ORNL-TM-3595

Contract No. W-7405-eng-26

Reactor Division

INDEXED ABSTRACTS OF SELECTED REFERENCES ON MOLTEN-SALT REACTOR TECHNOLOGY

D. W. Cardwell and P. N. Haubenreich

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DECEMBER 1971

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

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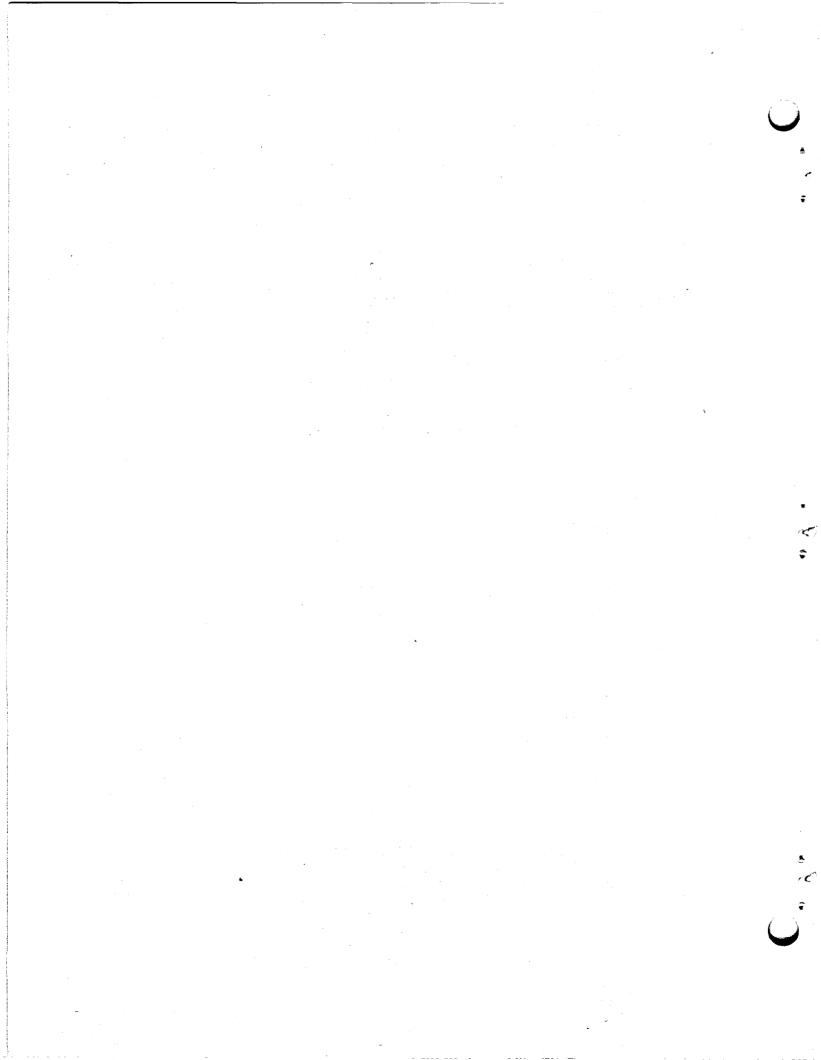


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ACKNOWLEDGMENTS

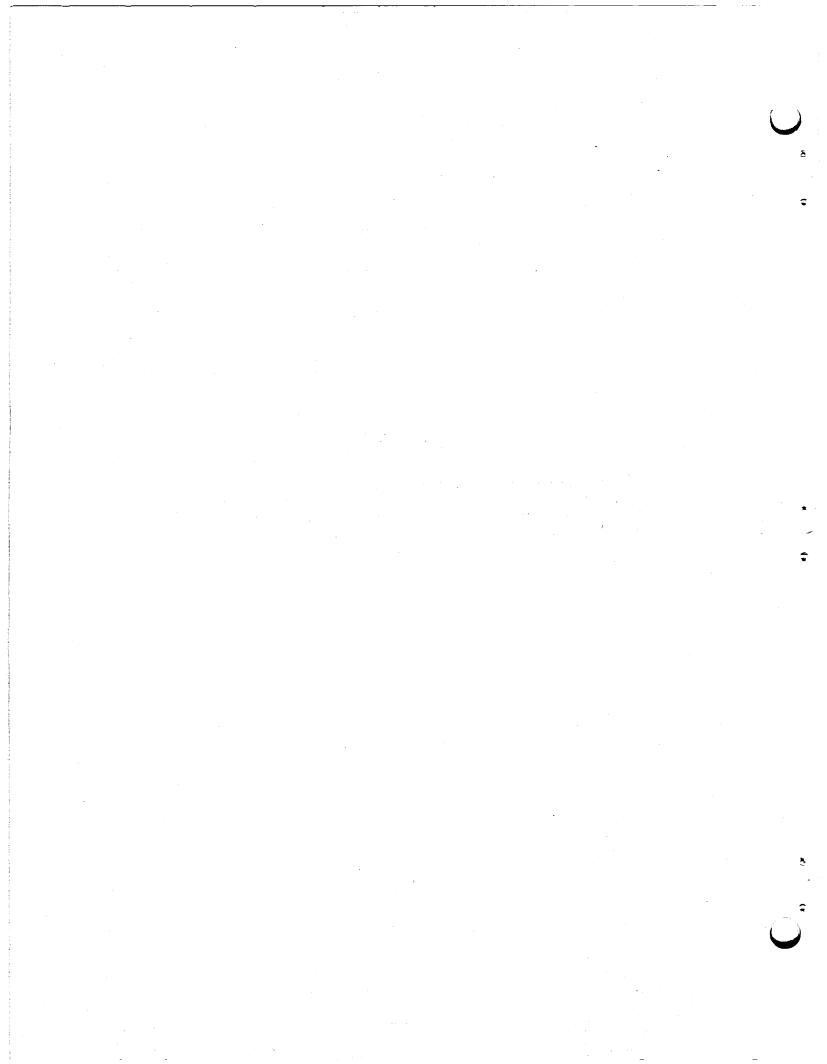
The authors express their appreciation to Ruth Hofstra, John A. Carpenter, and A. F. Joseph of the ORNL Mathematics Division for providing technical guidance and programming required for file formatting, computer entry and automated printout of the MSR publication abstracts contained in this report. 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.



INTRODUCTION

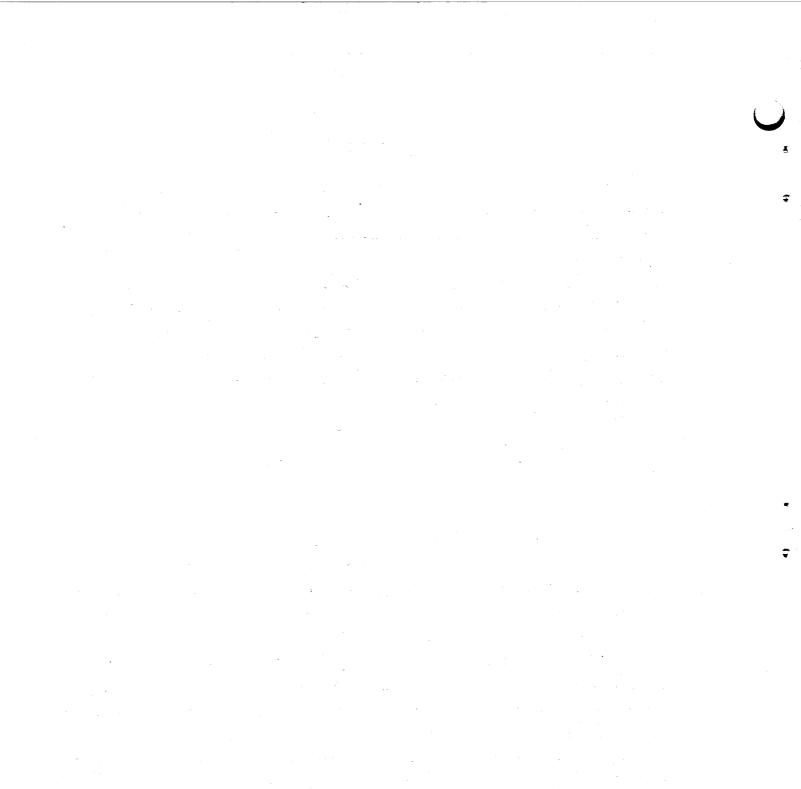
This document contains abstracts of 321 selected reports and papers which collectively provide a good, basic introduction to molten-salt reactor technology and describe major developments in the field since the initiation of the MSRE in 1960. As an aid in locating specific information, three indexes 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 <u>primary</u> 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.



AAX670003

Briggs RB

SUMMARY OF THE OBJECTIVES, THE DESIGN, AND A FROGRAM OF DEVELOPMENT OF MOITEN-SALT EREEDER REACTORS

Oak Ridge National Laboratory, Tenn.

ORNL-TM-1851 (June 1967), 84 F, 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) staticns 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 The present status of the breeder technology fast breeders. 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 ccsts + MSBE +
MSBR + natural resources + performance + power costs + technology
''

AAX670004

Carter WL + Whatley ME

FUEL AND EIANKET PROCESSING DEVELOPMENT FOR MOLTEN SALT BREEDER REACTORS

Cak Ridge National Laboratory, Tenn.

ORNL-IM-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 shculd 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

AAX67C005 Grimes WR CHEMICAL RESEARCH AND DEVELOPMENT FCB MCLTEN-SALT EFEFCER

Accession Number AAX670003 to AAX670005

AAX67C005 *Continued* REACT ORS

Oak Ridge National Laboratory, Tenn.

CRNL-TM-1853 (June 1967), 140 p, 26 fig, 69 ref. Results cf 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-EeF2-ThF4 similarly appear suitable as blankets for such machines. Several possible secondary coolant mixtures are available; NaF-NaEF4 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 MSBR 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 MSBR with a uranium-bearing fluoride fuel salt, a thorius-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 tukes, which must withstand 10(23rd) neutrons/sg-cm 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 descrited 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 tules 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 MSER.)

modified Hastelloy N + graphite + *development + *materials + inspection + *MSER + brazing + compatibility + mechanical properties + costs + reviews + *MSEP + *plans OTHER CATEGORIES: EDX + FCX

AAX670007

Accession Number AAX670005 to AAX670007

Page 6

AAX670007 *Ccntinued*

Scott D + Grindell AG

COMPONENTS AND SYSTEMS DEVELOPMENT FOR MOLTEN SALT ERFECER 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 ccolant-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 + *MSRF + *plans + *reviews + control-rod drives + control rods + cores + heaters + heat exchangers + hydraulics + pumps + steam systems + two-fluid reactor + valves + thermal insulation CTHER CATEGCRIES: HXX

AAX67C008

Tallackson JR + Moore RL + Litto SJ

INSTRUMENTATION AND CONTROLS DEVELOPMENT FCR MOLTEN-SALT BREEDER FEACTORS

Oak Ridge National Laboratory, Tenn. CRNL-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 + flcw measurement + MSRE + MSFP + plans + measurement + radiation measurement + temperature measurement + weigh cell OTHER CATEGORIES: JXX

AAX670009

Category A Molten-Salt Reactor Programs

AAX670009 *Continued* Ferry AM

PHYSICS PROGRAM FOR MOLTEN-SAIT BREEDEP REACTCRS Cak 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 Mclten-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 cutlined which should provide a sound tasis for design of a reactor experiment. The program includes thecretical 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 FOF 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 tehavior, changes in fluid flow conditions, and control rod movement. Reactivity coefficients which require evaluation include these associated with temperature, voids, pressure, fuel concentration, and graphite concentration. The integrity of rlant 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 MSER conditions, as they relate to reactor safety, need to be determined experimentally. To delineate and resolve the basic safety problems associated with

Accession Number AAX670009 to AAX670010

MSBR systems, about \$1.3 million is required over

AAX67C010 *Ccntinued*

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 CTHER CATEGCFIES: EGX

AAX67C011

Blumberg 5

MAINTENANCE DEVELOPMENT FOR MOLTEN-SALT BREEDER BEACTORS Cak 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 landling 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 CATEGCRIES: KEE

ABX 580001 MacPherson HG MOLTEN-SALT REACTORS Cak Ridge National Laboratory, Tenn. Part II of Fluid-Fuel Reactors, Addison-Wesley (1958), pp. 563-697.

The early history and 1958 development status of moltensalt reactors is covered in 7 chapters of this book, prepared for the second Geneva Conference. Chapter topics include chemistry, materials, nuclear aspects, heattransfer equipment, the Aircraft Feactor 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 AAX670010 to AEX580001

Category A Molten-Salt Reactor Programs

ABX5800C1 *Continued* molten salts

ABX64C004 Briggs RB MOLTEN-SALT POWER REACTORS AND THE RCLE OF THE MSBE IN THEIF DEVELCEMENT (FART 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 mclten-salt reactor to be the most promising thermal-neutron thorium-U233 treeder 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 MSE 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-SAIT REACTOR SHOWS MOST PROMISE TO CONSERVE NUCLEAR FUELS Cak 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 MSEE 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-SALI REACTORS Cak Ridge National Laboratory, Tenn. Proc. Intl. Conf. on Constructive Uses of Atomic Energy, Washington, Nov. 1968, pp. 111-121, 7 tig, 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 ccre. Advanta ce s

Accession Number ABX580001 to ABX680035

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 Cak 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 The fluid-fuel molten-salt investments can be low. 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 ccre with circulating salt containing both uranium and thorium, processed by reductive extraction into hismuth. *treeding performance + *economics + *electrical power +

#MSBR + *natural resources + *reviews + experience +
fuel cycle costs + MSRE + MSRE + processing

ABX690056

Rosenthal MW + Robertson RC + Bettis ES MCLTEN-SAIT BRFEDER REACTORS Oak Ridge National Laboratory, Tenn. Nucl. Engrg. Int. Vol. 14, No. 156 (May 1969), pp. 420-425, 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 solter-salt fuel. Brief descriptions are given or MSBF materials, core design, components, and processing schere. 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

Accession Number ABX680035 to AEX700054

Category A Molten-Salt Reactor Frograms

Continued ABX700054 Rosenthal MW + Kasten PR + Briggs RB MOLTEN-SAIT BEACTORS -- 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 CENL promise safe, low-cost power while extending rescurces 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 corresion 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 MSB plants of increasing size. *MSRP + *reviews + ARE + breeding performance + capital costs + design + development • fuel cycle ccsts + MSBR + materials + processing + safety + technology ABX700055 Shaw M + Landis JW + Laney RV + Rosenthal EW + Layman WH U. S. SUBVEY: 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 Freeder, Molten-Salt Ereeder, High-Temperature Gas, and Gas-Cocled 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 REACTEUES A SELS FONCUS Euratcm Energie Nucleaire, Vol. 13, No. 2 (Mar.-Apr. 1971) pp. 86-93, 10 fig.

Accession Number ABX700054 to AEX710020

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 treeder worthy of multinational consideration. They describe the concept, early development, recent progress, protlems, advantages and possible future development. (An English translation, ORNI-tr-2508, is available from ORNL.)

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

ACA65C004

Haubenreich PN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRE FFUGF. FFFT. 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 prenuclear 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 + *startur + *testing +
drying + freeze valves + krypton + loading + molten salts +
thermal insulation
OTHER CATEGORIES: MXX + KAB

ACA650010

Haubenreich PN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRP PROGR. BEP1. 8/31/65)

Cak Ridge National Laboratory, Tenn.

CRNL-3872 (Dec. 1965), pp. 7-78, 34 fig, 4C ref.

Prenuclear 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 refore additions of enriched U-235 tegan, 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 Frograms

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 + leading + measurement + molten salts + reactivity + operators + training + testing OTHER CATEGORIES: MXX + KAB

ACA660008

Hautenreich PN

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

Cak Ridge National Laboratory, Tenn.

CRNL-3936 (June 1966), pr. 7-92, 41 fig, 43 ref.

Preparations for high-power operation were completed. These included modifying coolant line archor 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 +
contrcl rcds + dynamics tests + heat treatments +
off-gas systems + piping + pumps + remote maintenance +
stability + startup + stress + testing
OTHER CATEGCENIES: MXX + KAE + KBA

ACA66C014 Haubenreich PN MOLTEN-SALT REACTOR EXPERIMENT (PAET 1 MSRF FEGGE. FEFT. 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 cil 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 menon 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.)

Accession Number ACA650010 to ACA660014

Category A Molten-Salt Reactor Frograms

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. Hut 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 + *cperation +
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-SAIT REACTOR EXPERIMENT (PART 1 MSRP PROGR. BEP1. 2/28/67)

Cak Ridge National Laboratory, Tenn.

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

Replacement blowers were received in Cotober 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 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 C.05%. Full-rower operation was resumed and continued through February.

*experience + *MSRE + *operation + analysis + blowers +
off-gas systems + reactivity + remote maintenance +
disconnects
OWURD CONFECTER: MAX + KER + KER

OTHER CATEGORIES: MXX + KAE + KBA

ACA670023

Accession Number ACA660014 to ACA670C23

Category A Molten-Salt Reactor Programs

ACA67C023 *Continued*

Haubenreich FN

NOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRE SEMIANN PECG 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 rever, 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. Euring 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. Iccls were developed and the latch was retrieved, but not the capsule. Operations analysis included long-term reactivity effects, thermocourle 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 CATEGORMES: MXX + KAE + MDX + MEC

ACA680012

Haubenreich FN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRF SEMIANN PECG 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 purp, 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 rearing was replaced on a main flower. Otherwise no equipment problem interfered with operation. Operation at various fuel levels, temperatures and pressures showed effects on xenon stripping, neutron 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

ACA680012 *Continued*

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 CTHER CATEGORIES: MXX + KAE • MDX + MEC

ACA68C019

Haubenreich PN

MOLTEN-SALT REACTOR EXPERIMENT (FART 1 ESRF SEMIANN FACG BEFT 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-scram 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 + *flucrination + *maintenance + *MSFE + *operation +
analysis + dynamic characteristics + freeze flanges +
samplers + uranium-233 + cff-gas systems
GTHER CATEGORIES: MXX + KAE + MDX + MEC + LHX

ACA69C021

Haubenreich FN

MOLTEN-SALT REACTOR EXPERIMENT (FART 1 MSBP SEMIANN EFCG REET 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

Accession Number ACA680012 to ACA690021

Category A Molten-Salt Reactor Frograms

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 towl (Mark-2) begar. *experience + *maintenance + *MSRE + operation + analysis + control rods + dynamics tests + freeze flanges + off-gas systems + reactivity + uranium-233 + ncise analysis OTHER CATEGORIES: MXX + KAB + MDX + MEC

ACA690028

Haubenreich PN

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

Cak Ridge National Laboratory, Tenn.

CRNL-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 ratic 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 cifgas 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 + xeron
OTHER CATEGORIES: MXX + KAB + MDX + MEC

ACA700021

ACA700021 *Continued*

Hautenreich PN

MOLTEN-SAIT REACTOR EXPERIMENT (PART 1, MSRP SEMIANN PROG REPI 2/28/70)

Cak Ridge National Laboratory, Tenn.

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

Nuclear operation of the MSRE was concluded on Iec. 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

CTHER CATEGORIES: MXX + KAB + MDX

ACA700035

Haubenreich FN

MOLTEN-SALT REACTOR EXPERIMENT (PART 1 MSRE SEMIANN FRCG REPT 8/31/70)

Oak Ridge National Laboratory, Tenn.

CRNL-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 piging. 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

ACA7 10028

Hautenreich PN

MOLTEN-SAIT REACTOR EXPERIMENT (PART I, MSRP SEMIANN PROG REPT 2/28/71)

Cak 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 Frograms

ACA710028 *Continued*

Pertions of the fuel system were removed for examination as planned. Control rods, rod thimbles and one mederator bar were taken out and the interior of the reactor vessel was viewed. The fuel sampler cage was cut cut 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 value 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 CTHER CATEGORIES: MEX

ACB660009

(Staff Report)

MSBR DESIGN STUDIES (CHAP 6, ESRP SEMIANN FRCG REFT 2/28/66) Cak Ridge National Laboratory, Tenn.

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

A reference design concept is described for a $1000-MW \in two-fluid MSER$ with fuel and coolant salts separated by graphite tubes in a 14-ft reactor vessel. Flowsheet, layouts of the radioactive systems, and processing by flucride 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

MCLTEN-SAIT BRFEDER REACTOR STUDIES (PART 3 MSRP SEMIANN PROG REPT 8-31-66)

Cak 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) MSBE included adoption of a modular concept, using four small reactors to facilitate maintenance, and revision of the primary heat exchangers to use tent tubes rather than bellows. Nuclear performances with and without Fa 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-cocled MSBE, and eipthermal treeder, 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

Accession Number ACA7 10028 to ACE660015

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 CTHER CATEGORIES: IAC

ACB670017

Briggs RB

MOLTEN-SALT BREEDER REACTOR DESIGN STUDIES (PART 3 MSFF SEMIANN FRCG REFT, 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 The 250 MW(e) reactor module has a vessel 12 ft continued. diam with 4-in. diam graphite talls between the ccre 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 fractics 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 flucrinator 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 cn varcrizing surfaces are presented.

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

ACB67C024

Briggs BB

MSBR DESIGN AND DEVELOPMENT (FAST 2 MSEE SEMIANN FACG FEPT 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) MSBB

Accession Number ACB660015 to ACB670024

Category A Molten-Salt Reactor Frograms

ACB670024 *Continued*

using four 250 MW(e) reactor modules involved new cell layouts to accommodate stresses in piping and redestalmounted 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 ccre 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 Dounreay Fast Reactor data. Flux-flattening is discussed and the temperature coefficients of reactivity calculated. The xencn-135 phèsgnafhàteiphoisedalcDèseèdpuedtaftebheat genetheidnelp blanket and coolant-salt pumps are outlined, farticularly with regard to the molten-salt journal bearing.

*MSBR + *progress report + *conceptual design + *ruffs + *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)

Cak Ridge National Laboratory, Tenn.

CRNL-4254 (Aug. 1968), pr. 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-tc-metal joints in the two-fluid concept and because means for chemical processing of a

single salt new 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 twofluid 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 hearing as in the two-fluid concept. A salt-hearing experimental test loop and program are described, however. Remote maintenance problems in an MSBR plant are discussed. Preliminary

Accession Number ACB670024 to ACB680013

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 +
&gammigschabretdingtjesftrdewedopmenintenance +
CTHER CATEGORIES: IAE

ACB68C02C

Briggs RB

MSBR DESIGN AND DEVELOPMENT (FART 2 MSRF SEMIANN FRCG FEPT 8-31-68)

Cak 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 EW(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 MSBE pumping requirements and first operation of the sodium flucroborate test loop are discussed, as were the requirements for a steam generator test facility. Analyses of the dynamic response of the MSER system (and the steam generator) indicates general feasibility. Neutron decay after shutdown was calculated. **Besistance** thermometers possibly can be used in the MSBR. Test equipment for measuring heat transfer properties of the salt are described and data given for thernal conductivity and heat capacity. Test equipment and first data on simulated mass transfer of xencn 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

Category A Molten-Salt Reactor Frograms

ACB690022 *Continued*

Briggs RB MSBR DESIGN AND DEVELOPMENT (PART 2 MSRP SEMIANN PRCG REP1 2-28-69)

Cak 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) MSEE continued with emphasis on the reactor core and vessel design, flow and temperature distributions, fission-product distribution in the systems, krypton and xencn 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 flctted. 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. Opera icn cf the scdium flucrcborate test lcor is described. A successful remotelyoperated orbital welder for piping is reported. Ine controls system studies continued. Freliminary drawings and descriptions of the MSEE are included.

*MSBR + *progress report + *conceptual design + *reactors +
*ESBE + *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 + xencn +
krypton + fluoroborates + decay + heat + graphite +
*steam systems
onumber of the structures =

OTHER CATEGORIES: IAD

ACB690029

Briggs RB

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

Cak Ridge National Laboratory, Tenn.

CRNL-4449 (Feb. 1970) pp 39-95, 41 fig, 12 tatles, 38 ref. Conceptual study of a single-fluid 1000 MW (e) reference design MSBE 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. I waste storage cell is also provided. Seismic disturbances were considered in the design. Laycut drawings are shown for all building

Accession Number ACB690022 to ACB690029

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 fluorotorate 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 + *MSFF +
*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 REPI 2-28-70)

Cak 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) singlefluid MSER were completed and the first draft of a report circulated. Princifal design data are presented. Studies are being made of first-generation types of solten-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 tules 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 ancunts 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-crerated orbital welder for riging discussed in some detail. Simulation studies of dynamic response of MSBR controls systems are presented. Development work was continued on

Accession Number ACB690029 to ACB700022

Category A Molten-Salt Reactor Frograms

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 + *MSEE +
*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
CTHER CATEGCRIES: IAC

ACB70C036

Briggs BB MSBR DESIGN AND DEVELOPMENT (FABT 2 MSRF SEMIANN FRCG FEPT 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 MSBF technical problems but some studies continued on a demonstration plant and plans progressed for an industrial study of a large MSER station. Flowsheets and laycut 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. Eatch processing was also considered. Capture cross section ratics for alpha for U-235 were determined experimentally. Euble generator and gas separator testing is reported. Plans procressed 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 tuttles 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

ACB700036 *Continued* graphite + rumrs + containment + earthquakes + fluoroborates + bubbles + gas separation + components + development CTHER CATEGCRIES: IAC

ACB710029

Briggs RB

MSBR DESIGN AND DEVELOPMENT (FART II, MSRP SEFIANN IFCG 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 coclant 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 + *ESEE + *ESEF + analysis + converters + coolants + design data + flowsheets + gas separation + industrial studies + progress report + steam generators + tritium OTHER CATEGORIES: HEX + IAD + IAE

ACC650006

Lindauer RB FUEL FROCESSING (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 molter 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 + *prccessing + *construction + absorption + corrosion products + design + hydrogen compounds + cxides + sodium flucride + uranium CTHER CATEGORIES: LHX + MEX

ACC650012

Category A Molten-Salt Reactor Frograms

ACC650012 *Continued* Lindauer RB FUEL FRCCESSING (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 FFE. *MSRE + *oxides + *processing + construction + operation + plant OTHER CATEGORIES: LHX + MCD ACC660010 (Staff Report) MOLTEN-SALT REACTOR PROCESSING STUDIES (Part 7 MSEE Frogr. Rept 2/28/66) Oak Ridge National Laboratory, Tenn. CRNL-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 MSBB system. Fuel will be processed on a 40-day cycle. Uranium will be flucrinated 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 bydrogen. Flucrinator corrosion can probably be eliminated by a layer of frozen salt on the wall. Experimental work with a small ccuntercurrent continucus fluorinator gave recoveries of 90 to 96% of the uranium. Volatile chronium flucrides can be trapped with negligible uranium losses on sodium fluoride beds. A preliminary design study of the above facility has illuminated protlems among which is handling high-heatgenerating materials. The fixed capital cost for the conceptual plant was \$5.3 million; the salt inventory cost was \$C.196 million, and the direct operating cost was **\$787,790** per year. *MSBR + *processing + corrosion protection + costs + design + distillation + fluorination + lithium + rare earths + sodium flucride + uranium + volatility + two-fluid reactor OTHER CATEGORIES: LJX ACC660016 (Staff Report) MCLTEN-SAIT REACTOR PROCESSING STUDIES (PART 9 MSRP PROG REPT 8/31/66)

Accession Number ACC650012 to ACC660016

ACC66C016 *Continued*

Cak Ridge National Laboratory, Tenn.

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

The MSBB 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. Corresion 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-SAIT REACTOR PROCESSING STUDIES (PART 10 MSRP PROG REPT 2/28/67)

Cak Ridge National Laboratory, Tenn.

ORNL-4119 (July, 1967), pp. 204-213, 5 fig, 2 ref. Studies on the flucrination-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 $3 \times 10 (-3rd)$, $3 \times 10 (-4th)$, $6 \times 10 (-4th)$ and $2 \times 10 (-4th)$ 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 + MSFE + rare earths + uranium + volatility

ACC670025

(Staff Report)

MOLTEN-SALT PROCESSING AND PREPARATION (PART 6 MSRF PROG REPT 8/31/67)

Oak Ridge National Laboratory, Tenn. CRNL-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 Frograms

ACC670025 *Continued*

step in the fluorination-distillation flowsheet. Relative volatilities of rare earth fluorides were reasured 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 teing studied. Modifications are being made to the MSRE fuel processing facility to permit uranium recovery by flucrination after only 35 days decay. Design and equipment fabrication is in progress for preparing 40 kg cf uranium-233 as the uranium-lithium flucride eutectic for replacement of the present MSBE uranium fuel. *MSBR + *processing + *distillation + volatility + rare earths + reductive extraction process + MSRE + uranium-233 + fluorination + lithium + bismuth CTHER CATEGORIES: LCA ACC680014 (Staff Report) MOLTEN SAIT PROCESSING AND PREPARATION (PARI & MSRP PROG REPT 2/29/68) Cak 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 lithiumbisguth 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 or the wessel wall. Relative volatility measurements were made for uranium, rubidium, ceaesium 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 flucrination tests were made with simulated MSFF 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 + thcrium + -

uranium + uranium-233 + volatility

ACC680021

ACC680021 *Continued*

(Staff Report)

MOLTEN SAIT PRCCESSING AND PREPARATION (PAET 6 MSEF FEGG REPT 8/31/68)

Oak Ridge National Laboratory, Tenn.

CRNL-4344 (February 1969), pp. 291-326, 22 fig, 13 ref. Measurement of distribution coefficients for the reductive extraction process continued. The solutility of protactinium and therium in bismuth was determined. Simulated molten salt-liquid tismuth 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 frozer salt wall for corresion protection during fluorination. Equipment is being installed at the MSRE for demonstration of fuel salt distillation. Relative volatility measurements were made with thorium fluoride. Preparaticn 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 PRCCESSING AND PREPARATION (PART 6 MSRP PROG REPT 2/28/69)

Cak 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 rareearth 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 guartz 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 + MSFE + solubility + thorium

ACC69C023 *Continued* CTHER CATEGORIES: LKX

ACC69C030

(Staff Beport)

MOLTEN-SALT PROCESSING AND PREPARATION (PART 6 MSBE FECG REPT 8/31/69)

Oak Ridge National Laboratory, Tenn.

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

Measurement of distribution coefficients in multen saltmetal systems continued and data is presented for transuranium elements, rare earths and thorium. The sclubility of plutcnium fluoride was measured in lithiumberyllium fluoride salt. Flowsheet analyses were made of protactinium isclation, rare-earth removal, thorium stripping, fission product concentrations and heat generation rates. Four runs were made with the semicontinuous system for contacting hisruth with aclten salt. Electrolytic cell and salt-metal contactor development Axial mixing was studied in both facked and continued. bubble cclumns. Abcut 11 liters of the MSFE fuel carrier salt was distilled. Design studies were carried cut cn: (1) heat transfer through the frozen salt walls of an electrolytic cell, (2) a continuous salt purification

system and (3) plutcnium capsules for refueling the MSRE. *MSBR + *processing + bismuth + columns + design + distillation + distribution + electrolysis + flowsheets + heat transfer + MSRE + plutonium + protactinium + reductive extraction process + solutility + rare earths + thorium

ACC700023

(Staff Report)

MOLTEN-SAIT PRCCESSING AND PREPARATION (PART 6 MSBP PRCG REPT 2/28/70)

Oak Bidge National Laboratory, Tenn.

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

A new processing flowsheet for a single-fluid MSEF 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 tisruth is being studied in support of the metal transfer process. Four more runs were made with the semicontinuous system for contacting bisruth with molten salt. Equipment is being prepared for a demonstration of the metal transport process. Contactor and electrolytic cell development is Data from the distillation of MSRE fuel carrier continuing. salt is presented. Material and energy balance calculations and calculations on the effect of chemical processing on

Accession Number ACC690023 to ACC700023

Category A Molten-Salt Reactor Frograms

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-SAIT PRCCESSING AND PREPARATION (PART 5, MSRP SEMIANN PROG REPT 8/31/70)

Cak Ridge National Laboratory, Tenn. ORNL-4622 (Jan. 1971), pp 199-224, 20 fig, 27 ref.

Cal^Culations were ^made fo^r 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 CRNI 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 ro radiation-induced incompatibility. *examinations + *in-pile tests + *radiclysis +

Accession Number ACC700023 to ACD650007

Category A Molten-Salt Reactor Programs

ACD65C007 *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. Varcr 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.) Prerarations 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 + flucrides + fluorchorates + iodire + molten salts + oxides + precipitation + protactinium OTHER CATEGORIES: MCD + CXX + DXX ACD650013 (Staff Report) RADIATICN CHEMISTRY (CHAP 5, MSRP SEMIANN PROG REP1 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 wolter-salt experiment to go in the CRR. It consists of a compact thermal-circulation loop of INOR-& including a 2-inch graphite core and 85 cc of fuel salt. description + development + in-pile tests ACD660011 (Staff Report) CHEMISTRY (CHAP 5, MSRP SIMIANN 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 nc anomalies and excellent purity. Flugging material in the offgas line proved to be oil decomposition products. Studies cf 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 +.

Accession Number ACD650007 to ACD660011

Category A Molten-Salt Reactor Programs

ACD66C011 *Ccntinued* 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 SIMIANN 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. Behavicr cf Fuel and Coclant Salts in MSBE. Physical Chemistry of Fluoride Melts: Viscosity and Censity of Molten Beryllius Fluoride: Transpiration Studies in Support of the Vacuum Listillation Process; Estimated Thermophysical Properties of MSBR Coclant Salt. Separation in Molten Fluorides: Extraction of Rare Earths from Molten Fluorides into Molter 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 Molter Metals; Protactinium Studies in the High-Alpha Molten-Salt Laboratory. Radiation Chemistry: Xenon Diffusion and Possible Formation of Cesium Carbide in an MSEF; Fission Product Eehavior in the MSRE. Development and Evaluation of Analytical Methods for Molten-Salt Beactors: Eetermination of Cxide 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 MSEE 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 + borch trifluoride + carbides + cells + circulation + compatibility + concentration + cores + corrosicn + corrosicn products + cover gas + decay + density + deposition + diagrams + dissolving + distillation + electrical properties + entrainment + equilibrium + examinations + fission + fluorides + fluorchorates + gamma radiation + gamma spectrometry + gases + Hastelloy N + inert gases + lithium iluoride + molten salts + oxide precipitation process + phase equilibria + progress report + protactinium + protactinium flucrides + rare gases + research + sampling + specific heat + uranium fluorides + vapor pressure + viscosity

ACD670019

Category A Molten-Salt Reactor Programs

ACD67C019 *Ccntinued* (Staff Report) CHEMISTRY (CHAP. 7, MSRP SEMIANN. FROGR. REPI. 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-BeF2-ZrF4 mistures, Sclubilities cf SmF3 and NdF3 in Molten LiF-BeF2 (66-34 mole %), Possitle MSBR tlanket-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 flucrides into molten metals, Frotactinium studies in the high-alpha molten-salt latoratory, Preliminary study cf the system LiF-ThF4-PaF4, Development and Evaluation of analytical methods for molten-salt reactors, Determinations of cxide in MSRE salts, Determination of U(3+)/U(4+) ratios in radioactive fuel by a hydrogen reduction method, EMP measurements on the Nickel-Nickel(II) couple in Molten fluorides, Studies of the anodic uranium wave in molten LiF-EeF2-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 + *notle metals + *cxides + *protactinium + *protactinium fluorides + *reductive extraction process + *sampling + *surveillance + beryllium flucride + bismuth + capsules + chemical properties + chemical reactions + compatibility + concentration + corrosion products + electrolysis + gamma spectrometry + hot cells + in-pile tests + liquid metals + lithium fluoride + materials + materials testing + metals + mclytdenum + MSRE + nickel + nickel alloys + progress report + rare earths + secondary salts + testing + thorium fluorides + uranium fluorides + zirconium fluoride CTHER CATEGCRIES: CXX + IXX + MCD

ACD67C020

Accession Number ACD67C019 to ACD67CC2C

Category A Molten-Salt Reactor Programs

ACD67C020 *Continued* (Staff Report) CONVECTION LOOP IN ORR (CHAP. 8, MSEP SEMIANN. PROGR. REPI. 2/28/67) Cak 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 Bidge Besearch Reactor was terminated Aug. 8, 1966, after development of $1.1 \times 10(1\xi)$ fissions/cc (0.27% U-235 burnup) in the LiF-EeF2-2rF4-UF4 (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, Sarpling and addition, Salt circulation, Second in-rile 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. GRNL-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 cr molybdenum. Cxide flucride equilibria in fuel systems was studied in a research for separation processes. Phase behavior, decomposition pressure and corrosiveness of flucroborate coolants is described. Recovery of protactinium and removal of fission products by reductive extraction is discussed. Developmental studies in analytical chemistry

Accession Number ACD670020 to ACD670026

directed primarily to improvements in analyses of

Category A Molten-Salt Reactor Programs

ACD67C026 *Ccntinued*

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

*analytical chemistry + *boron trifluoride + *coolants + *corrosion + *fission products + *fluoroborates + *molten salts + *graphite + *cxides + *rare earths + *reductive extraction process + analysis + behavior + beryllium fluoride + bismuth • blanket + capsules + cells + chemical reactions + chemistry + chrcmium + ccmpatibility + cores + ccrrosion 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 + cride precipitation process + phase equilibria + physical properties + protactinium + protactinium fluorides + radiclysis + rare gases + reduction + sampling + solubility + solidus + spectrophctcmetry + stability + surveillance + testing + thorium + thorium fluorides + uranium + uranium flucrides + vapor pressure OTHER CATEGORIES: CXX + EXX + MCD

ACD67C027 Bohlmann EG IRRADIATION EXPERIMENTS (FART 4, MSRF SEMIANN, FECGE, 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 + *notle 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 + MSBE + progress report +
radiation damage + rare earths + research + sampling +
stress + stress rupture + uranium fluorides

ACD680015

Accession Number ACD670026 to ACD680015

Category A Molten-Salt Reactor Frograms

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 stand point 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. Cther topics are "Proton Reaction Analysis for Lithium and Flucrine in MSR Grathite" and "Surface Ihenomena in Molten Salts". Items pertaining to the physical chemistry of molten salts are the thermodynamics of LiF-EeF2 melts from EMF measurements, and electrical properties cf melts. The chemistry cf silica in IiF-BeF2 melts is presented. A Molten Salt Chemistry Information Center is described. Synthesis and properties of molybdenum and nicbium flucrides is discussed. Reprocessing of fuel by reductive extraction into molten bismuth is described, with special emphasis on protactinium recovery. The behavior of BI3 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 + corresicn + cover gas + decomposition + distribution + electrical properties + electrical conductivity + electrolysis + equilibrium + examinations + experiment + fissile materials + fuel preparation + camma spectrometry + Hastellcy 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 + therium + thorium flucrides + uranium + uranium fluorides + vapor pressure + viscosity + volatility + zirconium fluoride + *bismuth + *boron trifluoride + *chemical properties + *chemical reactions + *chemistry + *compatibility + *corrosion products + *fluorides + *flucroborates + *molten salts + *graphite + *noble metals + *physical properties + *processing + *rare earths + *reductive extraction process OTHER CATEGORIES: CXX + MCD

ACD680016

Category A Molten-Salt Reactor Programs

ACD68C016 *Continued*

Bohlmann EG

IRRADIATION EXPERIMENTS (PART 4, MSRP SEMIANN. PRCG. FEFT. 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, MSRP PROGR. REPT. 8/31/68) Cak Ridge National Laboratory, Tenn.

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

MSRE chemistry topics include Feasibility of Fueling with PuF3, 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 Pheromena in Molten Salts. Items under Fhysical Chemistry of Molten Salts include Molybdenum Fluoride Chemistry, Alkali Flucroborates, Physical Properties of ThF4-containing Melts, Electrochemical Studies, Spectroscopy, Oxide Chemistry, and the Chemistry of Silica in LiF-EeF2. Fuel Reprocessing was studied in experiments on the Reductive Extraction of Pa and of Bare Earths into Bismuth. Analytical Studies included oxide determination, U(3+) and total reducing power, U(5+) in LiF-BeF2-ZrF4, Ni(0)/Ni(+) couple, Cr(2+), Hot Cell Spectrophotometer, Spectra of U(5+) and U(6+), a Gas Chromatograph for the Off-gas system, Hydrocarbons in MSRE Helium, Gamma Spectroscopy, Fission Froduct Penetration in Graphite, U-235 Analyses, and Determination of U.

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

Accession Number ACD680016 to ACD680022

Category A Molten-Salt Reactor Programs

白色菜

ACD68C022 *Continued*

*reductive extraction process + *solubility + *surveillance + beryllium + beryllium fluoride + butbles + 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 + cxidation + phase equilibria + plutonium fluorides + precipitation + progress report + protactinium + protactinium flucrides + rare earths + rare gases + reaction rates + reduction + research + sampling + sodium fluoride + solidus + specific heat + spectrophctometry + test facilities + testing + thermal conductivity + thermal properties + thorium + thorium fluorides + uranium + uranium fluorides + vapor pressure + viscosity + volatility + zirconium fluoride OTHER CATEGORIES: CXX + CXX + MCD

ACD68C023 Bohlmann EG IRRADIATION EXPERIMENTS (PART 4, MSFF FROGL. FEFT. 8/31/68) Cak Ridge National Laboratory, Tenn. ORNL-4344 (Aug. 1968), pp. 200-210, 1 fig, 8 ref.

Examinations of the graphite from an CFF convection lcop 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, grathite 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; nc 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, soctlike film, and no other deposits were found. A group of fission products, largely "noble metals" (Mc, 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 Frograms

ACD680023 *Continued*

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

ACD69C024

Grimes WR

CHEMISTRY (PARI 3. MSRP PROGR. REFT. 2/28/69) Cak Ridge National Laboratory, Tenn.

ORNL-4396 (Feb. 1969), pr. 129-196, 32 fig, 122 ref. MSBE 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 Froduct Inventory, Off-gas Inalyses, and Material Recovered from the cff-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 CeF3 (a standin fcr FuF3) solubility, Zone Melting, Phase Relations, Sclubility of Th(c) in LiF-ThP4, Densities, Crystal Structure, Spectroscopy in a Diamond-Windowed Cell, Distribution of U(4+) between fuel and (U-Th)C2 Solid Solution, Feference 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 Letermination of Oxide and Oxidation State, Enf, Voltammetric, ard Spectrographic Studies, Wetting Behavior, Contaminants in Blanket Gas from NaEF4 tests, and the Determination of Bi in MSRP Salts.

*analytical chemistry + *tismuth + *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 flucride +
beryllium oxide + blanket + boron trifluoride + bubbles +
carbon + capsules + cells + compatibility + concentration +

Accession Number ACD68C023 to ACD690024

Category A Molten-Salt Reactor Programs

ACD69C024 *Ccntinued* coolants + cover gas + density + deposition + diagrams + distribution + electrical conductivity + electrolysis + equilibrium + experiment + foaming + freezing + fuel preparation + fuels + gamma spectrometry + gas analysis + gases + hydrofluorinaticn + hydrogen + hydrogen compounds + inert gases + in-pile tests + interfacial tension + ions + laboratory equipment + lithium flucride + melting + metal transfer process + mists + MSBE + 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 flucride + spectrophotometry + surface tension + thorium + thorium fluorides + uranium + uranium fluorides + volatility + zirconium + zirconium fluoride OTHER CATEGORIES: CXX + DXX + MCD ACD690025 Bohlmann EG IRRADIATION EXFERIMENTS (PART 4, MSRP SEMIANN PROG REPI 2/28/69) Cak Ridge National Laboratory, Tenn. CRNL-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 ORE 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 (FART 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 talance, and gas Febavicr. Fission product behavior is deduced from surveillance specimers 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, hetrogeneous equilibria, liquidus temperature; solubility of thorium, U(3+)/U(4+)

Accession Number ACD690024 to ACD690031

Category A Molten-Salt Beactor Frograms

ACD690031 *Continued*

ratic, 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 carths 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 + *coclants + *fission products + *flucrides + *fluoroborates + *molten salts + *graphite + *physical properties + *surveillance + beryllium + beryllium fluoride + bismuth + toron trifluoride + bubbles + cells + concentration + corrosion products + cover gas + density + distribution + electrical conductivity + equilibrium + examinations + fuel preparation + gamma spectrometry + helium + hot cells + hydroflucrination + 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 + solutility + spectrophctcmetry + surface tension + testing + thorium fluorides + uranium fluorides + viscosity + void fractions + zirconium OTHER CATEGORIES: CXX + EXX + MCD

ACD70C024 Grimes WR CHEMISTRY (PART 3, MSRP PROGR. REFT. 2/28/70) Cak Ridge National Laboratory, Tenn. ORNL-4548 (Feb. 1970), pp. 93-187, 50 fig, 153 ref. MSRE chemistry topics discussed are corrosicn, arrearance 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 bcwl. 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 Flucicbcrates. These include the oxide chemistry of Pu in molten fluorides, and the solubility of the corrosicn product, Na3CrFf, in flucroborate melts. Basic chemistry work in support of

Accession Number ACD690031 to ACD700024

Page 44

Category A Molten-Salt Reactor Programs

ACD700024 *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: Tetermination 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, Femoval of Cxide 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 flucrides + 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 PRCG FEFT 8/31/70) Cak 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 molytdenum and nichium flucrides 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 NaFF4, coolant salt, and in-line analyses.

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

Accession Number ACD700024 tc ACD700038

Category A Molten-Salt Reactor Frograms

ACD700038 *Continued*

*electrical conductivity + *fission products + *fluorides + *fluoroborates + *graphite + molten salts + *ncble metals + *oxides + *processing + *physical properties + *rare earths + *reductive extraction process + *solutility + *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 + Hastellcy N + hydrogen compounds + inert gases + inventories + irradiation + liquidus + lithium fluoride + materials + neasurement + 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 + scdium fluoride + spectrophotometry + test facilities + testing + thermal properties + therive + thorium flucrides + uranium + uranium fluorides + vapor pressure + volatility + zirconium fluoride

ACE650008

(Staff Report)

METALLURGY (Chap. 4 MSRP Prog. Rept. 2/28/65) Cak Bidge National Laboratory, Tenn.

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

Reaction of Hastellcy 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 cood creep and tensile properties. In a poorly weldable heat silicon and aluminum concentrated in a grain-houndary eutectic. Brazing alloys were sought for joints between various refractory metals and graphite or Hastelloy N. Density, gas evolution, and porosity were measured on ESEE graphite. Cxidation 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 + sealing + 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 CTHER CATEGORIES: EAX + FEX + GEX

ACE65C014

Category A Mclten-Salt Reactor Frograms

ACE650014 *Continued* (Staff Report) METALIURGY (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 airmelted 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 + corresion + creep + development +
ductility + embrittlement + examinations + failures +
fluorides + molten salts + microstructure + graphite +
Hastelloy N + heat treatments + impurities + in-pilc 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)

METALIUBGY (Chap. 4 MSRP Prog. Rept. 2/28/66) Oak Ridge National Laboratory, Tenn. CRNL-3936 (June 1966), pp. 95-121, 18 fig, 10 ref.

Molten lead was circulated in NB-1% Zr and in a chromiummolybdenum steel; molten fluorides in Hastelloy N and type 304 stainless steel. MSRE surveillance specimens were examined after prenuclear operation. Irradiation darkened grain boundaries of Hastelloy N tut did not induce reaction with nitrogen. Development of graphite-to-Hastelloy N brazed joints included pressure testing with molten fluorides. Procurement, characterization, and MSER requirements of graphite are described. Irradiation decreased rupture life and creep ductility of Hastelloy N. Effects of pre- and postirradiation heat treatments or 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 tc ACE660012

Category A Molten-Salt Reactor Frograms

ACE660012 *Continued*

brazing • graphite + Hastellcy N + heat treatments + impurities + in-pile tests + inspection + ircr allcys + irradiaticn + joints + lead + liquid metals + loop + mass transfer + materials + materials testing + mechanical properties + metallography + metallurgy + metals + molybdenum + nitrogen + physical properties + procurement + progress report + materials testing + rupture + specifications + stainless steels + stress rupture + surveillance + testing + thermal convection + welding + tensile properties OTHER CATEGORIES: EXX + FXX + GXX

ACE660018

Adamson GM MSRP MATEFIALS (Chap. 6 MSRP Prog. Rept 8/31/66) Oak Ridge National Laboratory, Tenn.

ORNL-4037 (Jan. [967), 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 Hastellcy N. Grathite studies included characterization of several grades, measurement of creep under irradiation at 700 and 10C0 deg C, brazing to molybdenum and Hastelloy N, and molten-salt corrosion of brazed joints. Observations on Hastellcy N include good weldability with titanium modifications but poor with zirconium, postirradiation ductility improved by titarium additions but not tungsten or niobium, and extension of creep studies to 982 deg C. Molten flucrides 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 Crolcy 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 + fluorcborates + graphite + Hastelloy N + heat treatments + in-pile tests + inspection + irradiation + joints + loop + mass transfer + materials + materials testing • measurement + mechanical properties + metallccrarhy + 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 CTHER CATEGORIES: EXX + FXX + GXX

ACE67C021

Accession Number ACE660012 to ACE670021

Category A Molten-Salt Reactor Programs

ACE67C021 *Ccntinued*

Cook WH + McCoy HE + Kennedy CR + Werner WJ + Litman AP + Canonicc DA + Haseltine DM

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

Cak 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 corresion 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 + locg + mechanical properties + metallography + MSFF + 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.

CRNL-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 zirconiummodified 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, fatrication, coating with molybdenum, gas impregnation, and start of HFIF irradiation. Molten salt corrosion to type 3C4 I stainless steel, Crolcy 9M, and a graphite-to-molybdenum brazed joint were also studied.

diffusion + iron alleys + zirconium + elasticity + alloys + brazing + carbides + ccatings + 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 + lcop + mass transfer + materials + materials testing +

Accession Number ACE670021 to ACE670028

Category A Mclten-Salt Reactor Programs.

ACE670028 *Ccntinued*

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.

CBNL-4254 (Aug. 1968), pp. 183-240, 43 fig, 33 ref. Hastelloy N investigations include creep of MSRF surveillance specimens, effect of strain rate on ductility of irradiated and control specimens, creep of modified allcy, 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 varicus molten salts. Graphite studies include procurement, density, microstructure, increase in porosity by exidaticr, x-ray diffraction, gas impregnation, sealing with mclybdenur, 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 acdified Hastelloy N.

alloys + brazing + carbides + coatings + compatibility + corrosicn + creep + density + deposition + development + electrical conductivity + embrittlement + ductility + equipment + examinations + fabrication + failures + fluorides + molten salts + fluoroborates + graphite + Hastellcy N + heat treatments + impregnation + in-pile tests + inspection + irradiation + joints + locg + mass transfer + materials + materials testing + measurement + mechanical properties + metallography + metallurgy + metals + modified Hastelloy N + mclybdenum + ESRE + 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 DEVELOFMENT (Part 5 MSRP Prog. Rept. 8/31/68) Oak Ridge National Laboratory, Tenn.

Accession Number ACE670028 to ACE680024

Category A Mclten-Salt Beactor Frograms

ACE680024 *Continued*

ORNL-4344 (Feb. 1969), pp. 211-290, 62 fig, 32 ref. Graphite and Hastelloy N surveillance specimers 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, HFIF irradiation, and small-angle x-ray scattering. Effects of titanium content and irradiation temperature on creep, effect of aging on tensile properties, welcability, and molten salt corresion were studied on modified Hastelloy N. Stan card Hastelloy N studies included resistance to salts and air, transition joints with graphite, mechanical properties of velds in irradiated specimens, and fluted tubing. Precipitate morphologies were studied in both. Cther corrosion studies included Haynes alloy No. 25 in fluorchorates, stainless steel, and a chemical separation

still. Bearing coatings were studied by x-ray diffraction. alloys + tearings + brazing + carbides + cermets + coatings + columns + compatibility + corrosion + creef + 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 + cxidaticn + 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 OTHEB CATEGORIES: EXX + FXX + GXX

ACE690026

Eatherly WF + McCoy HE + Weir JR MATERIALS DEVELOPMENT (PART 5, MSRF SEMIANN FFCG FEFT 2/28/69)

Oak Fidge National Laboratory, Tenn.

ORNL-4396 (Aug. 1969), pp. 211-268, 54 fig, 44 ref. Magnetic particles from the MSRE pump bowl were guite diverse as shown by microprobe analysis and metallography. Graphite topics include radiation damage fundamentals, binder chemistry, hct 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 Mclten-Salt Reactor Frograms

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. Cther 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 + fluorcborates + 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

ACE69C032

Eatherly WF + McCoy HE + Weir JR MATERIAL DEVELOPMENT (PART 5, MSRP SEMIANN FRCG FEFT &/31/65) Cak 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 Hastellcy N are reported. Graphite studies include x-ray diffraction determination of anisctropy, 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, Ef, Nt, and Y additions improved postirradiation ductility; titanium plus hafnium was lest. 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 remcte welding.

bismuth + coatings + compatibility + ccolants + corresion + creep + defects + density + deposition + development + ductility + embrittlement + equipment + examinations +

Accession Number ACE690026 to ACE690032

Category A Mclten-Salt Reactor Frograms

ACE690032 *Continued* expansion + fabrication + failures + fluorides + fluoroborates + forming + microstructure + graphite + hafnium + heat treatments + impregnation + impurities + Hastelloy N + in-file 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

ACE70C025

Weir JR + McCoy HE

MATERIALS DEVELOPMENT (PART 5, MSRP SEMIANN FECG FEIT 2/28/70)

Oak Ridge National Laboratory, Tenn.

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

Microstructural changes and creep properties were studied on Hastellcy 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 cf aging, irradiation, and composition on mechanical properties, electron miscroscopy, weldatility, and corresion by varicus molten salts were studied for modified Eastelloy N. Back extrusion, welding, and trazing cf molybderum, compatibility of alloys with bismuth, coating with tungsten, and oxidation of steels were studied in surrert 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 + MSBE + molybdenum + physical properties + procurement + progress report + pyrccarbon + radiation damage + remote maintenance + rupture + sealing + stairless steels + steam generators + stress + surveillance + test facilities + testing + tungsten + welding + brazing

ACE70C039

Category A Molten-Salt Reactor Programs

ACE70C039 *Ccntinued*

Weir JR

MATERIALS DEVELOPMEN1 (PART 4 MSRP FBCGE. FEFT., 8/31/70) Cak Ridge National Laboratory, Tenn.

ORNL-4622 (Jan. 1971) pp. 119-198, 75 fig, 51 ref. Tubing and ther soccurle wells from the LSFF 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 cf titanium content and aging on hardness, tensile properties, creep, and postirradiation creep, postirradiation creep of alloys containing various combinations of Ti, Nb, and FF, weldability, creep of commercially melted alloys, and microstructure. Lccr studies of corrosion included type 3041 stainless steel, Hastelloy N and several acdified alleys exposed to fuel, blanket, and coolant salts. The electron microprobe was used to study the corresion of Hastelloy N by power plant steam, and study was started on a duplex steam-generator tube made of Incolcy 800 and Nickel 280. Development of processing equipment included back-extrusion, welding, and trazing cf rolybderur components, compatibility of Mo, TZM, Nb, NE-1% Zr, Ta, T-111, graphite, FE-5% Mo, and krazed joints in Mc with molten bismuth, and deposition of coatings by reduction of MoF6 and WF6 vapers 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)

MCLTEN-SAIT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING JANUARY 31, 1964

Cak 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 Frocessing.

Accession Number ACE700039 to ACX64CCC8

Category A Molten-Salt Reactor Programs

ACX640008 *Continued* MSRP + progress report

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ACX 64 CO 14
(Staff Report)
MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL EROGRESS FEECFT
  FOR PERICE 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 CENL, 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 FREGRESS FEFCFT FOR
  PERIOD ENDING FEERWARY 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-SAIT 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 Frocessing.
MSRP + progress report
ACX660007
(Staff Report)
MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL EROGFESS EFECET FOR
  PERICC ENDING FEERUARY 28, 1966
Oak Ridge National Laboratory, Tenn.
CRNL-3936 (June, 1966), 216 p.
      Status and progress are reported. Contents are abstracted
      and filed in 5 parts: MSRE, Metallurgy, Cheristry,
      MSBE Design Studies, and MSR Processing Studies.
MSRP + progress report
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ACX660013

Category A Molten-Salt Reactor Programs

ACX660013 *Ccntinued* (Staff Report) MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT FOR FERIOD ENDING AUGUSI 31, 1966 Cak 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 **ESRP** + progress report ACX67C015 (Staff Report) MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL FROGRESS SEFECET FOR FERIOD ENDING FEERUARY 28, 1967 Oak Ridge National Laboratory, Tenn. ORNL-4119 (July 196p), 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-SAIT REACTOR PROGRAM SIMIANNUAL PROGRESS REPORT FOR PERIOD ENDING AUGUSI 31, 1967 Cak Ridge National Laboratory, Tenn. ORNL-4191 (Dec. 1967), 260 p. Status and progress are reported in six parts with these titles: Mclten-Salt Reactor Experiment, MSER Design and Development, Chemistry, Nolten-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 FRCGBESS FFFCFT FOR PERICE ENDING FEERUARY 29, 1968 Oak Ridge National Laboratory, Tenn. CRNL-4254 (Aug. 1968), 282 p. Status and progress are reported in six parts with these Molten-Salt Reactor Experiment, MSBR Design titles: and Development, Chemistry, Molten-Salt Irradiation Experiments, Materials Cevelopment, and Molten-Salt Processing and Pregaration. Separate abstracts are filed for each part. MSRP + progress report ACX680018 (Staff Report) MOLTEN-SAIT REACTOR PROGRAM SEMIANNUAL PROGRESS REPORT FOR Accessicn Number ACX660013 to ACX680018

Category A Molten-Salt Beactor Frograms

ACX680018 *Continued* PERIOD ENDING AUGUST 31, 1968 Cak Ridge National Laboratory; Tenn. CRNL-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 Molter-Salt Processing and Pregaration. Separate abstracts are filed for each part. MSRP + progress report ACX690020 (Staff Report) MOLTEN-SAIT REACTOR PROGRAM SEMIANNUAL PROGRESS REPCR1 FOR PERIOD ENDING FEBRUARY 28, 1969 Cak 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, MSER Design and Development, Chemistry, Molten-Salt Irradiation Experiments, Materials Development, and Molten-Salt Processing and Preparation. Separate alstracts are filed for each part. MSRP + progress report ACX690027 (Staff Report) MOLTEN-SALI REACTOR PROGRAM SEMIANNUAL FRCGFESS FEFCFT FOR PERICD ENDING AUGUST 31, 1969 Oak Ridge National Laboratory, Tenn. CRNL-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 Levelopment, and Molter-Salt Processing and Preparation. Separate abstracts are filed for each part except Irradiation Experiments, where there was no activity. MSRP + progress report ACX70C018 (Staff Report) MOLTEN-SALT REACTOR PROGRAM SEMIANNUAL FROGRESS FEFCET FOB PERICO ENCING FEERUARY 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 Levelopment, and Molter-Salt

Accession Number ACX680018 to ACX700018

Category A Molten-Salt Reactor Programs

ACX70C018 *Ccntinued* Processing and Preparation. Separate abstracts are filed for each part except Irradiation Experiments, where there was no activity. MSRP + progress report ACX700034 (Staff Report) MOEDENPERITCRENCTOR ARGGRAM 35 FA 18 NOUAL PROGRESS REPORT Oak Ridge National Laboratory, Tenn. CRNL-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 Mclten-Salt Processing and Preparation. Separate abstracts are filed for each part. MSRP + progress report ACX710027 (Staff Report) MOLTEN-SAIT REACTOR PROGRAM SIMIANNUAL PROGRESS REPORT FOR PERIOD ENDING FEBRUARY 28, 1971 Cak Bidge National Laboratory, Tenn. ORNL-4676 (Aug. 1971), 277 p. Status and progress are reported in five parts with these titles: Molten-Salt Reactor Experiment, MSEE 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 MCSEL BEACTOR 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 corcept features a core fluid of UF4-NaF-BeF2 and a blanket fluid of ThF4-NaF-BeF4 separated by nickel alloy tubes. Processing is by flucride volatility. Farasitic 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 + *rolten salts + *reactors + costs + foreign + flucrination + Hastelloy N + inconels + inventories + neutron spectra + nickel +

Accession Number ACX700018 to ACX640021

Category A Mclten-Salt Reactor Frograms

ADX640021 *Continued* sodium flucride CTHER CATEGCRIES: IAF

ADX67C046 Gat U COOLING CONCEPTS FOR A COMPACT MOSEL (MCLTEN SALT) FEACTOR Kernforschungsanlage Julich, Cermany Nucl. Engrg. and Design 5 (1967), pp. 113-122, 10 fig. 23 ref. This review of engineering possibilities of the MCSEL 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: TAF ADX690063 Jensen RJ + Swanson E UTILITY AFFLICATION OF MOLTEN-SALT BREEDER REACTOR Northern States Power Co, Minnearolis + 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, cr Pu as initial fuel. On-site processing is required. A schedule for 1976 startup of a 30C- to 500-MWe prototype is presented. *electrical power + *industry + *MSBR + *protctyres + *utilities + architect-engineering + breeding performance + converters + economics + natural resources

Category B Reactor Analysis

BAX68C006 Perry AM INFLUENCE OF NEUTRON DATA IN THE DESIGN OF CTHER TYPES OF FOWER REACTOFS 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-sectior 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 crosssection uncertainties is less than plus or minus C.C2, 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 CTHER CATEGORIES: EFX BAX700008 Kasten PR + Craven CW + Wright RC CROSS-SECTION AND NUCLEAR-CONSTANT DATA FOR HEAVY METAL NUCLIDES (FUELS) Cak Ridge National Laboratory, Tenn. ORNL-IM-2851 (Rev.) (Apr. 1970) 21 p, 18 fig, 0 ref. Cross sections and nuclear constants of fissile and fertile materials and cf higher isotopes are summarized in graphical form, tased 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 Nuclides considered are Th-232, U-238, materials. Pa-233, Np-239, U-233, U-235, Eu-239, Eu-241, U-234, U-236, Pu-240, and Fu-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 Cak Ridge National Laboratory, Tenn.

Accession Number BAX680006 to BEX670012

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 tworegion, two-fluid MSER. The review covered cross section selection, fission product treatment, multigroup cell calculations, two-dimensional reactor criticality calculations, equilibrium depletion calculations and startup depletion calculations. Inscfar as possible, the cross sections and calculations methods were made independent of those used previously. The reference composition gave a k(eff) of 0.95. This discrepancy was traced to use of a low value for therium resenance integral in previous calculations. When the reactor was made critical by the addition of 14% mcre 0-233, the breeding ratio was 1.062 compared with 1.054 in the previous calculations. Beoptimization of the composition would probably decrease this difference in breeding ratio.

*calculations + MSER + *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 DEVELCEMENTS 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 sirgle-fluid MSEF. 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 C.1 tc 0.3 mils/kwh(e) below those for comparable uniform cores. Temperature coefficients of reactivity in zched-ccre single-fluid reactors are such that dynamic characteristics are expected to be acceptable. (Freprints of this paper are not available but similar, and more recent, data are presented in Nucl. Appl. and Tech. 8, 208 (Fet. 1970). See BFX700016.)

*MSBR + *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 *Ccntinued* Perry AM + Bauman HF REACTOR PHYSICS AND FUEL CYCLE ANALYSES Cak 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 crerating 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 cr plutonium may be used as a startup fuel. If chemical processing for Pa isolation and rare-earth reacval is omitted, the design has a conversion ratio of 0.8 to 0.9. The fuel cycle cost penalty for cperaticn 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 invertory + rare earths + ncble metals + fission products + rare gases + protactinium BFX70C056 Carlsmith FS + Lane JA POWER REACTORS FOR THE FUTURE -- AN EVALUATION Cak Bidge 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 varicus 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 ratic 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 ccsts + economics + fuel cycle costs + optimizations + natural resources BGX67C045 (Staff Report) SAFETY STUDIES FOR MSBR (PART 5 NUCL SAFEY FRCG ANN FECG **REFT 12/31/67**) Oak Ridge National Laboratory, Tenn.

Accession Number BFX700016 to EGX670045

Category B Reactor Analysis

BGX670045 *Continued* ORNL-4228 (April 1968), pr. 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 MSBR materials are compatible. *analysis + *design criteria + *MSBF + *safety + afterheat • compatibility + deposition + fission products + molten salts + reactivity + stability + dynamic characteristics

Category C Reactor Chemistry

CAX68C032 Thoma RE CHEMICAL FEASIBILITY OF FUELING MOLTEN SAIT FFACTORS 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 te made using the MSRE. Although separation of plutorium 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 MOLIEN MIXTURES OF LIF, BEF2, AND THF4 Cak Ridge National Laboratory, Tenn. ORNL-TM-2335 (Jan. 1969), 25 g. 9 fig, 7 ref. The Solubility of CeF3 was determined at varicus 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 sclvents 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 solutility data as a function of solvent composition were best correlated by a model that assumes BeF2 to be complexed as the BeF4(2-) icr and 1bF4 as the ThF5(1-) icn. *beryllium fluoride + *dissolving + *fluorides + *liquidus + *lithium flucride + *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 FC + Scott D +

Thoma RE

ASSESSMENT OF MOLTEN SALIS AS INTERMEDIATE COCLANTS FOR IMPERS

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 flucride, chloride, carbonate, nitrate-nitrite and fluoroborate salts. Chemical reactions that could cccur between scdium and flucroborates 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 + *fluoretorates + *liquid metals + *LMFBR + *NaK + *physical properties + *secondary salts + accidents + afterheat + applications + behavior + borch 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

> > Accession Number CAX690053 to CAX690061

Category C Reactor Chemistry

CAX690061 *Continued* extraction reprocessing of molten salt freeder reactor fuels. crystallization + data + fluorides + freezing + measurement + processing + separations CAX710023 Mailen JC + Smith FJ + Ferris IE SOLUBILITY OF FLUTONIUM TRIFLUORIDE IN MOLTEN 2 LITHIUM FLUCRIDE-BERYLLIUM FLUORIDE Cak Ridge National Laboratory, Tenn. J. Chem. and Eng. Data, 12 (Jan. 1971), 2 p, 1 fig, 7 ref. The solubility of plutonium trifluoride in molter 2 lithium flucride-beryllium fluoride was determined over the temperature range of 55C-660 deg C. The results can be expressed by the least-squares equation: log S(mole \$ plutonium trifluoride) = 3.2305 - 3096/1(deg K). The solid phase present at equilibrium was probably pure plutonium trifluoride. *molten salts + actinides + fluorides + *plutonium fluorides + *solutility + MSRE CCX680033 Kohn HW BUBBLES, DECES, AND ENTRAINMENT IN MOLIEN SALIS 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 rumr bowl. *bubbles + *entrainment + *fission products + *molten salts + *gas injection + *gas separation + *interfacial tensicn + *mists + *MSRE + *noble metals + *surface tension + beryllium fluoride + chemistry + circulation + cover cas + experiment + fissile materials + fluorides + foaming + fuels + gases + inert gases + lithiur fluoride + molybdenum + off-gas systems + rumps + sprays + viscosity + void fractions OTHER CATEGORIES: CFX + CJX CCX680038 Cantor S PHYSICAL PROPERTIES OF MSR FUEL, CODLANT, 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

Accession Number CAX690061 tc CCX680038

Category C Reactor Chemistry

CCX68C038 *Continued* conductivity, phase transition behavior, specific heat, heat of fusion, density, expansivity, compressibility, vapor pressure, surface tension, and gas sclubilities. *data + * fluorides + *fluoroborates + *physical properties + density + solubility + specific heat + surface tensicr + thermal conductivity + viscosity CDX670035 Malinauskas AP + Rutherford JI + Evans(III) RE GAS TEANSFORT IN MSRE MODERATOR GRAPHITE. I. REVIEW OF THEORY AND COUNTERDIFFUSION EXPERIMENTS Cak Ridge National Laboratory, Tenn.

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

The authors develop equations describing gas transport in porcus media. The experimental findings are limited but significant. Under MSRF conditions it appears guite justifiable to ignore normal diffusion effects in gas transport computations so that all the casecus-diffusion information necessary to correlate fission-productmigration 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 + concentratior +
deposition + design criteria + distribution + films +
fluids + inert gases + materials + measurement +
moderators + physical properties + rare earths + testing
CTHER CATEGORIES: EEX

CEX640018

Blankenship FF

EFFECTS OF RADIATION ON THE COMPATIBILITY OF MSRE MATEFIALS (PAET OF MSRE SEMIANN PROG REPT 7/31/64) Oak Bidge 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 MTE and later examined at CENL. Enhanced attack and other ancmalcus 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 frozer 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 CFX640018

Category C Reactor Chemistry

CEX640018 *Continued* recombination

CLX700010 Hautenreich PN FLUORINE FRODUCTION AND RECOMEINATION IN FROZEN MSR SAI1S AFTER REACTOR OPERATION Cak Ridge National Laboratory, Tenn. ORNL-TM-3144 (Sept. 1970) 36 F, 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/10C 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 guestions involved in storing and disposing of irradiated salt from the MSRF 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 surrourdings are kept at about 200 deg F.

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

CXX640020

Grimes WR

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

Cak 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 MSRF fuel and coolant salts are discussed.

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

CXX70C049 Grimes WB MOLTEN-SALT REACTOR CHEMISTRY Cak Ridge National Laboratory, Tenn. Nucl. Appl. Tech. 8, 137 (Feb. 1970) 19 p, 8 fig, 58 ref.

Accession Number CEX640018 to CXX700049

Category C Reactor Chemistry

CXX700049 *Continued*

Considerations leading to the choice of MSR fuel composition are discussed under the headings: Fhase Behavior Among Fluorides, Oxide Fluoride Phase Behavicr, MSRE and MSBR Fuel Compositions and Choice of Coolant. Physical properties of fuels and coolant are tabulated. In connection with the chemical compatibility of MSER materials, topics included are: Thermodynamic Data for Molten Flucrides, 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 Corresien Products and the Fission Separations chemistry is treated in terms cf Products. Separation of Protactinium and of Fission Froducts 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 + *corresion + *corresion products + *equilibrium + *fission products + *fluorides + *fluoroborates + *fuels + *graphite + *ESRF + *noble metals + *oxide precipitation process + *cxides + *protactinium flucrides + *serarations + actinides + beryllium fluoride + beryllium cxide + boron trifluoride + chromium + components + concentration + density + deposition + dissolving + experiment + fissile materials + gases + Hastelloy N + liquid metals + liquidus + lithium flucride + mass transfer + melting + mists + MSFE + 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 flucrides + uranium-235 + vapor pressure + viscosity + zirconium fluoride

Accession Number CXX700049 to CXX700049

Category E Graphite

EBX69C039 Greenstreet WL + Smith JE + Yahr GT + Valachovic RS THE MECHANICAL BEHAVIOR OF ARTIFICIAL GEAFHITES 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 lcading and unlcading. 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 Cak 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 econcmical enough for screening candidate materials. Thin disks of graphite are beated 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, cr 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 normal 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 + Barbancv VN + Kurakin VK + Zaitsev GG +

Strokov VI + Abrakhimuv U SHORT-TERM STRENGTH, CREEP, AND DUCTILITY CF GRAFHITE AT

Accession Number EBX690039 to EBX700042

EBX70C042 *Ccntinued*

300 TC 3500 DEG K

Not given.

LA-4462-TB (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. Ihe tensile and compressive properties depended in a complex fashion on grain size, temperature, and orientaticn; the coarse-grained material was weakest. At about 3000 deg K, the tensile behavior changed from brittle tc 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, Ukranian SSR, December 1967. creep + ductility + *graphite + experiment + *equipment +

microstructure + tensile properties + compressive properties

EBX70C043

Fontana A + Winand R

STUDY OF THE WETTABILITY OF GFAFHITE BY DIFFEFENT MCLTFN SCDIUM-FLUCRIDE-ZIRCONIUM TETRAFLUORIDE-ZIRCONIUM DIOXIDE MIXTURES IN THE PRESENCE OF VARIOUS GASECUS ATMOSEHEFES Universite libre de Eruxelles, 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 flucride 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 hct-stage microscope in the presence of different gases. The wettability of graphite by these mixtures become less as the ZIF4 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 2rC2 the change of atmosphere had no effect. CF4 and CO did not affear to have any significant influence on the wettability of graphite by the mixtures examined. Polished graphite surfaces were much less well wetted by NaF-2rF4-2rC2 mixtures than machined graphite surfaces. (auth)

wetting + graphite + cxides + molten salts

ECX710011

Chang SJ + Carpenter JA + Altom DW VISCOELASTIC ANALYSIS OF IRRACIATED GRAPHITE WITH VARIABLE CREEP COEFFICIENI Cak Ridge National Laboratory, Tenn.

Accession Number EBX700042 to ECX710011

Category E Graphite

ECX710011 *Continued*

ORNL-TM-3242 (May 1971), 31 p, 5 fig, 3 ref. This report is an addendum to ORNI-TM-2407 ccrcerning 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. Tc 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 It is shown that the problem reduces to the rate. solution of several associated (fictitious) elastic problems which have a common elastic modulus inversely propertional 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. computer program is written for the purpose and can include the previous solution as a special case. stress + analysis + graphite + radiation damage + elasticity + MSER + creep

EDX640016

Cook WH

MSRE GRAPHITE (IN MSRP PROGR. REPT., 7/31/64) Cak Ridge National Laboratory, Tenn.

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

The graphite purchases 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 ingregnations 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 x 10(20th) to 1.40 x 10(20th) 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 ECX7 100 11 to FIX640016

Category E Graphite

EDX640016 *Continued* procurement + reviews + specific heat + specifications + thermal conductivity + heat treatments + tensile properties + flexural properties + compressive properties + microstructure CTHER CATEGORIES: ACE EDX68C031 Kasten PR + Bettis ES + Cook WH + Fatherly WP + 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 + Ccck WH + Eatherly WF + Holmes DK + Kedl RJ + Kennedy CR + Kirslis SS + McCcy HE + Perry AM + Robertson RC + Scott I + Strehlow FA GRAPHITE BEHAVIOR AND ITS EFFECTS CN MSER FERFORMANCE

Cak 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 product 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 CBNL-IM-2316.

Accession Number EDX640016 to EfX690051

Category E Graphite

EDX69C051 *Ccntinued* *MSBR + *graphite + *reviews + irradiation + mechanical properties + creep + development + breeding performance + xenon

EXX70C048

Engle GB + Frice RJ + Bokros JC + White JL RESEARCH ON GRAPHITE -- THREE-YEAR SUMMARY FEFCET 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 Fyrolysis 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

Accession Number EDX690051 to EXX700048

Category F Hastelloy N and Related Allcys

FAX620004

DeVan JH + Evans(III) RB

CORROSION BEHAVIOR OF REACTOR MATERIALS IN FLUCRICE SALT MIXTURES

Oak Ridge National Laboratory, Tenn.

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

The report discusses (1) corresion experiments dealing with fluoride salts in support of the ESRE, 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 corresion testing of commercial alloys and in fundamental interpretations of the corrosion process, ORNL developed a high-strength nickel-base alloy containing 17% to, 7% Cr, and 5% Fe. Several long-term corrosion loops and in-pile carsule tests completed with this alloy demonstrate the excellent corrosion resistance to fluoride salt mixtures at high Thermodynamic methods are presented for temperatures. predicting corrosion rates. Radiotracer studies confirmed the corresion model. Also published as pp. 557-579 in Corrosion of Peactor Materials, IAEA, Vienna, 1962, Vol. II.

alloys + compatibility + *corrosion + *development + fluorides + *Hastelloy N + locp + nickel + fuels + thermodynamics + molten salts OTHER CATEGORIES: FBD

FAX620005 DeVan JH + Evans(III) PB CORROSICN EEHAVIOR OF REACTOR MATERIALS IN FLCORIDE SALT MIXTURES Cak 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 OdNL-1M-328 (FAX6200C4). alloys + compatibility + *corrosion + *development + fluorides + *Hastelloy N + locp + nickel + fuels + thermodynamics + *molten salts OTHER CATEGORIES: FBD

FAX690035 DeVan JH EFFECT GF ALLCYING ACDITICNS ON CORROSION BEHAVIOR OF NICKEL-MOLYBDENUM ALLOYS IN FUSED FLUCRINE MIXTUFES

Accession Number FAX620004 to FAX690035

Category F Hastelloy N and Related Allcys

FAX690035 *Continued*

Cak Ridge National Laboratory, Tenn.

ORNL-IM-2021 Vcl. 1 (May 1969), 45 p, 13 fig, 16 ref. Corrosion properties of nickel-molybdenum alleys 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 K. 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 crder: 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 chronium. A nickel-base allcy containing 17% Mo, 7% Cr, and 5% Fe, designated Hastelloy N, had the test combination of strength and corrosion resistance among the compositions tested.

alloys + compatibility + corresion + development + fluorides + fuels + molten salts + Hastelloy N + lccp + nickel allcys + allcy composition CTHER CATEGORIES: FEC

FAX69C045

DeVan JH

EFFECT OF ALLOYING ADDITIONS ON CORFOSION BEHAVIOF OF NICKEI-POLYBORNUM ALLOYS IS FUSED FLUORIDE MIXIURES Oak Ridge National Laboratory, Tenn.

ORNL-TM-2021 Vol. 1 (May 1969), 45 p. 13 fig, 16 ref. Corresier 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 tc 20% Mc and various percentages of Cr. Al, Ti, V, Fe, No, 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 5CC 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 nickelbase alloy containing 17% Mo, 7% Cr, and 5% Fe, desigrated Hastelloy N, had the best combination of strength and corrosion resistance among the compositions tested. compatibility + corresion + development + fluorides + fuels + Hastelloy N + loop

OTHER CATEGORIES: FBD

FBA660020

Accession Number FAX630035 to FEA660020

Category F Hastellcy N and Belated Alloys

FBA660020 *Continued* McCoy HE

STUDIES OF THE CARBON DISTRIEUTION IN HASTELLCY N Oak Ridge National Laboratory, Tenn. ORNL-TM-1353 (Feb. 1966), 24 p, 14 fig, 6 ref.

A small heat of Hastellcy 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 cculd be correlated with the observed changes in the carton Although marked segregation resulted, distribution. 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 sclubility of rolybdenum i n nickel so that the precipitate is basically nickel-mclybdenum intermetallic compounds. *heat treatments + welding + *metallography + *Hastellcy 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-Terperature Allcys Committee, Micromet Latoratories, West Lafayette, Ind., 21 p, 14 fig, 0 ref.

Though Hastelloy N is tasically a solid-solution alloy, thermcmechanical 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 grainboundary films on extraction replicas without interference from the matrix. The microstructure is characterized by stringers of massive primary precipitates of the Ni3Mc3C type. Exposure between 500 and 1000 deg C precipates 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, carlides go into solid solution. The only precipitates that form in

Accession Number FBA660020 to FEA680029

Category F Hastellcy N and Belated 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-NiKo 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.

CRNL-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 coclart salt on the tube side of the heat exchanger. The plugs had a slight interference fit (0.0000 to C.CCC2 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 cave 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 PRCFERTIES CF WELES IN A Ni-Mo-Cr-Fe ALLCY 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-11C-s (1966) AC FBB660022. *Hastelloy N + *welding + heat treatments + creep + ductility + tensile properties CTHER CATEGORIES: FEC

FBB66C022 Gilliland FG + Venard JT ELEVATED TEMPERATURE MECHANICAL PROFERTIES OF WELCS IN A

Accession Number FBA680029 to FBB660022

Category F Hastelloy N and Related Allcys

FBB66C022 *Ccntinued*

Ni-Eo-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 The contents of this paper also appear as temperature. CRNI-TM-1341, AC-FEE660022.

*Hastelloy N + *welding + heat treatments + creep +
ductility + tensile properties
OTHER CATEGORIES: FBC

FBB690040

McCoy HE + Cancnico DA

PREIRRACIATION AND POSTIRRADIATION MECHANICAL PROPERTIES OF HASTELLOY N WELDS

Cak Ridge National Laboratory, Tenn.

ORNL-IM-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-fatricated 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-FBE690041.

*Hastelloy N + *welding + creep + ductility + *irradiation + tensile properties CTHER CATEGCFIES: FEE

FBB69C041 McCoy HE + Canonico DA

Accession Number FBB660022 to FEE690041

Category F Hastelloy N and Related Allcys

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 (FBE690040) 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 FBB70C028 McCoy HE + Gunkel RW + Slaughter GM TENSILE PROPERTIES OF HASTELLCY N WELDED AFTEF IFFAILATION Cak 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 x 10 (20th) neutrons/sq-cm. All cf the unirradiated samples and 67% of the irradiated samples were satisfactorily welded by a specialized technique. Sufface 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 defcrmed 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 dec 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 fusicr 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 + Canchicc DA THE MEASUREMENT OF RESIDUAL STRESSES Oak Ridge National Laboratory, Tenn. ORNL-TM-3113 (Cct. 1970) 32 p, 12 fig, 16 ref. A mcdification 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. ir diareter 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

Accession Number FBB690041 to FEE700031

Category F Hastellcy 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, Chio BMI-1354 (June 1959), 16 p. 8 fig, 1 ref.

The temperature and frequency dependence of fatigue properties of Hastelloy N were determined by rotatingbeam 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 + fatique

FBC61C001

Swindeman FW

THE MECHANICAL PROPERTIES OF INOR-9

Cak Ridge National Laboratory, Tenn.

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

Tensile, creep, and relaxation tests were perferred or 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 cut where additional information is desirable.

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

FBC64C017 McCoy HE

Category F Hastellcy N and Related Alloys

FBC640017 *Continued*

INFLUENCE OF SEVERAL METALLURGICAL VARIABLES ON THE

TENSILE FROPERTIES OF HASTELLOY N

Oak Ridge National Laboratory, Tenn.

ORNL-3661 (Aug. 1964), 63 p, 35 fig, 10 ref. The tensile properties of Hastelloy & were measured after various heat treatments. One vacuum-melted and four airmelted 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 110C to 120C deg F range material that had been previously annealed at 2150 deg F significantly reduced the ductility. The se 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 fcrmation of this layer is associated with the presence of trace alloying elements.

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

FBC65C017

Venard JT

TENSILE AND CREEP PROPERTIES OF INCE-8 FOR THE MOLTEN-SALT REACTOR EXPERIMENT

Oak Ridge National Laboratory, Ienn.

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 these used for the The primary aim was to collect strength information MSRE. 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 P). Creef-rufture behavior was investigated at 493, 7C4, 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 crder.

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

FBD690036 Koger JW + Litman AP

Accession Number FBC640017 to FBD690036

Category F Hastelloy N and Related Allcys

FBD69C036 *Continued*

COMPATIBILITY CF HASTELLOY N AND CROLOY 9M WITH NaBF4-NaF-KBF4 (90-4-6 mole %) FIUCFCEORATE SALT Cak Ridge National Laboratory, Tenn. ORNL-TM-2490 (Apr. 1969), 41 F, 20 fig, 15 ref.

The compatibility of relatively impure (greater than 3000 ppm impurities) NaBF4-NaF-KEF4 (90-4-6 mole %) tested with Hastelloy N and Crcloy 9M was tested in natural circulation loops at a maximum temperature of £C5 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-guarters 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 fluctobcrate salt at 460 deg C. Initially, soluble metal flucride compounds formed in the hot leg. The reverse reaction in the cold leg causes the metal to deposit and to diffuse into the cold leq. 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 cr the equilibrium constant of the reaction changes so much with temperature that the pure metal is deposited.

*iron allcys + *Hastellcy N + compatibility + *corrosion +
*fluoroborates + *impurities + mass transfer + sclubility +
coolants + thermal convection + molten salts
CTHER CATEGORIES: GEX

FBE650015

Martin WR + Weir JR

EFFECT OF ELEVATED TEMPERATURE IRRADIATION CN THE STBENGTH AND DUCTILITY OF THE NICKEL-EASE ALLOY HASTELLOY N

Oak Ridge National Laboratory, Tenn.

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

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

#Hastelloy N + ductility + *tensile properties + irradiatior OTHER CATEGORIES: FBC

FBE650016

Martin WR + Weir JR

EFFECT OF ELEVATED TEMPERATURE IRRADIATION ON THE STRENGTH AND DUCTILITY OF THE NICKEL-BASE ALLCY HASTELICY N Cak Bidge National Laboratory, Tenn,

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

The tensile properties of Hastellcy N have been determined after irradiation at 600 deg 2 to 7 x 10(20th) n/sq-cm (E greater than 1 MeV) and 9 x 10(20th) n/sq-cm (thermal). The

Accession Number FBD690036 to FBE650016

Category F Hastelloy N and Related Allcys

FBE650016 *Ccntinued*

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. Euctility, 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. Fost 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 + irradiatior OTHER CATEGORIES: FBC

FBE660019 Martin WR + Weir JR PCSTIFRADIATION CREEP AND STRESS RUPTURE OF HASTELLCY N Oak Ridge National Laboratory, Tenn. ORNL-TH-1515 (June 1966) 31 p, 12 fig, 15 ref. The contents of this IN are the same as an article with the same title in the March 1967 Nuclear Arrlications, which see. #Hastelloy N + *creep + ductility + irradiation + microstructure CTHER CATEGORIES: FEC FBE67C029 Martin WR + Weir JR POSTIRRADIATION CREEP AND STRESS RUFTURE OF HASTELLCY K Cak Ridge National Laboratory, Tenn. Nucl. Appl. 3, 167 (Mar. 1967), 11 p, 10 fig, 17 ref. The creep ductilities of irradiated Hastelicy N at 650 deg C have been determined at several neutron exposures. Elevated-temperature irradiation emtrittlement greatly reduces the stress-rupture strength as measured in postirradiation uniaxial stress tests. The reduction in ductility to values as low as C.4% is due to an irradiation effect related to the process of intergranular fracture. Intergranular cracks, once formed, propaga te with greater ease in the irradiated alloy as compared with a sample exposed to a lesser radiation exposure. (Also published as CRNL-TM-1515.) *Hastelloy N + *creep + ductility + irradiation + microstructure OTHER CATEGORIES: FBC FBE670030 McCoy HE + Weir JR

IN- AND EX-REACTOR STRESS-RUPTURE PROPERTIES OF HASIELLOY N TUBING Oak Ridge National Laboratory, Tenn.

Accession Number FBE650016 to FEE670030

Category F Hastellcy N and Related Alloys

FBE670030 *Continued*

ORNL-TM-1906 (Sept. 196p), 27 p, 14 fig, 28 ref.

The stress-rupture properties of two heats of Hastelley 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 FEE68CC25).

Hastelloy N + creep + ductility + irradiation OTHER CATEGORIES: FBC

FBE670031

McCoy HE

AN EVALUATION OF THE MOLTEN SALT REACTOR EXPERIMENT HASTELLOY N SURVEILLANCE SPECIMENS -- FIRST GROUF Cak 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 x 10(20th) neutrons/sg-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

Cak 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 CTHER CATEGORIES: FEC

FBE68C026 MCCOY HE EFFECTS OF IRRADIATION ON THE MECHANICAL PROPERTIES OF TWO VACUUM-

Accession Number FBE670030 tc FEE680026

Category F Hastellcy N and Related Alloys

FBE680026 *Continued*

MELTED HEATS OF HASTELLCY N Cak 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 x 10 (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 creef 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 helitm 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 GEOUE Cak Ridge National Laboratory, Tenn.

ORNL-IM-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 x 10(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 CRE. Two rods of modified Hastelloy N containing small additions of titanium and zirconium from the core with a thermal fluence of 4.1 x 10(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 + creef + rupture + ccrrosion + irradiation + irradiation + tensile properties + microstructure OTHER CATEGORIES: FCE

FBE690044 McCoy HE

Category F Hastelloy N and Related Allcys

FBE69C044 *Ccntinued*

VARIATICN OF THE MECHANICAL PROPERTIES OF IRRADIATED HASTELLOY N WITH STRAIN RATE

Cak 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 x 10(20th) neutrons/sq-cm. At strain rates normally encountered in tensile tests, the fracture strain is guite sensitive to strain rate in the range of 500 to £50 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 dec 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 + allcy composition
OTHER CATEGORIES: FBC

FBE700027

McCoy HE

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

Cak Ridge National Laboratory, Tenn.

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

We examined the third group of hastelloy N surveillarce samples from the MSRE. Standard Hastelloy N was exposed in the core to a thermal of 9.4 x 10(20th) neutrons/sg-om over 15,289 hr at 650 deg C and outside the reactor vessel to 2.6 x 10(19th) neutrons/sg-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 2n 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 x 10(19th) to 9.4 x 10(20th) neutrons/sq-cm. TWO heats of modified Hastelloy N were irradiated in the core to a thermal fluence of 5.3 x 10 (20th) 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 FEE700027

Category F Hastellcy N and Related Alloys

FBE700027 *Continued* OTHER CATEGORIES: FCE

FBE710017

McCoy HE + Gehlbach RE

INFLUENCE CF 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 65C deg C, the postirradiation properties were equivalent to those of the air-melted material. Irradiation at about 76C deg C

inferior postirradiation properties. creep + ductility + Hastelley N + irradiation + thermal effects + alloy composition + microstructure + carbides CTHER CATEGORIES: FOR

FBE710018

McCoy HE

AN EVALUATION OF THE MOLTEN-SAIT REACTCE EXFERIMENT HASTELLCY 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 x 10(21) neutrons per square centimeter, and a fast fluence (greater than 50 keV) of $1.1 \times 10(21)$ 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. Sche heats with modified composition have been exposed to the core of the MSRE and show improved resistance to irradiation. Ine corresion of the Hastelloy N has been largely due to the selective removal of chronium. 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 hands that were formed during

Accession Number FBE700027 to FEE710018

Category F Hastellcy N and Related Alloys

FBE710018 *Continued* machining. modified Hastelloy N + MSRF + surveillance + corrosicn + creep + ductility + fluorides + Hastelloy N + irradiation + alloy composition + molten salts + microstructure + tensile properties OTHER CATEGORIES: FEC + FBC + FCE FBX640015 Taboada A METALLURGICAL DEVELOPMENTS (IN MSRF FRCGR. BEIT., 7/31/64) Cak Ridge National Laboratory, Tenn. ORNL-3708 (Nov. 1964), pp. 330-372, 27 fig, 8 ref. Metallurgical developments in support of the MSFF show that Hastelloy N is satisfactory. Physical, tensile, cree; and fatigue properties are given. Compatibility is excellent with circulating molten fluorides, graphite ir wolten 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 gadoliniaalumina 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 CTHER CATEGORIES: ACE FCC 69 CO 48 Sessions CE + Lundy TS DIFFUSION OF TITANIUM IN MODIFIED HASTELLOY N Cak Ridge National Laboratory, Tenn. ORNL-IM-2392 (Jan. 1969), 24 F, 7 fig, 13 ref. Diffusion coefficients of titanium-44 in titarium-modified Hastelloy N were determined over the range 800 to 1250 deg C by serial sectioning by latheing cr 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/FT) sq-cm/sec. Results were used to predict the maxirum 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 Hastellcy N and Belated Alloys

FCC690048 *Continued*

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

diffusion + titanium + modified Hastelloy N

FCC69C049

Sessions CE + Lundy TS

DIFFUSION OF TITANIUM IN MODIFIED HASTELLOY N Cak Ridge National Laboratory, Tenn.

J. Nucl. Mater. 31, 316 (July 1969), 7 p, 5 fig, 13 ref. Diffusion coefficients of titanium-44 in titariummodified Hastellcy N were determined over the range of 800 to 1250 deg C ty serial sectioning by latheing cr grinding and counting by gamma-spectroscopy. The data were fitted to D = (15.3 plus or minus 2.2) exp(-73,CCC plus or minus 33CO/RT) 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 CFNL-TM-2392. diffusion + titanium + modified Hastelloy N

FCC70C040

Sessions CE

INFLUENCE OF TITANIUM ON THE HIGH-TERFEFATURE DEFCEMATION AND FRACTURE FEHAVIOR OF SOME NICKEL BASED ALLOYS (THESIS) Oak Ridge National Laboratory, Tenn.

CRNL-4561 (July 1970), 189 p, 44 fig, 88 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 varicus stresses. Fracture at 650 deg (changed from intergranular at low carter to mixed transand 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 Loundaries 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 +

Accession Number FCC690048 to FCC700040

Category F Hastelloy Nand Related Allcys

FCC70C040 *Continued* *alloy composition OTHER CATEGORIES: FCA

FCC700044 Sessions CE

INFLUENCE OF TITANIUM ON THE HIGH-TEMPERATURE DEFORMATION AND FRACTURE BEHAVIOR OF SCHE NICKEL-EASED ALLOYS Cak Ridge National Laboratory, Tenn.

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

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

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

FCC710010

Sessions CE + Stansbury EE THERMAL STABILITY OF TITANIUM-MODIFIED HASTELLOY N AT 650 and 760 DEG C

Cak 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 cf 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 preciritation heat treatments were conducted at 650 and Titanium increases the stability of a complex 760 deg C. 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 cn 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 cr 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

FCD71C016

Evans (III) FE + Koger JW + DeVan JH CORROSION IN POLYTHERMAL LOOP SYSTEMS II. A SCLID-STATE DIFFUSION MECHANISM WITH AND WITHOUT LIQUID FILM EFFECTS

Accession Number FCC700040 to FCD710016

Category F Hastellcy N and Related Alloys

FCD710016 *Continued*

Oak Ridge National Laboratory, Tenn.

ORNL-4575, Vol. II (June 1971) 74 p, 16 fig, 49 ref. The corresion of alloys exposed to nonisothermal circulating liquids is important in systems with liquid coolants or coclant-fuel combinations. Mathematical descriptions were developed to explain transport of constituents of allcys. This report specializes to cases in which solid-state diffusion in the alloy deminates the corresion. 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 hct-tc-ccld-zone transfer of nickel in Inconel 600 pumped loops circulating socium. Actual corrosion is much higher than predicted; this suggests that the true corrosion reaction overrides a slow diffusior process. The second system is transfer of chromium in Hastelloy N thermal convection locps with molten salt. Three examples are considered; (1) corrosion at all points, transfer to salt only; (2) hot-tc-cold-zone trarsfer; 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.

*corrcsion + *models + *diffusion + analysis +
computer codes + coolant loops + fluorides + Hastellcy 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 CAMAGE Gak Bidge National Laboratory, Tenn. Irradiation Effects in Structural Alloys for Thermal and Fast Beactors, 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-terperature mechanical properties of Hastelloy N are generally that the creep-rupture life and ductility are reduced. Ine 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 cn specimens irradiated to a thermal fluence of 5 x 10(20th) 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 Allcys

FCE690043 *Continued*

0.5% for the standard alloy. In-reactor creef-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 IBRADATION DAMAGE AT HIGH TEMPERATURE - FHASE I Cak 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 numercus, small, laboratory melts and several 100-lb melts from commercial Additions of Ti, Zr, and Hf isproved the vendors. properties of the alloy both unirradiated and after Irradiation temperature had a marked effect irradiation. 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 + Rennedy CF + Litman AF + Gehlbach RE + Sessions CE + Kcger JW MATERIALS FOR MOLTEN-SALT REACTORS Cak 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 Hastellcy N + alloy composition + mechanical properties + sealing + fluoroborates + corrosion + compatibility + iron alloys + *reviews + progress report CTHER CATEGCRIES: EIX + FCX

FCX70C026 McCoy HE + Beatty RL + Cook WH + Gehlbach RE + Kennedy CR + Koger JW + Litman AP + Sessions CE + Weir JF NEW DEVELOFMENTS IN MATERIALS FOR MOLTEN-SALT REACTORS Oak Ridge National Laboratory, Tenn. Nucl. Appl. Tech. 8, 156 (Feb. 1970), 14 p, 13 fig, 34 ref.

Accession Number FCE690043 to FCX7CCC26

Category F Hastelloy N and Related Allcys

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 coclant 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 GRNL-TX-2511.)

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

FXX69C047 McCoy HE THE INOR-E STORY Cak 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. Mcre recent studies of microstructure and mechanical properties as influenced by irradiation show the need for reducing siliccn and molybdenum and adding small amounts of titanium, hafnium, and niotium. reviews + Hastelloy N + mcdified Hastelloy N + development + alloy composition

OTHER CATEGORIES: FCX

Accession Number FCX700026 to FXX690047

Category G Materials Other than Hastelloy N and Graphite

GAX670033

Stiegler JO + Weir JR

EFFECTS OF IBRADIATION ON CUCTILITY

Oak Ridge National Laboratory, Tenn.

Chap. 11, p. 311 in Euctility, Papers Presented at a Seminar of the American Society for Metals Oct 14-15, 1967, ASM, Metals Fark, 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 CRNI-TM-2019, AC-GAF680028.

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

GAX680028

Stiegler JO + Weir JR EFFECTS CF IRRADIATION ON EUCTILITY 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 Cot 14-15, 1967, ASM, Metals Park, Chio, 1968, AC-GAE670033.

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

GAX700045

Koger JW + Litran AP

CATASTROPHIC CCRRCSION OF TYPE 304 STAINLESS STEEL IN A SYSTEM CIRCULATING FUSED SODIUM FLUCECBGEATE Cak Ridge National Laboratory, Tenn.

ORNL-IM-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 Hascelloy N and Graphite

GAX70C045 *Ccntinued*

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

Adamson GM + Crouse RS + Manly WD INTERIM REPORT ON CORROSICN BY ZIRCCNIUM-BASE FLUCFIERS Cak 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 hotleg temperature of 1500 deg F. In Inconel-600 loops, subsurface voids were formed ty 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 + McElrcy DL ELECTFICAL RESISTIVITY ANOMALY IN NICKEL-BASE ALLOYS Oak Ridge National Laboratory, Tenn. Trans. ASB (Am. Soc. Metals) 61, 730 (Dec. 1968), 12 F. 16 fig, 17 ref. The electrical resistivity of eight nickel-base allcys containing ircn, chromium, and molybdenum was measured to Alloys with more than 50 wt % Ni showed a rapid 1000 C. 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 6CC C. The effects cf annealing and cold working were relatively minor for these

Accession Number GAX700045 to GCX68003C

Category G

Materials Other than Hastelloy N and Graphite

GCX68C030 *Ccntinued*

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 ALLCY TZM WITH LITEIUM FLUCRIDE-EERYLLIUM FLUORICE-THORIUM FLUORIDE-URANIUM(IV) FLUCRIDE (68-20-11.7-0.3 mcle %) AT 1100 deg C Cak Bidge National Laboratory, Tenn. ORNL-TM-2724 (Dec. 1969), 16 F, 2 fig, 9 ref.

The TZM alloy (Mo-0.5% Ti-0.08% 2r-0.02% C) showed very little attack by the fused salt (LiF-BeF1-ThF4-UF4, 68-20-11.7-0.3 mole %) at 1100 deg C for 1011 hr. Corrosion manifested itself as leaching of titarium 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 + molytdenum + prccessing + distillation + equipment + capsules + MSBR + molten salts

GDX710025

Nicholson EL

CCNCEFTUAI EESIGN AND DEVELOPMENT PROGRAM FOR THE MOLYBDENUM REDUCTIVE EXTRACTION EQUIPMENT TEST STANE Cak Bidge 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

Accession Number GCX68003C tc GDX710C25

Category G Materials Other than Hastelloy N and Graphite

GDX710025 *Ccntinued*

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 + *fatrication + *bismuth + molten salts + *test facilities + MSBE + development + plans + materials OTHER CATEGORIES: LDB

GFX660023

Tolson GM + Taboada A

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

Cak Ridge National Laboratory, Tenn.

ORNL-IM-1437 (Apr. 1966) 19 p, 10 fig, 5 ref. Thermal-convection loop tests of several structural alloys were cperated using circulating molten lead. Screening tests included carbon steel Letween 900 and 1100 deg F. type 410 stainless steel between 910 and 1210 deg F, Croloy 2-1/4 under both conditions, and nicbium 1% zirconium between 1000 and 1400 deg F. Iwo loors 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 flug in the cold regions because dentritic 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 NE-1% Zr alloy showed no measurable attack; however, nicbium crystals were found in the ccld 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 CONTECL ELEMENTS: MANUFACTURE, INSPECTION, DRAWINGS, AND SPECIFICATIONS Cak Ridge National Laboratory, Tenn. ORNL-4123 (July 1967) 53 r, 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*

Gd2C3-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 + incomels + *fabrication + *control rods +
ceramics

GXX68C039

Metzger GE

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

Wright-Patterscn Air Force Base, Chio.

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

HAX70C050 Kedl FJ FLUID DYNAMIC STUDIES OF THE MOITEN-SAIT BEACTOR EXFERIMENT CORE Oak Ridge National Laboratory, Tenn. ORNL-TM-3229 (Nov. 19, 1970), 33 p, 16 fig, 1C 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 mcdel, 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 DEVELCEMENT 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 MSFE. 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-SAIT THERMAL BREEDER REACTORS Oak Ridge National Laboratory, Tenn. CRNL-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 sclten-salt systems. The short-shaft pump is favored for the coolant-salt system because of reliability, the long

Accession Number HAX700050 to HEX670042

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 FUMF TEST STAND FOR THE MOLTEN SALT ERFEDER EXPERIMENT Oak Ridge National Laboratory, Tenn.

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

A stand is required to test the salt pumps for the Molten Salt Ereeder Experiment (MSBE). It will be designed to accommodate pumps having capacities rancing 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 beat 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

PRELIXINARY SYSTEMS LESIGN DESCRIPTION (TITLE I DESIGN) OF THE SALT PUMF TEST STANE FOR THE MOLTEN SALT BREEDER EXPERIMENT

Cak Ridge National Laboratory, Tenn.

ORNL-IM-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 + *MSEE + *pumps + *test facilities +
design + development + molten salts + plans +
quality assurance

HBX70C012

Smith PG

DEVELOPMENT OF FUEL- AND COOLANT-SALT CENTFIFUGAL FUMES FOR THE MSRE

Oak Ridge National Laboratory, Tenn.

CRNL-TM-2987 (Cct 1970), 50 p, 18 fig, 15 ref.

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

Accession Number HBX67C042 to HEX7CCC12

Category H Reactor Component Development

HBX70C012 *Ccntinued*

The fuel pump delivers 1200 gpm and the coclant 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 rever 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

CTHER CATEGORIES: MAE

HCX68C037

Kedl FJ + 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 Mclten 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 + *ESRE +
experience + hydraulics + testing + vibration
OTHER CATEGORIES: MAB

HCX710022

Bettis CE + Crcwley 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. Thε programs are: for the primary heat exchangers, PRIME1; for the reheaters, RETEX; and for the steam generatorsuperheaters, 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 ir this report. The programs developed were used in designing the four 556 MW primary exchangers, eight 36.6 MW reaheaters, and sixteen 121 MW steam generator superheaters. All are basically baffled shell and tuke exchangers; the stear 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

Accession Number HBX700012 to HCX710022

OMENCIALAL

Category H Reactor Component Development

HCX710022 *Ccntinued* programs as well as example computer input and cutput 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: HEX HFX62C007 Richardson M DEVELOPMENT OF FREEZE VALVE FCR USE IN ESRE Cak Ridge National Laboratory, Tenn. OBNL-IM-128 (Feb. 1962), 24 r, 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 10C 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 Cak Ridge National Laboratory, Tenn, ORNL-IM-1623 (Sept. 1966) 27 r, 9 fig, 3 tab. Various filter and adsortent materials were examined for possible use in the removal of oil mists and hydrocarton A controlled flow of oil was injected into a heated vapors. nickel reaction vessel to cause vaporization and some cracking of the oil. Helium flowing through the reaction vessel carried the cil mist and hydrocarbon vapor through a filter system. Filter effectiveness was determined by the use of a combination of felted metal fibers and ceramic fibers in a configuration procosed for use in the MSRE. Granulated charcoal removed hydrocarbon vapors (C-6 and above) in a manner consistent with the ϵ stablished adscrption isotherms for this material. *development + *filters + *off-gas systems + adsorption + charceal + components + filtration + hydrocarbons + materials + mists + MSRE + testing HXX64C019 Scott D COMPONENT DEVELOPMENT IN SUPPORT OF MSRE (FART OF MSRF SEEIANN FROG BEPT 7/31/64)

Accession Number HCX710022 to HXX640C19

Category H Reactor Component Development

HIX640019 *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 maintainatility 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 DB + Van Winkle B

MOLTEN-SALT CONVERTER REACTOR -- DESIGN STUDY AND COST 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 + Eauman BF + Carter WL + McDcnald WB + Rcbertson 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 MSBB 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 + thcrium IAB67C043 Briggs RB EFFECTS OF IRRADIATION ON THE SERVICE LIFE CF THE

MOLTEN-SALT FEACTOR EXPERIMENT Oak Ridge National Laboratory, Jenn.

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

Thermal neutron irradiation adversely afrects the high-temperature stress-rupture life of the Hastelley that was used in the MSBE. 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

Continued IAB670043 would have a minimum service life of 20,00C hr. *analysis + *Hastelloy N + *MSRE + *operation + *radiation damage + *stress rupture + design + limits + plans + reactor vessel OTHER CATEGORIES: MAA IAC66C024 Kasten FR + Bettis ES + Eauman HF + Carter WL + HcDcnald WB + Rcbertson RC + Westsik JH SUMMAFY OF MOLTEN-SALT ERFECER REACTOR DESIGN STUDIES Oak Ridge National Laboratory, Tenn. ORNL-TM-1467 (March 1966), 31 p, 7 fig, 11 talles, 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 MK(e) capacity. A much more detailed report on the same studies was subsequently published as ORNL-3996, MSRIS accession IAC660025, which see. *MSBR + *conceptual design + *performance + *power ccsts + reactors + containment + structures + molten salts + processing + neutron physics + *two-fluid reactor IAC660025 Kasten PR + Bettis ES + Robertson RC DESIGN STUDIES OF 1000-MW(e) ECITEN-SALT BREELER FEACTCES Cak Ridge National Laboratory, Tenn. ORNL-3996 (Aug. 1966), 150 p, 43 fig, 52 tables, 3C 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 ccolant salt transports the heat from the primary heat exchangers to steam generators and reheaters. The reference design fuel salt is LiF-BeF2-UF4 (68.3-31.2-0.5 mole %), the blanket salt is

LiF-ThF4-BeF2 (71.0-27.0-2.0 $mcl \in \Re$), and the ccclant salt is NaF-NaBF4 (61.1-38.9 mole %). On-site fuel recycle processing was assumed, with flucride 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/NW(e), the fuel doubling time about 13 yrs and the estimated fuel-cycle ccst is 0.35 mills/kWhr. General flowsheets and conceptual designs for the reactor, primary heat exchangers, salt circulating purps and steam generators are presented. Cost and performance estimates are also given. Several alternate designs are triefly described: (a) a modified primary heat exchanger design: (b) a system using 580 deg F rather than 7CC deg F feedwater; (c) a modular concept using four small reactors rather than one large reactor; (d) an MSBR crerating without Pa removal in the chemical processing plant; (e) a concept

Accession Number IAB670043 to IAC660025

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 singleregion, 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 FS TWO-FLUID MOLTEN-SALT BREEDER REACTOR DESIGN STULY (STATUS AS CF JANUARY 1, 1968).

Oak Ridge National Laboratory, Tenn.

CRNL-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-LeF2 (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 sufficien 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 treeding ratic cf 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 rcds + drain tarks +
heat exchangers + structures + steam generators + pumps +

Accession Number IAC660025 to IAC700047

Fage 108

Category I Reactor Design

IAC70C047 *Ccntinued*

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(ϵ) 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 0-233 carried it a LiF-BeF2-ThF4 mixtume 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 PWB 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 ccst 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

IAC710013

Robertson FC (editor) CONCEPTUAL DESIGN STUDY OF A SINGLE-FLUID ECLTEN-SAIT BREEDER FEACTOR 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-BeF2-ThF4-UF4 (71.7-16.C-12.C-(.3 mole %). The salt is pumped through a 22-ft diam x 2C ft high graphite-moderated and reflected reactor vessel and then through primary heat exchangers where it is cooled

Accession Number IAC700047 to IAC710013

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 Hastellcy N improved by additives to increase the resistance to irradiation damage. Iritium, Xe and Kr are sparged from the circulating fuel salt by helium An off-gas system removes the fission-products bubbles. for storage and decay and recycles the helium. A 1-grm 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, NaEF4-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(ϵ), the breeding ratio is 1.06 and the annual yield is abcut The net thermal efficiency is 44%, and the 3.3%. 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 Hastelley N behavier under irradiation, suitability of coolant salt, maintenance procedures, and the behavior of fission-product particulates.

*MSBR + *conceptual design + *performance + *jower ccsts + *capital equipment + coolants + physical properties + graphite + Hastelloy N + reactors + cores + control rcds + drain tanks + heat exchangers + structures + pumps + steam generators + steam systems + cff-gas systems + containment + neutror flux + processing + breeding performance + fission products + ncble 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) MSBB Cak Ridge National Laboratory, Tenn.

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

Controlling tritium migraticn 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 Incolcy could be used. The major expenses associated with the modification are the costs of the additicral 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 (abcut \$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 ccsts + conceptual design + loop + coolants + heat exchangers + pumps + power costs +

fuel cycle costs + steam systems

IAD700052

Bettis ES + Bauman HF

MOLTEN-SAIT CONVERTER REACTORS

Oak Ridge National Laboratory, Tenn. Power Engrg. Vol. 74 No. 8, 42 (Aug 1970) 3 F, 2 fig. Development of rapid fuel-salt processing and longer-lived graphite is needed before molten-salt breeder reactor power staticns 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 ccst would be only about 0.8 mills/kwh and the construction costs are expected to be attractively lcv. 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 pettle-ted type of converter reactor has a structure that easily accommodates dimensional changes in the graphite and the shares 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 Fu 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 + MSEE + *fuel cycle costs + *materials + plutonium + *converters

IAE70C059

McWherter JF

MOLTEN SALT BREEDER EXPERIMENT DESIGN EASES Cak 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 MSBE. The pumps and heat exchangers in the MSBE are similar to those project of the MSBE.

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 REACTCF CONCEFTS 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 envisioned. 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 toiling 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 KCNZEFTION DER SALZSIFDERFAKTOREN SAWA UND WARS Inst. of Nuclear Research, Warsaw, Foland Kernenergie 10 (1967), pp. 184-186, 12 ref.

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

Accession Number IAD700052 to IAF670048

Category I Reactor Design

IAF670048 *Continued* envisioned. 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 UC13 and PuC13 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.) *toiling + *cores + *conceptual design + *reactors + breeding performance + chlorides + fast neutrons + foreign + mercury + molten salts IAF690014 Perry AM A HIGH-YIFID MCLTEN-SALT FURST REACTOR Oak Ridge National Laboratory, Tenn. Proceedings of the National Topical Meeting on Past Burst Reactors, Albuquergue, 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 10(16th) nvt in neutron-irradiation specimens in single bursts with widths cf less than 1 msec. A reactor design is presented which achieves these goals, using as fuel lithium-uranium fluoride (73-27 scle %) 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 MEBR HEAT EXCHANGERS Cak 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 rcrmal 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 NW(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 he about 2100 deg F and 1800 deg F in the 565-MW and 94-MW Accession Number IAF670048 to IEA710005

Category I Reactor Design

IBA710005 *Continued*

units respectively. The calculated temperatures are believed to be conservatively high. Elmination 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 + ccoling + design + radiation heating + heat exchangers + heat transfer + MSBE + MSBR + nolle metals OTHER CATEGCENES: HCX

IBB67C039 (Staff Report) DESIGN STUDY OF A HEAT-EXCHANGE SYSTEM FOR CNE MSEE CONCEFT

Oak Ridge National Laboratory, Tenn,

CRNL-TM-1545 (Sept. 1967), 201 p, 9 fig, 12 tab, 4C 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) MSBE 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-E 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 coclant-salt system are raised above these 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 CTHER CATEGORIES: IEC

IBB71C015 Fraas AF A NEW APPROACH TO THE DESIGN OF STEAM GENEFATCRS FCF MOLTEN SAIT BEACTOF POWER PLANTS Oak Ridge National Laboratory, Tenn. CRNL-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

Accession Number IBA710005 to IEB710015

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 tc dryress. The slightly superheated steam emerging from the top of the small central tube then flows tack downward through the annulus between the central tube and the outer tube. A portion of the heat transferred from the high-terrerature 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 Further, it appears to make possible a construction. simple plant control system with exceptionally good plant response to changes in load demand.

*stear generators + *design + steam systems + control

IED680036

Peebles FN REMOVAL OF XENCN-135 FROM CIRCULATING FUEL SALT OF THE MSBR BY MASS TRANSFER TO HELIUM BUBBLES

Cak Ridge National Laboratory, Tenn.

ORNL-TM-2245 (July 1968), 33 F, 8 fig, 2 tab, 21 ref.

Removal of dissolved xenon-135 by mass transfer to he lium bubbles cffers 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 cn rates cf mass transfer to gas bubbles were reviewed. Father extensive literature references point to relialle equations for prediction of mass transfer rates to single buttles rising in stationary liquids under the two extreme cases of a rigid bubble interface and of a perfectly mobile butble interface. In general, experimental data are available which support these predictions. No reliable criterion for predicting the transition from one type tehavior to another is available. *tubbles + *mass transfer + *xenon + analysis + data + design + models + MSER + reviews OTHER CATEGORIES: HEX

Accession Number IBB710015 to IBD68CC36

Category J Instrumentation and Controls

JAA710009

Chang SI A SYSTEMATIC PROCEDURE FOR DETERMINING SYSTEM FAFAMETERS FY 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 IBE-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 nineteerth order model of the Molten Salt Breeder Feactor. 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 Cak Ridge National Laboratory, Tenn. ORNL-TM-2489 (June, 1969) 43 F, 16 fig, 7 ref.

A preliminary study was made of the dynamics and control of a 1000 Mw(e), single-fluid MSEB by an analog computer simulation. An abbreviated, lumred-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 + tehavior + computers + excursions + instrumentation + plant + stability + systems CTHER CATEGCRIES: BCX

JAB70C017 Sides WH CONTROL STUDIES OF A 1000-Mw(e) MSBR Cak 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 JAE700017

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 cutlet 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 CONTFOL 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 MSER 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 CBNI-TM-2927 (MSRIS accession JAB700C17).

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 USEF Cak Ridge National Laboratory, Tenn.

ORNL-IM-2405 (Jan. 1969), 42 F, 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 cutput quantities of changes in certain input quantities were noted. The output quantities were

Accession Number JAB700017 to JCX690019

Category J Instrumentation and Controls

JCX690019 *Continued*

heat-exchanger water outlet temperature and pressure, salt outlet temperature, and enthalpy cutrut. 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
CTHER CATEGCRIES: JAE

JDX67C037

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 sourd caution alarms at 100 counts/min and high level alarms at 400C counts/min. The gamma monitrons sound tells at 7.5 wF/hr. Building evacuation alarms are actuated from coinciderce modules. This report describes the system and triefly 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 FISSICN-PROEUCT MCNITORING IN HIGH-TEMPERATURE GAS-COCLED REACTORS

Cak Ridge National Laboratory, Tenn.

ORNL-IM-2791 (Dec. 1969), 33 F, 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. Eiscussed is the applicability of ionization chambers, beta and gamma spectrometers, charged-wire precipitators, Cerenkov detectors, filters, diffusion tutes, thermal gradient tubes, deposition tubes, and impactors as plateout monitors. It is recommended that the deposition-tute and gamma-spectrometer systems be further developed as component

Accession Number JCX690019 to JDX690060

Category J Instrumentation and Controls

JDX 69 CO 60 *Ccntinued* plateout monitors and tested first in a high-terperature gas loop and then in reactor service. *development + *fission products + *instrumentation + *off-gas systems + components + coolant loops + HTGE + plans + testing + monitors OTHER CATEGORIES: IBD JEX650020 Engel JR APPLICATION OF AN ON-LINE DIGITAL COMPUTER TO A REACTOR EXPERTMENT Cak Bidge 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 tare every 10 min. Alarms are taken from 171 signals. A number of calculations using current reactor data are rerformed periodically and on demand. Summaries of the calculations are recorded by typewriters. The Bunker-Ramc Mcdel-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 CRNI Gak 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. { **F** later, more detailed reference on this subject is JAC 67C036.) description + equipment + MSRE + remote maintenance + viewing devices **OTHER CATEGORIES:** KEA + MEE JFX670036 Moore RL CLOSED-CIRCUIT TELEVISION VIEWING IN MAINTENANCE OF RADIOACTIVE SYSTEMS AT ORNL Oak Ridge National Laboratory, Tenn. CRNL-TM-2032 (Nov. 1967), 13 p, 7 fig. This report discusses factors affecting the use of closed-circuit television in radicactive systems, ther 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. Accession Number JDX690060 to JFX670036

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 Bidge National Laboratory, Tenn. ORNL-IM-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 explcit these techniques. CRNI has started to adapt an orbital cut-and-weld system (developed for the Air Force) to permit completely remote work applications. This describes factors involved in radioactive system This report 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 regain work. Progress of the ORNI 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 radicactive system equipment replacement by cutting and welding. *maintenance + *reactors + *remote welding + cutting tools + development

Category L Fuel Preparation and Frocessing

LAX690010

Chandler JM + Bolt SE

PREPAFATION OF ENRICHING SALT (LITHIUM-7 URANIUM-233 TETRAFLUORIDE) FOR REFUELING THE MOLTEN SALI REACIDE Cak 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 call of the Thorium-Dranium Recycle Facility at CRNL 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 dicxide by treatment with hydrocen and converted to uranium tetraflucride 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 corresion preducts to metal and subsequently allowed their removal by The quality of the product was well within filtration. the requirements established for the MSFE. The fuel concentrate, containing 39 kg cf uranium (91.4% uranium-233), was packaged in nime containers of various sizes (0.5 to 7 kg of uranium) for addition to the reactor fuel drain tank and in 45 enrichment carsules, each containing 96 g of vranium, for addition to the howl of the fuel circulating pump. The tuel 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 flucride + MSRE + uranium-232
CTHER CATEGORIES: MCD

LAX70C013

Chandler JM + Eolt SE URANIUM-233-BEARING SALT PREPARATION FOR THE ESRE Cak Ridge National Laboratory, Tenh. 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 mcle %). 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 tricxide 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 IAX690010 to IAX700013

Category L Fuel Preparation and Frocessing

LAX700013 *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 CRNI-4371. See LGX6900100.)

*fuel preparation + *MSRE + *uranium-233 +
corrosion products + hydrcfluorination + lithium fluoride +
reduction + uranium-232
OTHER CATEGORIES: MCD

LAX710019

Shaffer JH

PREPABATION AND HANDLING OF SALT MIXTURES FOR THE MOLIEN-SALT REACTOR EXPERIMENT

Cak Ridge National Laboratory, Tenn.

ORNL-4616 (Jan. 1971), 41 p, 30 fig, 36 ref.

A molten mixtuure of LiF, BeF2, ZrF4, and UF4 served as the circulating fuel for the Moltan-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 prenuclear operations and subsequently to flush the reactor core after a fuel drain. Approximately 26,000 lb of these fused flucride 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 +
hydrofluorinaticn + MSRE + loading + beryllium fluoride +
lithium fluoride + uranium fluorides + zirccnium flucride

LBX680027

Mailen JC + Cathers GI

FLUORINATION OF FALLING DROPLETS CF MCITEN FLUCRICE SALT AS A MEANS CF RECOVERING URANIUM AND PLUTONIUM Oak Ridge National Laboratory, Tenn,

ORNL-4224 (November 1968) 21 p, 7 fig, 10 ref.

A flucrimation 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.

Accession Number LAX7 J0013 to LEX680027

Category L Fuel Preparation and Processing

1BX680027 *Continued*

Experimental equipment was developed, and small-scale fluorinations were made using several salt sclutions. 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 + corresion products +
molten salts + MSBR + plutonium + protactinium + uranium

ICA670014

McNeese LE

CONSIGERATIONS OF LOW PRESSURE DISTILLATION AND ITS APPLICATION TO PROCESSING OF MOLTEN-SALT EBFEDER REACTOR FUELS

Oak Ridge National Laboratory, Tenn.

CRNL-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 simultanecus 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 kuildup of rare earth fluorides at the varorization 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, NECDYMIUE, SAMAFIUM, EURCPIUM, EARIUM, STRONTIUM, YTRIUM, AND ZIRCONIUM IN MIXTURES OF LITHIUM-FLUCRIDE AND EERYILIUM-FLUCFILE Oak Ridge National Laboratory, Tenn. ORNL-TM-2058 (Jan. 1968, 43 p, 8 fig, 9 ref. Cne step in processing the fuel stream of a molten-

Accession Number LBX680027 to LCA680008

Category L Fuel Preparation and Processing

ICA680008 *Continued*

salt breeder reactor is removal of rare earth fission product fluorides from the lithium flucride-beryllium flucride carrier salt by low pressure distillation. For designing the distillation system we have measured relative volatilities of the fluorides of cerium, lanthanum, praesodymium, neodymium, samarium, europium, barium, strontium, ytrium, 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 +
eguilibrium + fission products + lithium fluoride +
measurement + MSBR + processing + volatility

ICA690037

Smith FJ + Ferris LM + Thempson CT

LIQUID-VAECE EQUILIBRIA IN LITHIUM FLUORIDE-BERYLLICM FLUORIDE AND LITHIUM FLUCRIDE-BERYLLIUM FLUCHICE-THOFIUM FLUORIDE SYSTEMS

Oak Ridge National Laboratory, Ienn.

ORNL-4415 (June, 1969) 18 p, 4 fig, 21 ref.

Liquid-vapcr equilibrium data for several lithium flucrideberyllium fluoride and lithium fluoride-beryllium flucridethorium flucride systems were obtained by the transpiration method over the temperature range of 900 to 1050 deg (. 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 flucrides. 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 flucrides

LCB680007

Carter WL + Lindauer RB + McNeese LE

DESIGN OF AN ENGINEERING-SCALE, VACUUM DISTILLATION EXPERIMENT FOR MOLTEN-SALT REACTOR FUEL

Cak Ridge National Laboratory, Tenn.

ORNL-IM-2213 (Nov. 1968), 133 p, 43 fig, 14 ref.

Experimental equipment has been designed for an engineeringscale demonstration of vacuum distillation of moltensalt reactor fuel. The distillation is carried out at about 1000 deg C and 1-mm mercury to separate carrier

Accession Number 1CA680008 to 1CB680CC7

Category L Fuel Preparation and Frocessing

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, rlus 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 shelltype, 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 uraniumfree spent fuel from the MSRE.

*distillation + *molten salts + beryllium fluoride +
Hastelloy N + lithium fluoride + MSRE + rare earths +
zirconium fluoride

ICB710007

Hightower JR + McNeese LE

LCW-PRESSURE DISTILLATION OF MOLTEN FLUORIDE MIXTURES: NONRADIOACTIVE TESTS FOR THE MSRE DISTILLATION EXFERIMENT Cak 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, orestage still reservcir, a condenser, and a condensate receiver. The equipment was tested in 1968 by processing six 48-liter batches of nonradioactive IiF-EeF2-ZrF4-NdF3 (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 Evidences of concentration polarization and/or tests. entrainment were ncted 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 MSER 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 LCB680007 to LCB710007

Fage 126

Category L Fuel Preparation and Frocessing

ICB710007 *Continued* MSRE + separations + testing

ICC710024 Hightower JR + McNeese LE + Hannaford EA + Cochran FC LCW PRESSURE DISTILLATION OF A PORTION OF THE FUEL CARRIER SALT FROM THE MOLTEN SAIT BEACTOR EXFERIMENT Cak Ridge National Laboratory, Tenn.

ORNL-4577 (August 1971), 56 p, 17 fig, 10 ref. High-temperature low-pressure distillation of irradia ted MSRE fuel carrier salt was demonstrated. Twelve liters of this salt was distilled in 23 hr with still pet temperatures of 90C-980 deg C and condenser pressures of 0.1-0.8 torr. Eleven condensate samples taken during the rur were aralyzed for Li, Be, Zr, CS-137, ZR-95, CE-144, FM-147, FU-155, Y-91, SR-90, and SR-89. Effective relative vclatilities, 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 rolarization may have also been a contributor. The effective relative volatility for CS-137 was only 20%, cr less, cr previously measured value; no explanation of this discrepancy is available. Althcuch 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 distillaticn. *distillation + *experience + *molten salts + *MSRE +

beryllium fluoride + cesium + entrainment + fluorides +
processing + rare earths + liquid level measurement +
lithium fluoride + zirccnium fluoride + separations +
operation + volatility
OTHER CATEGORIES: LIX

IDA690012

Perris LM

SCME ASPECTS OF THE THERMODYNAMICS OF THE EXTRACTION OF URANIUM, THORIUM, AND RARE EARTHS FROM MOITEN LITHIUM FLUCRIDE-PERYLLIUM FLUORIDE INTO LIQUID LITHIUM-BISMUTH SOLUTIONS

Cak Ridge National Laboratory, Tenn.

ORNL-IM-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,

Accession Number LCB7100C7 to LDA69CC12

Category I 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 flucride + rare earths + thorium + uranium + *reductive extraction process + *design data OTHER CATEGORIES: CBX

IDA690013

Mailen JC + Ferris LN + Ncgueira ED ESTIMATE CF THE SCLUBILITY OF PROTACTINIUM IN LIQUID EISMUTH Oak Ridge National Laboratory, Tenn.

Inorg. Nucl. Chem. Letters, 5 (1969) pp 869-872, 8 ref. The sclubility 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

LDA690038

Dahlke O + Gans W + Knacke O + Muller F

DISSOCIATION PRESSURE OF LISMUTH IN THE SYSTEM: BISNUTH-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 tismuth 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
CTHER CATEGORIES: CEX

LDA70C014

Schilling CE + Ferris LM THE SOLUBILITY OF THORIUM IN IIQUID EISMUTH Cak 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

Accession Number LDA690012 to LDA700014

Category L Fuel Preparation and Processing

LDA70C014 *Ccntinued*

temperature range 450 - 900 degrees C, using a filtration technique. Data obtained using mild steel equipment can be represented by the expression: $\log S(ppm Th) = 7.7085 -$ 3852/T(deg K). Several data points were obtained using molybdenum apparatus. These values are in gccd agreement with those determined in mild steel. *solubility + thorium + bismuth + filtration + *data

LDA700015

Smith FJ + Ferris LM

MUTUAL INTEFACTIONS OF THORIUM, NICKEL AND BISMUTH IN Th-Ni-Bi SOLUTIONS

Cak Ridge National Laboratory, Tenn.

J. Incry. Nucl. Chem. 32 (1970) pp 2363-2868, 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 thoriumnickel-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 sclubility of thorium and nickel could be expressed as a mole fraction product. S = mole fraction thorium x mole fraction nickel. **Ihe** variation of the sclubility with temperature can be expressed as log S = 1.115 - 6397/T (deg K). The standard free energy of formation of ThNi2Bi was estimated to le -51 kcal/mole at 650 deg C.

*solubility + thorium + nickel + bismuth + *data OTHER CATEGORIES: CBX

1DA700046

Ferris LM + Mailen JC + Lawrence JJ + Smith FJ + Noqueira ED

EQUILIBRIUM DISTRIBUTION OF ACTINIDE AND LANTHANICE FLEMENTS BETWEEN MOLTEN FLUCRIDE 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 wolter fluoride salts were determined between 500 and 700 deg C. At each temperature, the distribution coefficients (mode fraction in bisruth phase divided by mole fraction in salt) obeyed the relationship D = (EnLi)K(1st) in which τ is the valence of the element in the salt phase. Cver the range of conditions investigated, thorium, protactinium and zirccrium existed as tetravalent species in the salt; uranius, neptunium, plutcnium, americium, curium, californium, lanthanum, and neodymium were trivalent; and europium was divalent. The distribution behavior of each element was affected by salt

Accession Number LDA70C014 to LDA7CCC46

Category L 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 flucride (72-16-12)mole %) as the salt phase, the values of log K91st) 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 + teryllium fluoride + bismuth + lithium fluoride + protactinium fluorides + rare earths + thorium fluorides + zirconium fluoride

LGX650002

McNeese LE + Scott CE

RECONSTITUTION OF MSR FUEL BY REDUCING URANIUM HEXAFLUCRIDE GAS TO URANIUM TETRAFLUORIDE IN A MOLIEN SALI Oak Ridge National Laboratory, Tenn

OBNL-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 This step would replace the conventional reactor. method of reduction in which yranium hexofluoride is reduced to uranium tetrafluoride power in a hydrogenflucrine 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 cf abscrition 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 abscrpticr step are presented and information concerning the reduction of intermediate fluorides is considered.

*hydrcgen + *reduction + absorption + chemical reactions +
fluorine + molten salts + hydrogen + uranium fluorides

LHX69C011

Lindauer BE + McGlothlan CK

DESIGN, CONSTRUCTION, AND TESTING CF A LARGE MOLTEN-SALT FILTER

Oak Ridge National Laboratory, Tenn.

ORNL-TM-2478 (March 1969), 35 p, 9 fig, 12 ref.

The Molten Salt Peactor 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

Accession Number IDA700046 to IHX690011

Category L Fuel Preparation and Frocessing

IHX690011 *Continued* of 1200 deg F. *filters + *MSFE + construction + ccrrcsicn products + design + development + fluorides + testing LIX650023 Lindauer RB MSRE DESIGN AND OPERATIONS REPORT, PART VII, FUEL HANDLING AND PROCESSING PLANT Oak Bidge National Laboratory, Tenn. ORNL-TH-907 (May 1965), 96 p, 20 fig, 8 ref. The on-site plant is designed to remove cxides 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 compcunds + operation + oxides + plans + plant + safety CTHER CATEGORIES: MAA LIX67C013 Lindauer FP MSRE DESIGN AND OPERATIONS REFORT, FART VII, FUEL HANILING AND PRCCESSING PLANT Oak Ridge National Laboratory, Tenn CRNL-TM-907 Revised (Dec. 1967), 65 p, 19 fig, 7 ref. Flowsheets and equipment for the ESEE Fuel Frocessing 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 + *frocessing + accidents + equipment + flowsheets + fluorination + hydrogen + hydrogen compounds + operation + plant + safety + ura nium OTHER CATEGORIES: MAA LIX69C008 Lindauer FB PROCESSING OF THE MSRE FLUSH AND FUEL SALTS Cak Ridge National Laboratory, Tenn. ORNL-IM-2578 (Aug. 1969), 75 p, 25 fig, 12 ref. The ESRE Fuel Processing Plant, shakedown tests cf equipment and procedures, and the uranium recovery operation are described. The MSRE flush and fuel salt batches were flucrimated to recover 6.5 and 216 kg of uranium, respectively. Known losses during processing were less than 0.1%. Gross beta and gamma decontamination

Accession Number LHX690011 to LIX690008

Category I Fuel Preparation and Processing

LIX69C008 *Ccntinued*

factors of 1.2 x 10 (9th) and 8.6 x 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

IIX690009

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 CBBL. The preparation process involved reducing uranium tricxide (uranium-232 content, 222 ppm) to uranium dioxide by treatment with hydrogen, converting the dioxide to tetraflucride by hydroflucrination, and fusing the tetraflucride with lithium-7 fluoride. The original MSRE fuel salt, which contained 220 kg of uranium (35% uranium-235), was flucrinated to volatilize the uranium as the hexafluoride which was absorbed on sodium flucride. The urarium was decentaminated from fission products by a factor of almost ten to the ninth. Fluorine utilization averaged 39%. Corresion products were removed from the barren carrier salt by reduction and filtration. Corrosion rates for surfaces exposed to flucrine 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

PRELIMINATY DESIGN STUDY OF A CONTINUOUS FLUORINATION-VACUUM-DISTILLATION SYSTEM FOR REGENERATING FUEL AND FERTILE STREAMS IN A MSER

Oak Ridge National Laboratory, Tenn.

CRNL-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

Accession Number LIX690008 to LJX660006

Category L Fuel Preparation and Frocessing

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 flucrinated to remove bred uranium sufficiently fast to keep a low concentration in the tlanket. 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 ccst 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

1JX660032

thorium

Whatley ME + Carter WL + Lindauer RB + McNeese LE + Scott CE + Hightower JR ENGINEERING DEVELOPMENT OF ON-SITE FRCCESSING FOF

MOLTEN-SALT EBFEDER 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-7hF4) is fluorinated to recover bred U-233. *conceptual design + *distillation + *fluorination + *processing + costs + flowsheets + molten salts + MSBR +

LJX670032 Carter WL + Whatley ME FUEL AND FLANKET FROCESSING DEVELOPMENT FOR MOLTEN SAL1 BREEDER REACIORS Cak Ridge National Laboratory, Tenn. ORNL-IM-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

Accession Number IJX660006 to LJX670032

Fuel Preparation and Frocessing

IJX670032 *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 + ccsts +
distillation + fuels + protactinium + two-fluid reactor

LKX620003

Carter WL + Milford RP + Stockdale WG DESIGN STUDIES AND COST ESTIMATES OF TWO FLUORIDE VOLATILITY PLANTS

Cak Ridge National Laboratory, Tenn.

ORNL-IM-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 frcm a 1000-MWE one-region fused salt converter reactor. Two conditions were considered for the smaller plant: (1) retention of the waste salt fcr 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,COC for the larger plant and \$1,103,000 for the smaller plant. With protactinium discard the capital ccst is \$10,188,000 for the smaller plant. Processing consists of fluorization, abscrption on sodium fluoride, condensation in cold traps, reduction in a hydrogen-fluorine flame, dissolution in makeur 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 Cak 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 char 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 mare

Accession Number LJX670032 to LKX700C3C

Category I Fuel Preparation and Processing

LKX70C030 *Ccntinued*

earths, and (4) a computer code that calculates the nuclear, chemical, and physical processes occurring in the fuel stream of an MSBB. This work was carried out in the Chemical Technology Division during the period Cotober through December 1968.

*development + *processing + *MSBR + bismuth +
computer codes + contactors + electrolysis + rare earths +
reductive extraction process + thorium

LKX710001

McNeese LE

FNGINEERING DEVELOPMENT STUDIES FOR MOLTEN-SALT BREEDER REACTOR PROCESSING No. 2

Oak Ridge National Laboratory, Tenr.

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 reductiveextraction flowsheet for a single fluid MSEE, (2) material-balance calculations that show the effects of the removal times for zirccnium, 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 reductiveextraction facility, (7) experiments related to the development of electrolytic cells for use with molten salt and bismuth, and (8) installation of equiprent at the MSRE for demonstrating low-pressure distillation of molten salt using irradiated MSBE fuel carrier salt. This work was carried cut by the Chemical Technology Livision during the period January - March 1969.

*development + *processing + *MSBR + bismuth + electrolysis +
heat generation + MSBE + protactinium + rare earths +
reductive extraction process + zirconium

LXX660031

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 ir molten flucride 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

> > Accession Number LKX7CC030 to LXX660031

Fuel Preparation and Processing

LXX66C031 *Ccntinued*

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 prcducts + *molten salts +
*separations + beryllium + chemical reactions +
distillation + experiment + icdine + oxides + processing +
protactinium + rare earths + reductive extraction process

LXX70C029

Whatley ME + McNeese LE + Carter WL + Ferris LM + Nicholscn 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 CFNI requires continuous processing of the fuel salt, lithium flucride-beryllium flucride-thorium fluoride (72-16-12 mole %) containing approximately 0.3 mole % uranium-233 tetraflucride. The reactor and processing plant are planned as an integral The main functions of the processing plant will be system. to isclate 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 kismuth solutions at approximately 600 deg C, utilizing multistage countercurrent 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. Bare 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 +
hismuth + chemical reactions + extraction columns +
fission products + flowsheets + fuel cycle + molten salts +
MSBR + processing + separations +
reductive extraction process
OTHER CATEGORIES: LDX

IXX710021 McNeese LE ENGINEERING DEVELCPMENT STUEIES FOR MSER PROCESSING Nc. 3 Oak Ridge National Laboratory, Tenn, CRNL-TM-3138 (May 1971), 97 p, 35 fig, 13 ref. This report describes (1) calculated steady-state performance of a protactinium isclation system for

Accession Number LXX660031 to LXX7 10021

Category L Fuel Preparation and Frocessing

IXX710021 *Continued*

reactors fueled with uranium or plutonium; (2) materialbalance 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 immiscitle liquids having large density differences; (8) calculated heat generation rates and temperatures in a protactinium extraction column; and (9) low-pressure distillation of irradiated MSEE 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

IXX710026

McNeese LE ENGINEERING DEVELCPMENT STUDIES FOR MSBR PROCESSING NC. 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 sild-steel reductive extraction facility, (2) measurement of axial dispersion in packed columns in which immiscitle 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 calibraticn cf an orifice-head pot 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

MAC68C034

Tallackson JB

NUCLEAR AND PROCESS INSTRUMENTATION -- FART IIA, MSFF DESIGN AND OFERATIONS 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 neutror 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 Cak Ridge National Laboratory, Tenn.

ORNL-IM-2743 (Dec. 22, 1969) 23 p, 7 fig, 0 ref.

A new MSEE 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 captureto-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

*MSRE + *irradiation + design + fatrication + *uraniun-233 +
cross sections + fission products + adsorption + graphite +
*Hastelloy N + surveillance + pyrocarbon + intrusion
OTHER CATEGORIES: MBX

MAX650019

Accession Number MAC680034 to 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-IM-728 (Jan. 1965), 567 F, 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 ccolant salt used to transport heat from the primary heat exchanger to the air-cooled radiator (heat sink) was Lif-EeF2. All parts of the systems in contact with the salts were fabricated of standard Hastelloy N. Cperating 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 varicus reactor system electrical circuits, are also (Nuclear calculations, operational procedures, described. performance data, and maintenance aspects are given in companion reports ORNL-TM-729, 73C, 731, 732, 733, 908, 9C9, 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 ANE CONSTRUCTION (PART OF MSRP SEMIANN PROG REPORT 7/31/64) Cak Ridge National Laboratory, Tenn. ORNL-3708 (Nov. 1964), pr. 22-82, 44 fig. This paper gives a brief description of the MSEE 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

Accession Number MAX650019 to BEX700002

MBX700002 *Continued* MOLTEN-SALT REACTOR EXPERIMENT Cak 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 cutside 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 CTHER CATEGORIES: MEX

MCA660001

Guymon RH + Haubenreich PN + Engel JR MSRE DESIGN AND OPERATIONS REFORT PART XI, TEST FRCGRAM Cak Bidge National Laboratory, Tenn.

OBNL-IM-911 (Nev. 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 + *cperation + primary system + secondary systems + containment + components + experiment + surveillance + procedures CTHER CATEGORIES: KAE

MCA68C004

Engel JF

MSRE DESIGN AND OPERATIONS REFORT, FART XI-A, TEST INCOMAN FOR UNANIUM-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

Accession Number MBX700002 to MCA680004

Category M MSRE

MCA680004 *Continued* of the reactivity behavior, noise analysis, measurement of the capture-to-fission ratic for uranium 233, and additional investigation of salt, uranium, fissionproduct, and materials tehavior. *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 CESIGN AND OPERATIONS REPORT, PART IX, SAFETY **PROCEDURES AND EMERGENCY FIANS** Cak Ridge National Laboratory, Tenn. ORNL-TM-909 (June 1965), 46 p, 16 fig, 3 ref. Contains brief description of tasic plan, emergency philcsophy, 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 CESIGN AND OPERATIONS REPORT, PART VIII, OPERATING PROCEDURES (VOL. I) Cak Ridge National Laboratory, Tenn. ORNL-IM-9C8 (Vcl. 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) Cak Ridge National Laboratory ORNL-IM-908 (Vcl II) (Jan. 1966), 539 p. Contains procedures for reactor startur, 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 MCB69C054 Guymon BH + Haubenreich PN OPERATING SAFETY LIMITS FOR THE MSRE Accession Number MCA680004 to MCB690054

MCB69C054 *Continued* Cak 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 + *creration + *safety limits + procedures + safety CTHER CATEGORIES: KAB

MCB710012

Guymon RH

MSRE PROCEDURES FOR THE PERIOD BETWEEN EXAMINATION AND ULTIMATE DISFOSAL (PHASE ITT OF DECOMMISSION PROGRAM) Oak Ridge National Laboratory, Tenn. CRNL-TM-3253 (Feb. 1971), 41 p, 3 fig, 9 ref.

This document describes the condition of the MSFE 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 access will be controlled by the perscnnel. 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

Cak Ridge National Laboratory, Tenn,

ORNL-IM-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 Feactor Experiment (MSRE) in nuclear startup and power-level operating procedures. Both simulators were set up on general purpose, portable 3lectronic 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

Accession Number MCB690054 to MCC660005

Category M MSRE

MCC660005 *Continued* control and instrumentation system, the operating procedures, and the rod and radiator-docr drives were checked out. Scre minor modifications were made to the system as a result of the experience with these sigulators. analog systems + *MSRE + *training + sigulation + *operators + startup + testing MCC670044 Guymon RH MSRE -- TFAINING, PREPARATION FOR OPERATION AND OPERATING TECHNIQUES Cak Ridge National Laboratory, Tenn. Suppl. to ANS Trans (10), Conf. on Beactor Cp. Exp. (July 1967), p. 35. Three periods of training were given: Initial (system familiarity prior tc checkout), Frecritical (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 PN + Engel JR RECENT EXFERIENCE WITH THE MOLTEN-SALT REACTOR EXPERIMENT Oak Ridge National Laboratory, Tenn. Summary of paper presented at ANS Winter Meeting, Washington D. C. Nev. 10-15, 1968, ANS Transactions 11 (2) 619, 1 p. Operating experience with U-235 fuel in the MSRF 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 tehavior, nuclear characteristics, and reactivity behavicr. 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 remotemaintenance 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 CTHER CATEGORIES: KAE

MCD69C017

MCD69C017 *Ccntinued*

Engel JR + Hautenreich PN

OPERATION OF THE MOLTEN SALT REACTOR EXFERIMENT WITH U-233 FUEL

Oak Ridge National Laboratory, Tenn.

Summary of paper presented at ANS Conference on Reactor Operating Experience, San Juan, Fuerto 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 chaacteristics of the reacter were essentially as predicted. Special fuel samples were taken to evaluate U-233 integral cross-section ratics in an MSR spectrum. Experiments were conducted to measure cover-gas entrainment in the fuel loop and its effect or fissiorproduct 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. (Freprints 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 + *cperation + *experience + fuels + *uranium-233 + *plutonium + *nuclear analysis + dynamic characteristics + reactivity + fission products + buttles + cross sections + corrosion + gamma spectrometry + inert gases + noble metals + rare gases + control rods + pumps + off-gas systems + helium + argon + cover gas + void fractions CTHER CATEGORIES: KAB

MCD69C055 Haubenreich PN MOLTEN-SALT REACTOR PROGRESS. Cak, Ridge National Laboratory, Tenn. Nucl. Engrg. Int. Vol. 14, No. 155 (April 1969), pp. 325-329, 3 fig. This article touches briefly on earlier MSE technology development, then describes the MSRE and its crerating experience. (A more recent, lengthier paper along the same lines is in the Fet. 1970 Nucl. Appl. Tech.) *experience + *MSRE + *operation + design + MSRP + startup CTHER CATEGORIES: KAB MCD690062 Blumberg F + Dyer FF + Houtzeel AMSRE USES RENOTE GAMMA SPECTRCMETRY FCR FISSION FROIUCT

DEFCSITICN STUDIES Oak Ridge National Laboratory, Tenn.

ANS Trans. 12(2) (Dec. 1969), p. 842.

Accession Number MCD690017 to MCD690062

Category M MSBE

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 spectrom@try + *MSRE + analysis +
deposition + experience + fission products + cperaticr +
remote maintenance
OTHER CATEGCRIES: JEX + MCE

MCD70C001

Haubenreich PN + Engel JR EXPERIENCE WITH THE MOLTEN-SAIT REACTOR EXPERIMENT Cak 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 flucride 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 Barch 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 resured in October 1968. Cver 2500 EFFH 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 + flucrimation + performance + reactivity + reliability

OTHER CATEGORIES: KAB + MEC

MDA620001 Fngel JR + Haubenreich PN TEMPEFATURES IN THE MSRE CORE DURING STEADY-STATE POWER OPERATION Cak Ridge National Laboratory, Tenn. ORNL-TM-378 (Nov. 5, 1962) 58 p, 14 fig, 8 ref.

Accession Number MCD690062 to MCA620001

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 erriched 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 COFFFICIENT 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 (-5th) deg F for the fuel and -7.3×10 10 (-5th) for the graphite. These values refer to a reactor fueled with salt which does not contain therive. They were about 5% larger than the values obtained from a cne-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 CTHER CATEGORIES: EEX

MDA63C002 Haubenreich FN INHERENT NEUTRON SOURCE IN CLEAN MSFE FUEL SAIT

Accession Number MDA620001 to MDA63C0C2

Category M MSRF

MDA63C002 *Continued* Cak Ridge National Laboratory, Tenn. ORNL-TM-611 (Aug. 1963), 17 F, 6 ref.

Alpha particles from uranium interact with beryllium and flucrine to produce an inherent source of neutrons in the MSRE fuel salt. The spontaneous fission scurce is relatively insignificant. Calculations are described which predict an inherent source of 3 to 5 x 1C (5th)

neutrons/sec in the 25 cu ft of salt in the MSFE core. calculations + MSRE + neutron sources + fu \in 1s + molten salts

MDA640001

Engel JR + Haubenreich PN + Prince EE MSRE NEUTRON SOURCE REQUIREMENTS Cak 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 servodriven fission chambers are installed in the instrument shaft to monitor the filling operation, the calculations indicate that the required source strength car be reduced from 4 x 10(7th) n/sec to 7 x 10(6th) n/sec. An antimonyberyllium source with an initial strength cf 4 x 10 (8 th) n/sec would still produce 7 x 10(6th) n/sec one year after Abstractor's note: The external source installation. ultimately selected for use in MSRE was americium-curiumberylliuz.

*calculations + *neutron sources + *MSRE + safety +
neutron flux + nuclear analysis

MDA640002

Engel JR + Prince BE

CRITICALITY FACTORS IN MSRE FUEL STORAGE AND ERAIN 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 +

Accession Number MDA63C0C2 to MIA64CCC2

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.

OBNL-IM-73C (Feb. 1964), 199 p, 63 fig, 52 ref. Early calculations of effects of core size and fuelto-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 erriched 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 CPERATIONS REPORT, PART V -- REACTOR SAFETY ANALYSIS REPORT Cak Ridge National Laboratory, Tenn.

ORNL-IM-732 (Aug. 1964), 300 F, 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 CTHER CATEGORIES: BGX + JEX

MDA65C001 Ball SJ + Kerlin TW STABILITY ANALYSIS OF THE MOLTEN-SALT REACTCE EXFERIMENT Cak Ridge National Laboratory, Tenn. ORNL-IM-107C (Dec. 1965), 80 F, 23 fig, 20 ref. A detailed analysis shows that the Mclten-Salt Reactor Experiment is inherently stable with uranium-235 fuel.

Accession Number MDA640002 to MIA650001

Category M MSRE

HDA650001 *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, frequence 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

PERICD MEASUREMENTS ON THE MOLTEN SALT REACTOR EXPERIMENT DURING FUEL CIRCULATION: THECRY AND EXTERIMENT Cak Ridge National Laboratory, Tenn. ORNL-TM-1626 (Oct. 1966), 36 F, 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 otserved 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 letween the time-independent flux conditions.

*analysis + *circulation + *dynamics tests + *experiment +
*MSRE + control rods + criticality + delayed reutrons +
experience + models + reactivity + startup + uranium-235
CTHER CATEGOFIES: ECX + MDC

MDA66C004

MDA66C004 *Ccntinued* Engel JR + Haubenreich PN + Ball SJ ANALYSIS OF FILLING ACCIDENTS IN MSRE Cak 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. Varicus restulated 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 Abstractor's note: It was subsequently be prevented. 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 CTHER CATEGORIES: BGX

MDA67C038

Haubenreich PN SAFETY ASPECTS OF THE MSRE Cak Ridge National Laboratory, Tenn.

Nucl. Safety Vcl. 8 Nc. 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 treeder. 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 treeder reactors. *accidents + *MSRE + *reviews + *safety + afterheat + analysis + containment + experience OTHER CATEGORIES: BGX

MDA670040 Kedl RJ + Houtzeel A

Accession Number MDA660004 to MDA670040

Category M MSRE

MDA670040 *Continued*

DEVELOPMENT OF A MODEL FCR COMPUTING XE-135 MIGRATICN 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. 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 butbles and well below 1% with bubbles. Preliminary measurements made on the critical reactor show xenon poisoning of C.3 to C.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 + *xencn + computer codes + diffusion + graphite + krypton

OTHER CATEGORIES: BFX + IBA

MDA670041

Kedl RJ

A MODEL FCR COMPUTING THE MIGRATION OF VERY SHORT LIVED NOBLE GASES INTO MSRE GRAPHITE

Cak Ridge National Laboratory, Tenn.

ORNL-IM-1810 (July 1967), 26 p, 4 fig, 1 tab, 6 ref.

A model describing the migraticn of very short-lived roble gases from the fuel salt to the graphite in the ESRE 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-14C, CE-141, SR-E9, 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 + menon CTHER CATEGORIES: IEA

MDA68C003 Haubenreich FN + Engel JR + Gabtard CH + Guymen RH + Prince BE MSRE DESIGN AND OFERATIONS REPORT PART V-A, SAFETY ANALYSIS OF OPERATION WITH U-233 Cak Ridge National Laboratory, Tenn. ORNL-TM-2111 (Feb. 1968), 80 p, 24 fig, 36 ref. This report presents data and analyses that suffert

Accessicn Number MDA670040 to MEA680003

MDA680003 *Continued*

the conclusion that it is safe to load and operate the MSRE with uranium-233. It summarizes pertiment experience with the MSRE and new information on materials through December, 1967. Procedures for producing, hardling 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.

CRNL-IM-2685 (Aug. 10, 1969) 22p, 2 fig, 11 ref. After about three years of nuclear operation, the MSRE fuel, enriched 0-235, was replaced with a 0-233 fuel In this new mixture there are quantities mixture. 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 scurce is time-dependent because of the buildur 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 x 10(8th) neutron/sec, primarily from the reactions Be9(Alpha,N)C12 and F19 (Alpha, N) Na 22. Alpha-N reactions with lithium will produce less than 3 x 10 (6th) neutrons/sec. Spontaneous fission will produce less than 10(2rd) neutrons/sec.

*MSRE + *neutron sources + *fuels + *uranium-233 +
*uranium-232 + decay + beryllium + fluorine + lithium +
calculations

MDA69C002 Steffy RC • Wood FJ THEORETICAL DYNAMIC ANALYSIS CF THE MSFE WITH U-233 FUFL Cak 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

Accession Number MDA680003 tc MDA690002

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 critericn. *MSRE + *dynamic characteristics + *stalility + *mcdels + *uranium-233 + fuels + nuclear analysis + reactivity + delayed neutrons + computer codes + mixing + calculations + methods

OTHER CATEGOFIES: BCX

MDA 690005 Burke CW + Clark FH ANALYSIS OF TRANSIENTS IN THE MSRE SYSTEM WITH UBANIUE-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 CENL 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 cut of the fuel at some point in the system external to the core could be swept back into the core in a concertrated 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 te required.

analysis + dynamic characteristics + computers + feedback + MSRE + analog systems + stability + mathematics + models + simulation + control + reactivity + safety + startur + heat transfer + ccntrcl rcds + vranium-233 + accidents + control-rod drives + delayed neutrons + dynamics tests + excursions + kinetic equations

MDA700006

Accession Number MDA690002 to MEA700006

MDA700006 *Continued*

Ulrich WC

AN EXTENDED HYDRAULIC MODEL OF THE MSRE CIRCULATING FUEL SYSTEM (THESIS)

Cak 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 rump 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 propertional 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

CALCUIATED RADIOACTIVITY OF MSRE FUEL SALT Oak Ridge National Laboratory, Tenn.

CRNI-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 isctopes 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 flucrination of the fuel salt after the period of D-235 operation. Results are presented which give the inventory and radioactivity of individual isctopes in the salt up to January 1, 1975. After storage for 5 years, the gamma-ray shielding required for shipping fuel is determined by thalium-208 and neutrons from uranium-232 daughters produce the controlling radiation dese through a lead shield.

*MSRE + *fission products + *isotopes + stcrage + fuels + *decay + uranium-235 + uranium-233 + uranium-232 + disposal + processing + shielding + *inventories + rare cases + radioactivity + calculations

MDA700032

Category M MSRE

MDA700032 *Continued*

Robinson JC

ANALYTICAI DETERMINATION OF THE NEUTRON FLUX-10-PRESSURE FREQUENCY: APPLICATION TO THE MCITED-SAIT FEACTOF EXPERIMENT

Oak Ridge National Laboratory, Tenn.

Nucl. Sci. and Eng. 42(3), 382-396 (December 197C), 15 F. 7 fig, 2 tables, 18 ref.

The neutron flux-to-pressure frequency response for a molten-salt-fueled reactor with a small amcunt cf gas entrained in the molten salt was determined analytically. The one-dimensional conservation equations describing the flow cf the corpressible 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. Ihe matrix exponential technique was selected to cttain a solution for the hydraulic equations over the techniques ncrmally employed (nodal or modal) 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) Results from the computed neutron flux-tois small. pressure frequency response for the molten-salt-fueled reactor under study show that the shape of the acdulus 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 Therefore, we conclude that the amount of void fracticn. 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

M DA 710003

Kerlin TW + Fall SJ + Steffy RC

THEORETICAL DYNAMICS ANALYSIS OF THE MOITEN-SAIT FEACTCE EXPERIMENT

Oak Ridge National Laboratory, Tenn.

Nuclear Technology, Feb. 1971, 15 p, 24 fig, 12 ref.

The dynamic characteristics of the MSFE were calculated for operation with U-235 and U-233 fuels. The analysis

Accession Number MDA700032 to MEA710003

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 MOLTEN SALT REACTOR EXPERIMENT

Cak Ridge National Laboratory, Tennessee ORNL-TM-3CO2 (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 talance. A calibration check of the instruments is planned. The heat-removal capabilities of the fuel-salt tc coolant-salt heat exchanger and coolant-salt tc air radiator were telow 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 FOF THE MOLTEN-SALT REACTOR EXPERIMENT: DETERMINATION OF VOII 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-

Accession Number MDA710003 to MDB700033

Category M MSRE

MDB700033 *Continued*

pressure frequency response was then obtained from a crosspower 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-tc-pressure frequency response and that properly designed dynamic tests involving small reactivity perturbations (introduced by means other than rcd 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 CTHER CATEGCRIES: MCD

MDB71C002

Kerlin TW + Ball SJ + Steffy RC + Buckner MR EXPERIENCES WITH DYNAMIC TESTING METHODS AT THE MCITEN-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 CTHER CATEGCRIES: MCD

MDC66C002 Reclin TW + Ball SJ EXPERIMENTAL DYNAMIC ANALYSIS OF THE MCITEN-SAIT BEACTOR

Accession Number MDB700033 to MDC660002

MDC66C002 *Continued*

EXPERIMENT Oak Ridge National Laboratory, Tenn.

ORNL-TM-1647 (Cct. 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 rewer-to-reactivity frequency response. Three types of input disturbances were used: the pseudcrandcm 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 cscillation 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 EALANCE IN THE MSRE Oak Ridge National Laboratory, Tennessee CRNL-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 aromalous reactivity changes. Sensitivity for detecting shortterm 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 CFNI-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 syncriss of OBNI-TE-1796, same title, accession number MEC670001. *MSRE + *reactivity + *experience + *xenon + *nuclear analysis + operation + data processing + bubbles + control rods + rare earths + models + fissicn products + fuels + uraniur-235 MDC680002 Prince BE + Ball SJ + Engel JF + Haubenreich FN + Kerlin TW ZERO-FOWEF FHYSICS EXPERIMENTS ON THE MOLTEN-SALI REACIOR EXPERIMENT Cak 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 controlrcd calibraticn (period-differential worth and rod drop-integral worth measurements), determinations of the reactivity loss due to fuel circulation, the 'static' reactivity coefficients of excess 0-235 concertration 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 + mcdels + startup CTHER CATEGORIES: KAE + MCE

Fry DN + Kryter RC + Robinson JC

MDC68C005

Accession Number MDC670002 to MIC680005

Category M MSRE

MDC680005 *Continued* MEASUREMENT OF HELIUM VOID FRACTION IN THE MSFE FUEL SALT USING NEUTEON-NOISE ANALYSIS Oak Ridge National Laboratory, Tenn. CRNL-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 cculd be measured using neutron noise analysis. The neutron power spectral density (NPSC) 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 C.1%. It is concluded that changes in the circulating void fraction can be inferred with good sensitivity directly from reutron noise measurements, and, consequently, NFSE can complement and enhance the value of the MSRE reactivity balance calculations.

*analysis + *experience + *models + *MSRE + *ncise analysis + *reactivity + *void fractions + dynamic characteristics + experiment + uranium-235 OTHER CATEGORIES: MCD

MDC690003

ROBINSON JC + Fry DN DETERMINATION OF THE VOID FRACTICN IN THE MSBE USING SMALL INDUCED FFESSURE PERTUREATIONS

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-jump 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-tc-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

Accession Number MDC680005 to MIC690003

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 (Abstractor's note: This work was subsequently model. reported in Nucl. Sci. & Tech., see MDB 700009.) *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 Cak 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 chamter 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 cifers a non-perturbing method of determining the circulating void fraction with the reactor operating at power. (This work is described in detail in ORNI-TM-2315. See MCC680005.) *MSRE + *noise analysis + *void fractions + measurement + neutron flux + helium + cover gas CTHER CATEGORIES: MCD MDC69C016 Robinson JC + Fry DN THE FREQUENCY RESPONSE OF THE NEUTRON FIUX TO FRESSUFF 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) 252, 2 p, 3 ref. An analytical model was developed to compute the neutronflux-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 fltx, perturbations. Analysis indicated that the neutron-flux-topressure frequency response is proportional to the

Accession Number MDC690003 to MIC690016

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-TE-2318. See MDC-690003.)

*MSRE + *dynamic characteristics + *luttles + *mcdels + *nuclear analysis + experiment + measurement + *void fractions + cover gas + helium + calculations OTHER CATEGORIES: MCD

MDC700004

Steffy RC

FREQUENCY RESPONSE TESTING OF THE MOLTEN SALT REACIOR EXPERIMENT (THESIS)

Cak Ridge National Laboratory, Tenn.

ORNL-TM-2823 (Mar. 1970) 118 p, 27 fig, 31 ref.

Tests to determine the neutron flux-tc-reactivity frequency response were performed on the MSFE 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 11-233 fuel mixture. Test patterns employed were rseudorandcm binary sequences (PRBS) and pseudorandom ternary sequences (PRTS) of various sequence lengths and minimumpulse-duration times. In some tests reactivity (controlrod 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 thecretical 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

MDC700005 Steffy RC EXPERIMENTAL DYNAMIC ANALYSIS OF THE MSRE WITH U-233 FUEL Oak Ridge National Laboratory, Tenn. CRNL-TM-2997 (April 1970), 28 p, 10 fig, 11 ref. During the startup with U-233 fuel, tests showed that the

Accession Number MDC690016 to MDC700005

system time response to stop changes in reactivity, the

OTHER CATEGORIES: MCD

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 tinary 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) Cak 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 beccme radicactive were designed and located so that they can be disconnected by the use of longhandled 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 A1 THE MSRE Oak Ridge National Laboratory, Tenn. ANS Trans 9(2) (1966), p. 530.

Maintenance operations are performed at MSEE with long tools manipulated through access holes provided in a pertable 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 MFE680001

MEB680001 *Continued* EQUIPMENT AND PROCEDURES Cak Ridge National Laboratory, Tenn. ORNL-TM-910 (June 1968), 80 p, 26 fig, 3 ref.

A record of the methods developed for maintaining the radicactive 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 radicactive 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 + *Flans + design + development +
equipment + procedures + shielding + tools

MEC70C053

Haubenreich FN + Elumberg R + Richardson M MAINTENANCE OF THE MOLTEN-SAIT REACTCR EXPERIMENT Cak Ridge National Laboratory, Tenn. Paper, ANS 197C Winter Meeting, Washington, Nov. 1970,

29 p, 8 fig, 8 ref.

The MSRE was designed for maintenance of radicactive 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, GRNI; 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 on when the maintenance + maintenance + reliability +

OTHER CATEGORTES: KBA + EEA

MFX700020

Hautenreich PN + Richardson M PLANS FOR FOST-OPERATION EXAMINATION OF THE MOLTEN-SALT REACTOR EXPERIMENT Cak 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

Accession Number MEB680001 to MFX7CCC20

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 charccal 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. Frocedures and tools are available for some jcts; for others, they are currently being developed.

decommissioning + examinations + MSRE + plans + reacte main tenance

Accession Number MFX700020 to MFX700020

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: aquecus horcgerecus, 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 gair. Maintenance was identified as the most important factor influencing the practicability of any of the three. Ihe molten-salt reactor was judged to have the highest probability of achieving technical feasibility. *AEC + *development + *plans + *reactors + breeding performance + fuels + LNR + maintenance + MSBR + optimizations + reviews CTHER CATEGORIES : AEX NXX630001 Voznick HF + Uhl VW MOLTEN SALT FOR HEAT TRANSFER Atlantic Besearch Corp + Irexel Institute of Technology Chem. Engrg. Vcl. 70, 129 (May 27, 1963) 8 p, 4 fig, 37 ref. Heat-transfer salt (HIS) composed of 40% scdium mitrite, 7% sodium nitrate and 53% potassium nitrate has been used widely since 1937 fcr 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 HIS 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 + rumps + 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 DFD1 by a task force from industry, national laboratories and the AEC, and contains a fcreword by ERDT Director M. Shaw. The reject treats first the general features of the thorium cycle (resources, nuclear characteristics of thorium and 0-233 in thermal- and fast-neutron spectra). Then it discusses thorium utilization in specific reactor types: hightemperature gas-cocled, molten-salt, light-water, and heavy-water thermal reactors and a fast reactor. The MSBR considered is a single-fluid treeder 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 + treeding performance + natural resources + development + MSER CTHER CATEGORIES: AEX NXX69C057 (Staff Report) COST-BENEFIT ANALYSTS OF THE U.S. BREEDER FEACTOF FECCEAM 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 LMFER plus converters. Large tenefit/ccst 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 + LMFER + MSER + power costs NXX70C011 Bond VP EVALUATION OF FOTENTIAL HAZARIS 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 thecretical and experimental standpoints all

Accession Number NXX690046 to NXX700011

Category N Miscellaneous

NXX700011 *Continued*

factors involved. Factors conceivably increasing toxicity are considered in detail. These include selective concentration (actually discrimination) in the human lody, 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 radiotiological meaning as the same dose of x-rays; the ICRP-AEC max permissible body burden of 1000 microcuries is guite 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 HTC and other sources of radiation.

*health physics + *tritium + beta decay + concentration +
environment + reactors + reviews + safety + wastes

NXX70C057

Deonigi DE

A SIMULATION OF THE UNITED STATES FOWEF ECONCRY Facific 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
CTHER CATEGCRIES: EFX

NXX70C058 (Staff Report) POTENTIAL NUCLEAR POWER GROWIH FATTERNS United States Atomic Energy Commission DRDI WASH-1098 (Dec. 1970) 249 F, 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 competition included all reactor types, over half of the reactors built

Accession Number NXX700011 to NXX700058

Category N Miscellaneous

NXX70C058 *Ccntinued* 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 + LMFER + MSER + natural rescurces + power costs

NXX700060 (Staff Report) REPORT OF THE FFI REACTOR ASSESSMENT PANEL Edison Electric Institute, N.Y.

EEI Publication 70-30 (April 1970), 53 p, 14 fig, 7 ref. A panel cf 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 IMFBR 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 AFC development deter utility involvement. *development + *electrical power + *reactors + *utilities + costs + energy + industry + FTGR + LMFBR + LWBR + MSBR +

natural rescurces + rower costs + processing + reviews

Accession Number NXX700058 to NXX700060

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		AAX670010	IAB670043
absorption		ACA660008	IBD 680036
ACB690029	LG X 6 50 00 2	ACA6EC014	JAB 690018
ACCESCOOE	LKX620003	AC A6 700 16	JAE700017
BAX700008	BRACECCCC	ACA670023	JCX690019
accidents		ACA680012	MCD690062
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	MDA660004	ACA6E0019	
CAX 690053	MDA670038	ACA690021	MEA640007
IBA710005	MDA 690005	ACA690028	MEA 65000 1
LIX670013	NXX630001	ACA70021	MDA 660003
MDA E4CCO7		ACA7 000 35	M EA 660004
actinides		ACB7 10029	MDA 670038
ACD660017	ACD700038	ACD660017	MEA670040
ACD670020	CAX680032	AC D6 700 19	MDA 670041
ACD670027	CAX690052	ACD670020	MDA 680003
ACD680015	CA X710023	ACD670026	MCA 690005
ACD680016	CXX700049	ACD670027	MDB 700003
ACD 69C024	LDA700046	AC D7 000 38	MCC660002
administration		BGX67C045	MDC 680002
MCB710012		CDX670035	MEC 68 00 05
adsorption		CL X7 000 10	MDC 690003
HIX660026	MAD690004	ECX7 100 1 1	MDC 700004
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	T107 100 13		HLC 100005
BGX 670045	IAC7 100 13	HCX710022	
CAX690053	IBA710005	analytical chemistry	
C1X700010	MCA670038	ACD650011	ACD680022
IAC700047		ACD660011	ACD €90024
aging		ACD6EC017	ACD 690031
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alloy composition		ACD6 700 26	ACD 700038
ACE660018	FBE7 100 17	ACD670027	
ACE670028	FBE710018	applications	
ACE680017	FCC700040	CAX690053	NXX630001
ACE68C024	FCC700044	architect-engineering	g ·
ACE690026	FCC7 100 10	ADX690063	Pillion and the second second
ACE700025	FCE690043	argon	÷
ACE700039	FCE7 10004	MCD690017	
FAX69C035	FC X690033	barium	
FBC610001	FCX700026	ACD7(0024	
FBC 650017	FXX690047	bearings	1. A
FBE690044	GXX680039	ACA670023	ACE 680024
alloys	JAN 00005.7	ACA6 800 12	ACE690026
ACE650008	ACE680017	ACB670024	ACE 700025
ACE650014	ACE680024	ACB6E0013	NXX630001
ACE66CC12	FA X620004	tehavior	MAACJUUUT
			C V V 7000/0
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ACE 67C028	FA X6 90035	ACD670026	JAB690018
aluminum		ACD6E0023	JAB 700017
ACE66C018	ACE670021	CAX690053	NXX 630001
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JAB690018	JCX690019	beryllium	
JAB70C017	MCC660005	ACC660016	ACD 6800 22
JAB710008	MDA690005	ACD680015	ACD 690031

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			•	
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*Conti			ACEGEOOIE	ACE700025
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	LXX660031		ACE670028	FBX640015
tervll	lium fluoride		ACE6 800 17	
	ACD660017	CAX690052	treeding performance	÷
	ACDE7C019	CC X680033	AAX670009	BEX 670012
	ACD670020	CXX700049	ABX640004	BFX680009
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	CAX680032	LDA700046	ACB660015	IAC 700047
terv11	lium oxide		ACE670024	IAC710013
Leryt	ACD690024	CXX700049	ACB68C013	IAF 670047
teta d			ADX640021	IAF670048
reca	NXX70C011		ACX690063	NXX 590002
tismut			BAX6ECOC6	NXX690046
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	ACD680022	LDA700046	turnup	N RY (70040
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	ACD700038	LXX7 10021	BBX67C012	MDA 690001
	ACE690026	LXX710026	MDA6 20001	M CA 690002
	ACE690032		MDA620002	MDA 700006
tlanke			EDA6 300 02	MEA700007
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	BFX 680009		BFX700056	LKX 620003
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	ACA EEOC14	ACA670016	capital equipment	
boilin	-		IAC660025	I AC 700051
	IAF67C047	IAF670048	IAC700047	IAC710013
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_	ACD EECO23	CXX700049	AC 26 700 26	CEX 640018
	ACD690024		ACD680015	GDX690042
traziı			AC 1680022	
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	ACE650008	ACE690026	AC B6 900 29	EAX 700008

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carbides		ACC7C0037	IAF 670047
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	FCC/ 100 10	circulation	
ACE680024			CCX680033
carbon		ACD660017	
ACD 690024	FBA660020	ACD670020	M DA 66 0003
EXX700046		ACD670027	MDC 680002
carbonates	, •	coatings	
CAX690053	· · · ·	ACE680013	ACE680024
carriers		ACE670028	ACE690032
CAX680032		ACE6 800 17	ACE700025
casting		cotalt	
FBX64C015		ACE680024	
cells		cake	
ACB 690022	ACD680016	EXX700048	
ACB690029	ACD680022	columns	
ACD66C017	ACD690024	ACC690030	ACE660018
ACD670019	ACD690031	ACC7C0023	ACE680024
ACD 670026	ACD700038	ACC700037	L EX 680027
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FBX640015	GGX670034	AAX6700C6	ACE 700025
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		ACD670019	CAX680032
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charcoal		ACD670026	CAX690053
HIX66C026		AC D6 700 27	CFX 640018
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ACD690024	LXX660031	ACE670028	GDX690042
ACD 690031	LXX7C0029	ACE680017	GFX 660023
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AAX 670005	ACD690031	AC E6 900 26	
A BX 58000 1	ACD700024	components	
ACD650011	ACD700038	AAX6700C7	HBX700012
ACD 660011	CAX680032	A A X 6 700 08	HIX 660026
ACD670019	CAX690052	ABX69C056	H XX 64 0019
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ACD680015	CXX640020	ACA 67C023	JDX690060
ACD 680016	CXX7C0049	AC A6 800 12	M EX 640003
ACD680022	LXX660031	ACB700022	MCA 66 0001
ACD 680023	MCA680004	AC E7 000 36	MCD €800 10
ACD690024		CXX7C0049	MCD 700001
chlorides		HEX620006	MEB700003

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2	•		
components		CAX690053	MDA 670038
Continued		IAC660024	M EA 680003
NXXE3COO1		contamination	
compressive Propert		JDX670037	MEC 700053
EBX69C039	EDX640016	control	
EBX700042		AAX6700C8	JAA 710009
computer codes		ACE680020	JABE90018
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ACC 690023	EDA 670040	ACB690029	JAB710008
BBX670012	MDA 670041	ACE700022	MAC 680034
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HCX710022	MDA700006	IAC7 100 13	M CA E90005
IBB670039	MDC700004	IBB710015	
JAA 710009		control rods	
computers		AAX670007	MAC 680034
JAB690018	JEX650020	ACA650010	MCD690017
JAB700017	MAC680034	ACA660008	MDA €40006
JAB710008	MDA690005	ACA690021	MDA 660003
JCX69C019	MDC70004	ACA690028	M LA 690005
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ACD670019	CAX680032	GGX670034	MDC €80002
ACD680015	CAX 690053	IAC7C0047	MDC 700004
ACD 680022	CDX670035	IAC7 100 13	
ACD690024	CXX700049	control-rod drives	
ACD 690031	NXX700011	AAX670007	M CA 690005
ACD700024		MAC6E0034	MDC 700004
conceptual design		converters	
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ACB660015	111660030	ADX690063	LKX €20003
ACB670017	IAC660024	GXX6ECO39	N XX 700057
ACB670024	TAC660025	IAA650024	NXX 700058
ACB680013	IAC700047	coclant locps	
ACB 68002 C ACB 690022	IAC700051	FCE7 100 16 ccolants	J EX 690060
ACB 69002 9	IAC7 100 13 IAC7 10014	ACE7 100 29	CAN COODE 3
ACB700022	IAD700052		CAX 690053
ACB700022	IAE700059	ACD670026 ACC680015	C XX 64 002 0 C XX 70 00 4 9
ACB700050	IAF670047	ACD6EC022	FBD 690036
ADX64C021	IAF670047	ACD680023	GFX €60023
ADX670046	IAF690014	ACD69C024	IAC 66 0025
GDX71C025	LJX660032	ACD630024	IAC700047
construction	LUAUUUUJZ	ACD7(0024	IAC 710013
ACA 650004	LH X6 90011	ACD700024	IAC710014
ACC6 5000 6	MEX640003	ACE6 900 26	JCX 6900 19
ACC650012	MEX700002	ACE69C032	N XX 630001
contactors	11 EX 7 00 0 0 2	ACE7 000 25	N AA CJUUU I
ACE700039	LKX700030	cooling	
containers	BUX 100030	IAC660025	I FA 710005
NXX630001		COLES	TTULIOUND
containment		AAX670007	ACD670020
AAX670010	IAC700047	ACA710028	ACD 670026
ACA 66 C008	IAC7 100 13	ACE680013	ACD670028
ACB 68002 C	MCA660001	ACE680020	A EX 670046
ACB690029	MCB7 100 12	ACB690022	HAX700050
ACB700036	MDA 640007	ACD660017	IAC710013
		#CD000047	**********

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cores	· · · ·	ACD680023	MDC 690016
Continued	8 (1) (1) (1) (1) (1) (1) (1) (1) (1) (1)	ACC690024	NXX 630001
IAF670047	MDA620001	cracks	
IAF670048		ACA6E0014	EDX640016
corrosion	and the second second	ACE6 500 08	
A BX 58000 1	CXX700049	creep	
ACD660017	FA X620004	AC E6 50008	FEC € 1000 1
ACD670019	FAX 6 2000 5	ACE650014	FBC 650017
ACD670020	FAX690035	ACE660012	FEE660019
ACD670026	FAX690045	ACE66C018	FBE 670029
ACD 670027	FBD690036	AC E6 700 28	FEE670030
ACD680015	FBE670031	ACE680017	FBE 670031
ACD680022	FBE690034	ACE680024	FEE680025
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ACD 700024	FBE710018	ACE690032	FEE690034
ACD700038	FBX640015	ACE7C0025	FBE 690044
ACE650008	FCD710016	ACE700039	FEE700027
ACE650014	FCX690033	EBX7C0042	FBE710017
ACEEECC12	FC X7 00026	ECX7 100 1 1	FEE710018
ACE660018	GAX700045	EDX680031	FBX640015
ACE670021	GC X6 10002	EDX690051	FCC700040
ACE670028	GDX690042	EXX7C0048	FCC 700044
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ACE690032	LIX690008	FBB65C041	
ACE70C025	MCD690017	criticality	
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CAX690053		BEX670012	M E A E 6 0 0 0 3
corrosion products		MCA6E00C4	MDC 680002
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ACD670019	CXX700049	AAX67COC9	BBX670012
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ACD68C015	LE X680027	crystallization	
ACD680022	LHX690011	CAX69CO61	
ACD 690024		cutting tools	
corrosion protection		ACA7 10028	KBB 690006
ACB67C017	ACC680021	data	the second second second
ACC660010	ACE660018	ACB680013	CAX690061
ACC EEC016		ACE690029	CCX €80038
costs		ACC7C0037	IBD 680036
AAX 670004	ACC660010	ACE650011	LEA700014
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AAX670009	IAA650024	BAX7COOCE	
AAX67C01C	LJX660006	data acquisition	systems
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ABX 690056 ACB660009		MAC680034 data processing	
	LJX670032		MIC 670002
ACB660009	LJX670032	data processing	MEC 670002 MDC 700004
ACB660009 cover gas	IJX670032 NXX700060	data processing JEX650020	
ACB660009 cover gas ACA690028	LJX670032 NXX700060 ACD690031	data processing JEX650020 MDC6700C1	
ACB660009 cover gas ACA690028 ACD660C17	LJX670032 NXX700060 ACD690031 ACD700024	data processing JEX650020 MDC6700C1 decay	MDC 700004
ACB660009 cover gas ACA690028 ACD660C17 ACD67C026	LJX670032 NXX700060 ACD690031 ACD700024 CCX680033	data processing JEX650020 MDC6700C1 decay ACE690022	MDC 700004 ACD 6700 27

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decommissioning		design criteria	
MFX700020.		ACB660015	IAE 700059
decomposition		BGX670045	MAC680034
ÅCD670026	CAX 690053	CDX670035	
ACD680015		design data	
decontamination		AAX670009	HCX710022
LIX69C008	LJX660006	ACE670024	IAA 650024
LIX690009		ACB7 10029	IAE700059
defects		BAX6E00C6	LDA 690012
ACE690026	ACE700025	BEX670012	MEB700003
ACE690032		development	
delayed neutrons		Ā A X 6 7 0 0 0 3	E CX 680031
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ACD690024	ACE700039	ACB700036	HIX €60026
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HBX690058	MAX650019	FCC690048	D LAC/0041
	NCD690055	disconnects	
HBX 690059	MCD690062		
H BX 700012			the second second second
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IAB670043	MEA640005	ACD670020	ACD 670027
IAE700059	NEB680001	dispersion	
IBA710005	MEC700053	LXX710026	

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disposal		AAX670010	M EA 680003
MDA700007	and the second sec	ACA6E0012	MDA 690002
dissolving		ACA6E0019	MEA 690005
ACD660017	CAX690052	ACE6 800 13	MEA 700006
CAX680032	CXX700049	ACB690029	MDA 700032
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ACC670018	LCA680008	JAB7C0017	MDC 680005
ACC 670025	LCA690037	JAE7 100 08	MCC690003
ACC680014	LCE680007	JCX690019	MDC 690016
ACC 680021	LCB710007	MCA680004	MEC700004
ACC690023	LCC7 10024	MCD650017	MDC 700005
ACC 690030	LJX660006	dynamics tests	
ACC700023	LJX660032	ACA650010	MDB 710002
ACD660011	LJX670032	ACA660008	MEC €60002
ACD660017	LXX660031	ACA690021	MDC 680002
ACE66C018	LXX710021	ND A6 60003	M EC 690003
distribution	MART TOUL T	MDA 690005	MDC 700005
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ACC680021	ACD690024	earthquakes	
ACC690023	ACD690031	ACB690029	ACB 700036
ACC 690030	ACD700024	economics	TCD 100030
ACC700023	ACD700024	ABX640004	CAX690053
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ACD670026	LDA690012	ABX690007	I AA 660030
ACD 680015	LDA 7 00046	AEX7 10020	NXX €90057
drain tanks	LDA / 00048	ADX690063	N XX 700057
ACB690029	IAC710013	BFX700056	NXX 700058
ACB700036	MAX650019	elasticity	NAA 700030
ACE650008	MD A640002	ACE670028	EDX640016
IAC 700047	HDA040002	EEX700041	F EX 640015
		ECX7 100 1 1	r ex 0400 IJ
drying ACA 650004		electrical circuit:	~
ductility		MAC6E0034	MAX650019
ACEEECODE	FBC650017	electrical conduct:	
ACE650014	FBE650015	ACD6E0015	ACE 66 0018
ACE660012	FBE650016	ACD680022	ACE670028
ACE660018	FBE660019	ACD690022	ACE680017
ACE67C021	PBE670029	ACD690031	ACE680024
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ACE680024	FEE680026	ACE660012	GCX680030
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ACE700039	FBE710018	AEX700055	NXX 700057
EBX700042	FBX 6400 15	ACB66C015	N XX 700058
and the second	FCC7C0040	ADX690063	NXX 700058
FBB 660021	FCC700044		
FBB660022		electrical propert.	
FBB690040	FCC710010	ACD660017	ACD 680022
FBB690041	FCE690043	ACD680015	ACD 700038
FBB70C028	FCE710004	electrolysis	100070040
FBC610001	GAX670033	ACC690023	ACD 670019
FBC 640017	GA X680028	ACC690030	ACD 6800 15
dynamic character		ACC7C0023	ACD 680022
AAX670009	MDA650001	ACC7 000 37	ACD690024

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LKX710001	LXX710026	ACE650014	
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ACD670027	ACE670028	IAF690014	MDA 660004
ACE650008	ACE680017	JAE6 900 18	M CA 680003
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ACE660018	ACE690032	ACB67C024	ACE 690032
ACE 670021	ACE700025	ACD680022	ACE700025
emergency cooling		ACE660012	E XX 700048
CAX 690053		ACE6600 18	FEX 640015
energy		ACE680024	HCK710022
NXX700060	1	ACE69C026	
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MDA680003	" ·	ABX670049	MCD 680010
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LXX700029		ACA650004	MCD €90055
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ACD660017	LCB710007	ACA660008	MCD700001
ACD700038	LCC7 10024	ACA660014	MDA 66 0003
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ACD680015	CXX700049	ACA70021	MDC 680002
ACD 680022	LCA680008	ACA700035	MEC €80005
ACD690024	LCA690037	ACA710028	MDC 690003
ACD690031	LDA 690012	ACE650011	M CC 700005
equipment	104070012	ACD660011	MEB 66 0028
ACD 67 C02 6	GDX690042	HEX700012	MEC700053
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ACE680024	MEE680001	ACD66C017	IAF 690014
ACE69C032	MEC700053	ACD670020	LXX 660031
E EX700042		ACD670026	MAX650019
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FBX640015		ACD68C015	MCA 680004
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ACD 680015	ACE690026	extraction columns	· · ·
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ACD 680022	ACE700025	LXX700029	LXX 710026
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ACE670028	FBB700028	ACE700025	FEE670031
ACE680017	FEX640015	ACE7C0039	FBE 680025
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ACE670028	ACE690026	FAX690045	FBX640015
ACE680017	ACE690032	FBA660020	FCD710016
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				ACC700037	
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	GAX680028	MAX650019		BBX670012	N XX 690046
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	HTGR			BFX7C0016	N X X 700058
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	LMFBR		MSRE		
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	LWBR			AEX690007	ACE700025
	NXX700060			ACA650004	CAX710023
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	A A X 670008	ED X690051		ACE7 10029	FEE700027
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	ABX64C004	HC X710022		ACC670025	FCX 700026
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	ABX 680035	IAC660024		ACC680021	HAX 700050
	A BX690007	IAC660025		ACC690023	HBX620006
	ABX690056	IAC700047		ACC690030	H EX 7000 12
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	ACB 690029	JAB690018		ACD6 800 16	J FX 670036
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	ACB700036	JAB710008		ACD680023	L AX 700013
	ACB710029	JCX690019		ACD690024	LAX710019
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	ACC 670018	LCA680008		ACE660018	LCC710024
	ACC670025	LCA690037		ACE670021	LHX690011
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Page 196	Keyword	Index	
MSRE		MDA6200C2	MDC 700005
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NCD70001	MDC690016	CAX690053	NXX 63000 1
MDA620001	MDC700004		

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 secondor 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	Mol	ten-Salt Reactor Programs	D	Ana
	AA	MSRP - Plans and Organizations		
	AB	MSRP - Technical Summaries	E	Gra
	AC	MSRP - Progress Reports		EA
		ACA MSRE		EB
		ACB Large MSR's		EC
		ACC Salt Processing		ED
		ACD Chemistry	F	Has
		ACE Materials	•	FA
	AD	MSR Activities Outside MSRP	•	FB
В	Rea	ctor Analysis		
	BA	Nuclear Data		
	BB	Static Neutronics		
	BC	Dynamics		
	BD	Thermal Effects	•	
	BE	Activation, Radiation and Shielding		FC
	BF	Fuel Cycle and Economics		
	BG	Safety	-	
	BH	Computer Programs	x	
С	Rea	ctor Chemistry		
	CA	Phase Relations		
	CB	Thermodynamics and Equilibria	G	Mat
	CC	Physical Properties	•	a
	CD	Rates and Diffusion		GA
	CE	Corrosion Reactions		GB
	CF	Fission Product Behavior		GC
	CG	Tritium Behavior		
	CH	Oxide Behavior		GD
	CI	Crystal Studies		GE
	CJ	Surface Effects		GF
	CK	Electrochemistry		GG
	CI.	Radiolysis		

J	Analytical	Cnemistry	r
_			-

- phite
 - Fabrication
 - Unirradiated Properties
 - Irradiation Effects
 - Applications
 - telloy N and Related Alloys
 - Alloys Leading to Hastelloy N
 - Standard Hastelloy N
 - FBA Microstructure
 - FBB Fabrication
 - FBC Mechanical and Physical **Properties**
 - FBD Corrosion
 - FBE Radiation Damage
 - Modified Hastelloy N
 - FCA Microstructure
 - FCB Fabrication
 - FCC Mechanical and Physical Properties
 - FCD Corrosion
 - FCE Radiation Damage
- erials Other Than Hastelloy N ind Graphite
 - Stainless Steels
 - Steels other than Stainless
 - Nickel and Ni-Base Alloys other than Hastelloy N
 - Molybdenum and Mo-Base Alloys
 - Brazing Alloys
 - Other Metals
 - Nuclear Control Materials

Subject Categories in MSRIS (continued)

H	Rea	ctor Component Development	\mathbf{J}	Ins	trumentation and Controls
	HA	Core		JA	General
	HB	Pumps			JAA Instrument Development
	HC	Heat Exchangers			JAB Plant Control
	HD	Steam Generators		JB	Nuclear Control and Plant Safe
	HE	Gas Injection and Removal		JC	Process
	HF	Valves		JD	Radiation and Contamination
		HFA Freeze Valves			Monitoring
		HFB Mechanical Valves		JE	Data Collection and Analysis
	HG	Flanges		JF	Communication and Surveillance
	HH	Heaters		JG	Electrical and Pneumatic
	HI	Other Components			Systems
			K	0pe	ration and Maintenance
Ι	Rea	ctor Design		KA	Operation
	IA	Reactor Plant	. *		KAA ARE
		IAA Early Molten-Salt Reactors			KAB MSRE
		IAB MSRE			KAC Other Molten-Salt Systems
		IAC One-Fluid MSBR (Reference Design)		KB	Maintenance
		IAD Other Thermal Molten-Salt			KBA MSRE Maintenance
		Reactors			KBB Other Molten-Salt and Radioactive Systems
		IAE MSBE			Radioactive Systems
	 	IAF Fast and Epithermal Molten-Salt Reactors		• • • • •	
i at i ta	IB	Systems			
		IBA Fuel			
· ··	-	IBB Coolant			
	en e	IBC Steam			
		IBD Gas			
	- 1997 -	IBE Containment			

Safety

Subject Categories in MSRIS (continued)

L Fue	el Preparation and Processing	MM	SRE
LA	Salt Procurement and	M	A Design
	Preparation		MAA Plant
LB	Fluorination		MAB Major Component
LC	Distillation		MAC Instrumentation and Controls
•	LCA Experimental Basis		MAD Auxiliary Systems and
	LCB Engineering Development		Components
	LCC Operating Experience	M	B Construction
LD	Reductive Extraction	. M	C Operation
	LDA Experimental Basis		MCA Program
	LDB Engineering Development		MCB Procedures
LE	Metal Transfer		MCC Training
	LEA Experimental Basis		MCD Experience
	LEB Engineering Development	M	D Analysis
LF	Oxide Precipitation		MDA Theoretical
	LFA Experimental Basis		MDB System Performance
	LFB Engineering Development		MDC Nuclear Performance
LG	Adsorption and Reduction	M	E Maintenance
LH	Salt Purification		MEA Principles
LI	MSRE Salt Processing		MEB Procedures
LJ	Plants for Two-Fluid MSBR		MEC Experience
LK	Plants for One-Fluid MSBR	M	F Decommissioning

N Miscellaneous

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The Followirg Index States The Category And Gives Feferences To Each Article Which Was Keyed To It

AAX				ACX6400C8	ACX680011
	AAX670003	AA X670008		ACX640014	ACX 6800 18
	AAX670004	AAX670009		ACX6500C3	ACX690020
	AAX 670005	AA X670010		ACX650009	ACX 690027
	AAX670006	AAX670011		ACX660007	ACX700018
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ABX				ACX670015	ACX710027
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ACX

Page	202	Category	Index		•
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	ACD 660011	ACD690031		FEE670029	FEE710018
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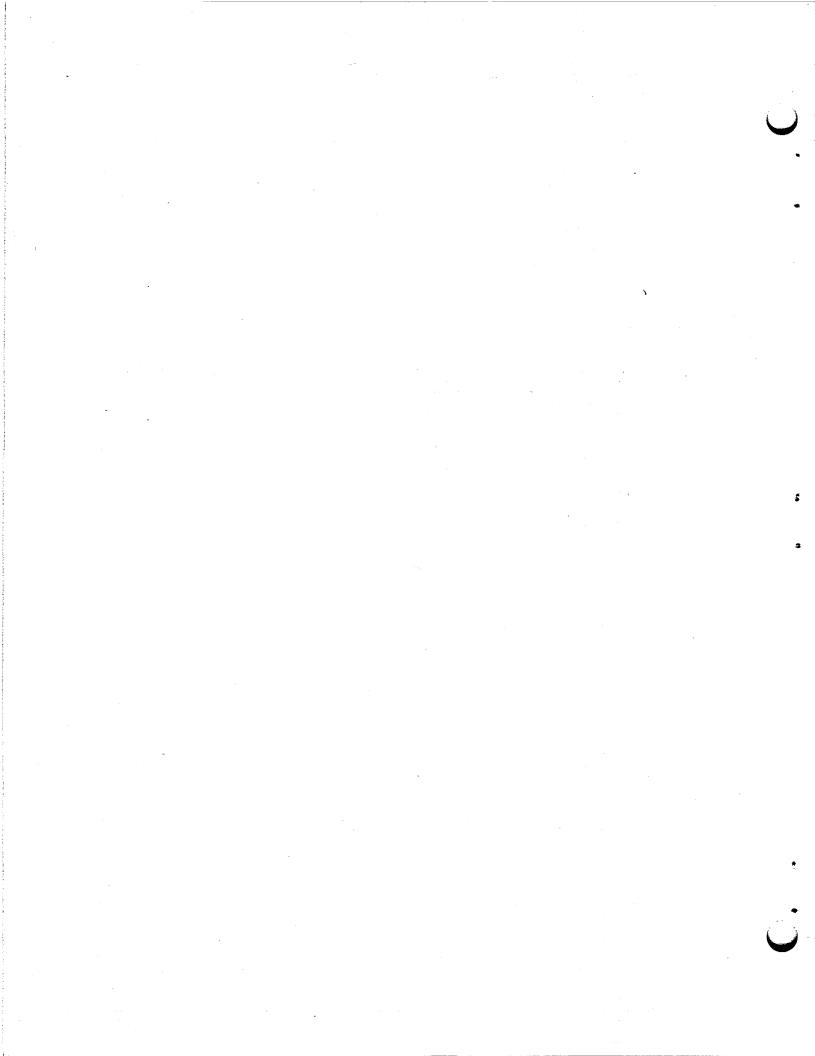
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	LJX670032	LKX 6 2000 3		MDA620001	MDA 670040
LKX		- · · · · ·		MDA6 20002	MEA 670041
	ACC690023	LKX700030		MDA6300C2	MDA 680003
	ACC700023	LKX7 10001		MDA640001	M EA 690001
	LKX620003			MDA640002	MDA 690002
LXX				MDA640006	MEA 690005
	LXX660031	LXX7 10021	(MDA 6400C7	MDA 700006
	LXX70CC29	LXX710026		MDA650001	M CA 700007
MAA	TYY 100052	LAA7 10020		NDA6600C3	MDA 700032
DAA	IAB 670043	LI X670013		MDA660004	MEA710003
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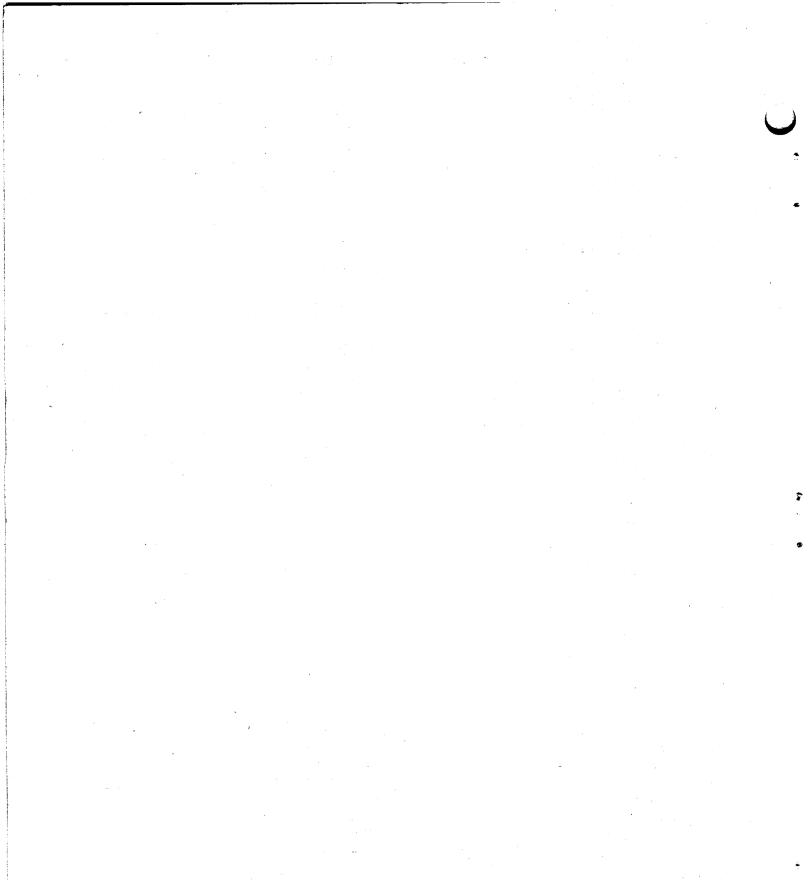
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