-pulse-duration times. In some tests reactivity (control-rod position) was forced to follow the test pattern, and in other tests the neutron flux was forced to follow the test pattern. The experimental results were analyzed by several different methods and the results were compared. The frequency response of the uncontrolled reactor system was found to be in good agreement with theoretical predictions for both the U-235 and U-233 fuel loadings. There were no indications of response characteristics that might cause control or safety problems. Advantages and disadvantages of various testing and analytical methods are discussed.

*MSRE + *dynamic characteristics + *measurement + *experiment +
*testing + reactivity + computer codes + data processing +
nuclear analysis + *analysis + control rods +
control-rod drives + computers + data acquisition systems +
methods

OTHER CATEGORIES: MCD

MDC700005

Steffy RC

EXPERIMENTAL DYNAMIC ANALYSIS OF THE MSRE WITH U-233 FUEL

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2997 (April 1970), 28 p, 10 fig, 11 ref.

During the startup with U-233 fuel, tests showed that the system time response to step changes in reactivity, the

Category M
MSRE

MDC700005 *Continued*

flux-to-reactivity frequency response, and the outlet temperature-to-power frequency response agreed favorably with theoretical predictions. Time-response tests 1, 5, and 8 Mw verified the prediction that, although after a perturbation the reactor returned to its original power level more rapidly when the initial power was high than when it was low, the system was load-following at all significant power levels. Flux-to-reactivity frequency response was effectively measured using periodic pseudorandom binary and ternary sequences. As predicted, for the MSRE, the degree of stability increased with increasing power level.

*analysis + *dynamic characteristics + *dynamics tests +
*experiment + *MSRE + *reactivity + *stability + experience +
models + uranium-233 + methods

OTHER CATEGORIES: MCD

MEA640005

Blumberg R

REMOTE MAINTENANCE OF THE MSRE (PART OF MSRP SEMIANN PROG
REPT 7/31/64)

Oak Ridge National Laboratory, Tenn.

ORNL-3708 (Nov. 1964), pp. 190-200, 5 fig, 5 ref.

Maintainability was a primary consideration in the design and planning of the MSRE. Components that will become radioactive were designed and located so that they can be disconnected by the use of long-handled tools inserted through a work shield set up on top of the containment cell. Large items will be disconnected this way, then will be removed by a crane operated from a shielded control room. The work shield, tools, remote viewing equipment, and procedures have been developed and tested.

*maintenance + *MSRE + *plans + design + development +
equipment + procedures + shielding + tools

OTHER CATEGORIES: MEB + KBA

MEB660028

Blumberg R

MAINTENANCE OF RADIOACTIVE SYSTEMS AND COMPONENTS AT THE MSRE

Oak Ridge National Laboratory, Tenn.

ANS Trans 9(2) (1966), p. 530.

Maintenance operations are performed at MSRE with long tools manipulated through access holes provided in a portable shield. Experience has been good. (A later, more detailed reference on this subject is MEC700053.)

experience + MSRE + procedures + remote maintenance + tools

OTHER CATEGORIES: KBA

MEB680001

Blumberg R + Hise EC

MSRE DESIGN AND OPERATIONS REPORT, PART X -- MAINTENANCE

Accession Number MDC700005 to MEB680001

Category M
MSRE

MEB680001 *Continued*
EQUIPMENT AND PROCEDURES

Oak Ridge National Laboratory, Tenn.

ORNL-TM-910 (June 1968), 80 p, 26 fig, 3 ref.

A record of the methods developed for maintaining the radioactive portions of the MSRE is presented. The maintenance system utilizes long-handled tools operated through a movable shield for most of the in-cell manipulations. For some radioactive transfer and setup tasks that cannot be handled otherwise, a crane that is operated remotely from a shielded control room is used. Overall descriptions are given of the components and the methods of maintenance. Some detailed procedures, written from the standpoint of the people who perform the work, are also presented. Reference material that will be useful when detailed information is required is included.

*maintenance + *MSRE + *plans + design + development +
equipment + procedures + shielding + tools

MEC70C053

Haubenreich PN + Blumberg R + Richardson M

MAINTENANCE OF THE MOLTEN-SALT REACTOR EXPERIMENT

Oak Ridge National Laboratory, Tenn.

Paper, ANS 1970 Winter Meeting, Washington, Nov. 1970,

29 p, 8 fig, 8 ref.

The MSRE was designed for maintenance of radioactive systems by simple tools inserted through a portable shield. This system proved practical for radioactive maintenance and inspection jobs arising in 5 years of MSRE operation. Delays in the program due to maintenance were not excessive and activity releases and personnel exposures were minimal. This paper, given at a special session on Maintenance of Radioactive Systems describes the MSRE design for maintenance, lists jobs done, and discusses the experience. Copies are available from MSRP Director's Office, ORNL; a summary is in ANS Trans. Vol. 13, No. 2, p. 789.

*design + *experience + *MSRE + *remote maintenance +
*tools + contamination + examinations + equipment +
health physics + maintenance + performance + reliability +
shielding

OTHER CATEGORIES: KBA + MEA

MPX700020

Haubenreich PN + Richardson M

PLANS FOR PCST-OPERATION EXAMINATION OF THE MOLTEN-SALT REACTOR
EXPERIMENT

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2974 (April 1970), 30 p, 0 fig, 3 ref.

In December 1969, after more than 4 successful years, the

Category M
MSRE

MFX70C020 *Continued*

nuclear operation of the MSRE was concluded and the plant was placed in standby. Work planned for early in FY-1971 includes removal of some core graphite; viewing inside the reactor vessel and inside the fuel-pump bowl, inspection of portions of the salt piping, the offgas charcoal bed, the coolant salt pump, and the control rods; and testing the coolant salt flowmeter. Each study is justified by its benefit to the Molten-Salt Reactor Program. Procedures and tools are available for some jobs; for others, they are currently being developed.

decommissioning + examinations + MSRE + plans + reactor maintenance

Category N
Miscellaneous

NXX590002

(Staff Report)

REPORT OF THE FLUID-FUEL REACTORS TASK FORCE

United States Atomic Energy Commission DRDT

TID-8507 (Feb. 1959), 188 p, 7 fig.

A critical evaluation was made of the 3 fluid-fuel concepts under development by the USAEC: aqueous homogeneous, molten-salt, and liquid-metal-fuel. The task force concluded that all 3 could breed in the thorium-U-233 cycle, with the AHR having the greatest potential gain. Maintenance was identified as the most important factor influencing the practicability of any of the three. The molten-salt reactor was judged to have the highest probability of achieving technical feasibility.

*AEC + *development + *plans + *reactors +
breeding performance + fuels + LMR + maintenance + MSER +
optimizations + reviews

OTHER CATEGORIES: AEX

NXX630001

Voznick HF + Uhl VW

MOLTEN SALT FOR HEAT TRANSFER

Atlantic Research Corp + Irexel Institute of Technology

Chem. Engrg. Vol. 70, 129 (May 27, 1963) 8 p, 4 fig, 37 ref.

Heat-transfer salt (HTS) composed of 40% sodium nitrite, 7% sodium nitrate and 53% potassium nitrate has been used widely since 1937 for heating and cooling in the petroleum and chemical industries. HTS is inexpensive, has good heat transfer properties, has a very low vapor pressure and is non-toxic. The freezing point of dry HTS is 290 deg F, and is depressed by water in the salt. HTS is not highly reactive with air, but a blanket of steam or inert gas is recommended. Carbon steel is satisfactory to 850 deg F, with stainless steel recommended for applications to 1100 deg F. HTS is commercially available from several sources and at least two firms manufacture complete heat transfer systems using this salt. Pumps up to 17,000 gpm, valves, and steam generators up to 20 MW are in service.

*coolants + *molten salts + *heat exchangers +
*heat transfer + *heat treatments + *nitrates +
*physical properties + *safety + *secondary salts +
*steam generators + *thermal conductivity +
accidents + applications +
bearings + behavior + compatibility + components +
containers + corrosion + cover gas + density +
experience + failures + flanges + fluids +
freezing + inert gases + liquidus + materials +
melting + NaK + piping + pumps + reviews +
specific heat + stability + thermal insulation +
tubing + valves + vapor pressure + viscosity

NXX69C046

Category N
Miscellaneous

NXX69C046 *Continued*
(Staff Report)

THE USE OF THORIUM IN NUCLEAR POWER REACTORS

United States Atomic Energy Commission DRDT

WASH-1097 (June 1969), 144 p, 28 fig, 65 ref.

This report identifies the factors involved in thorium utilization and describes the status as of mid-1968. It was prepared under the direction of DRDT by a task force from industry, national laboratories and the AEC, and contains a foreword by DRDT Director M. Shaw. The report treats first the general features of the thorium cycle (resources, nuclear characteristics of thorium and U-233 in thermal- and fast-neutron spectra). Then it discusses thorium utilization in specific reactor types: high-temperature gas-cooled, molten-salt, light-water, and heavy-water thermal reactors and a fast reactor. The MSBR considered is a single-fluid breeder with reductive extraction processing for Pa and rare earths. No conclusion as to the relative merits of various reactor types is explicitly presented.

reviews + thorium + uranium-233 + breeding performance +
natural resources + development + MSBR

OTHER CATEGORIES: AEX

NXX69C057

(Staff Report)

COST-BENEFIT ANALYSIS OF THE U.S. BREEDER REACTOR PROGRAM

United States Atomic Energy Commission DRDT

WASH-1126 (Apr. 1969), 98 p, 8 fig.

This report weighs the quantifiable benefits of breeder reactors against the costs incurred by the government in their development. A model of the U.S. electrical power economy was used to compare cases without a breeder and with the LMFBR plus converters. Large benefit/cost ratios for LMFBR development were found in all cases.

Development of a parallel breeder was also considered and appeared desirable under most sets of assumptions. The light-water breeder, the molten-salt breeder and the gas-cooled fast breeder are mentioned as candidates for development.

*economics + *electrical power + *optimizations +

*reactors + AEC + LMFBR + MSBR + power costs

NXX70C011

Bond VP

EVALUATION OF POTENTIAL HAZARDS FROM TRITIUM WATER

Brookhaven National Laboratory, N.Y.

Paper IAEA SM 146/13, IAEA Symposium on Environmental Aspects of Nuclear Power Stations, New York, Aug. 10-14, 1970, 21 p, 44 ref.

This paper analyses possible biological effects of tritium, reviewing from theoretical and experimental standpoints all

Category N
Miscellaneous

NXX700011 *Continued*

factors involved. Factors conceivably increasing toxicity are considered in detail. These include selective concentration (actually discrimination) in the human body, and possible effects due to incorporation into molecules such as DNA. Conclusions are: these factors do not significantly increase the dose expected from tritium in the environment or effects of that dose; a dose of radiation from tritium has the same radiobiological meaning as the same dose of x-rays; the ICRP-AEC max permissible body burden of 1000 microcuries is quite conservative; the MPC for water is conservative by a large factor. Anticipated population exposure from reactor-produced HTO is very small compared to that from existing HTO and other sources of radiation.

*health physics + *tritium + beta decay + concentration + environment + reactors + reviews + safety + wastes

NXX700057

Deonigi DE

A SIMULATION OF THE UNITED STATES POWER ECONOMY

Pacific Northwest Laboratory, Washington

Proc. American Power Conf., Vol. 32 (1970), pp. 105-115,

10 fig, 3 ref.

A comprehensive simulation of a typical U.S. utility system was made and the optimal growth pattern, using fossil-fuel plants and various reactor types, was calculated using projected availability dates, fuel utilization performance, and costs estimated in 1967. In the case where all reactor types were allowed, by the year 2010 over half of the capacity was in molten-salt converter reactors using excess plutonium from fast breeders.

*electrical power + *economics + *natural resources + *reactors + *optimizations + *systems + converters + power costs + simulation

OTHER CATEGORIES: BFX

NXX700058

(Staff Report)

POTENTIAL NUCLEAR POWER GROWTH PATTERNS

United States Atomic Energy Commission DRDT

WASH-1098 (Dec. 1970) 249 p, 33 fig.

This report describes the development and application of a model of the U.S. electrical power economy by the Systems Analysis Task Force, whose activities centered at the Pacific Northwest Laboratories. Input data, including costs appropriate for 1967, were provided by other task forces established by DRDT to evaluate various reactor concepts. Most consideration was given to combinations of a few reactor types considered most likely to be developed in the U. S. In the one case in which the competitor included all reactor types, over half of the reactors built

Category N
Miscellaneous

NXX70C058 *Ccontinued*

after the year 2000 were plutonium-fueled molten-salt converters (Fig. 6.9).

*economics + *electrical power + *optimizations +
*reactors + AEC + capital costs + converters + fuel cycle +
fuel cycle costs + LMFBR + MSER + natural resources +
power costs

NXX700060

(Staff Report)

REPORT OF THE EEI REACTOR ASSESSMENT PANEL

Edison Electric Institute, N.Y.

EEI Publication 70-30 (April 1970), 53 p, 14 fig, 7 ref.

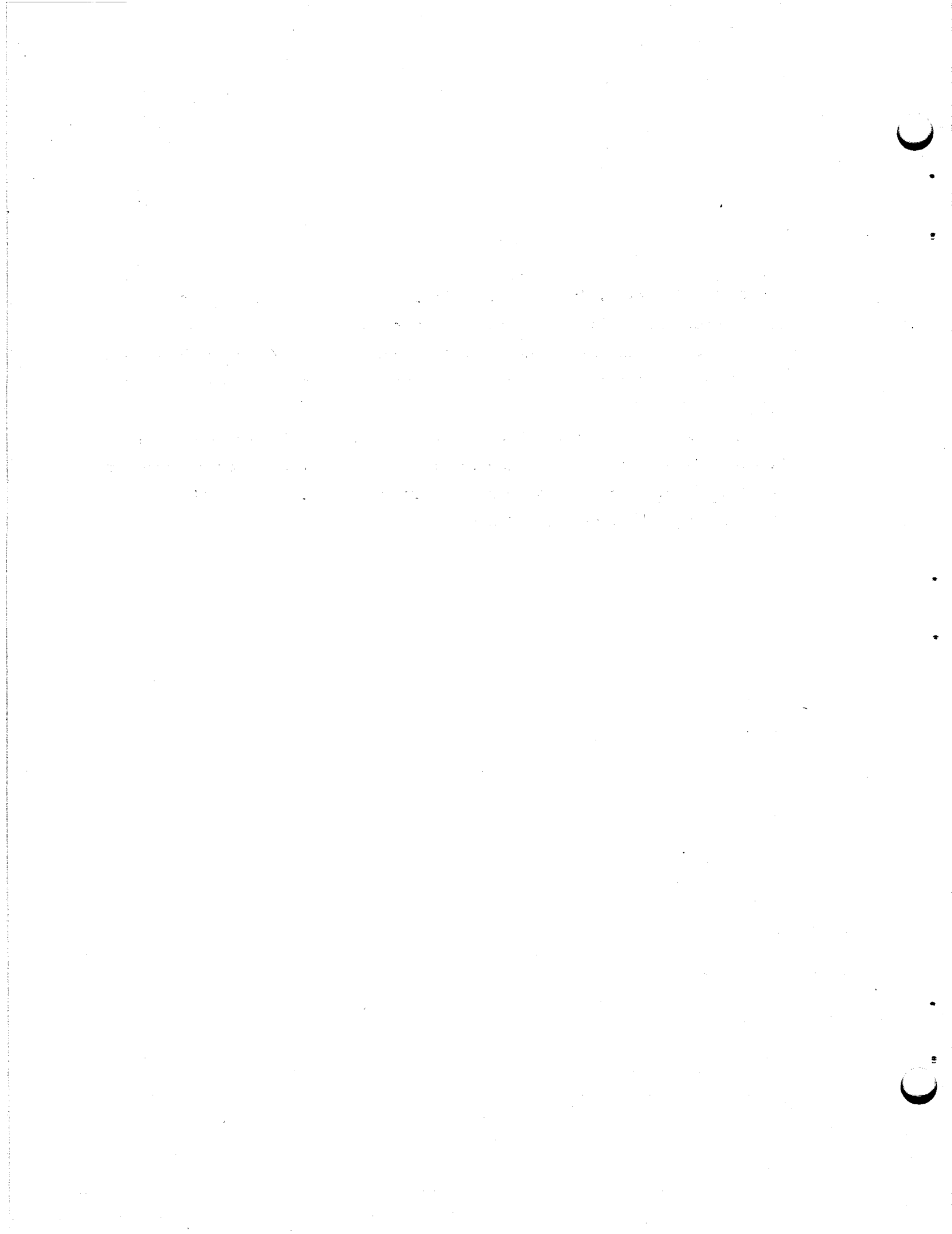
A panel of 6 utility executives (plus a working group) reviewed power reactor developments and suggested the direction, priorities, financial requirements, and timing of utility involvement. Major emphasis on the LMFBR is indicated. With regard to molten-salt reactors, the panel concludes that they promise low costs in the future, but the current lack of a supplier and the small scale of AEC development deter utility involvement.

*development + *electrical power + *reactors + *utilities +
costs + energy + industry + HTGR + LMFBR + LWBR + MSBR +
natural resources + power costs + processing + reviews

KEYWORD INDEX

The reviewers who prepared the abstracts for MSRIS had a list of about 600 keywords from which to select a set for each abstract. In the pages which follow are listed the keywords that were actually used, each followed by the identification numbers of the abstracts to which that keyword was assigned.

The user of this index should be aware of a peculiarity of the listing: all keywords beginning with a capital letter (Hastelloy N, for example) are listed after all the other keywords. This is a consequence of the preparation and print-out of the index by the computer.



The Following Index States The Key Term And Gives References To Each Article Which Was Keyed To It

absorbers		analysis	
MAX65C019		AAK670010	IAB670043
absorption		ACA660008	IBD680036
ACB690029	LGX650002	ACA66C014	JAB690018
ACC65C006	LKY620003	ACA670016	JAB700017
BAX700008		ACA67C023	JCY690019
accidents		ACA680012	MCD690062
AAX670010	MDA660004	ACA68C019	MDA620001
CAX69C053	MDA670038	ACA69C021	MIA640007
IBA710005	MDA690005	ACA690028	MIA650001
LIX670013	NXX630001	ACA7C0021	MDA660003
MDA64C007		ACA700035	MIA660004
actinides		ACB710Q29	MDA670038
ACD660017	ACD700038	ACD660017	MEA670040
ACD670020	CAX680032	ACD670019	MDA670041
ACD670027	CAX690052	ACD67C02C	MDA680003
ACD68C015	CAX710023	ACD670026	MEA690005
ACD680016	CX700049	ACD67C027	MDB700003
ACD69C024	LDA700046	ACD700038	MEC660002
administration		BGX67C045	MDC680002
MCB710012		CDX670035	MEC680005
adsorption		CLX700010	MDC690003
HIX660026	MAD690004	ECX710011	MDC700004
afterheat		FCD710016	MEC700005
BGX670045	IAC710013	HCX71C022	
CAX69C053	IBA710005	analytical chemistry	
CLX700010	MEA670038	ACD65C011	ACD680022
IAC700C47		ACD660011	ACD690024
aging		ACD66C017	ACD690031
FCC71001C		ACD670019	ACD700024
alloy composition		ACD670026	ACD700038
ACE660018	FBE710017	ACD67C027	
ACE67C028	FBE710018	applications	
ACE680017	FCC700040	CAX69C053	NXX630001
ACE68C024	FCC700044	architect-engineering	
ACE690026	FCC710010	ADX69C063	
ACE70C025	FCE690043	argon	
ACE700039	FCE710004	MCD690017	
FAX69C035	FCX690033	barium	
FBC610001	FCX700026	ACD7C0024	
FBC65C017	FXX690047	bearings	
FBE690044	GX680039	ACA670023	ACE680024
alloys		ACA680012	ACE690026
ACE650008	ACE680017	ACB67C024	ACE700025
ACE65C014	ACE680024	ACB680013	NXX630001
ACE66C012	FAX620004	behavior	
ACE660018	FAX620005	ACD66C017	CXX700049
ACE67C028	FAX690035	ACD670026	JAB690018
aluminum		ACD66E0023	JAB700017
ACE66C018	ACE670021	CAX690053	NXX630001
analog systems		CDX670035	
JAB690018	JCY690019	beryllium	
JAB70C017	MCC660005	ACC660016	ACD680022
JAB710008	MDA690005	ACD66C015	ACD690031

beryllium

Continued

ACD700038 MDA690001
LXX660031

beryllium fluoride

ACD660017 CAX690052
ACD67C019 CCX680033
ACD670020 CXX700049
ACD67C026 LAX710019
ACD670027 LCA680008
ACD68C015 LCA690037
ACD680022 LCB680007
ACD69C024 LCB710007
ACD690031 LCC710024
ACD70C038 LDA690012
CAX680032 LDA700046

beryllium oxide

ACD690024 CXX700049

beta decay

NXX70C011

bismuth

ACC66C016 ACE7C0025
ACC670025 ACE700039
ACC690023 CXX700049
ACC690030 GDX710025
ACC70C023 LDA690012
ACC700037 LDA690013
ACD67C019 LDA690038
ACD670026 LDA700014
ACD68C015 LDA7C0015
ACD680022 LDA7C0046
ACD690024 LKX7C0030
ACD690031 LKX710001
ACD700024 LXX700029
ACD700038 LXX710021
ACE69C026 LXX710026
ACE690032

blanket

AAX670004 IAC660025
ACB67C017 IBB670039
ACD67C026 LJX660006
ACD690024 LJX670032
BFX68C009

blowers

ACA66C014 ACA670016

boiling

IAP67C047 IAF670048

boron trifluoride

ACD66C017 ACD690031
ACD670026 ACD700038
ACD680015 CAX690053
ACD66C023 CXX700049
ACD690024

brazing

AAX670006 ACE680024
ACE65C008 ACE690026

ACE660012 ACE690032
ACE66C01E ACE700025
ACE670021 ACE700039
ACE67C02E FBX640015
ACE680017

breeding performance

AAX670009 BEX670012
ABX640004 BFX680009
ABX670049 BFX700016
ABX66C035 BFX700056
ABX690007 EDX680031
ABX69C056 EDX690051
ABX700054 IAA650024
ABX71C02C IAA660030
ACE660009 IAC660025
ACB66C015 IAC700047
ACE670024 IAC710013
ACB66C013 IAF670047
ADX640021 IAF670048
ADX690063 NXX690002
BAX66C006 NXX690046

bubbles

ACB66C02C IAC710013
ACE690022 IED680036
ACB7C0036 MCD690017
ACD680015 MCA700032
ACD66C022 MDB700033
ACD690024 MLC670001
ACD690031 MDC67C002
CCX680033 MDC690016

turnup

ACD680022 BEX670012

calculations

AAX670009 MCA650001
BBX67C012 MDA690001
MDA620001 MCA690002
MDA62C002 MDA700006
MDA630002 MCA700007
MDA6400C1 MDC690016
MDA64C0C2

capital costs

ABX7C0054 IAC710014
BFX700056 LKX620003
IAA66C03C NXX700058

capital equipment

IAC66C025 IAC700051
IAC700047 IAC710013

capsules

ACD650007 ACD680023
ACD660017 ACD690024
ACD67C019 ACD700038
ACD670026 CEX640018
ACD66C015 GDX690042
ACD680022

capture

ACB690029 FAX700008

carbides		ACC7C0037	IAF670047
ACD660C17	FBE710017	CAX690053	IAF670048
ACD680022	FCC700040	chromium	
ACE67C028	FCC700044	ACD670026	CXX700049
ACE680017	FCC710010	ACD6E0022	
ACE68C024		circulation	
carbon		ACD660017	CCX680033
ACD690024	FBA660020	ACD670020	MCA660003
EXX70004E		ACD670027	MDC680002
carbonates		coatings	
CAX690053		ACE680013	ACE680024
carriers		ACE67C02E	ACE690032
CAX680032		ACE680017	ACE700025
casting		cobalt	
FBX64C015		ACE680024	
cells		ccke	
ACB69C022	ACD680016	EXX700048	
ACB690029	ACD680022	columns	
ACD66C017	ACD690024	ACC690030	ACE660018
ACD670019	ACD690031	ACC7C0023	ACE680024
ACD67C026	ACD700038	ACC700037	LEX680027
ACD680015		compatibility	
ceramics		AAX670005	ACE690032
FBX640015	GGX670034	AAX67C0C6	ACE700025
cermets		ACD650007	ACE700039
ACE68C024		ACD660017	BGX670045
cesium		ACD67C019	CAX680032
ACD70C024	LCC710024	ACD670020	CAX690052
charcoal		ACD670026	CAX690053
HIX66C026		ACD670027	CFX640018
chemical properties		ACD680015	CXX700049
ACD670019	ACD700024	ACD6E0016	FAX620004
ACD68C015	ACD7C0038	ACD680022	FAX620005
ACD680022	CAX680032	ACD6E0023	FAX690035
ACD69C024	CAX690053	ACD690024	FAX690045
ACD690031	CXX700049	ACD7C0024	FBD690036
chemical reactions		ACD700038	FEE670031
ACD670019	ACD700024	ACE6500C8	FBE690034
ACD67C026	ACD700038	ACE650014	FEE700027
ACD680015	CAX690053	ACE660012	FEX640015
ACD680022	CXX700049	ACE66001E	FCX690033
ACD68C023	LGX650002	ACE670021	FCX700026
ACD690024	LXX660031	ACE67002E	GDX690042
ACD69C031	LXX7C0029	ACE680017	GFX660023
chemistry		ACE6E0024	NXX630001
AAX67C005	ACD690031	ACE690026	
ABX580001	ACD7C0024	components	
ACD650011	ACD700038	AAX6700C7	HBX700012
ACD6E0011	CAX680032	AAX670008	HIX660026
ACD670019	CAX690052	ABX69C056	HXX640019
ACD67C026	CCX680033	ACA660014	IEB670039
ACD680015	CXX640020	ACA67C023	JDX690060
ACD68C016	CXX7C0049	ACA680012	MEX640003
ACD680022	LXX660031	ACB7C0022	MCA660001
ACD680023	MCA680004	ACE700036	MCD680010
ACD690024		CXX7C0049	MCD700001
chlorides		HEX620006	MLB700003

components		CAX69C053	MDA 670038
Continued		IAC660024	MLA 680003
NXX63C001		contamination	
compressive Properties		JDX67C037	MEC 700053
EBX69C039	EDX640016	control	
EBX700042		AAX6700C8	JAA 710009
computer codes		ACE680020	JAB690018
AAX670009	LKX700030	ACB690022	JAB 700017
ACC69C023	MDA 670040	ACE690029	JAB710008
BBX67C012	MDA 670041	ACE700022	MAC680034
FCD710016	MDA690002	ACB7C0036	MDA 640007
HCX71C022	MDA700006	IAC710013	MLA 690005
IBB670039	MDC700004	IBB710015	
JAA 71C009		control rods	
computers		AAX67C007	MAC 680034
JAB69C018	JEX650020	ACA650010	MCD690017
JAB700017	MAC680034	ACA660008	MDA 640006
JAB710008	MDA690005	ACA690021	MDA 660003
JCX69C019	MDC7C0004	ACA690028	MLA 690005
concentration		ACB6E002C	MDC 670001
ACD66C017	ACD7C0038	FBX640015	MEC670002
ACD670019	CAX680032	GGX670034	MDC 680002
ACD680015	CAX690053	IAC7C0047	MDC 700004
ACD680022	CDX670035	IAC710013	
ACD690024	CXX700049	control-rod drives	
ACD69C031	NXX700011	AAX670007	MLA 690005
ACD700024		MAC6E0034	MDC 7C0004
conceptual design		converters	
ABX670049	HCX710022	ACB7C0022	IAC 660025
ACB66C009	IAA650024	ACE710029	IAD700052
ACB660015	IAA660030	ADX690063	LKX 620003
ACB670017	IAC660024	GXX6E0039	NXX700057
ACB67C024	IAC660025	IAA650024	NXX 700058
ACB680013	IAC700047	coolant loops	
ACB68002C	IAC7C0051	FCE710016	JEX 690060
ACB690022	IAC710013	coolants	
ACB69C029	IAC710014	ACE710029	CAX 690053
ACB700022	IAD700052	ACD67C026	CXX640020
ACB700036	IAE700059	ACD680015	CXX 700049
ACB710029	IAF670047	ACD6E0022	FBD690036
ADX64C021	IAF670048	ACD680023	GFX 660023
ADX670046	IAF690014	ACD69C024	IAC 660025
GDX71C025	LJX660032	ACD690031	IAC700047
construction		ACD7C0024	IAC 710013
ACA 65C004	LHX690011	ACD700038	IAC710014
ACC650006	MEX640003	ACE690026	JCX 690019
ACC650012	MEX7C0002	ACE69C032	NXX630001
contactors		ACE700025	
ACE700039	LKX700030	cooling	
containers		IAC660025	IEA 710005
NXX630001		cores	
containment		AAX670007	ACD670020
AAX670010	IAC700047	ACA710028	ACD 670026
ACA 66C008	IAC710013	ACE6E0013	ACD670027
ACB68C02C	MCA660001	ACE680020	AEX 670046
ACB690029	MCB710012	ACB690022	HAX700050
ACB700036	MDA640007	ACD660017	IAC710013

cores

Continued

IAF670047
IAF670048

corrosion

ABX580001
ACD660017
ACD670019
ACD67C02C
ACD670026
ACD67C027
ACD680015
ACD680022
ACD680023
ACD70C024
ACD700038
ACE65C008
ACE650014
ACE66C012
ACE660018
ACE67C021
ACE670028
ACE68C017
ACE680024
ACE69C026
ACE690032
ACE70C025
CAX680032
CAX69C053

corrosion products

ACC65C006
ACD660017
ACD670019
ACD67C026
ACD670027
ACD68C015
ACD680022
ACD69C024

corrosion protection

ACB67C017
ACC660010
ACC66C016

costs

AAX67C004
AAX670006
AAX670009
AAX67C01C
ABX680035
ABX69C056
ACB660009

cover gas

ACA690028
ACD66C017
ACD67C026
ACD680015
ACD680016
ACD680022

MDA620001

CXX700049
FAX620004
FAX620005
FAX690035
FAX690045
FBD690036
FBE670031
FBE690034
FBE700027
FBE710018
FBX640015
FCD710016
FCX690033
FCX700026
GAX700045
GCX610002
GDX690042
GFX660023
GXX680039
LBX680027
LIX690008
MCD690017
NXX630001

ACD690031
ACD700024
CXX700049
LAX690010
LAX700013
LFX680027
LHX690011

ACC680021
ACE660018
ACC660010
ADX640021
IAA650024
LJX660006
LJX660032
LJX670032
NXX700060

ACD690031
ACD700024
CCX680033
MCD690017
MDA700006
MDC690015

cracks

ACA66C014
ACE650008

creep

ACE650008
ACE650014
ACE660012
ACE66C018
ACE670028
ACE68C017
ACE680024
ACE69C026
ACE690032
ACE7C0025
ACE700039
EBX7C0042
ECX710011
EDX66C031
EDX690051
EXX7C0048
FBB660021
FBB66C022
FBB690040
FBB69C041

criticality

ACA65001C
BEX670012
MCA66C004
MDA640002

cross sections

AAX67C009
BAX680006
BAX7C00C8

crystallization

CAX69C061

cutting tools

ACA710028

data

ACB66C013
ACE690029
ACC7C0037
ACD650011
ACD660011
BAX7C00CE

data acquisition systems

JEX65C02C
MAC680034

data processing

JEX650020
MDC67C0C1

decay

ACE690022
ACB7C0036
ACE660017
ACD67C02C

MDC690016
NXX630001

EDX640016

FEC610001
FBC650017
FEE660019
FBE670029
FEE670030
FBE670031
FEE680025
FBE680026
FEE690034
FBE690044
FEE700027
FBE710017
FEE710018
FBX640015
FCC700040
FCC700044
FCC710010
FCE690043
FCE710004

MDA640006
MDA660003
MDC680002

BBX670012
MAD690004
MCD690017

KBB690006

CAX690061
CCX680038
IBD680036
LEA700014
LDA700015

MDC700004

MIC670002
MDC700004

ACD670027
MDA690001
MEA700007

decommissioning		design criteria	
MPX700020		ACB660015	IAE700059
decomposition		BGX670045	MAC680034
ACD670026	CAX690053	CDX670035	
ACD680015		design data	
decontamination		AAX670009	HCX710022
LIX69C008	LJX660006	ACE670024	IAA650024
LIX690009		ACB710029	IAE700059
defects		BAX6E0006	LDA690012
ACE690026	ACE700025	BBX670012	MLB700003
ACE690032		development	
delayed neutrons		AAX670003	ECX680031
ACD680023	MDA690002	AAX670004	EDX690051
MDA640006	MDA690005	AAX670005	FAX620004
MDA660003	MDC680002	AAX670006	FAX620005
density		AAX670007	FAX690035
ACD660017	ACE700039	AAX670008	FAX690045
ACD680022	CAX690053	AAX670011	FCC710010
ACD690024	CCX680038	ABX5E0001	FXX690047
ACD690031	CYX7C0049	ABX640004	GDY710025
ACE660012	EDX640016	ABX6E0035	HAX700050
ACE660018	EXX700048	ABX690056	HEX620006
ACE670028	FEX640015	ABX7C0054	HBX670042
ACE680017	MDA640002	ABX700055	HEX690058
ACE680024	NXX630001	ABX710020	HBX690059
ACE690032		ACE670024	HEX700012
deposition		ACB6E0013	HCX680037
ACD660017	ACE700025	ACE700022	HFX620007
ACD690024	ACE700039	ACB700036	HIX660026
ACD700038	BGX670045	ACB710029	HXX640019
ACE670028	CDX670035	ACC700037	JLX690060
ACE680017	CYX700049	ACD6E0013	KBB690006
ACE680024	MCD690062	ACE650008	LHX690011
ACE690032		ACE6E0014	LJX670032
description		ACE660012	LKX700030
ACD650013	JFX660027	ACE6E0018	LKY710001
HBX69C05E	MBX640003	ACE670028	LXX700029
HBX690059	MCD700001	ACE680017	MEA640005
IAP690014		ACE6E0024	MEB680001
design		ACE690026	NXX690002
ABX640004	IEB670039	ACE6E0032	NXX690046
ABX700054	IBB710015	ACE700025	NXX700060
ACC6E0006	IBD680036	diagrams	
ACC660010	JAA710009	ACD6E0017	ACD700038
ACC69003C	LHX690011	ACD690024	
ACC700023	LJX660006	diffusion	
HAX70005C	LKX620003	ACE670028	FCD710016
HBX620006	MAC680034	CDX670035	MDA670040
HBX670042	MAD690004	FCC690048	MCA670041
HBX690058	MAX650019	FCC6E0049	
HBX69C059	MCD690055	disconnects	
HBX700012	MCD690062	ACA670016	
HCX680037	MDB700003	dismantling	
IAB670043	MEA640005	ACD670020	ACD670027
IAE700059	MEB680001	dispersion	
IBA710005	MEC700053	LXX710026	

disposal		AAX670010	MLA680003
MDA700007		ACA680012	MDA690002
dissolving		ACA680019	MLA690005
ACD660017	CAX690052	ACE680013	MLA700006
CAX680032	CXX700049	ACB690029	MDA700032
distillation		ACE700022	MLA710003
AAX670004	ACE680024	BFX680009	MDB700033
ACC660010	GDX690042	BGX670045	MLE710002
ACC660016	LCA670014	JAE690018	MLC660002
ACC670018	LCA680008	JAB700017	MDC680005
ACC670025	LCA690037	JAE710008	MDC690003
ACC680014	LCB680007	JCX690019	MDC690016
ACC680021	LCB710007	MCA680004	MDC700004
ACC690023	LCC710024	MCD690017	MDC700005
ACC690030	LJX660006	dynamics tests	
ACC700023	LJX660032	ACA690010	MDB710002
ACD660011	LJX670032	ACA660008	MDC660002
ACD660017	LXX660031	ACA690021	MDC680002
ACE660018	LXX710021	MDA660003	MDC690003
distribution		MDA690005	MDC700005
ACC680014	ACD680022	MDE700033	
ACC680021	ACD690024	earthquakes	
ACC690023	ACD690031	ACB690029	ACB700036
ACC690030	ACD700024	economics	
ACC700023	ACD700038	ABX640004	CAX690053
ACC700037	CDX670035	AEX670049	IAA650024
ACD670026	LDA690012	ABX690007	IAA660030
ACD680015	LDA700046	AEX710020	NXX690057
drain tanks		ADX690063	NXX700057
ACB690029	IAC710013	BFX700056	NXX700058
ACB700036	MAX650019	elasticity	
ACE650008	MDA640002	ACE670028	EDX640016
IAC700047		EBX700041	FEX640015
drying		ECX710011	
ACA650004		electrical circuits	
ductility		MAC680034	MAX650019
ACE650008	FBC650017	electrical conductivity	
ACE650014	FBE650015	ACD680015	ACE660018
ACE660012	FBE650016	ACD680022	ACE670028
ACE660018	FBE660019	ACD690024	ACE680017
ACE670021	FBE670029	ACD690031	ACE680024
ACE670028	FBE670030	ACD700024	ACE700039
ACE680017	FBE680025	ACD700038	FEX640015
ACE680024	FEE680026	ACE660012	GDX680030
ACE690026	FBE690044	electrical power	
ACE690032	FBE710017	ABX650007	NXX690057
ACE700039	FBE710018	AEX700055	NXX700057
EBX700042	FBX640015	ACB660015	NXX700058
FBB660021	FCC700040	ADX690063	NXX700060
FBB660022	FCC700044	electrical properties	
FBB690040	FCC710010	ACD660017	ACD680022
FBB690041	FCE690043	ACD680015	ACD700038
FBB700028	FCE710004	electrolysis	
FBC610001	GAX670033	ACC690023	ACD670019
FBC640017	GAX680028	ACC690030	ACD680015
dynamic characteristics		ACC700023	ACD680022
AAX670009	MDA650001	ACC700037	ACD690024

electrolysis		ACD690031	EDX640016
Continued		ACD7C0038	MEC700053
LKX70003C	LXX710021	ACE650008	MFY700020
LKX710001	LXX710026	ACE65C014	
embrittlement		excursions	
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ACE650014	ACE680024	JAB7C0017	MDA690005
ACE66C012	ACE690026	expansion	
ACE660018	ACE690032	ACB67C024	ACE690032
ACE67C021	ACE700025	ACD680022	ACE700025
emergency cooling		ACE660012	EXX700048
CAX69C053		ACE660018	FEX640015
energy		ACE680024	HGX710022
NXX700060		ACE69C026	
engineered safeguards		experience	
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engineering		ABX690007	MCD690017
LXX700029		ACA65C004	MCD690055
entrainment		ACA65001C	MCD690062
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ACD700038	LCC710024	ACA66C014	MDA660003
CCX680033		ACA670016	MCA670038
environment		ACA67C023	MDA67C041
JDX67C037	NXX700011	ACA680012	MEB700003
equilibrium		ACA680019	MDB710002
ACD660017	ACD700024	ACA69C021	MDC670001
ACD67C026	ACD700038	ACA690028	MFC670002
ACD680015	CXX700049	ACA7C0021	MDC680002
ACD680022	LCA680008	ACA700035	MFC680005
ACD690024	LCA690037	ACA710028	MDC690003
ACD69C031	LDA690012	ACD650011	MFC700005
equipment		ACD660011	MEB660028
ACD67C026	GDX690042	HEX700012	MFC700053
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ACE660012	JFX670036	LCC710024	
ACE66C018	LIX650023	experiment	
ACE670028	LIX670013	ACC7C0037	EBX700041
ACE680017	MEA640005	ACD650011	EEX700042
ACE680024	MEB680001	ACD66C017	IAF690014
ACE69C032	MEC700053	ACD670020	LXX660031
EBX700042		ACD67C026	MAX650019
erosion		ACD670027	MCA660001
FBX640015		ACD66C015	MCA680004
errors		ACD660016	MCA660003
BAX680006	EBX670012	ACD680022	MFC660002
examinations		ACD69C024	MDC680002
ACA700035	ACE660012	ACD690025	MFC680005
ACA710028	ACE660018	ACD7C0038	MDC690003
ACD65C007	ACE670021	CCX680033	MFC690016
ACD660017	ACE670028	CEX64C018	MDC700004
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ACD670027	ACE680024	CXX7C0049	
ACD680015	ACE690026	extraction columns	
ACD680016	ACE690032	GDX710025	LXX710021
ACD680022	ACE700025	LXX700029	LXX710026
ACD690024	ACE700039		

fabrication		ACD670027	LXX700029
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ACE68C017	FBB650018	ACD680022	MAD690004
ACE680024	FBX640015	ACD680023	MCA680004
ACE69C026	GDX710025	ACD690024	MCD680010
ACE690032	GGX670034	ACD69C031	MCD690017
ACE700025	GXX680039	ACD700024	MCD690062
ACE700039	MAD690004	ACD7C0038	MDA670041
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failures		BFX7C0016	MDC67C001
ACA66C014	ACE680024	BGX670045	MDC670002
ACE650008	ACE690026	flanges	
ACE650014	ACE690032	NXX6300C1	
ACE66C012	ACE700025	flexural properties	
ACE660018	CAX690053	EDX64C016	
ACE67C028	NXX630001	flow measurement	
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fast neutrons		ACA710028	MCB700003
IAF670047	IAF690014	HAX7C0050	
IAF67C048		flowsheets	
fatigue		ACB66C0C9	ACC700037
EBX69C039	FBX640015	ACE660015	IAA660030
FBC590001		ACB680013	LIX650023
feedback		ACB69C029	LIX670013
JCX69C019	MDA710003	ACE710029	LJX660032
MDA690005	MDB710002	ACC66C016	LXX700029
ferroalloys		ACC690023	MAX650019
ACE690026	ACE700025	ACC69003C	MCB710012
fertile materials		ACC7C0023	
BAX7CC008		fluid flow	
films		HAX7C0050	MDA620001
CDX67C035		HGX710022	
filters		fluids	
ACA660014	LHX690011	CDX670035	NXX630001
HIX660026		fluorides	
filtration		AAX67C0C5	ACE700039
HIX66C026	LIX690008	ACD650011	CAX680032
LAX690010	LIX690009	ACD66C017	CAX690052
LDA7C0014		ACD670020	CAX690053
fissile materials		ACD67C027	CAX690061
ACD67C026	BAX7C0008	ACD680015	CAX710023
ACD680015	CAX690052	ACD680016	CCX680033
ACD680016	CCX680033	ACD68C022	CCX680038
ACD68C022	CYX700049	ACD690024	CXX640020
ACD700038		ACD69C031	CYX700049
fission		ACD700024	FAX620004
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ACD67C02C	BAX7C0008	ACE650014	FAX690035
fission products		ACE660012	FAX690045
AAX670005	CCX680033	ACE66C018	FBE670031
AAX67C009	CDX670035	ACE670021	FEE690034
ACA660014	CYX700049	ACE67C028	FBE700027
ACB690022	IAC7C0047	ACE680017	FEE710018
ACB700036	IAC710013	ACE68C024	FBX640015
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ACD670020	LCA680008	ACE690032	GCX610002
ACD67C026	LXX660031	ACE700025	LAX690010

fluorides		CXX7C0049	NXX700058
Continued		fuel cycle costs	
LCB710007	LHX690011	AAX67C003	IAC660025
LCC710024		ABX690007	IAC700047
fluorination		ABX7C0054	IAC700051
ACA680019	ADX640021	ACE660015	IAC710013
ACB670017	IAA650024	ACB670024	IAC710014
ACC660010	LBX680027	BFX6E00C9	IAD700052
ACC660016	LIX650023	BFX700016	LJX660006
ACC670018	LIX670013	BFX7C0056	NXX700058
ACC67C025	LIX690008	fuel preparation	
ACC680014	LJX660006	ACD6E0015	LAX690010
ACC68C021	LJX660032	ACD680022	LAX700013
ACC700037	MCD700001	ACD65C024	LIX690009
fluorine		ACD690031	MDA680003
ACA700035	CLX700010	ACD7C0038	
ACD65C007	LGX650002	fuels	
CEX640018	MDA690001	AAX67C004	IAC660025
fluoroborates		ACC670018	IAF690014
AAX670005	ACE670028	ACD670019	LCA670014
ACB68C02C	ACE680017	ACD6E0016	LJX670032
ACB69C022	ACE680024	ACD690024	MCA680004
ACB700036	ACE690026	ACD7C0024	MCD680010
ACD65C011	ACE690032	CAX680032	MCD690017
ACD660017	ACE700025	CAX69C052	MDA630002
ACD67C026	ACE700039	CCX680033	MDA640002
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ACD690031	FCX690033	FAX65C035	MDC670002
ACD7C0024	FCX7C0026	FAX690045	NXX590002
ACD700038	GAX700045	gamma radiation	
ACE660018	HCX710022	ACD660017	ACD670027
ACE670021		ACD67C02C	ACD680023
foaming		gamma sources	
ACD690024	CCX680033	ACD6E0023	
foreign		gamma spectrometry	
ABX700055	ADX670046	ACA67C023	ACD680015
ABX71002C	IAF670047	ACA690028	ACD680022
ADX64C021	IAF670048	ACA700021	ACD690024
forming		ACD6E0017	ACD690031
ACE69C026	ACE700025	ACD670019	ACD700038
ACE690032		ACD67C02C	MCD690017
freeze flanges		ACE670027	MCD690062
ACA680012	ACA690028	gas analysis	
ACA680019	ACA700021	ACD670026	ACD690024
ACA690021		ACD6E0016	
freeze valves		gas flow	
ACA650004	IAC710013	CDX67C035	
HFX62C007	MAX650019	gas injection	
freezing		ACB6E002C	ACB700022
ACD69C024	MDA660004	ACE690022	CCX680033
CAX690061	NXX630001	ACB690029	
MDA640002		gas separation	
fuel cycle		ACB680020	ACB690029
ACB680013	LXX700029	ACB65C022	ACB700022

gas separation

Continued

ACB700036
ACB700036
ACB710029

gases

ACD660017
ACD670026
ACD680016
ACD680023

glass

JFX670036

graphite

AAX670006
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ACB670024
ACB680013
ACB680020
ACB690022
ACB700022
ACB700036
ACD650007
ACD660017
ACD670019
ACD670020
ACD670026
ACD670027
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ACD700038
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ACE650014
ACE660012
ACE660018
ACE670021
ACE670028
ACE680017

hafnium

ACE680024
ACE690032

hardness

ACE700039

health physics

MAC680034
MEC700053

heat

ACB690022

heat balance

ACA710028
MDB700003

heat exchangers

AAX670007

CCX680033
IAC710013

ACD690024
ACD700038
CCX680033
CXX700049

ACE680024
ACE690026
ACE690032
ACE700025
ACE700039
CAX680032
CDX670035
CEX640018
CXX700049
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EBX700041
EBX700042
EBX700043
ECX710011
EDX640016
EDX680031
EDX690051
EXX700048
FBX640015
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FCX700026
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MDB700003

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ACE690022
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ACE660018
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FBB650018
FBX640015

heat generation

IAC710013
LKX710001
LXX710021

heat transfer

ACA660014
ACA670023
ACE660015
ACB680020
ACB690022
ACB690029
ACB700022
ACB700036
ACC690030
ACD680015

heat treatments

ACA660008
ACE650008
ACE650014
ACE660012
ACE660018
ACE670021
ACE670028
ACE680017
ACE680024
ACE690026
ACE690032
ACE700025
ACE700039
EDX640016
EXX700048

heaters

AAX670007

helium

ACD670026
ACE690031
IAC710013

hot cells

ACD670019
ACD670020
ACD670026
ACD670027

hydraulics

AAX670007
HBX620006

HCX710022
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IAC700051
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MEB700003
NXX630001

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IEA710005
JCX690019
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FEE680026
FBX640015
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FCC710010
GCX680030
NXX630001

MCD690017
MIC690015
MIC690016

ACD680015
ACD680022
ACD690031

HEX700012
HCX680037

hydraulics		inert gases	
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MDA700006		ACD670026	ACD700038
hydrocarbons		ACD680016	CCX680033
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ACD670026	HI X660026	ACD690024	MCD690017
hydrodynamics		ACD690025	NXX630001
MDA700006		inspection	
hydrofluorination		AA X670006	ACE670028
ACD680015	LAX690010	ACE650008	ACE680017
ACD680022	LAX700013	ACE650014	ACE680024
ACD690024	LAX710019	ACE6E0012	EDX640016
ACD690031	LIX690009	ACE660018	MEX700002
hydrogen		instrumentation	
ACD680015	LGX650002	AA X670008	JEX650020
ACD680022	LGX650002	ACB680020	JFX670036
ACD690024	LIX650023	IAC710013	MAC680034
ACD690031	LIX670013	JAB690018	MAX650019
ACD700024	LIX690009	JAB700017	MDA640007
LAX690010		JDX670037	MFB700003
hydrogen compounds		JDX690060	
ACC650006	ACD700038	interfacial tension	
ACD680022	LIX650023	ACD6E0015	ACD690031
ACD690024	LIX670013	ACD680022	CCX680033
impregnation		ACD690024	
ACE670028	ACE690032	intrusion	
ACE680017	ACE700025	EDX640016	MAD690004
ACE680024	ACE700039	inventories	
ACE690026		ACD670026	ADX640021
impurities		ACD680023	AIX670046
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ACD690031	ACE690032	iodine	
ACE650008	ACF700025	ACD650011	LXX660031
ACE650014	FED690036	icns	
ACE6E0012		ACD690024	ACD700024
in-pile tests		iron	
ACD650007	ACE660012	ACE700039	GFX660023
ACD650013	ACE660018	GCX610002	
ACD660011	ACE670021	iron alloys	
ACD670019	ACE670028	ACE650014	ACE700025
ACD670027	ACE680017	ACE6E0012	FBD690036
ACD680015	ACE680024	ACE660018	FCX690033
ACD680016	ACE690026	ACE670028	FCX700026
ACD680023	ACE690032	ACE690026	
ACD690024	ACE700025	irradiation	
ACD690025	CEX640018	ACD670026	ACE690032
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ACE650014		ACD700038	ACE700039
inconels		ACE650008	CEX640018
ABX580001	GAX700045	ACE650014	ELX640016
ADX640021	GCX610002	ACE660012	EDX680031
FBX640015	GCX680030	ACE6E0018	EDX690051
FCD710016	GGX670034	ACE670021	EXX700048
industrial studies		ACE670028	FBB690040
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industry		ACE680024	FBB700028
ADX690063	NXX700060	ACE690026	FBE650015

irradiation		ACD670019	CXX700049
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FBE670031	FCE710004	liquidus	
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FBE69C034	IAE700059	ACD680022	CAX690053
FBE690034	IAF690014	ACD690024	CXX700049
FBE690044	MAD690004	ACD700024	NXX630001
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isotopes		ACC660010	LCA690012
BAX700008	MDB700003	ACC66C016	MDA690001
MDA700007		ACC67C025	
junctions		lithium chloride	
ACB680013	ACE670028	ACD7C0024	CAX690053
ACE65C008	ACE680017	lithium fluoride	
ACE650014	ACE680024	ACD66C017	CCX680033
ACE66C012	FBX640015	ACD670019	CXX700049
ACE660018		ACD67C02C	LAX690010
kinetic equations		ACD670027	LAX700013
ACD690031	MDA710003	ACD66C015	LAX710019
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krypton		ACD690024	LCA690037
ACA650004	MDA670040	ACD69C031	LCB680007
ACB69C022	MDA670041	ACD700024	LCB710007
laboratory equipment		ACD7C0038	LCC710024
ACD680016	ACD690024	CAX680032	LCA690012
lattice		CAX690052	LDA700046
EXX700048		CAX690053	LIX690009
layout		loading	
ACB660009	ACB690029	ACA65C0C4	MCA680004
ACB68C013	ACB700036	ACA650010	MTA680003
lead		LAX710019	
ACE65C014	GFX660023	loop	
ACE660012	IAC660025	ACE65C014	FAX620005
ACE66C01E		ACE660012	FAX690035
lead cooling		ACE66C018	FAX690045
ACB660015	GXX680039	ACE670021	FEX640015
ADX670046		ACE670028	FCD710016
leakage		ACE68C017	GAX700045
CAX69C053		ACE680024	GCX610002
leaks		ACE7CC039	IAC710014
ACA700021	ACD670027	FAX620004	
ACD670020	CAX690053	losses	
limits		LIX690008	
IAB670043		machining	
linings		FBX640015	
GFX660023		maintenance	
liquid level measurement		AAX670011	ACA690028
GAX700045	LXX710026	ACA66C014	ACB680013
LCB71C007	MAX650019	ACA67C023	ACB680020
LCC71C024		ACA680012	ACB690022
liquid metals		ACA66C019	ACB690029
ACD66C011	CAX690053	ACA690021	ACE700022

maintenance

Continued

ACB700036
 HBX700012
 HXX640019
 IAC700051
 IAC710013
 JFX67C03E
 KBB690006
 MAX65C019

manipulators

JFX67C036

mass transfer

ACB670024
 ACB68002C
 ACB690022
 ACB69C029
 ACB700022
 ACB70C03E
 ACE650014
 ACE66C012
 ACE660018
 ACE67C02E
 ACE680017

materials

AAX670006
 ABX700054
 ACD670019
 ACD67C02E
 ACD680015
 ACD68C01E
 ACD680022
 ACD680023
 ACD690024
 ACD69C031
 ACD700024
 ACD70003E
 ACE650008
 ACE650014
 ACE660012

materials testing

ACB680020
 ACD67C019
 ACD67C02C
 ACD670026
 ACD67C027
 ACD680023
 ACE65C00E
 ACE650014

mathematics

FCD710016
 JCX69C019

measurement

AAX670008
 ACA65C01C
 ACD700024
 ACD70C03E

MEX700002
 MCB710012
 MCD680010
 MCD700001
 MEA640005
 MEB680001
 MEC700053
 NXX590002

ACE680024
 ACE690026
 ACE690032
 ACE700025
 CXX700049
 FBD690036
 FCD710016
 GFX660023
 IBD680036
 MDA670040
 MDA670041

ACE660018
 ACE670028
 ACE680017
 ACE680024
 ACE690026
 ACE690032
 ACE700025
 CDX670035
 CEX640018
 GDX710025
 HIX660026
 IAC700051
 IAD700052
 MCA680004
 NXX630001

ACE660012
 ACE660018
 ACE670028
 ACE680017
 ACE680024
 IAF700059
 IAF690014

MDA690005

CDX670035
 LCA680008
 MDA700006
 MDB700033

ACE66C018

ACE670028

ACE68C017

ACE680024

CAX65C061

mechanical properties

AAX670006

ACE650008

ACE65C014

ACE660012

ACE660018

ACE670021

ACE670028

ACE68C017

ACE680024

melting

ACD690024

ACD69C031

CXX700049

mercury

IAF670047

metal transfer process

ACC700023

ACC7C0037

ACD690024

metallography

ACE650008

ACE65C014

ACE660012

ACE66C018

ACE670021

ACE67C02E

ACE680017

ACE68C024

metallurgy

ACE65C008

ACE650014

ACE660012

ACE660018

ACE680017

metals

ACD67C019

ACE65C008

ACE650014

ACE660012

ACE660018

ACE67C02E

methods

AAX67C009

MDA620002

MDA65C002

microstructure

ACE650008

ACE65C014

ACE660012

ACE690026

ACE690032

MDC660002

MIC690015

MDC690016

MIC700004

ACE690026

ACE690032

ACE700025

EIX680031

EDX690051

FEA660020

FCX690033

FCX700026

GXX680039

FEX640015

NXX630001

IAF670048

ACD690031

ACD700024

ACE690026

ACE690032

ACE700025

FBA660020

FEC640017

FBX640015

GCX610002

ACE680024

ACE690026

ACE690032

ACE700025

ACE680017

ACE680024

ACE690026

ACE690032

ACE700025

MDC660002

MIC700004

MDC700005

ACE700025

ACE700039

EEX700042

EDX640016

EXX700048

microstructure

Continued

FBA660020	FBE710018
FBA680029	FBX640015
FBC590001	FCC700040
FBE660019	FCC700044
FBE67C029	FCC710010
FBE690034	FCE710004
FBE700027	GAX670033
FBE710017	GAX680028

mists

ACD670026	ACD700038
ACD68C015	CCX680033
ACD680022	CXX700049
ACD680023	HIX660026
ACD69C024	

mixing

MDA690002

mixtures

CAX690052	CAX690053
-----------	-----------

models

FCD710016	MDA690005
FCD710016	MDA700006
HAX700050	MDC670001
IBD68C036	MDC670002
JCX690019	MDC680002
MDA620002	MDC680005
MDA650001	MDC690003
MDA66C003	MDC690016
MDA670040	MDC700005
MDA69C002	

moderators

BFX68C009	CDX670035
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modified Hastelloy N

AAX67C006	FBE700027
ACE650014	FBE710018
ACE660018	FCC690048
ACE670028	FCC690049
ACE680017	FCC700040
ACE680024	FCC700044
ACE690026	FCC710010
ACE69C032	FCE690043
ACE700025	FCE710004
ACE700039	FCX690033
FBE690034	FCX700026
FBE69C044	FXX690047

modular design

ACB660015	ACB670024
ACB670017	

molten salts

AAX670005	FAX620005
ABX58C001	FAX690035
ACA65C004	FBD690036
ACA650010	FEE710018
ACD65C007	FBX640015
ACD650011	FCX700026
ACD66C011	GAX700045

ACD660017
ACD670020
ACD670026
ACD670027
ACD680015
ACD680016
ACD680022
ACD680023
ACD690024
ACD69C025
ACD690031
ACD700024
ACD700038
ACE650014
ACE660012
ACE660018
ACE67C021
ACE670028
ACE680017
ACE68C024
ACE690026
ACE690032
ACE700025
ACE700039
ADX640021
BGX670045
CAX710023
CCX68C033
CEX640018
CLX700010
CXX640020
EBX700043
FAX620004

GCX610002
GEX690042
GDX710025
GXX680039
HEX670042
HBX690058
HEX690059
HBX700012
IAC660024
IAC700047
IAF670047
IAF670048
IAF690014
IEB670039
LAX710019
LEX680027
LCA670014
LCB680007
LCB710007
LCC710024
LCA690013
LGX650002
LJX660032
LKX620003
LXX660031
LXX700029
LXX710021
LXX710026
MAX650019
MDA630002
MLA670041
MDB700003
NXX630001

nclytdenum

ACD670019	ACE690032
ACD670026	ACE700025
ACD670027	ACE700039
ACE700038	CCX680033
ACE68C008	CEX640018
ACE660012	GAX670033
ACE66C018	GAX680028
ACE670028	GCX610002
ACE680017	GDX690042
ACE680024	GEX710025
ACE69C026	

monitors

JDX67C037	MAC680034
JDX690060	

natural resources

AAX670003	NXX690046
ABX67C049	NXX700057
ABX690007	NXX700058
ADX690063	NXX700060
BFX700056	

neptunium

BAX700008

neutron fluence		AAX670009	MDA 64 0002
ACB670024	IAF690014	BAX680006	MCA 640006
neutron flux		BBX670012	MDA 650001
ACB670024	MDA640001	BFX680009	MCA 66 0004
IAC700047	MDA640006	BFX700016	MDA 690002
IAC710013	MDC690015	IAF690014	MDA 700006
MDA 62C001		MCA680004	MIC 670001
neutron physics		MCD690017	MDC 670002
ACB680013	BFX680009	MDA620001	MIC 690016
ACB680020	IAC660024	MDA620002	MDC 700004
ACB690022	IAC660025	MDA640001	
ACB690029	IAC700047	off-gas systems	
ACB700022	IAF690014	AAX670010	ACD 680023
ACB700036	MDA620002	ACA660008	ACD 690024
BAX700008		ACA660014	ACD 700038
neutron sources		ACA670016	CCX 680033
IAF690014	MDA640001	ACA670023	HIX 660026
MDA 63C002	MDA690001	ACA680012	IAC 660025
neutron spectra		ACA680019	IAC 700047
ADX640021	IAF690014	ACA690021	IAC 700051
neutron yield		ACA690028	IAC 710013
AAX670009	BAX700008	ACB670017	JDX 670037
nickel		ACD660011	JLX 690060
ACD670019	FCC700040	ACD670026	MAC 680034
ACE700039	FCC700044	ACD680015	MAX 650019
ADX640021	GCX610002	ACD680022	MCD 690017
FAX620004	GCX680030	operating costs	
FAX620005	LDA700015	LKX620003	
nickel alloys		operation	
ACD670019	GGX670034	ACA650010	LIX 690008
FAX690035		ACA660008	MCA 660001
niobium		ACA660014	MCA 680004
ACE700039	GXX680039	ACA670016	MCB 650021
GCX610002		ACA670023	MCB 650022
nitrates		ACA680012	MCB 660029
NXX630001		ACA680019	MCB 690054
nitrogen		ACA690021	MCB 710012
ACE650008	FEE690034	ACA690028	MCD 680010
ACE660012	FEE700027	ACA700021	MCD 690017
noble metals		ACC650012	MCD 690055
ACB690022	ACD690031	IAB670043	MCD 690062
ACD660017	ACD700024	JEX650020	MCD 700001
ACD670019	ACD700038	LCC710024	MLB 700003
ACD670020	EFX700016	LIX650023	MDC 670001
ACD670027	CCX680033	LIX670013	MIC 670002
ACD680015	CXX700049	operators	
ACD680016	IAC700047	ACA650010	MCC 660005
ACD680022	IAC710013	MCB650022	MCC 670044
ACD680023	IBA710005	optics	
ACD690024	MCD690017	JFX670036	
noise analysis		optimizations	
ACA680012	MDB700033	BFX700056	NXX 700057
ACA690021	MDB710002	NXX590002	NXX 700058
ACA690028	MDC680005	NXX690057	
ACA700021	MDC690015	oxidation	
MDA 70C032		ACD680015	ACD 690024
nuclear analysis		ACD680022	ACD 700038

precipitation

Continued

FCC700044

pressure

MDC690003

primary salt

ACB680020

ACB690029

ACD700038

CAX680032

primary system

MCA660001

procedures

MBX700002

MCA660001

MCB650021

MCB650022

MCB660029

MCB690054

processing

AAX670004

AAX670010

ABX690007

ABX700054

ACB660009

ACB660015

ACB670017

ACC650006

ACC650012

ACC660010

ACC660016

ACC670018

ACC670025

ACC680014

ACC680021

ACC690023

ACC690030

ACC700023

ACC700037

ACD680015

ACD680022

ACD690024

ACD700038

ACE690026

BFX700016

CAX680032

CAX690061

procurement

ACE660012

ACE660018

ACE670028

ACE680017

ACE680024

ACE690026

production

LAX710019

progress report

CLX700010

CXX700049

HCX710022

MDA700006

MCB710012

MCC670044

MDC660002

MEA640005

MEB660028

MEB680001

GDY690042

IAA660030

IAC660024

IAC700051

IAC710013

LCA680008

LCB710007

LCC710024

LIX650023

LIX670013

LIX690008

LIX690009

LJX660006

LJX660032

LJX670032

LKX620003

LKX700030

LKX710001

LXX660031

LXX700029

LXX700029

LXX710021

LXX710026

MAX650019

MDA700007

NXX700060

ACE690032

ACE700025

ACE700039

FDX640016

FBX640015

LAX710019

ACE660015

ACB670017

ACE670024

ACB660013

ACE680020

ACB690022

ACE690029

ACE700022

ACB700036

ACE710029

ACD650007

ACD650011

ACD660017

ACD670019

ACD670020

ACD670027

ACD680015

ACD660022

ACD690025

ACD700024

ACD700038

ACE650008

ACE650014

ACE660012

ACE660018

ACE670021

ACE670028

protactinium

AAX670004

ACE660015

ACC660014

ACC680021

ACC690023

ACC690030

ACD650011

ACD660011

ACD660017

ACB670019

ACD670026

ACB680015

ACD660022

ACD690024

protactinium fluclides

ACD660017

ACD670019

ACB670026

ACD660015

ACD680022

ACD690024

prototypes

ADX690063

HBX620006

pumps

AAX670007

ACA660008

ACA710028

ACB670017

ACE680017

ACE680024

ACE690026

ACE690032

ACE700025

ACE700039

ACX640008

ACX640014

ACX650003

ACX650009

ACX660007

ACX660013

ACX670015

ACX670022

ACX680011

ACX680018

ACX690020

ACX690027

ACX700018

ACX700034

ACX710027

EDX640016

FBX640015

FCX690033

FCX700026

MEX640003

ACD690031

ACD700038

BFX700016

CXX700049

IAA660030

IAC660025

LBX680027

LCA690013

LJX670032

LKX620003

LKX710001

LXX660031

LXX710021

ACD690024

ACD690031

ACD700038

CXX700049

LLA700046

HFX620007

HXX640019

ACB670024

ACB680013

ACB680020

ACB690022

pumps

Continued

ACB690029 IAC660025
 ACB700022 IAC700047
 ACB700036 IAC700051
 CCX680033 IAC710013
 HBX620006 IAC710014
 HBX670042 MAX650019
 HBX690058 MCD690017
 HBX690059 NXX630001
 HBX700012

pyrocarbon

ACE700025 MAD690004
 EXX700048

quality assurance

HBX690059 MBX700002

radiation damage

ACB670024 ACE680017
 ACD670020 ACE680024
 ACD670027 ACE690026
 ACD680023 ACE690032
 ACE650008 ACE700025
 ACE650014 CXX700049
 ACE660012 ECX710011
 ACE660018 IAB670043
 ACE670028 MDA680003

radiation heating

ACB690029 IEA710005

radiation measurement

AAX670008 MAC680034
 JDX670037

radioactivity

MDA700007

radiolysis

ACA660014 ACD680023
 ACA700035 CEX640018
 ACD650007 CLX700010
 ACD670026

rare earths

AAX670009 EFX700016
 ACC660010 CAX690052
 ACC670018 CDX670035
 ACC670025 CXX700049
 ACC680014 FBX640015
 ACC690030 GGX670034
 ACC700023 LCA670014
 ACD660011 LCA680008
 ACD660017 LCA690037
 ACD670019 LCB680007
 ACD670026 LCB710007
 ACD670027 LCC710024
 ACD680015 LDA690012
 ACD680022 LDA700046
 ACD690024 LKX700030
 ACD690031 LKX710001
 ACD700024 LXX660031
 ACD700038 MDC670001

ACD700038

MIC670002

rare gases

ACD660017 ACD690031
 ACD670026 BFX700016
 ACD680015 CXX700049
 ACD680022 MCD680010
 ACD680023 MCD690017
 ACD690024 MDA700007

reaction rates

ACD680015 ACD700024
 ACD680022 ACD700038
 ACD690024 CLX700010
 ACD690031 CXX700049

reactivity

AAX670010 MDA620001
 ACA650010 MDA620002
 ACA670016 MDA640006
 ACA670023 MDA660003
 ACA670023 MDA680003
 ACA680012 MDA690002
 ACA690021 MDA690005
 ACA700021 MDA700006
 BAX680006 MIC660002
 BBX670012 MDC670001
 BFX680009 MDC670002
 BGX670045 MDC680002
 MCA680004 MDC680005
 MCD680010 MDC690003
 MCD690017 MIC700004
 MCD700001 MIC700005

reactor vessel

ACE670017 ACB690022
 ACB680013 HAX700050
 ACB680020 IAB670043

reactors

ABX700055 IAE700059
 ACE680020 IAF670047
 ACB690022 IAF670048
 ACE700036 IAF690014
 ADX640021 KBB690006
 ADX670046 LKX620003
 BAX680006 MAX650019
 BFX700056 MDB700003
 CAX680032 NXX590002
 IAC680024 NXX690057
 IAC660025 NXX700011
 IAC700047 NXX700057
 IAC700051 NXX700058
 IAC710013 NXX700060

recombination

CEX640018 CLX700010

reduction

ACD660011 ACD690031
 ACD670026 ACD700038
 ACD680015 LAX700013
 ACD680022 LGX650002
 ACD690024 LIX690009

reductive extraction process		rupture	
ACC660016	ACD690031	ACE650008	ACE680024
ACC670025	ACD700024	ACE650014	ACE690026
ACC680014	ACD700038	ACE660012	ACE690032
ACC680021	GDX710025	ACE660018	ACE700025
ACC690023	LDA690012	ACE670028	EBX700041
ACC690030	LDA690038	ACE680017	FEE690034
ACC700023	LKX700030		
ACC700037	LKX710001	safety	
ACD670019	LXX660031	AAX670010	MCB650021
ACD670026	LXX700029	ABX660035	MCB690054
ACD680015	LXX710021	AEX690056	MCA640001
ACD680022	LXX710026	ABX700054	MDA640007
ACD690024		BGX670045	MDA670038
		CAX690053	MDA680003
reliability		LIX650023	MDA690005
ACE660018	ACE690032	LIX670013	NXX630001
ACE670028	MCD700001	MAC660034	NXX700011
ACE680017	MEC700053		
ACE680024		safety limits	
		MCB650054	
remote maintenance		samplers	
ACA660008	JFX660027	ACA660014	ACA680012
ACA660014	JFX670036	ACA670023	ACA680019
ACA670016	MCD690062		
ACA670023	MEB660028	sampling	
ACA710028	MEC700053	ACD660017	ACD690024
ACE690032	MFY700020	ACD670019	ACD690031
ACE700025		ACD670020	ACD700024
		ACD670026	ACD700038
remote welding		ACD670027	CXX700049
AAX670011	ACE680024	ACD680015	JLX670037
ACE680017	KEE690006	ACD680016	MAX650019
		ACD660022	
replacement		sealing	
ACB670017	CAX690052	ACE650008	ACE690032
CAX680032		ACE670028	ACE700025
		ACE660017	PCX690033
research		ACE660024	PCX700026
AAX670005	ACD680022	ACE690026	
ACD660017	ACD690031	secondary salts	
ACD670020	ACD700024	ACD670019	HGX710022
ACD670027	ACE690032	ACD650024	JCX690019
ACD680015		ACC700024	NXX630001
		CAX690053	
reviews		secondary systems	
AAX670003	EDX640016	GFX660023	MCA660001
AAX670005	EDX680031	JCX650019	
AAX670006	EDX690051		
AAX670007	FBX640015	separations	
ABX580001	FCX690033	CAX660032	LCB710007
ABX640004	FCX700026	CAX690061	LCC710024
ABX670049	FXX690047	CXX700049	LXX660031
ABX680035	HBX670042	LCA690037	LXX700029
ABX690007	IED680036		
ABX690056	MDA670038	shielding	
ABX700054	MDE700003	MDA640006	MEB680001
ABX700055	NXX590002	MDA700007	MEC700053
ABX710020	NXX630001	MEA640005	
ACX640014	NXX690046		
BBX670012	NXX700011	shrinkage	
CXX640020	NXX700060	ACE670024	

simulation		CCX66C033	
JAB69C018	MDA660004	stability	
JAB700017	MDA690005	AAX670009	MCA680004
JAB71C008	MDA710003	AAX670010	MCA650001
JCX690019	NXX700057	ACA66C008	MDA690002
MCC66C005		ACD670026	MCA690005
single-fluid reactors		BGX67C045	MDA710003
ACB68C013	BFX700016	CAX680032	MLB710002
ACB680020	CXX700049	CAX690053	MDC660002
ACB690029	HCX710022	JAB69C018	MDC700005
BFX68C009		JAB700017	NXX630001
sites		JCX69C019	
IAC71C013		stack	
sodium		JDX67C037	
FCD71C016		stainless steels	
sodium fluoride		ACE65C014	ACE700025
ACC65C006	ACD700024	ACE660012	ACE700039
ACC660010	ACD700038	ACE660018	GAX670033
ACD680015	ADX640021	ACE67C021	GAX680028
ACD68C022	CAX690053	ACE670028	GAX700045
ACD680023	LIX690009	ACE68C017	GCX610002
ACD690024	LJX660006	ACE680024	GFX660023
ACD690031	LKX620003	ACE69C026	
solidus		startup	
ACD670026	ACD700024	ACA65C004	MCC660005
ACD68C022	CAX690052	ACA65C01C	MCD690055
ACD69C024	CAX690053	ACA660008	MEA660003
ACD690031	CXX700049	ACA66C014	MDA690005
solubility		MCE650022	MDC680002
ACC690023	CAX690052	steam cycle	
ACC69C03C	CAX710023	ACE660015	IEB670039
ACD670026	CCX680038	HGX710022	
ACD680015	CXX700049	steam generators	
ACD680022	FED690036	ACB68C02C	CAX690053
ACD69C024	LDA690013	ACE690022	HGX710022
ACD690031	LDA700014	ACB69C029	IAC660025
ACD700024	LDA700015	ACE700022	IAC700047
ACD700038		ACB71C029	IAC710013
specific heat		ACE690032	IEB670039
ACB680020	CCX680038	ACE7C0025	IBB710015
ACD66C017	EDX640016	ACE7C0039	NXX630001
ACD68C015	FBX640015	steam systems	
ACD680022	MDB700003	AAX67C007	IAC700051
ACD7C0024	NXX630001	ACE660015	IAC710013
CAX690053		ACB69C022	IAC710014
specific inventory		IAC660025	IEB710015
BFX680009	EFX700016	IAC7C0047	JCX690019
specifications		storage	
ACE660012	FBX640015	CLX7C001C	MDA640002
EDX64C016	GGX670034	MCE710012	MCA700007
spectrophotometry		stress	
ACD670026	ACD690031	ACA66C008	ACE690032
ACD68C022	ACD700024	ACE670024	ACE700025
ACD690024	ACD700038	ACD67C02C	ECX710011
spheres		ACD670027	FEB700031
ACB680020	ACB700022	ACE69C026	HGX710022
sprays			

stress rupture		FBC61C0C1	GAX670033
ACD67C02C	ACE660018	FBC640017	GAX680028
ACD67C027	ACE670028	FBC650017	
ACE650008	ACE680017	test facilities	
ACE650014	ACE680024	ACE680020	ACE690026
ACE660012	IAB670043	ACB690022	ACE690032
structures		ACE690029	ACE700025
ACB680020	IAC660024	ACB7C0022	GDX710025
ACB69C022	IAC700047	ACE7C0036	HEX690058
ACB69C029	IAC710013	ACD680015	HEX690059
ACB700036		ACD6E0022	IAE700059
surface tension		ACD700038	
ACD690024	CCX680033	testing	
ACD690031	CCX680038	ACA650004	CDX670035
ACD700024		ACA65C01C	EBX690039
surveillance		ACA660008	EEX700041
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ACD67C026	ACE690026	ACD67C026	FEX640015
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technology		MDC69C0C3	
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tensile properties		CAX690053	
ACE660012	FBE650015	thermal convection	
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ACE70C039	FBE670031	ACD680016	GCX610002
EEX690039	FBE680026	ACE65C014	GFX660023
EBX700042	FBE690034	thermal effects	
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MAX650019		ACC670018	EAX700008
thermal shock		ACC680014	LBX680027
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thorium		ACD660023	LXX710021
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tungsten		MDA650001	MDC690003
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viewing devices		ACB670017	IBD680036
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viscosity		ACE680013	MDA670041
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Hastelloy N

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HTGR

JDX690060

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LMR

NXX590002

LWBR

NXX700060

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MSRE

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NXX630001

CATEGORY INDEX

The category structure used in the MSRIS appears in outline form on the next three pages, followed by the index.

The significance of the letter X, which does not appear in the outline but does appear in category designations elsewhere, requires some explanation.

There are two meanings or uses of the letter X. One is simply as a "filler." For various reasons associated with the computer, every category designation, whether it be the primary category in the identification number or an "other category" in the last line of the entry, must have exactly 3 letters. Thus for category N, which is not subdivided, every abstract is designated NXX. Similarly, all abstracts falling into second-order categories which are not subdivided have X as the last letter in their category designation (ABX700054, for example).

The other use of X is to denote a general or broad treatment. This is the meaning if X appears in place of the letters for established second- or third-order categories. To illustrate, MAX690019 is a comprehensive description of the MSRE design (category MA) which discusses general considerations and embraces information in third-order categories MAA, MAB, MAC, and MAD.

SUBJECT CATEGORIES IN MSRIS

A Molten-Salt Reactor Programs
 AA MSRP - Plans and Organizations
 AB MSRP - Technical Summaries
 AC MSRP - Progress Reports
 ACA MSRE
 ACB Large MSR's
 ACC Salt Processing
 ACD Chemistry
 ACE Materials
 AD MSR Activities Outside MSRP

B Reactor Analysis
 BA Nuclear Data
 BB Static Neutronics
 BC Dynamics
 BD Thermal Effects
 BE Activation, Radiation and Shielding
 BF Fuel Cycle and Economics
 BG Safety
 BH Computer Programs

C Reactor Chemistry
 CA Phase Relations
 CB Thermodynamics and Equilibria
 CC Physical Properties
 CD Rates and Diffusion
 CE Corrosion Reactions
 CF Fission Product Behavior
 CG Tritium Behavior
 CH Oxide Behavior
 CI Crystal Studies
 CJ Surface Effects
 CK Electrochemistry
 CL Radiolysis

D Analytical Chemistry

E Graphite
 EA Fabrication
 EB Unirradiated Properties
 EC Irradiation Effects
 ED Applications

F Hastelloy N and Related Alloys
 FA Alloys Leading to Hastelloy N
 FB Standard Hastelloy N
 FBA Microstructure
 FBB Fabrication
 FBC Mechanical and Physical Properties
 FBD Corrosion
 FBE Radiation Damage
 FC Modified Hastelloy N
 FCA Microstructure
 FCB Fabrication
 FCC Mechanical and Physical Properties
 FCD Corrosion
 FCE Radiation Damage

G Materials Other Than Hastelloy N and Graphite
 GA Stainless Steels
 GB Steels other than Stainless
 GC Nickel and Ni-Base Alloys other than Hastelloy N
 GD Molybdenum and Mo-Base Alloys
 GE Brazing Alloys
 GF Other Metals
 GG Nuclear Control Materials

Subject Categories in MSRIS
(continued)

H Reactor Component Development

- HA Core
- HB Pumps
- HC Heat Exchangers
- HD Steam Generators
- HE Gas Injection and Removal
- HF Valves
 - HFA Freeze Valves
 - HFB Mechanical Valves
- HG Flanges
- HH Heaters
- HI Other Components

I Reactor Design

- IA Reactor Plant
 - IAA Early Molten-Salt Reactors
 - IAB MSRE
 - IAC One-Fluid MSBR (Reference Design)
 - IAD Other Thermal Molten-Salt Reactors
 - IAE MSBE
 - IAF Fast and Epithermal Molten-Salt Reactors

IB Systems

- IBA Fuel
 - IBB Coolant
 - IBC Steam
 - IBD Gas
 - IBE Containment
-

J Instrumentation and Controls

- JA General
 - JAA Instrument Development
 - JAB Plant Control
- JB Nuclear Control and Plant Safety
- JC Process
- JD Radiation and Contamination Monitoring
- JE Data Collection and Analysis
- JF Communication and Surveillance
- JG Electrical and Pneumatic Systems

K Operation and Maintenance

- KA Operation
 - KAA ARE
 - KAB MSRE
 - KAC Other Molten-Salt Systems
 - KB Maintenance
 - KBA MSRE Maintenance
 - KBB Other Molten-Salt and Radioactive Systems
-

Subject Categories in MSRIS
(continued)

L Fuel Preparation and Processing

- LA Salt Procurement and Preparation
 - LB Fluorination
 - LC Distillation
 - LCA Experimental Basis
 - LCB Engineering Development
 - LCC Operating Experience
 - LD Reductive Extraction
 - LDA Experimental Basis
 - LDB Engineering Development
 - LE Metal Transfer
 - LEA Experimental Basis
 - LEB Engineering Development
 - LF Oxide Precipitation
 - LFA Experimental Basis
 - LFB Engineering Development
 - LG Adsorption and Reduction
 - LH Salt Purification
 - LI MSRE Salt Processing
 - LJ Plants for Two-Fluid MSBR
 - LK Plants for One-Fluid MSBR
-

M MSRE

- MA Design
 - MAA Plant
 - MAB Major Component
 - MAC Instrumentation and Controls
 - MAD Auxiliary Systems and Components
 - MB Construction
 - MC Operation
 - MCA Program
 - MCB Procedures
 - MCC Training
 - MCD Experience
 - MD Analysis
 - MDA Theoretical
 - MDB System Performance
 - MDC Nuclear Performance
 - ME Maintenance
 - MEA Principles
 - MEB Procedures
 - MEC Experience
 - MF Decommissioning
-

N Miscellaneous

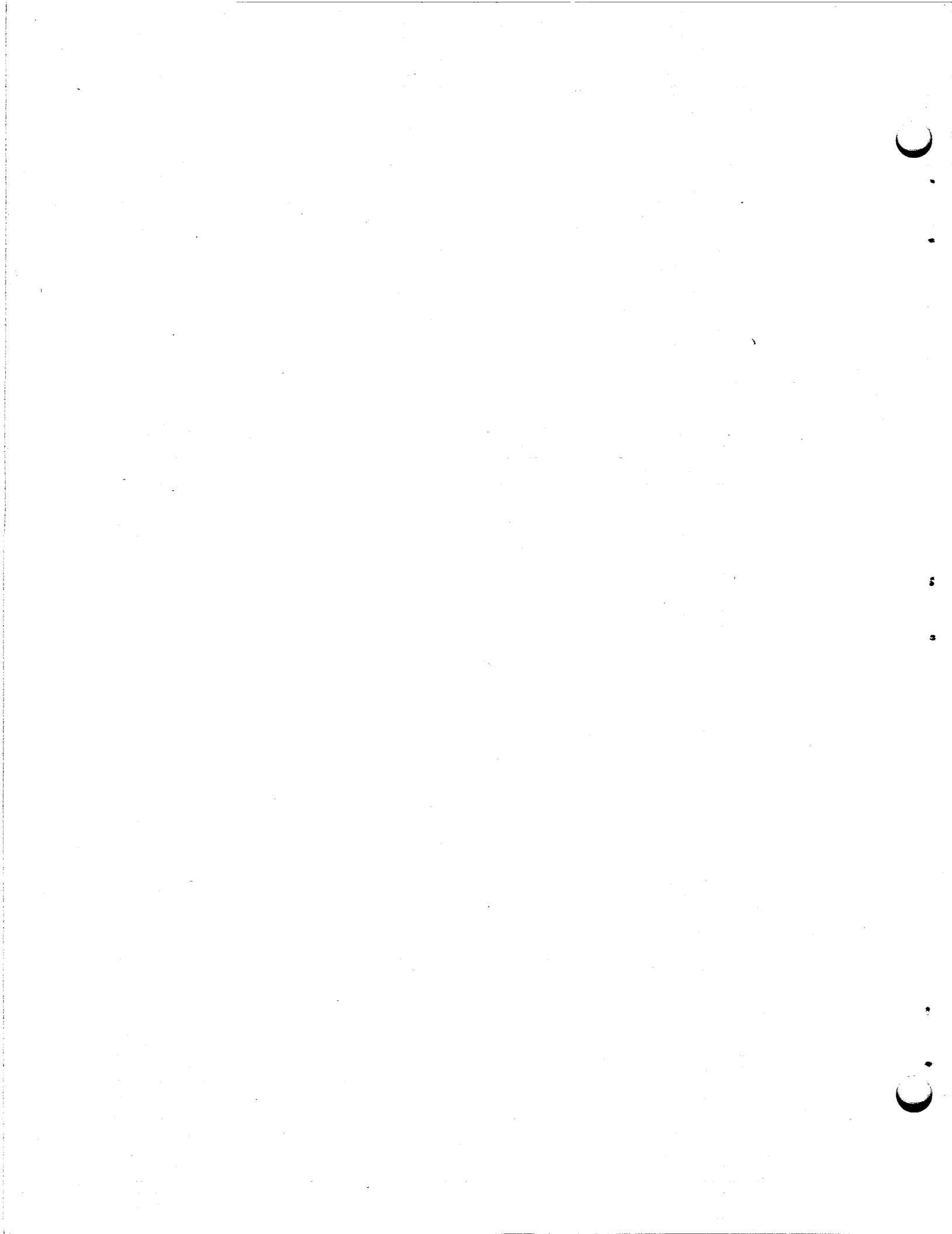
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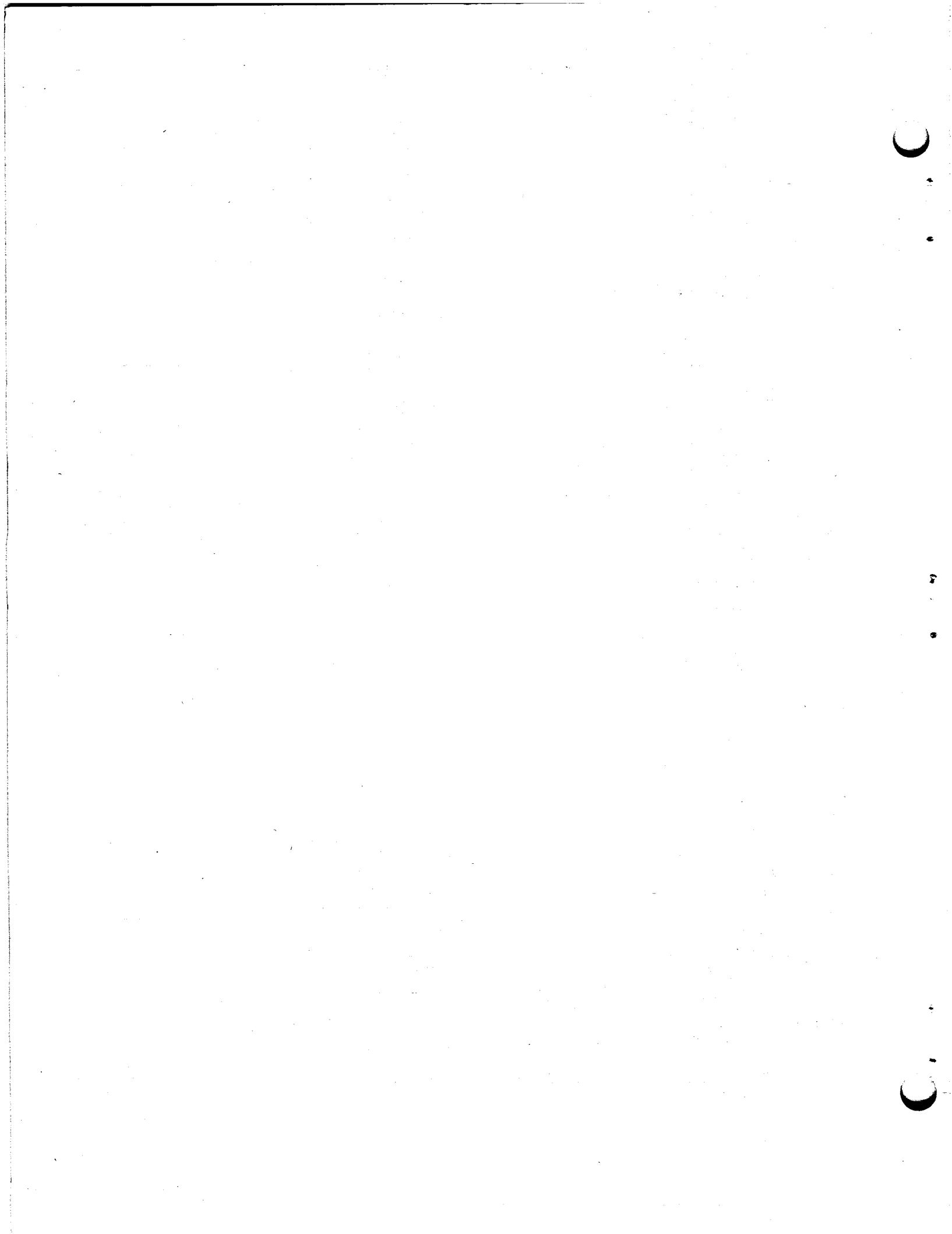
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