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Vol. I

LMFBR SAFETY

1. Review of Current Issues and Bibliography of Literature 1960-1969

J. R. Buchanan - G. W. Keilholtz

PREPARED FOR THE U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH
UNDER INTERAGENCY AGREEMENT ERDA NO 40-551-75 and 40-552-75

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LMFBR SAFETY

1. REVIEW OF CURRENT ISSUES AND BIBLIOGRAPHY
OF LITERATURE (1960-1969)

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PREPARED FOR THE U.S. NUCLEAR REGULATORY COMMISSION
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FOREWORD

The Nuclear Safety Information Center (NSIC), which was established in March 1963 at Oak Ridge National Laboratory, is principally supported by the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research. Support is also provided by the Division of Reactor Development and Demonstration of the U.S. Energy Research and Development Administration. NSIC is a focal point for the collection, storage, evaluation, and dissemination of safety information to aid those concerned with the analysis, design, and operation of nuclear facilities. A system of key words is used to index the information cataloged by the Center. The title, author, installation, abstract, and key words for each document reviewed are recorded at the central computing facility in Oak Ridge. The references are cataloged according to the following categories:

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7. Fission Product Release, Transport, and Removal
8. Sources of Energy Release under Accident Conditions
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17. Operational Safety and Experience
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PARTS AND METHOD OF INDEXING ABSTRACTS

LMFBR SAFETY

1. REVIEW OF CURRENT ISSUES AND BIBLIOGRAPHY OF LITERATURE (1960-1969)

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ABSTRACT

This report discusses the current status of liquid-metal fast breeder (LMFBR) development and one of the principal safety issues, a hypothetical core-disruptive accident (HCDA). Bibliographic information on worldwide LMFBRs relative to the development of the breeder reactor as a safe source of nuclear power is presented for the period 1960 through 1969. The bibliography consists of 1560 abstracts covering early research and development and operating experiences leading up to the present design practices that are necessary for the licensing of breeder reactors. Key-word, author, and permuted-title indexes are included for completeness.

INTRODUCTION

This bibliography of abstracts of literature relating to the liquid-metal fast breeder reactor (LMFBR) from the computer file of the Nuclear Safety Information Center (NSIC) provides valuable background information for those engaged in research, design, analysis, and licensing of breeder reactors. This is the first volume of a series on LMFBR safety and covers the literature for 1960 through 1969. This introduction to the bibliography includes a brief discussion of the current status of breeder development and one of the principal safety issues — the hypothetical core-disruptive accident (HDCA).

As attested to by ERDA and others, the development of the breeder reactor is essential to the assurance of adequate long-range supplies of nuclear fuel.^{1,2} The LMFBR is the favored fast breeder reactor system in the United States, Great Britain, West Germany, the USSR, Japan, Italy, and France. Current-generation LMFBRs use mixed oxide (plutonium-uranium)

fuel cooled by liquid sodium. Coolant temperatures range from 400°C (750°F) at the inlet to 540°C (1000°F) at the mixed core outlet while the system is kept at a low pressure of 2 to 10 atm.³ The feasibility of developing commercially attractive LMFBRs in the United States is founded on the extensive body of both light-water reactor (LWR) experience and sodium reactor technology.⁴⁻⁹ The latter includes several years of experience each with the Sodium Reactor Experiment, Hallam, Sefor, and Fermi reactors; 10 years of operating experience with the Experimental Breeder Reactor (EBR-II), plus experience with its predecessor EBR-I;¹⁰ the fuels and manufacturing technology developed for the Fast-Flux Test Facility (FFTF);¹¹ and the component designs for the Clinch River Breeder Reactor Plant (CRBRP);^{12,13} as well as the utilization of data and experience gained from the LMFBR development programs in Europe.¹⁴⁻²³ Selected plant characteristics are given in Table 1 for the world's LMFBRs.¹³

As for any nuclear reactor, the safety of the fast reactor depends upon ensuring the continuity of heat removal from the hot radioactive core. At the same time provisions must be made for shutting down the reactor (i.e., terminating the nuclear fission process) when required, so that heat production after shutdown is the decay heat. Thus a major task is to ensure that the reactor can be scrammed when required. All shutdown system designs of the present generation incorporate two independent neutron-absorber systems. Less than half the capacity of either system is usually sufficient to scram the reactor. Swelling and creep phenomena in materials due to radiation-induced void formation are receiving considerable attention. Special attention is paid to the removal of decay heat, generally through diverse heat-removal systems. Other areas of concern relate to the energetics and mechanical consequences of a core-disruptive accident. Major research and development programs are under way to provide further understanding of these safety phenomena, and commercial development of the breeder in the U.S. depends upon successful completion of this effort. The decision²⁴ as to commercialization is to be made by 1986.

The Comptroller General of the United States in a report²⁵ to Congress found that, "there are, ..., environmental and safety uncertainties

Table 1. Characteristics of various LMFBR plants

Reactor	Nation	Criticality	Power [MW(t)]	Fuel	Fuel loading (kgm)	Heat transport system ^{a,b}
EBR-1	U.S.	1951	1.2	U	64	1 loop
BR-5	USSR	1958	5→10	PUO ₂ (UC)	49.6	2 loop
DFR	U.K.	1959	60	UO ₂	200	24 loop
EBR-II	U.S.	1963	62.5	U	363	Pool
Enrico Fermi	U.S.	1963	300	U	2000	3 loop
Rapsodie	France	1966	20→40	Pu-UO ₂	300	2 loop
SEFOR	U.S.	1969	20	Pu-UO ₂	1920	1 loop + ACS
BOR-60	USSR	1970	60	UO ₂	176	2 loop
FARET ^c	U.S.	1972	50	Pu-UO ₂	260	1 loop + ACS
BN-350	USSR	1972	1000	Pu-UO ₂	2050	6 loop
Phenix	France	1973	560	Pu-UO ₂	3500	Pool
PFR	U.K.	1974	600	Pu-UO ₂	6000	Pool
FFTF	U.S.	1978	400	Pu-UO ₂	3000	3 loop
JOYO	Japan	1976	100	Pu-UO ₂	903	2 loop + ACS
BN-600	USSR	1978	1470	Pu-UO ₂	3900	Pool
SNR-300	West Germany	1980	736	Pu-UO _{2,2}	5000	3 loop + ACS
PEC	Italy	1979	~125	Pu-UO ₂	?	2 loop semipool
MONJU	Japan	1983	540	Pu-UO ₂	7900	3 loop
Super Phenix	France	1981	2900	Pu-UO ₂	?	Pool
CRBRP	U.S.	1983	975	Pu-UO ₂	7561	3 loop

^aACS = Auxiliary Cooling System.

^bThe loop system uses pipes to connect the reactor vessel, coolant pump, and heat exchanger; whereas the pool concept immerses the core, heat exchanger, and coolant pump in a single pool of sodium. The former has a smaller vessel, a smaller inventory of sodium, and greater ease in isolating components for maintenance. The pool-type system has a greater heat capacity because of the larger sodium inventory and less concern about leakage since there are no external pipes.

^cThe FARET project was cancelled in 1965.

regarding the LMFBR that remain to be resolved. These include hypothetical core disruptive accidents which may lead to ... recriticality. ... Only the recriticality issue is unique to the LMFBR and other fast reactors. The other issues are common to nuclear fission power in general, but their resolution is critical because the LMFBR would exacerbate the problems."

The LMFBR has one very important safety advantage over the LWR;²⁶ that is, there is no need for high-pressure components within the reactor containment. With no natural pressure source, the concern then becomes a question of whether there are any combinations of malfunctions that could lead to a power excursion of sufficiently high ramp rate to generate excessive pressure. Other advantages accrue from the good heat transfer characteristics of the sodium and its chemical reactivity, which make it efficient in retaining many fission products. On the other hand, the rapid oxidation or burning of sodium is important to safety assessments as is the reactivity associated with incidents that lead to voiding of sodium.

Design Philosophy

The approach to safety in the design of LMFBRs is similar to the "defense-in-depth" philosophy used in LWR design²⁷ (see Table 2). The philosophy, founded on the supposition that the probability of accidents decreases rapidly with severity, defines three levels of safety:¹³

1. Care in design, construction, and operation.
2. Standby engineered safety features such as redundant decay-heat-removal systems, emergency power systems, and two independent shutdown systems that are designed to control the consequences of equipment failures, operator error, and anticipated transients.
3. Other features, such as dynamic loading conditions for specific components, the reactor vessel, or containment structure, designed to control the consequences of events such as a major core-disruptive accident that have been judged to be of low probability.

The first level of safety emphasizes accident prevention through intrinsic design features such as quality assurance, redundancy, testability, inspectability, and fail-safe characteristics.

Table 2. Multilevel safety design of an LMFBR plant

Level	Objective	Operating category	Typical events within operating capacity	Typical safety design features
1	Provision of simple, reliable, and functional design free of defects, with inherent safe performance, fabricated and operated to the highest proven standards	Normal operation	Events that will normally occur: Full-power operation Startup and shutdown Random fuel-pin failure Refueling	Fuel assembly design to prevent flow blockage Fuel of proven performance through life Core restraint to provide negative power coefficient Adequate Doppler coefficient Low-pressure coolant system with wide margin to boiling Maximum use of proven technology and hands on maintenance Radioactive-waste-treatment system
2	Provision of protection systems to provide adequate response of system in the event of all identified transients	Anticipated faults	Events that, based on experience, are expected to occur at least once in the life of the plant: Loss of off-site power Loss of power to one pump Operator error Spurious scrams	Multiple reactor coolant loops Decay-heat-removal redundancy Battery power supplies for vital services Guard vessels for leak protection Inert atmospheres in sodium cells
		Unlikely faults	Events which are not expected to occur individually but which might occur once during the life of the plant, based on the total list of such events: Pump seizure Failure of dump system Steam-generator leak	Two independent shutdown systems Sodium-water reaction protection system
3	Provision of extra capability to cope with extremely unlikely events that are never expected to occur and additional design requirements to provide prudent margin for unforeseen events	Extremely unlikely faults	Events that are never expected to occur: Maximum flood, or earthquake, or tornado Large sodium fire Large sodium-water reaction	Site selection Flood barriers Containment isolation Low-leakage containment
		Hypothetical events	Design margins to provide for unforeseen events	Capability to accept extra thermal loads in the core support structure Capability to accept dynamic loads in vessel and primary system components Geometric requirements around the vessel Control-room radiological protection

Safety level 2 philosophy is implemented in the plant-protection systems, reliability programs, heat-removal-system protection, redundant power supplies, etc. An extensive reliability program is currently under way for the CRBRP to help set the design basis for the plant protection system.²⁸ The reliability criterion used is that stated in the regulatory guidance provided for the analogous accident in LWRs, "Anticipated Transient Without Scram" (ATWS),²⁹ as follows: "The safety objective is that the likelihood of all accidents with significant consequences not included in the design basis envelope should not be greater than one in one million per year, i.e., should not occur with a failure rate greater than 10^{-6} per year. For the particular potential failure path of ATWS, the staff believes that a failure rate of the order of one-tenth of the overall safety objective is an appropriate objective."

The third level of safety supplements the first two by providing plant response to low-probability faults. Analysis of the HCDA illustrates the present state of technology in treating the third level of safety and shows that the provision of design margin involves an evaluation of the entire reactor plant. The core-disruptive accident is the most controversial area of LMFBR reactor safety evaluation. The prime conceivable mechanistic initiator of a core-disruptive accident is the combination of an accident, such as complete loss of coolant flow, with a total inability to "scram" or shut the reactor down (i.e., a disabling failure in the second level of safety). For this reason a major effort is being put forth in the LMFBR development program to demonstrate that the reliability of scram as well as system structural integrity and post-shutdown core cooling is so high as to preclude choice of the HCDA as a design basis.

Hypothetical Core-Disruptive Accident

A number of views support the inclusion of low-probability HCDAs in the design basis of the reactor containment. Members of the NRC Regulatory Staff³⁰ (1975) have stated: "It is our opinion that the occurrence of low probability core disruptive accidents, leading to loss-of-coolable geometry, cannot be completely precluded, and the state of technology is

such that precise quantification of the accident consequences cannot be made at this time. It is therefore important that, from the viewpoint of the health and safety of the public, the consequences of core disruptive accidents must be safely accommodated in a fast breeder reactor through appropriate features incorporated into the containment design."

Relative to the CRBR design, the report²⁵ of the U.S. Comptroller General stated: "A major LMFBR design safety issue is whether a HCDA should be included as a Design Basis Accident in LMFBRs. On the basis of the current state of technology and experience, NRC safety experts told us that such an accident, although unlikely, was within the realm of possibility and should be provided for in the design of the first LMFBR demonstration plant on the Clinch River in Tennessee until a better understanding of the HCDA phenomenon is achieved."

However, in a May 1976 letter to the CRBRP, the NRC stated that "it is our current position that the probability of core melt and disruptive accidents can and must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum. We will therefore not consider CDAs as design basis accidents. ... Nevertheless, ... prudence dictates that additional measures be taken to limit consequences and reduce residual risk from potential CRBR accidents having a lower probability than design basis accidents to insure that the public health and safety is adequately protected."³¹ NRC will require that the function specifications for protecting the containment integrity provide for "a core mechanical work energy release of 1200 MW-sec based on fuel vapor as the working fluid and expansion to 1 atmosphere."³¹

Earlier (1974) Nicholson³² (of Ohio State University and formerly with Argonne National Laboratory) expressed the following view: "Over the past few years there has been an increasing body of opinion developing that the studies of hypothetical meltdown and disassembly accidents contribute nothing to the design of safe reactors and detract from the objective of licensing the fast breeder. ... Hypothetical accident studies are not only useful but in fact an essential ingredient in both the design process and licensing. In particular, one design criterion must be the containment of all effects resulting from a hypothetical disassembly accident. ... [Nicholson] ... believes this statement to be true for any fast breeder that is built within at least the next fifteen years and perhaps

longer. There is no way to avoid this design requirement in the foreseeable future. It is well within the capacity of the industry to design for containment, but not without the expenditure of considerable effort. It is essential that all designers accept this challenge and proceed to accommodate the necessary perturbations in the design process."

Nicholson suggested an energy release in the order of 1000-MWsec (~ 500 lb TNT) be used as a target value for the design basis of containments for a demonstration size plant until probabilities of failure below the 10^{-6} to 10^{-7} range can be demonstrated.³² As the design and detailed accident analysis progresses, the energy release value can be adjusted toward the end goal of an optimum design parameter. The design also depends on the choice of the degree of fuel-coolant interaction and of the working fluid since the mechanical work potential depends dramatically on whether the fluid is fuel vapor or sodium vapor. The damage potential is much lower if fuel vapor is the only working fluid.

The French³³ have taken the HCDA "into account for the dimensioning of the primary and secondary containment in a conservative way, even quite arbitrarily and regardless of the initial causes." For SUPER-PHENIX, they have assumed as a working hypothesis "a coherent gravity compaction of the core leading to a \$60/sec ramp rate." A design criterion provides that "the safety vessel, its upper closure structures, and structures around them must safely contain the consequences of whole core accidents." Containment response to the HCDA has been analyzed by calculations and explosive tests. For the PSAR, a mechanical energy release of at least 800 MWsec was assessed.³⁴

Smidt,³⁵ of West Germany, found "that the safety design basis of present fast reactors compare favorably with that of LWRs." However, for future larger plants, he called for "a better definition especially with respect to the whole core accident" and suggested that "this might even become a matter" that will determine the "general feasibility of large commercial LMFBRs." Hübel,¹⁴ of West Germany, reported that the SNR-300 will have a "provision of a two-stage defense line against HCDAs. The reactor vessel as a primary containment boundary has to be designed against a minimum mechanical energy of 150 MW-sec while 370 MW-sec are required for the integral containment system."

In the United Kingdom, Farmer²¹ suggests that among the "range of possible or hypothetical accidents it is not unusual to see a description of a power excursion caused by a reactivity change of \$100 per second. This implies the possible addition of a few dollars over a few tens of milliseconds which would seem to ... [be] ... a possibility which could never be entirely ruled out." Farmer suggests that the margin for effectiveness of the containment is not adequate if his current accident model is extrapolated to a 1200-MW(e) reactor. However, he hopes "to show that the model is excessively pessimistic on several counts" and "that the modest rearrangement of the core necessary to cancel the reactivity change can be brought about by moderate pressures exerted at lower temperatures by fission products contained in the fuel, or indeed by any other mechanism or materials added for that specific purpose." It is believed that "these effects would reduce the damaging work capability of the accident by at least an order of magnitude, would lead to low central pressures and only partial core melting." It would then be possible "to show a capability of engineering containment."

An optimistic view was expressed by ERDA (1975),³⁶ "Future LMFBRs can be expected to differ from early U.S. designs. Perhaps the most significant current change results from the conviction that we have essentially exhausted the search for sequences that could result in energetic HCDAs. Other than conceivable cladding motion or sodium void reactivity effects, we feel there remains no credible potential for damaging energetics from an LMFBR HCDA. It is likely that these exceptions could, if necessary, be controlled by design. Current design requirements for containment of energetics other than that implied by credible cladding or sodium voiding effects, then, represent no more than margin to cover for the fact that confirmatory experiments have not yet been completed."

ERDA has made conceptual plans for a Safety Research Experimental Facility (SAREF) to investigate the energetics of an HCDA under more prototypic conditions. The facility would perform tests to determine the potential for recriticality and associated energy releases.²⁵ Part of the SAREF package includes a test reactor that would be built in a remote

site to carry out tests involving core disruptive accidents that would include recriticality.³⁷

Instructions for Using Bibliography

The abstracts in this bibliography were selected from the NSIC computer file using REACTOR, LMFBR and REACTOR, LMCR as key words. The document descriptions added to the file prior to January 1970 are found in this first volume. Abstracts added since that time will be issued in follow-up volumes covering periods of approximately two years.

The abstracts in this volume are sorted into the NSIC category scheme and listed by accession numbers that are essentially in chronological number. The bibliography is followed by key-word, author, and permuted-title indexes.

A brief explanation of the scope of each category* into which the abstracts are presented follows:

1. General Safety Considerations — Deals with all safety aspects of radiation policy, safeguards, standards, codes, cost, financial liability, and insurance. Also covers the comparative risk to public health and safety from nuclear and nonnuclear hazards.

2. Siting of Nuclear Facilities — Considers factors used in evaluating sites, for example, facility design, proposed operation, population density, use characteristics of the environs, physical characteristics of the site, and earthquake considerations.

3. Transportation and Handling of Radioactive Materials — Pertains to shipping containers, shipping regulations, criticality safety, transportation accidents, design of tiedowns, shipping condition (impact, fire, vibration, etc.), heat-transfer capability, etc.

5. Heat Transfer and Thermal Transients — Considers all facets of the analysis of postulated accidents. Includes burnout heat flux, heat transfer, reliability analyses, thermal-mechanical effects, in-reactor experiments, coolant-activity buildup, and experiments such as in LOFT.

*Categories 4, 14, 15, and 19 through 21 are inactive; therefore they are not included.

6. Reactor Transients, Kinetics, and Stability — Includes the studies, analytical and experimental, such as those conducted in TREAT and SPERT, in which the transient behavior of reactors and criticality accidents are examined. Deals, for example, with reactivity effects due to voids and temperature, Doppler coefficient, etc.

7. Fission Product Release, Transport, and Removal — Pertains to the release of fission products from various materials and their movement within a nuclear-facility containment system. Transport of fission products involves the physical and chemical characterization of the released radionuclides and also includes mechanisms such as deposition, adsorption, filtration, fallout, plateout, etc.

8. Sources of Energy Release under Accident Conditions — Considers nuclear, Wigner, and gamma energies, energy related by chemical reactions, metal-water reactions, and all other types of energy that might be released as a result of a nuclear accident.

9. Nuclear Instrumentation, Control, and Safety Systems — Includes the design of control and safety systems for various nuclear processes, as well as the required instrumentation and hardware. The problems involved are the performance required of safety systems; the specification of instrumentation; the concepts of coincidence, redundancy, failure modes, and reliability; adequacy of shutdown margins; design features of mechanical devices; and related subjects.

10. Electrical Power Systems — Deals with routine and emergency means of supplying electrical power to nuclear facilities of any kind. Includes all phases of the generation and distribution of electrical power to and within a nuclear power plant. Considers the reliability of the power system and the effects of system disturbances on safety and operation of the reactor.

11. Containment and Structural Integrity of Nuclear Facilities — Encompasses all aspects of pressure containment for all types of nuclear facilities. Includes fuel cladding, piping, pressure vessels, containment, and reactor buildings. Major concerns are design considerations, penetrations, structural integrity, leak testing, quality assurance, and radiation damage.

12. Plant Safety Features — Reactor — Covers all engineered safety features of reactors except containment. Includes systems for reducing temperature and pressure, air cleaning, disposal of liquid waste, and emergency core cooling. Other topics include maintenance, decontamination, fuel-element protective devices, etc.

13. Plant Safety Features — Nonreactor — Deals with facilities associated with the external fuel cycle and with the use of radioactive material created by nuclear fission. Major divisions are mining, ore conversion, enrichment, fuel-element fabrication, hot cells, fuel processing, fuel conversion, waste disposal, and radioisotopes.

16. Meteorological Considerations — Considers not only diffusion and deposition of radioactive material near the surface of the earth in connection with reactor operations, but also the atmospheric transport and fallout in the troposphere and stratosphere as a result of nuclear-weapons tests.

17. Operational Safety and Experience — Covers the safety aspects of routine reactor operation and of incidents or unusual operating occurrences, large or small. Includes power, research, and test reactors, also fuel-processing plants.

18. Safety-Analysis and Design Reports — Deals largely with safety-analysis and design reports for reactors not yet operational. The reports are generated by facility designers and by the NRC Regulatory Staff.

References on mathematical models and codes usually appear in Category 5 or 18; however, some may be expected to occur in any of the categories depending on the subject matter of the model or code.

References

1. *Report of the Cornell Workshops on the Major Issues of a National Energy Research and Development Program*, Cornell University, Ithaca, N.Y. (December 1973).
2. *Report of the Liquid Metal Fast Breeder Reactor Program Review Group*, ERDA-1 (January 1975).
3. J. Graham, *Fast Reactor Safety*, Academic, New York, 1971.
4. D. Okrent, *Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors*, Oct. 11-14, 1965, ANL-7120.
5. D. Okrent, *Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design*, Nov. 7-9, 1968, ANL-7520, Parts I and II.
6. *Proceedings of the Fast Reactor Safety Meeting*, Beverly Hills, Calif., Apr. 2-4, 1975, CONF-740401-P1,P2,P3.
7. M. H. Fontana, "1974 ANL Topical Meeting on Fast Reactor Safety," *Nucl. Safety* 16(2), 141 (March-April 1975).
8. *Liquid Metal Fast Breeder Reactor Program Plan*, Vol. 10, *Safety*, WASH-1110, 2d Ed. (December 1972).
9. D. Okrent, "Design and Safety in Large Fast Power Reactors," *At. Energy Rev.* 7, 135 (1969).
10. L. J. Koch, "EBR Plus 10: Is the U.S. Formula Right for Effective LMFBF Development?" *Nucl. News* 17(10), 55 (August 1974).
11. D. E. Simpson, *FFTF Design for Safety, Proceedings of the Fast Reactor Safety Meeting*, Beverly Hills, Calif., Apr. 2-4, 1974, CONF-740401-P2.
12. *Clinch River Breeder Reactor Plant, Preliminary Safety Analysis Report*, Docket 50-537 (April 1975).
13. J. Graham, "Nuclear Safety Design of the Clinch River Breeder Reactor Plant," *Nucl. Safety* 16(5), 564 (September-October 1975).
14. H. J. Hübel, "The Safety Related Criteria and Design Features for SNR," *Proceedings of the Fast Reactor Safety Meeting*, Beverly Hills, Calif., Apr. 2-4, 1974, CONF-740401, p. 1.
15. D. Smidt, "Selection of Safety Design Basis of Fast Reactors in the Federal Republic of Germany," *ibid.*

16. *Proceedings of International Conference on Engineering of Fast Reactors for Safe and Reliable Operation, Karlsruhe, October 1972, Gesellschaft fur Kernforschung MBH, Karlsruhe, 1973.*
17. F. Justin et al., "Safety Criteria for the Super-Phenix Reactor," *Trans. Amer. Nucl. Soc.* 20, 464 (1975).
18. C. Girard, "French Program on LMFBR Safety," *Proceedings of the Fast Reactor Safety Meeting, Beverly Hills, Calif., Apr. 2-4, 1974, CONF-740401-P1.*
19. H. J. Teague, "Fast Reactor Safety Research Needs, What, Why, How, and When," *ibid.*
20. G. Kessler, "Fast Reactor Safety Research Needs Within the Debenelux-Project SNR," *ibid.*
21. F. R. Farmer, "Contribution to Panel Discussion on Selection of Safety Design Bases for Fast Power Reactors," *ibid.*
22. "USSR Details Design Criteria, Economics of Fast Breeder Reactors Now in Service," *Energy Digest* 3, 440-443 (Sept. 30, 1973).
23. E. Gilbey, "Breeder Reactors - Design for Safety," *New Scientist*, p. 418, Feb. 22, 1973.
24. "Commercialization Decision by 1986 - Seamans," *Nucl. News* 19(2), 31 (February 1976).
25. "The Liquid Metal Fast Breeder Reactor: Promises and Uncertainties," Issue Paper to the Congress by the Comptroller General of the United States, OPS-76-1, pp. 63-77 (July 31, 1975).
26. R. B. Nicholson, "The Role of Hypothetical Accident Analysis in Fast Breeder Design and Licensing," *Proceedings of the Fast Reactor Safety Meeting, Beverly Hills, Calif., Apr. 2-4, 1974, CONF-740401-P3*, p. 1660.
27. *The Safety of Nuclear Power Reactors (Light-Water-Cooled) and Related Facilities*, WASH-1250 (July 1973).
28. *CRBR Project, Reliability Program Plan*, Docket 50-537 (Jan. 9, 1976).
29. *Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors*, WASH-1270 (September 1973).
30. T. P. Speis et al., "Studies of Core Disruptive Accidents and Licensing Aspects of Fast Breeder Reactors," *Trans. Amer. Nucl. Soc.*, 20, 547 (1975).

31. R. P. Denise, U.S. Nuclear Regulatory Commission, letter to L. W. Claffey, CRBRP Office, May 6, 1976.
32. R. B. Nicholson, "The Role of Hypothetical Accident Analysis in Fast Breeder Design and Licensing," *Proceedings of the Fast Reactor Safety Meeting, Beverly Hills, Calif., Apr. 2-4, 1974*, CONF-740401-P3.
33. J. F. Petit, "Safety Design Basis for Power LMFBR in France," *ibid.*
34. J. F. Petit, "Creys-Malville Plant (Super Phenix) Main Inherent and Design Safety Characteristics," *Trans. Amer. Nucl. Soc.* 22, 479 (1975).
35. D. Smidt, "Selection of Safety Design Basis of Fast Reactors in the Federal Republic of Germany," *Proceedings of the Fast Reactor Safety Meeting, Beverly Hills, Calif., Apr. 2-4, 1974*, CONF-740401-P3.
36. W. H. Hannum, J. D. Griffith, and A. C. Millunzi, "Reactors - Safe at Any Speed," *Trans. Amer. Nucl. Soc.* 22, 483 (1975).
37. J. D. Allen et al., *Conceptual Design for a Fast Reactor Safety Test Facility, Preliminary Report*, LA-6031-MS (October 1975).

BIBLIOGRAPHY

CATEGORY 1
GENERAL SAFETY CONSIDERATIONS

1-004268 DATED 1964
DE VATHAIRE F + VERNIER P + PASCOUET A
THE CONCEPT OF NUCLEAR SAFETY IN FRANCE - THE INFLUENCE OF SAFETY REQUIREMENTS ON REACTOR DESIGN
COMMISSARIAT A L'ENERGIE ATOMIQUE
17 PAGES- IAEA- MAY 1964, A/CONF. 28/P/82- PRESENTED AT THIRD U. N. INTERNATIONAL CONFERENCE ON THE PEACEFUL
USES OF ATOMIC ENERGY, GENEVA, AUGUST 31-SEPTEMBER 9, 1964

A REVIEW IS PRESENTED OF THE REACTOR SAFETY EXPERIENCE IN FRANCE ON THE FOLLOWING REACTOR TYPES-
HEAVY WATER-GAS REACTORS, FAST NEUTRON REACTORS, AND WATER RESEARCH REACTORS. THE SAFETY RULES
AND THEIR EFFECTS ON THE REACTOR DESIGNS AND POWER LIMITS ARE DISCUSSED. THE CORRESPONDING
SAFETY STUDIES ARE ALSO DESCRIBED.

ADMINISTRATIVE CONTROL + CONTAINMENT + DESIGN CRITERIA + REACTOR, FAST + FRANCE + REACTOR, GCR + REACTOR, HWR
+ REACTOR, LMCR

1-004609 DATED 1963
RAPSDODIE - A VITAL FAST REACTOR PROJECT
1 PAGE- NUCLEAR ENGINEERING, 8(88)- 316 (SEPTEMBER 1963)

RAPSODIE IS TO BE A SODIUM-COOLED, EXPERIMENTAL FAST REACTOR WITH AN ALL-PLUTONIUM CORE. IT IS
DESIGNED TO HAVE A THERMAL POWER OUTPUT OF 20 MW. THE DESIGN OF RAPSDODIE WAS SUCH AS TO PROVIDE
INDUSTRIAL EXPERIENCE THAT COULD BE USED DIRECTLY IN THE DESIGN OF A PROTOTYPE POWER REACTOR.
THE FIRST CORE OF RAPSDODIE IS TO BE A CERAMIC MIXTURE OF OXIDES UO_2 - PUO_2 . A DESCRIPTION AND
SCHEMATICS OF THE REACTOR ARE GIVEN.

REACTOR, FAST + FRANCE + REACTOR, LMCR + REACTOR DESCRIPTION

1-006038 DATED 1965
TEITEL RJ + BROWN JB
LIQUID METAL BREEDER REACTOR (LMB) NUCLEAR EVALUATION
DOUGLAS AIRCRAFT CO., INC., SANTA MONICA, CALIFORNIA
12 PAGES, 4 TABLES, 8 FIGURES, 18 REFERENCES- NUCLEAR APPLICATIONS, 1(1)- 13-24 (FEBRUARY 1965)

THE LIQUID METAL BREEDER (LMB) REACTOR IS AN INTERNALLY-COOLED FLUID FUEL REACTOR BASED UPON A
TH232-U233 THERMAL BREEDER CYCLE. IT EMPLOYS A MOLTEN LEAD COOLANT, A URANIUM-BISMUTH SOLUTION
FUEL, A THORIUM BISMUTHIDE DISPERSION IN LEAD-BISMUTH BLANKET FLUID AND A GRAPHITE MODERATOR.
HEAT FROM THE FUEL IS TRANSFERRED THROUGH A GRAPHITE FUEL ELEMENT TO THE COOLANT WHICH TRANSPORTS
THE HEAT TO AN EXTERNAL BOILER AND PUMP. THIS ARRANGEMENT OVERCOMES THE MAJOR DISADVANTAGES
FOUND IN PREVIOUS EXTERNALLY-COOLED LIQUID-METAL-FUEL REACTORS. LMB REACTOR TECHNOLOGY CAN LEAD
TO AN EFFICIENT BREEDER, EVEN IN LARGE POWER SOURCES, AND WARRANTS FURTHER ENGINEERING
EVALUATIONS.

REACTOR, BREEDER + REACTOR, LMCR + REACTOR DESCRIPTION

1-006974 ALSO IN CATEGORY 6 DATED 1965
AMERICAN NUCLEAR SOCIETY FAST REACTOR TECHNOLOGY NATIONAL TOPICAL MEETING
363 PAGES, FIGURES, TABLES, REFERENCES- APRIL 28, 1965, ANS-100 (CONF-650417)- ANS NATIONAL TOPICAL MEETING ON
FAST REACTOR TECHNOLOGY, DETROIT, MICHIGAN, APRIL 26-28, 1965

TWENTY-EIGHT PAPERS IN SESSIONS ON REACTOR OPERATING EXPERIENCE, ADVANCED CONCEPTS, FUEL
TECHNOLOGY, PHYSICS AND SHIELDING, SAFETY PROGRAMS AND CORE DESIGN, AND MECHANICAL COMPONENTS AND
SODIUM TECHNOLOGY.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + RADIATION DAMAGE +
REACTIVITY EFFECT + REACTOR KINETICS + REACTOR STABILITY + POWER DISTRIBUTION + OPERATING EXPERIENCE SUMMARY
+ DESIGN STUDY + DOUNREAY (LMFBR) + SEFOR (RE) + SHIELDING + MAIN COOLING SYSTEM + FUEL ELEMENTS

1-007695 ALSO IN CATEGORIES 6 AND 9 AND 17 DATED 1965
OKRENT D
A LOOK AT FAST REACTOR SAFETY
ARGONNE NATIONAL LABORATORY
25 PAGES, 1 FIGURE, 2 TABLES, 131 REFERENCES- NUCLEAR SAFETY 6(4)- 317-342 (SUMMER, 1965)

DESIGN HISTORY (DOUNREAY REACTOR, EBR-II, ENRICO FERMI, PATTERNS AND TRENDS, SAFETY-RELATED DESIGN
AND OPERATING EXPERIENCE). DIFFERENCES IN NUCLEAR SAFETY FOR LARGE-SODIUM-COOLED FAST REACTORS
(DELAYED-NEUTRON FRACTION, DOPPLER COEFFICIENT, SODIUM COEFFICIENT AND VOID EFFECT, STABILITY,
MAXIMUM ACCIDENT CALCULATIONS). LARGE-SODIUM-COOLED-FAST REACTOR DESIGNS AND ACCIDENT STUDIES,
NON-NUCLEAR SAFETY CONSIDERATIONS, COOLANTS OTHER THAN SODIUM (STEAM, GAS, OTHER METAL COOLANTS),
SAFETY PHILOSOPHY AND SAFETY RESEARCH, REFERENCES.

REACTOR, BREEDER + DELAYED NEUTRON + DOPPLER COEFFICIENT + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LARGE +
REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR STABILITY + SODIUM + SODIUM COEFFICIENT + STEAM +
VOID COEFFICIENT + OPERATING EXPERIENCE SUMMARY + HELIUM + DOUNREAY (LMFBR) + SAFETY PRINCIPLES AND PHILOSOPHY

1-010160 DATED 1965
SMIDT D + FRISCH W + GIORDANO P + HEUSENER G + KESSLER G + KREWER KH + MERK W + MALMBERG T + SCHONFELD E

CATEGORY 1
GENERAL SAFETY CONSIDERATIONS

1-010160 *CONTINUED*
SAFETY AND COST ANALYSIS OF A 1000 MWE SODIUM COOLED FAST POWER REACTOR
KERNFORSCHUNGSZENTRUM, KARLSRUHE, INSTITUT FÜR REAKTORENTWICKLUNG
KFK-398 +. 38 PAGES, 14 FIGURES, 6 TABLES- OCTOBER 1965

THE KARLSRUHE GROUP HAS COMPLETED A DESIGN STUDY OF A 1000-MWE SODIUM-COOLED FAST-BREEDER REACTOR. THE MAIN FEATURES OF THE CORE ARE PLAIN CYLINDRICAL SHAPE, OXIDE FUEL OF 87 PERCENT THEORETICAL DENSITY, NO MODERATOR LIKE BED AND A MODERATE FLATTENING WITH H/D EQUALS 1/3. THE PRIMARY CIRCUIT IS OF THE LOOP TYPE WITH SOME FEATURES OF THE POOL TYPE. THERE ARE TWO INTERMEDIATE HEAT EXCHANGERS AND FOUR PUMPS, AND FOUR SECONDARY LOOPS. ANALYSIS HAS SHOWN THAT OUR PROPOSED 1000-MWE REACTOR IS VERY SAFE. INFINITE RAMPS OF THE ORDER OF 10 DOLLARS/SEC CAN BE COUNTERACTED BY A SPRING-DRIVEN SAFETY SYSTEM, MUCH MORE THAN WILL ACTUALLY OCCUR. COOLANT BOILING IN LARGER REGIONS OF THE CORE CANNOT OCCUR AS LONG AS THE SAFETY SYSTEM IS OPERATING.

ACCIDENT ANALYSIS + REACTOR, FAST + REACTOR, LMCR + SAFETY REVIEW + REACTOR SAFETY SYSTEM + SODIUM + DESIGN STUDY + GERMANY

1-010510 DATED 1966
BRITISH FAST REACTOR GOES AHEAD
3 PAGES, 5 FIGURES, EURONUCLEAR, 3(3), PP. 123-125 (MARCH 1966)

CONSTRUCTION ON THE UKAEA PROTOTYPE FAST REACTOR IS TO START IMMEDIATELY. THE PFR HAS BEEN DESIGNED TO HAVE GOOD INHERENT SAFETY. THE SODIUM IN THE REACTOR VESSEL IS AT ATMOSPHERIC PRESSURE AND THE PIPEWORK ON THE HIGH PRESSURE SIDE OF THE PRIMARY PUMPS IS CONTAINED WITHIN THE VESSEL. PUMP MOTORS ARE INCLUDED ON THE PRIMARY PUMPS AND EVEN IF THESE SHOULD FAIL THE NATURAL CIRCULATION OUGHT TO BE SUFFICIENT TO COOL THE SHUTDOWN CORE. IF THE SECONDARY COOLANT CIRCUIT SHOULD FAIL THERE ARE ALTERNATIVE THERMAL SYMPHON NAK HEAT EXCHANGERS INCORPORATED IN THE VESSEL AND CONNECTED TO AIR-COOLED HEAT EXCHANGERS IN THE MAIN VENTILATION STACK OF THE BUILDING.

REACTOR, FAST + REACTOR, LMCR + SODIUM + UNITED KINGDOM + NAK + DESIGN STUDY + DOUNREAY (LMFBR) + PFR (LMFR)

1-011398 DATED 1965
SMIDT D + FRISCH W + GIORDANO P + HEUSENER G + KESSLER G + KREWER KH + MERK W + MALMBERG T + SCHONFELD E
SAFETY AND COST ANALYSIS OF A 1000-MWE SODIUM-COOLED FAST POWER REACTOR
INSTITUTE FOR REACTOR DEVELOPMENT, KERNFORSCHUNGSZENTRUM, KARLSRUHE, GERMANY
ANL-7120 +. 13 PAGES, 6 TABLES, 14 FIGURES, OCTOBER 11-14, 1965, PAGES 33-45, PROCEEDINGS OF THE CONFERENCE ON SAFETY, FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS, CFSTI, \$9.25 CY, \$3.75 MN

THE MAIN FEATURES INCLUDE A CYLINDRICALLY SHAPED, OXIDE FUEL (87 PERCENT THEORETICAL DENSITY), NO MODERATOR, SUCH AS BED, AND A MODERATE FLATTENING WITH A HEIGHT-TO-DIAMETER RATIO OF 1 TO 3. THIS RESULTS IN A RELATIVELY LARGE NEGATIVE DOPPLER COEFFICIENT AND AN INTERNAL BREEDING RATIO CLOSE TO UNITY. THE VOID COEFFICIENT WILL THEN BE SOMEWHAT LARGER THAN THAT IN SEVERAL OTHER DESIGNS. ITS EFFECT ON OVERALL SAFETY IS ONE OF THE TOPICS OF THIS PAPER. THE CAPITAL COSTS, BASED ON STUDIES FROM INDUSTRY, WERE CALCULATED TO BE \$15/KW.

REACTOR, BREEDER + REACTOR, LMCR + DESIGN STUDY + GERMANY + ECONOMIC STUDY + SAFETY PROGRAM

1-012341 ALSO IN CATEGORY 17 DATED 1966
INDUSTRY, AEC LEADERS CLASH OVER FAST BREEDER PROGRAM
ATOMIC ENERGY COMMISSION
3 PAGES, NUCLEONICS WEEK 7(45), PAGES 1-3, (NOVEMBER 9, 1966)

INDUSTRIAL REPRESENTATIVES INDICATED THAT THEY NEEDED SYSTEMS-ENGINEERING EXPERIENCE WITH SODIUM-COOLED REACTORS, WHICH THEY CANNOT GET FROM JUST ONE OPERATING REACTOR (EBR-2). SECOND, AN INTENSIVE PROGRAM ON SODIUM-REACTOR SAFETY DEVELOPMENT IS NEEDED. FINALLY, THE PROGRAM NEEDS TO BE ACCELERATED. AEC COMMISSIONER RAMEY INDICATED THAT SEVERAL 200- TO 500-MW(E) DEMONSTRATION BREEDER PLANTS WOULD NEED TO BE BUILT, REQUIRING A MASSIVE DEVELOPMENT PROGRAM FOR WHICH PRESENT TECHNOLOGY IS NOT READY.

REACTOR, FAST + REACTOR, LMCR + REVIEW

1-015836 DATED 1967
SHAW M
NUCLEAR POWER - THE NEXT DECADE OF DEVELOPMENT IN THE U.S.
DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY, U.S. ATOMIC ENERGY COMMISSION
21 PAGES, 13 FIGURES, APRIL 25, 1967, ATOMIC POWER CONFERENCE, CHICAGO, ILLINOIS, APRIL 25-27, 1967

OUR OBJECTIVE IS TO PERMIT THE ECONOMICAL TAPPING OF THE UNLIMITED ENERGY AVAILABLE IN OUR URANIUM RESOURCES AND TO USE MOST EFFICIENTLY THE PLUTONIUM SUPPLIED BY THE LIGHT-WATER-REACTOR COMPLEX. THE DEVELOPMENT OF HIGH-GAIN FAST BREEDERS WILL INCREASE FUEL UTILIZATION FROM THE 1 TO 2% NOW AVAILABLE TO OVER 50% AND WILL PROVIDE AN EXCESS OF BURNABLE PLUTONIUM TO FUEL NEW FAST BREEDERS. THE SODIUM-COOLED LIQUID-METAL FAST-BREEDER REACTOR (LMFBR) HAS THEREFORE BEEN ESTABLISHED AS THE HIGHEST-PRIORITY PROGRAM IN OUR CIVILIAN POWER PROGRAM.

AVAILABILITY - AMERICAN POWER CONFERENCE, E. R. WHITEHEAD, SECRETARY, ILL. INSTITUTE OF TECHNOLOGY, CHICAGO, ILL.

REACTOR, FAST + REACTOR, LARGE + REACTOR, LMCR + PLUTONIUM + URANIUM + REACTOR, POWER + SURVEY

CATEGORY 1
GENERAL SAFETY CONSIDERATIONS

1-021390 DATED 1967
HOSIETLER DK + STEWART JE + HOLLAND CL + WOODS JJ
THE EFFECTS OF SAFETY ON ECONOMICS IN A 1000-MW(e) LIQUID-METAL-COOLED FAST-BREEDER REACTOR (LMFBR)
BABBUX AND WILCOX CO.
1 PAGE, ANS TRANSACTIONS 10(2), PAGE 694, (JUNE 1967), PRESENTED AT THE 1967 WINTER MEETING OF THE AMERICAN NUCLEAR SOCIETY, CHICAGO, ILLINOIS, NOV. 5-9, 1967

IN A LIQUID-METAL-COOLED FAST-BREEDER REACTOR THERE EXISTS A TRADE-OFF BETWEEN CORE SAFETY AND ECONOMICS. RESULTS INDICATE THAT THE PRICE PAID FOR SPOILING (ANY FEATURE OF A CORE THAT IS INTRODUCED SPECIFICALLY TO IMPROVE CORE-SAFETY CHARACTERISTICS AT THE EXPENSE OF BREEDING RATIO, DOUBLING TIME, AND ECONOMICS) IS HIGH, AND THAT NEITHER THE PANCAKE NOR ANNULUS HAS A CLEAR-CUT ADVANTAGE FROM EITHER A FUEL-COST OR SAFETY STANDPOINT. A STRONG INCENTIVE THEREFORE EXISTS TO DEVELOP ENGINEERED SAFEGUARDS THAT WILL PERMIT REDUCED EMPHASIS ON SODIUM VOIDING IN CORE DESIGN.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + DESIGN STUDY + ECONOMIC STUDY + SAFETY PROGRAM

1-021968 ALSO IN CATEGORY 17 DATED 1968
AEC TESTIMONY TO JCAE ON FERMI MELTDOWN CAUSE
ATOMIC ENERGY COMMISSION, US
1 PAGE, ATOMIC ENERGY CLEARING HOUSE 14(7), PAGE 45 (FEBRUARY 12, 1968)

ACCORDING TO MILTON SHAW, THE FLOW-BLOCKING OBJECT WAS A PIECE OF A CONICAL FLOW GUIDE WHICH HAD VIBRATED LOOSE. REP. HOSMER NOTED THAT THE DESIGNER IN 1959 COVERED THE VESSEL BOTTOM WITH ZIRCONIUM SHEET, AND SAID PADMA HAD RECEIVED A MEMO INDICATING IT WAS EASIER TO PUT ZIRCONIUM SHROUDS OVER THE CORE THAN TO JUSTIFY NOT PUTTING IT IN TO THE AGRS. SHAW OBSERVED IT WAS A LAST-MINUTE DECISION HURRIEDLY MADE, NOT SHOWN IN THE FINAL CONSTRUCTION DRAWINGS. HOSMER ASKED IF AGRS HAD PEOPLE SO FRIGHTENED. SHAW REPLIED, NO, AND THIS WAS NOT AN EXCUSE NOT TO DO A THOROUGH ENGINEERING JOB.

CONGRESSIONAL ACTIVITY + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + AGRS + FLOW BLOCKAGE + QUALITY ASSURANCE

1-023108 DATED 1967
REYNOLDS AD + STEWART SL
ANALYSIS OF THE SEFOR MOCKUP CRITICAL EXPERIMENTS IN ZPR-3
GENERAL ELECTRIC CO., SUNNYVALE, CALIF.
GEAP-5294 +. 54 PAGES, TABLES, REFERENCES, MARCH 1967

A DESCRIPTION OF THE RESULTS AND APPLICATIONS OF A SERIES OF CRITICAL EXPERIMENTS PERFORMED ON A MOCKUP OF SEFOR AT ZPR-3. CRITICAL MASS VALUES WERE DETERMINED FOR 1-, 2-, AND 3-SEGMENT FUEL DESIGNS. EFFECTIVENESS OF SEGMENTED FUEL IN REDUCING AXIAL EXPANSION REACTIVITY WAS DEMONSTRATED, AND RESULTS WERE AN IMPORTANT FACTOR IN SELECTION OF THE 2-SEGMENT DESIGN FOR SEFOR. MEASUREMENTS OF THE REACTIVITY WORTH OF THE RADIAL REFLECTOR ESTABLISHED THE ADEQUACY OF THE SEFOR REFLECTOR CONTROL SYSTEM. DOPPLER COEFFICIENTS WAS MEASURED. THE CALCULATED U-238 DOPPLER COEFFICIENT WAS IN AGREEMENT WITH EXPERIMENT. THE MEASURED PU-239 CONTRIBUTION TO THE SEFOR DOPPLER COEFFICIENT WAS NEAR ZERO. MAXIMUM POSITIVE REACTIVITY CAUSED BY LOSS OF SODIUM WAS MEASURED. THE MEASURED REACTIVITY WAS SMALL AND CLOSE TO THE CALCULATED VALUE. OTHER MEASUREMENTS MADE INCLUDED THE RATIO OF 1/BETA-EFFECTIVE, FISSION RATIOS, FISSION AND BORON TRAVERSES, AND PLUTONIUM WORTH DISTRIBUTION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.05 MICROFICHE

REACTOR, BREEDER + CRITICAL ASSEMBLY + DOPPLER COEFFICIENT + REACTIVITY EFFECT, EXPANSION + REACTOR, FAST + REACTOR, LMCR + SODIUM COEFFICIENT + COMPARISON, THEORY AND EXPERIENCE + CRITICALITY EXPERIMENT + MEASUREMENT, REACTIVITY + SEFOR (KE) + ZPR 3 (CAP)

1-025467 DATED 1967
FAST REACTOR SAFETY
2 PAGES, NUCLEAR ENGINEERING, 12(135), PAGES 933-934 (DECEMBER, 1967)

GIVES THE MOST STRIKING COMMENTS OF 11 PERSONS PRESENTED DURING A PANEL DISCUSSION. ONE MEMBER FROM THE USSR STATED THAT THE MAIN ATTENTION SHOULD BE DIRECTED TOWARD SMALL ACCIDENTS, WHERE THE PROBABILITY IS GREATEST. ON THE SUBJECT OF SODIUM BOILING, MANY PROBLEMS ARE PROBABLY FALSE. REAL PROBLEMS EXPERIENCED INCLUDED SODIUM LEAKAGE AND OXIDATION IN THE CORE. OPINIONS WERE EXPRESSED THAT (1) IT HAS BEEN FORTUNATE THAT ENERGY RELEASED IN ACCIDENTS COMPARES WITH WHAT CAN BE CONTAINED BY DESIGN, (2) SAFETY IS A PROBLEM OF INSTRUMENTATION, AND (3) ACCIDENTS ARE NOT FORESEEN BUT ARE FORESEEABLE.

ACCIDENT + DESIGN CRITERIA + REACTOR, FAST + REACTOR, LMCR + UNITED KINGDOM + SAFETY PRINCIPLES AND PHILOSOPHY

1-026406 DATED 1967
KELLY JC
GOOD BREEDING - HOW WILL FUTURE POWER NEEDS BE PROVIDED
WESTINGHOUSE ELECTRIC CORPORATION
6 PAGES, 10 FIGURES, IEEE TRANS. ON INDUSTRY AND GENERAL APPLICATIONS IGA-3(6), PAGES 476-81 (NOV.-DEC. 1967)

CATEGORY 1
GENERAL SAFETY CONSIDERATIONS

1-026406 *CONTINUED*

THE NUCLEAR POWER SYSTEM IN USE IN THE 1980S AND BEYOND CERTAINLY WILL INCLUDE THE HIGH-GAIN FBR. WITH THE SUCCESSFUL APPLICATION OF THE LMFBR, FUEL-CYCLE COSTS OF ALL SYSTEMS WILL BECOME STABILIZED AND PREDICTABLE. THE DEVELOPMENT TIME REQUIRED TO BRING A COMMERCIAL LMFBR INTO BEING IS ESTIMATED AT 12 TO 20 YEARS, DEPENDING ON THE APPROACH AND LEVEL OF RISK ACCEPTED. THE PRESENT LARGE-SCALE AEC PROGRAM APPEARS TO BE BASED UPON THE INTRODUCTION OF THE COMMERCIAL LMFBR IN THE 1980S. THE DEVELOPMENT OF THE LMFBR WILL NOT TRULY BE UNDERWAY UNTIL THERE IS A UTILITY COMMITMENT TO PLANT CONSTRUCTION COMPARABLE TO THAT UNDERTAKEN BY THE UTILITIES ON LWRS IN THE LATE 1950S.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, POWER + FORECAST + REACTOR, LMFBR

1-027827 ALSO IN CATEGORY 11 DATED 1968
MANUAL OF LECTURE NOTES - REACTOR SAFETY COURSE NO. 7
UNITED KINGDOM ATOMIC ENERGY AUTHORITY, RISLEY, WARRINGTON, LANCASHIRE
200 PAGES, FIGURES, TABLES, REFERENCES, SEPTEMBER 23, 1968

NINE SECTIONS. (I) INTRODUCTION, 5 LECTURES. (II) FISSION PRODUCT RELEASE, 5 LECTURES. (III) CONTAINMENT ENGINEERING, 7 LECTURES. (IV) CONTROL AND INSTRUMENTATION, 7 LECTURES. (V) GAS COOLED REACTORS, 11 LECTURES. (VI) WATER REACTORS, 12 LECTURES. (VII) GENERAL, 5 LECTURES. (VIII) FAST REACTORS, 7 LECTURES. (IX) SITING AND EMERGENCY PROCEDURES, 4 LECTURES. NEW LECTURES APPEAR TO BE (7) ASSESSMENT OF ENVIRONMENTAL HAZARDS FROM FISSION PRODUCT RELEASES. (8A) FISSION PRODUCT BEHAVIOR IN LIQUID COOLED REACTORS. (10A) CONTAINMENT OF FISSION PRODUCTS WITHIN FUEL ELEMENTS. (24A) METHODS AND PROBLEMS IN REACTOR ASSESSMENTS. (40) PROBABILITY TECHNIQUES APPLIED TO WATER REACTORS.

AVAILABILITY - AUTHORITY HEALTH AND SAFETY BRANCH, U.K.A.E.A., 11 CHARLES II STREET, LONDON, S. W.1.

REACTOR, GCR + SAFETY REVIEW + UNITED KINGDOM + REACTOR, LWR + SAFETY PRINCIPLES AND PHILOSOPHY + FISSION PRODUCT RETENTION + SAFETY ANALYSIS + REACTOR, HTGR + PROBABILITY + REACTOR, LMFBR + PARTITION COEFFICIENT + FUEL ELEMENTS

1-028163 ALSO IN CATEGORIES 6 AND 9 DATED 1968
GKEEBLER P + HENLEY EJ
ADVANCES IN NUCLEAR SCIENCE AND TECHNOLOGY
GENERAL ELECTRIC COMPANY + UNIVERSITY OF HOUSTON
391 PAGES, VOLUME 4, 1966, ACADEMIC PRESS, NEW YORK AND LONDON

THIS VOLUME IS CONCERNED WITH ADVANCED REACTOR CONCEPTS. TWO OF THE CHAPTERS DEAL WITH THE TECHNOLOGY OF, RESPECTIVELY, SODIUM-COOLED FAST BREEDER POWER REACTORS AND GAS-COOLED POWER REACTORS, THE LATTER INCLUDING BOTH THERMAL CONVERTERS AND FAST BREEDERS. COGNIZANCE IS TAKEN OF THE CENTRAL ROLE OF REACTOR SAFETY IN THE DEVELOPMENT OF FAST BREEDER REACTORS IN TWO CHAPTERS DEALING WITH, RESPECTIVELY, THE DOPPLER EFFECT IN FAST REACTORS AND THE ANALYSIS OF MAXIMUM ACCIDENTS. THREE CHAPTERS COVER RECENT DEVELOPMENTS IN AREAS OF MORE GENERAL APPLICABILITY, NAMELY, OPTIMAL CONTROL THEORY, MODERN PERTURBATION THEORY, AND INDUSTRIAL APPLICATION OF IONIZING RADIATIONS. MAIN HEADINGS FOR THE 7 CHAPTERS ARE - GAS-COOLED REACTOR TECHNOLOGY, SAFETY AND ECONOMIC CHARACTERISTICS OF A 1000-MWE FAST SODIUM-COOLED REACTOR DESIGN, AND THE DOPPLER EFFECT IN FAST REACTORS.

BETHE TAIT + DOPPLER EFFECT + ECONOMICS + REACTOR, FAST + REACTOR, GCR + REACTOR, LMCR + REACTOR DYNAMICS + REACTOR SAFETY SYSTEM + REACTOR CONTROL + THEORETICAL INVESTIGATION + MATHEMATICAL TREATMENT + REACTOR, LMFBR

1-029517 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION M-8 - OFF-SITE EMERGENCY PLANS, DETAILS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, PAGE M-21 THRU -23 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE DETAILS OF ANTICIPATED OFF-SITE EMERGENCY PLANS. (ANSWER IS A VERY BRIEF DISCUSSION OF ON-SITE ACTIONS.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + RADIATION PROTECTION, ORGANIZATION + EMERGENCY PROCEDURE + RADIATION SAFETY AND CONTROL + SEFOR (RE) + RESPONSE TO NRC QUESTION + OFF SITE

1-029722 ALSO IN CATEGORIES 5 AND 18 DATED 1967
BAKER L + EGGEN DT + IVINS RD + SZAWLEWICZ SA
US LMFBR SAFETY PROGRAM
ARGONNE NATIONAL LABORATORY + US ATOMIC ENERGY COMMISSION
25 PAGES, 2 FIGURES, 2 TABLES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPTEMBER 1967

DESCRIBES OBJECTIVES AND STATUS OF PLANNING EFFORT FOR SAFETY ANALYSES AND EVALUATIONS IN THREE TECHNICAL AREAS - (1) ANALYSIS OF ACCIDENTS AND ENERGY RELEASE, (2) PREVENTION AND MITIGATION OF THE CONSEQUENCES OF ACCIDENTS, AND (3) ESTABLISHMENT OF SAFETY-RELATED CRITERIA, CODES AND STANDARDS. BASED ON A TABULATION OF POSSIBLE KNOWN OR POTENTIAL OFF-NORMAL OPERATIONS, INSTABILITIES, AND MALFUNCTIONS, DISCIPLINE-ORIENTED REQUIREMENTS WERE DEVELOPED TO SPECIFY THE

CATEGORY 1
GENERAL SAFETY CONSIDERATIONS

1-029722 *CONTINUED*

RESEARCH AND DEVELOPMENT EFFORT. A BRIEF INDICATION OF THE OBJECTIVES AND CURRENT EFFORTS PROCEEDING UNDER EACH TASK ARE GIVEN.

AVAILABILITY - L. BAKER, JR., ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ACCIDENT ANALYSIS + CODES AND STANDARDS + SAFETY PROGRAM + R AND D PROGRAM + ENGINEERED SAFETY FEATURE + REACTOR, LMFBR

1-029865 ALSO IN CATEGORIES 12 AND 18 DATED 1968
LMFBR PROGRAM PLAN. VOL. 3 - COMPONENTS
ARGONNE NATIONAL LAB., ILL.
WASH-1103 +. 437 PAGES, FIGURES, TABLES, AUGUST 1968

TWELVE TASK AREAS INCLUDE (1) COMPONENT REQUIREMENTS, (2) DEVELOPMENT OF QUALITY AND RELIABILITY ASSURANCE PROGRAMS, (3) COMPONENT MATERIALS REQUIREMENTS, (4) DEVELOPMENT OF FABRICATION AND NDT PROCESSES, (5) FACILITIES FOR COMPONENT DEVELOPMENT. SPECIFIC COMPONENT DEVELOPMENT PROGRAMS ARE (6) STEAM GENERATOR, (7) CORE HANDLING EQUIPMENT (8) COOLANT TRANSPORT AND STORAGE SYSTEMS, (9) REACTOR VESSEL AND RELATED COMPONENTS, (10) REACTIVITY CONTROL DRIVES, (11) LIQUID METAL PUMPS, (12) HEAT EXCHANGERS. ***EACH SECTION BRIEFLY STATES THE GOALS, INTERACTION WITH OTHER PROJECTS, CURRENT STUDIES, AND GIVES A SCHEDULE OR FLOW SHEET FOR THE EFFORT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FABRICATION + PRESSURE VESSELS + RELIABILITY, COMPONENT + TEST, NONDESTRUCTIVE + FUEL HANDLING MACHINE + PROPERTY, PHYSICAL + QUALITY ASSURANCE + MAIN COOLING SYSTEM + R AND D PROGRAM + STEAM GENERATOR + REACTOR, LMFBR + PUMPS + PIPES AND PIPE FITTINGS + JACOBS + HEAT EXCHANGERS + CONTROL ROD DRIVE MECHANISMS

1-029994 ALSO IN CATEGORIES 7 AND 9 DATED 1968
LMFBR PROGRAM PLAN. VOL. 10 - SAFETY
ARGONNE NATIONAL LABORATORY, LMFBR PROGRAM, OFFICE, ARGONNE, ILLINOIS
WASH-1110 +. 256 PAGES, FIGURES, TABLES, AUGUST 1968

THE SAFETY ELEMENT OF THE LMFBR PROGRAM IS TECHNOLOGICALLY ORIENTED TO AID DESIGNERS, OPERATORS, AND REGULATORY GROUPS IN ACHIEVING THE REQUISITE LEVEL OF SAFETY FOR LMFBRs. THE OBJECTIVE IS TO PROVIDE REALISM AND CONFIDENCE IN THE UNDERSTANDING AND ANALYSIS OF ACCIDENT SITUATIONS, TO DEVELOP AND EVALUATE SAFETY SYSTEMS FOR THE PREVENTION OF ACCIDENTS AND MITIGATION OF THEIR CONSEQUENCES, AND TO DEVELOP STANDARDS AND CODES FOR THE SAFE DESIGN, SITING, CONSTRUCTION, AND OPERATION OF BOTH USAEC TEST FACILITIES AND COMMERCIAL LMFBRs. THE SCOPE INCLUDES EVALUATIONS AND EXPERIMENTATION RELATIVE TO ABNORMAL OPERATION, POTENTIAL ACCIDENTS, AND SAFETY SYSTEMS DESIGNED TO PREVENT ACCIDENTS AND TO LIMIT THEIR CONSEQUENCES. EMPHASIS IS ON PREVENTION, EARLY DETECTION, AND CONTROL OF ACCIDENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

INSTRUMENT, PROTECTIVE + REACTOR SAFETY SYSTEM + INSTRUMENT, SURVEILLANCE + SAFETY PRINCIPLES AND PHILOSOPHY + R AND D PROGRAM + REACTOR, LMFBR + JACOBS + INSTRUMENTS, MISC.

1-030719 DATED 1968
COHEN KP + ONEILL GL
SAFETY AND ECONOMIC CHARACTERISTICS OF A 1000-MWE FAST SODIUM-COOLED REACTOR DESIGN
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIFORNIA
ADVANC. NUCL. SCI. TECHNOL., 4, 67-107, 1968

THE SAFETY CRITERIA AND OPERATING LIMITS FOR FUEL, CLADDING, AND COOLANT ARE EXAMINED FOR LARGE FAST NUCLEAR REACTORS BY DESCRIBING A DESIGN APPROACH IN THE LIGHT OF PRESENT TECHNOLOGY. A PRELIMINARY REFERENCE REACTOR CORE, COOLANT SYSTEM, AND CONTAINMENT DESIGN ARE PRESENTED, AND HAZARDS ANALYSES ARE PERFORMED TO PREDICT THE DYNAMIC RESPONSE OF THE REACTOR AND PLANT TO CREDIBLE PLANT ACCIDENTS. THE HAZARDS ANALYSES ARE EXTENDED TO INCLUDE HYPOTHETICAL ACCIDENTS, AND A CONTAINMENT CONCEPT IS DESIGNED AND ANALYZED TO PROVE ITS ADEQUACY FOR THE HYPOTHETICAL ACCIDENT.

ECONOMICS + REACTOR, FAST + DESIGN STUDY + REACTOR, LMFBR

1-030761 ALSO IN CATEGORY 18 DATED 1967
DIECKAMP HM + BEELEY RJ + STOKER DJ + ASH EB
ATOMICS INTERNATIONAL DEMONSTRATION FAST BREEDER REACTOR
ATOMIC INTERNATIONAL, CANOGA PARK, CALIF.
CONF-670413-(AND SUPPL) +. 22 PAGES, PAGES 3.1-22 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO, CALIF., APRIL 10-12, 1967. HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY, 1967

DESCRIBES A 500-MWE NA-COOLED FAST-BREEDER CONCEPTUAL DESIGN TO ESTABLISH THE BASIS FOR EARLY DEMONSTRATION PLANT OPERATION. THE CONCEPTUAL DESIGN BEING OBTAINED BY SYSTEMATICALLY EXAMINING ALTERNATE SYSTEM AND SUBSYSTEM CHOICES IS GIVEN. COST-EFFECTIVENESS TECHNIQUES (CAPITAL, FUEL AND DEVELOPMENT COSTS, PRIOR EXPERIENCE, MAINTAINABILITY AND RELIABILITY) ARE EVALUATED. MAJOR SYSTEM EVALUATIONS HAVE BEEN OR ARE BEING CONDUCTED RELATIVE TO BASIC PLANT OPERATING PARAMETERS, CORE GEOMETRY, PRIMARY SYSTEM CONFIGURATION, FUEL HANDLING AND STEAM GENERATORS, AS WELL AS OVERALL PLANT DESIGN AND LAYOUT.

CATEGORY 1
GENERAL SAFETY CONSIDERATIONS

1-030761 *CONTINUED*
ECONOMICS + AI + DESIGN STUDY + REACTOR, LMFBR

1-030762 ALSO IN CATEGORY 18 DATED 1967
GAST K + SCHLECHTENDAHLE + SMIDT D + GUTHMANN E + HELD C + KOEHLER M + MORGENSTERN F
NA-2 DESIGN - 300 MWE GERMAN FCR
KERNFORSCHUNGSZENTRUM KARLSRUHE, GERMANY
CONF-670413-(AND SUPPL) +. 15 PAGES, PAGES 3,23-38 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO,
CALIF., APRIL 10-12, 1967, HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY

PRESENTS PRELIMINARY RESULTS OF THE CURRENT CONCEPTUAL DESIGN STUDY FOR A 300-MWE NA-COOLED FAST
BREEDER. DUE CONSERVATISM IS APPLIED IN THE DESIGN AND IN THE SAFETY ASSESSMENT TO ACCOUNT FOR
THE FIRST-OF-A-KIND-CHARACTERISTIC OF THE REACTOR. IT INCLUDES MOST OF THE BASIC FEATURES
PREVIOUSLY ESTABLISHED FOR LARGE COMMERCIAL BREEDER REACTORS. THOSE DESIGN FEATURES WHICH ARE OF
VITAL IMPORTANCE TO THE SAFETY AND OPERATING RELIABILITY OF THE REACTOR ARE DISCUSSED. EMPHASIS
IS LAID UPON THE CRITICAL DESIGN ASPECTS OF THE CORE AND OF THE PRIMARY COOLANT SYSTEM, AND
CONSEQUENCE-LIMITING SAFEGUARDS ARE DESCRIBED.

DESIGN STUDY + GERMANY + SAFETY ANALYSIS + REACTOR, LMFBR

1-032907 ALSO IN CATEGORY 18 DATED 1968
AEC QUESTIONS - 7 TOPICS
GENERAL ELECTRIC COMPANY
50231-21 +. 17 PAGES, 5 FIGURES, SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

RESPONSES TO 7 QUESTIONS CONCERNING EMERGENCY TREATMENT OF INJURED PERSONNEL BOTH ON AND OFF SITE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

NRC QUESTION + RADIATION INJURY TREATMENT + RADIATION PROTECTION, ORGANIZATION + EMERGENCY PROCEDURE +
PERSONNEL EXPOSURE, RADIATION + SEFOR (RE) + RADIOLOGICAL ASSISTANCE + REPORT, PSAR + OFF SITE + REACTOR,
LMFBR

1-032908 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 1 - INJURED PERSONNEL - ON SITE TREATMENT FACILITIES
GENERAL ELECTRIC CO., SAN JOSE
50231-21 +. 3 PAGES, PG. 1 THRU 3 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT., 1968, DOCKET 50-231

DESCRIBE ON-SITE FACILITIES FOR DECONTAMINATION AND IMMEDIATE EMERGENCY TREATMENT OF INJURED
PERSONNEL. *** ANSWER ON PAGES 1 THRU 3.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + ON
SITE + REACTOR, LMFBR

1-032909 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 2 - QUALIFICATION OF PHYSICIANS AND HEALTH PHYSICISTS
GENERAL ELECTRIC
50231-21 +. 2 PAGES, PAGES 4 AND 5 SEFOR, PSAR, SUPPLEMENT 20, SEPT., 1968, DOCKET 50-231

DESCRIBE QUALIFICATIONS, EDUCATION, AND TRAINING OF PERSONNEL RESPONSIBLE FOR SUPERVISION AND CARE
OF INJURED RADIOACTIVELY CONTAMINATED PERSONS AND OTHER OCCUPATIONAL HEALTH HAZARDS. *** ANSWER
ON PAGES 4 AND 5.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + CERTIFICATION + STAFFING

1-032910 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 3 - TRANSPORT OF INJURED PERSONNEL
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-21 +. 1 PAGE, PG. 6 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

DESCRIBE ARRANGEMENTS FOR TRANSPORT OF INJURED PERSONNEL. *** ANSWER ON PAGE 6. TWO VEHICLES
AVAILABLE. EMERGENCY PROCEDURE NO. 006 APPLIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR +
REACTOR, LMFBR

CATEGORY 1
GENERAL SAFETY CONSIDERATIONS

1-032911 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 4 - HOSPITALS - ACCEPTANCE OF CONTAMINATED PERSONNEL
 GENERAL ELECTRIC
 50231-21 +. 2 PAGES, PAGES 7 AND 8 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

DESCRIBE LOCATION OF HOSPITAL AGREEABLE TO ACCEPTING PATIENTS FOR FURTHER DECONTAMINATION, AND CONTAMINATED INJURED PERSONNEL. *** ANSWER ON PAGES 7 AND 8.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + OFF SITE + REACTOR, LMFBR

1-032912 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 5 - MEDICAL PERSONNEL QUALIFICATIONS
 GENERAL ELECTRIC
 50231-21 +. 1 PAGE, PG. 9 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

DESCRIBE QUALIFICATIONS OF PROFESSIONAL MEDICAL PERSONNEL AT THE SUPPORT FACILITY (HOSPITAL OR CLINIC) TO TREAT RADIATION AND CONTAMINATION VICTIMS, INCLUDING NUMBER AND TYPES OF PHYSICIANS AND A DESCRIPTION OF ANY SPECIALIZED TRAINING RELATED TO CONTAMINATION OR RADIATION INJURIES.
 *** ANSWER ON PAGE 9.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + TRAINING + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + OFF SITE + REACTOR, LMFBR + CERTIFICATION + STAFFING

1-032913 ALSO IN CATEGORY 18 DATED 1968
RESPONSES TO QUESTION 6 - OFF SITE MEDICAL FACILITIES
 GENERAL ELECTRIC
 50231-21 +. 1 PAGE, PG. 10 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

DESCRIBE LIMITATIONS THAT EXIST REGARDING AVAILABILITY OF OFF-SITE MEDICAL FACILITIES AND SUPPORT.
 *** (ANSWER) NONE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + OFF SITE + REACTOR, LMFBR

1-032914 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 7 - STANDING ORDERS AT OFF SITE HOSPITAL
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 50231-21 +. 1 PAGE, PG. 11 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

DESCRIBE WRITTEN PLAN AND STANDING ORDERS IN RECEIVING AREA OF HOSPITAL DETAILING ACTIONS TO BE TAKEN AND PROCEDURE TO BE FOLLOWED WHEN CONTAMINATED PERSON WITH OR WITHOUT INJURY IS BROUGHT TO HOSPITAL. *** (ANSWER) PLANS AND ORDERS ARE BEING DEVELOPED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + OFF SITE + REACTOR, LMFBR

1-033051 ALSO IN CATEGORIES 9 AND 12 DATED 1968
LIQUID METAL ENGINEERING CENTER - SEMIANNUAL TECHNICAL PROGRESS REPORT, JULY-DECEMBER 1968
 LIQUID METAL ENGINEERING CENTER
 LMEC-69-1 +. 281 PAGES, 87 FIGURES, 17 TABLES, REFERENCES, DECEMBER 1968

SUBJECTS COVERED - VALVES AND SODIUM TECHNOLOGY, PUMP STUDIES, PIPING STUDIES, LMFBR TEST AND FACILITY PLANNING, CODES AND STANDARDS FOR LMFBR AND RELATED ACTIVITIES (REACTOR VESSEL, CIRCULATING COLD TRAP, MECHANICAL FREE-SURFACE SODIUM PUMPS, ELECTRO-MAGNETIC PUMPS, SODIUM PIPING, VALVES, AND QUALITY), MAINTENANCE AND MALFUNCTION ANALYSIS, COMBINED UNIT REACTOR VESSEL STUDY, QUALITY ASSURANCE, OPERATION OF SODIUM COMPONENTS TEST INSTALLATION, GENERAL OPERATING ACTIVITIES, LIQUID METAL INFORMATION CENTER, SODIUM PUMP TEST FACILITY, SODIUM PUMP SEAL TEST PROGRAM, BEAMING TEST PROGRAM, STEAM GENERATOR AND INTERMEDIATE HEAT EXCHANGER PERFORMANCE EVALUATION, INSTRUMENTATION TESTING, EVALUATION, CALIBRATION, AND STANDARDS, OPERATION AND MAINTENANCE OF THE LCFL, CETS, AND HMPF PUMP TEST LOOP, MODIFICATIONS TO THE LCFL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
 COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + CODES AND STANDARDS + PRESSURE VESSELS + SODIUM + TESTING + STRESS ANALYSIS + EQUIPMENT + STEAM

CATEGORY 1
GENERAL SAFETY CONSIDERATIONS

1-033051 *CONTINUED*
GENERATOR + PUMP, MAGNETIC + REACTOR, LMFBR + PUMPS + VALVES + PIPES AND PIPE FITTINGS + INSTRUMENTS, MISC.

1-033429 ALSO IN CATEGORIES 17 AND 18 DATED 1968
LMFBR PROGRAM PLAN. VOLUME I - OVERALL PLAN
ARGONNE NATIONAL LAB.
WASH-1101 +. 350 PAGES, FIGURES, TABLES, AUGUST 1968

CONTENTS - THE OBJECTIVES, SCOPE, BACKGROUND INFORMATION, REQUIREMENTS AND GOALS, PLAN ORGANIZATION, NEEDS, PLAN OF ACTION, SCHEDULING AND PRIORITIES, AND FACILITIES. APPENDICES CONTAIN COLLECTION OF TASKS AND CONTRACT WORK, FACILITIES TABLE, GLOSSARY, AND PRIVATELY SUPPORTED WORK. NINE OTHER ELEMENTS OF THE PLAN, WHICH ARE PRESENTED IN SEPARATE VOLUMES, ARE PLANT DESIGN, COMPONENTS, INSTRUMENTATION AND CONTROL, SODIUM TECHNOLOGY, CORE DESIGN, FUELS AND MATERIALS, FUEL RECYCLE, PHYSICS, AND SAFETY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, BREEDER + REACTOR, FAST + SODIUM + REACTOR COOLANT + DESIGN STUDY + REVIEW + R AND D PROGRAM + REACTOR, LMFBR + JACOBS

1-033932 ALSO IN CATEGORY 18 DATED 1968
MEUSER MH + PERRY JA
FABRICATION, CONSTRUCTION, AND INSTALLATION QUALITY ASSURANCE REQUIREMENTS FOR THE FFTF
BATTELLE-NORTHWEST, RICHLAND, WASH.
BNWL-511 +. 56 PAGES, MARCH 15, 1968

STATES GENERAL REQUIREMENTS FOR QUALITY ASSURANCE FOR PACIFIC NORTHWEST LABORATORY (PNL) CONTRACTORS INVOLVED IN PROCUREMENT, FABRICATION, CONSTRUCTION, INSTALLATION, OR TESTING EFFORT FOR THE FFTF PROJECT. THE SECTION 1.0, INTRODUCTION, THROUGH SECTION 14.0, PROGRAM DOCUMENTATION, ARE AEC/PNL BASELINES. A GLOSSARY DEFINING TERMS IS INCLUDED AS APPENDIX A.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + CONTAINMENT CONSTRUCTION + FFTF (TR) + QUALITY ASSURANCE + R AND D PROGRAM + REACTOR, LMFBR

1-035995 DATED 1969
KEYFITZ IM
MOISTURE SEPARATION OR STEAM REHEAT VS SODIUM REHEAT PLANT CYCLE TECHNICAL AND ECONOMIC EVALUATION. 1000 MWE
LMFBR FOLLOW-ON STUDY
WESTINGHOUSE ELECTRIC CORP., MADISON, PA. ADVANCED REACTORS DIV.
WARD-2000-20 +. 58 PAGES, 25 FIGURES, 12 TABLES, 5 REFERENCES, JAN. 1969

AN EVALUATION OF A MOISTURE SEPARATION OR STEAM REHEAT CYCLE VERSUS A SODIUM REHEAT CYCLE FOR A NOMINAL 1000-MWE, SODIUM-COOLED, FAST-BREEDER REACTOR VERIFIES THAT, FROM BOTH A TECHNICAL AND ECONOMIC VIEWPOINT, MOISTURE SEPARATION OR STEAM REHEAT IS SUPERIOR TO SODIUM REHEAT. THE ECONOMIC ADVANTAGE IS ESTIMATED AT FROM 0.014 TO 0.06 MILLS/KWH POWER COST, DEPENDING UPON THE ASSUMED TOTAL CAPITAL PRICE IN THE RANGE OF 120 TO 160 \$/KW AND THE ASSUMED FUEL COST IN THE RANGE OF 7 TO 16 CENTS PER MILLION BTU. EXPENSES INCURRED BY THE CUSTOMER ARE NOT INCLUDED IN THE TOTAL CAPITAL PRICE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + SODIUM + ANL + ECONOMIC STUDY + REACTOR, LMFBR

1-044436 ALSO IN CATEGORY 3 DATED 1969
SEABORG, GT
LOOKING AHEAD IN NUCLEAR (GLENN SEABORG)
U.S. ATOMIC ENERGY COMMISSION
5 PAGES, MECHANICAL ENGINEERING, PP. 30-34 (AUGUST 1969)

REPORT OF AN ADDRESS AT THE 37TH CONVENTION OF THE EDISON ELECTRIC INSTITUTE WHEREIN SEABORG DISCUSSED THE BROAD ASPECTS OF THE PROSPECTS FOR N-POWER. HE REPORTS ON THE NUMBER OF PRESENT AND PROJECTED POWER REACTORS, THEIR POWER OUTPUT, AND THE PROGRESS IN AND NEED FOR BREEDER REACTORS. REGARDING THE LATTER, HE MENTIONS AEC REPORT -- COST-BENEFIT ANALYSIS OF U.S. BREEDER PROGRAM -- A STUDY BASED ON OPTIMIZATION OF U.S. ELECTRIC ENERGY ECONOMY FROM 1970 TO 2020. THE QUALITY ASSURANCE INFLUENCE ON THE AEC, THE NUCLEAR INDUSTRY, AND THE UTILITIES INTERACTION IS DISCUSSED. NUCLEAR PHOBIA, THERMAL POLLUTION, AND RADIOACTIVE WASTE MANAGEMENT ARE DISCUSSED, AND SEABORG IMPLIES THAT THESE ARE (PROBLEM) AREAS WHERE CONCERN IS REALLY NOT WARRANTED, BEING SOLVED OR SOLVABLE.

*N-POWER FORECAST + PROPONENT + *SPOKESMAN, AEC + *SOCIO/PHILOSOPHICAL CONSIDERATION + REACTOR POWER + REACTOR, BREEDER + REACTOR, LMFBR + REACTOR, HTGR + N-POWER, SAFETY OF + DATA COLLECTION + ECONOMICS + ENERGY PREDICTION + FUEL, NUCLEAR + FUEL, FOSSIL + QUALITY ASSURANCE + WASTE MANAGEMENT + PUBLIC RELATIONS + THERMAL POLLUTION + ATMOSPHERIC POLLUTION + WASTE HEAT MANAGEMENT + BENEFICIAL USE, AGRICULTURE + BENEFICIAL USE, COMMERCIAL + BENEFICIAL USE, RECREATIONAL + ESTHETICS + AGENCY, PROMOTIONAL + INDUSTRY, UTILITY + ENVIRONMENT + FISSION PRODUCT RELEASE

CATEGORY 1
GENERAL SAFETY CONSIDERATIONS

1-049354

DATED 1970

CIVILIAN NUCLEAR POWER

1 PAGE, ATOMIC ENERGY CLEARING HOUSE, NO. 25, PG 10 (JUNE 22, 1970)

IT IS GENERALLY ACCEPTED BY HOUSE APPROPRIATIONS COMMITTEE THAT N-POWER PLANTS SHOULD PLAY AN IMPORTANT AND INCREASING ROLE IN CRITICAL POWER SUPPLY SITUATION FACING THE NATION, THAT SUCH PLANTS HAVE BEST POTENTIAL OF ALL ALTERNATIVES. THE N-PLANTS HAVE BEEN PROJECTED AS PROVIDING 25 PERCENT OF TOTAL ELECTRIC POWER BY 1980 AND OVER 50 PERCENT BY 2000. HIGHEST PRIORITY IS BEING PLACED BY AEC ON DEVELOPMENT OF LMFBR. DEVELOPMENT WILL REQUIRE SIZEABLE INVESTMENT, BUT BENEFITS WILL RETURN INVESTMENT MANY TIMES OVER -- SAVINGS TO COUNTRY BY 1980 ESTIMATED AT \$3 BILLION. MUCH OF WORK WILL GO FOR NAUGHT UNLESS GROWING PUBLIC CONCERN IS MET HEAD ON. THE COMMITTEE NOTES SAFETY RECORD OF AEC IS 2ND TO NONE, BUT IT EXPRESSED CONCERN THAT AEC PURSUE STILL UNKNOWN HAZARDS ANALYSES, VIZ., ELIMINATING OR MINIMIZING EFFECT OF NUCLIDES AND THERMAL POLLUTION.

REACTOR, BREEDER + CONGRESSIONAL ACTIVITY + FISSION PRODUCT RELEASE + REACTOR, POWER + THERMAL POLLUTION +
REACTOR, LMFBR + DATA COLLECTION + N-POWER, SAFETY OF + PROPOONENT + PUBLIC RELATIONS + N-POWER FORECAST

CATEGORY 2
SITING OF NUCLEAR FACILITIES

2-002801 ALSO IN CATEGORY 11 DATED 1965
COTTRELL WB + SAVOLAINEN AW
U. S. REACTOR CONTAINMENT EXPERIENCE. A HANDBOOK OF CURRENT PRACTICE AND ANALYSIS, DESIGN, CONSTRUCTION, TEST, AND OPERATION. CHAPTER 1. - INTRODUCTION
OAK RIDGE NATIONAL LABORATORY
57 PAGES, 12 TABLES, 16 FIGURES, 41 REFERENCES, AUGUST 1965, ORNL-NSIC-5, VOL. I, CFSTI

THIS CHAPTER IS THE INTRODUCTION TO A COMPREHENSIVE REVIEW OF U.S. REACTOR CONTAINMENT EXPERIENCE. 1-1 PHILOSOPHY - 1-1.1 - DEFINITIONS - 1-1.2 - NEED FOR CONTAINMENT - 1-1.3 - SAFETY FACTORS, 1-2 RAMIFICATIONS OF REACTOR CONTAINMENT - 1-2.1 - REACTOR SITE CRITERIA - 1-2.2 - NUCLEAR MERCHANT SHIP SITE CRITERIA - 1-2.3 - RELIABILITY OF REACTOR SYSTEMS - 1-2.4 - NUCLEAR SAFETY ECONOMICS - 1-2.5 - LIABILITY AND INDEMNITY, 1-3 GENERAL DESCRIPTION OF REACTOR TYPES, 1-4 - GENERAL DESCRIPTION OF CONTAINMENT TYPES - 1-4.1 - PRESSURE CONTAINMENT - 1-4.2 - PRESSURE-SUPPRESSION CONTAINMENT - 1-4.3 - PRESSURE-RELEASE CONTAINMENT - 1-4.4 - MULTIPLE CONTAINMENT - 1-4.5 - STATE OF THE ART, 1-5 GUIDE TO THE USE OF HANDBOOK - 1-5.1 PURPOSE - 1-5.2 LIMITATIONS AND USE.

REACTOR, BWR + CODES AND STANDARDS + CONTAINMENT DESIGN + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + CONTAINMENT, MULTIPLE + CONTAINMENT, PRESSURE RELIEF + CONTAINMENT, PRESSURE SUPPRESSION + CONTAINMENT, PRESSURE VENTING + DOSE + ECONOMICS + REACTOR, FAST + REACTOR, GCR + REACTOR, HWR + INSURANCE + REACTOR, LMCR + REACTOR, MARITIME + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR, MOLTEN SALT + CONTAINMENT SYSTEM, OPERATION + REACTOR, ORGANIC COOLED + REACTOR, PWR + RELIABILITY ANALYSIS + REACTOR, RESEARCH + REACTOR SAFETY SYSTEM + SITING + SITING, REACTOR + REACTOR, TRAINING + ACCIDENT, PROBABILITY OF + CONTAINMENT STRUCTURE + CONTAINMENT LEAKAGE CONTROL + ENGINEERED SAFETY FEATURE

2-002811 ALSO IN CATEGORY 11 DATED 1965
BERNSEN SA + GRIFFIN RF
U. S. REACTOR CONTAINMENT EXPERIENCE- A HANDBOOK OF CURRENT PRACTICE AND ANALYSIS, DESIGN, CONSTRUCTION, TEST, AND OPERATION. CHAPTER 11.-ECONOMICS
BECHTEL CORPORATION
57 PAGES, 20 TABLES, 18 FIGURES, 30 REFERENCES- AUGUST 1965, ORNL-NSIC-5, VOL. II- CFSTI

THIS CHAPTER DISCUSSES THE ECONOMIC FACTORS IN DESIGNING AND EVALUATING CONTAINMENT SYSTEMS AND VARIOUS CONTAINMENT FEATURES. ITS PURPOSE IS TO PROMOTE AN UNDERSTANDING OF THE ASPECTS OF CONTAINMENT WHICH ARE OF GREATER ECONOMIC SIGNIFICANCE AND TO PROVIDE GUIDANCE IN MAKING ECONOMIC COMPARISONS OF ALTERNATE DESIGNS. CONTAINS THE FOLLOWING - (1) INTRODUCTION, (2) ECONOMIC IMPACT OF CONTAINMENT, (3) UNIT COSTS OF CONTAINMENT COMPONENTS, (4) COST EXPERIENCE ON CONSTRUCTED FACILITIES, (5) TYPICAL COST EVALUATIONS, (6) COST ESTIMATING, (7) SUMMARY, AND (8) REFERENCES.

BIG ROCK POINT (BWR) + BUDEGA BAY (BWR) + REACTOR, BWR + BONUS (ISR) + REACTOR, BREEDER + CODES AND STANDARDS + CONCRETE + CONTAINMENT, UNDERGROUND + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + CONTAINMENT, MULTIPLE + CONTAINMENT, PRESSURE RELIEF + CONTAINMENT, PRESSURE SUPPRESSION + CONTAINMENT, PRESSURE VENTING + DRESDEN 1 (BWR) + EBWR (BWR) + ECONOMICS + EGCR (AGR) + ELK RIVER (BWR) + REACTOR, FAST + REACTOR, GCR + REACTOR, HWR + HUMBOLDT BAY (BWR) + HWCTR (TR) + INDIAN POINT 1 (PWR) + REACTOR, LMCR + REACTOR, ORGANIC COOLED + PATHFINDER (ISR) + PEACH BOTTOM 1 (HTGR) + PIQUA (OCR) + REACTOR, PWR + SITING + SITING, REACTOR + REACTOR, INTERNAL SUPERHEAT + VBWR (BWR) + VENTILATION SYSTEM + YANKEE ROWE (PWR) + CONTAINMENT AIR LOCK + CONNECTICUT YANKEE (PWR) + CONTAINMENT CONSTRUCTION + SAN ONOFRE 1 (PWR) + STACK + CONTAINMENT FILTERING SYSTEM + CONTAINMENT STRUCTURE + RAVENSWOOD (PWR) + ENGINEERED SAFETY FEATURE + FILTERS

2-022328 ALSO IN CATEGORY 11 DATED 1967
MCCULLOUGH CR + BROWN GM + HANCOCK RP + LYERLY RL + MITCHELL W + TURNER SE
SITING STUDIES OF TWO LARGE DUAL-PURPOSE DESALINATION REACTORS
SOUTHERN NUCLEAR ENGINEERING, INC., DUNEDIN, FLORIDA
SNE-33 +. 87 PAGES, 18 FIGURES, 16 REFERENCES, SEPTEMBER 1, 1967

PRESENTS A PRELIMINARY STUDY ON THE SITING OF A 3-GW(T) AND A 10-GW(T) REACTOR FOR DESALINATION AND POWER PRODUCTION. TWO REACTOR CONCEPTS WERE TREATED, THE HTGR AND THE LMFR. CONTAINMENT AND ENGINEERED SAFETY FEATURES ARE DISCUSSED IN CONJUNCTION WITH THE DESIGN-BASIS ACCIDENT (DBA) AND MAXIMUM HYPOTHETICAL ACCIDENT (MHA). METHODS OF DETERMINING DOWNWIND DOSES IN THE EVENT OF ACCIDENTS ARE PRESENTED.

ATMOSPHERIC DIFFUSION + REACTOR, GCR + REACTOR, LMCR + SITING, REACTOR + REACTOR, DESALINATION + DISTANCE + DOSE CALCULATION, INTERNAL + DOSE CALCULATION, EXTERNAL + JACOBS

2-029433 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION A-1. DEFINITIONS AND FUNCTION OF CLASS I AND II ITEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIFORNIA
11 PAGES, 3 FIGURES, 2 TABLES, PAGE A-1 THRU A-11 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS THE EXTENT TO WHICH THE DEFINITIONS AND FUNCTIONS OF CLASS I AND CLASS II ITEMS HAVE BEEN MODIFIED IN THE COURSE OF FINAL DESIGN AND CONSTRUCTION. (ANSWER ON PGS A-1 THRU -11.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING

CATEGORY 2
SITING OF NUCLEAR FACILITIES

2-029434 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION A-2 - EARTHQUAKE EXPOSURE
 GENERAL ELECTRIC CO., SAN JOSE, CALIFORNIA
 4 PAGES, 1 TABLE, PAGE A-12 THRU A-15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

INDICATE THE EARTHQUAKE EXPOSURE, DESCRIBED BY RESPONSE SPECTRA, USED IN THE DESIGN FOR BOTH THE DESIGN AND MAXIMUM EARTHQUAKES. (ANSWER ON PGS A-12 THRU -15.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + RESPONSE SPECTRUM

2-029435 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION A-3 - EARTHQUAKE DESIGN CRITERIA
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 1 PAGE, PAGE A-15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE THE CRITERIA USED IN DESIGN OF THE CLASS I AND CLASS II ITEMS FOR BOTH THE DESIGN AND MAXIMUM EARTHQUAKES, INCLUDING CRITERIA FOR COMBINED LOADING CONDITIONS FOR ACCEPTANCE LIMITS IN TERMS OF STRESS, DEFORMATION, OR OTHER SELECTED CONDITION FOR THESE LOADING COMBINATIONS, AND FOR THE DAMPING COEFFICIENTS AND THE RESPONSE SPECTRA PLOTS OR ACCELERATION COEFFICIENTS TO BE USED FOR THE DYNAMIC COMPONENTS OF LOADING. (ANSWER) SEE APPENDIX A OF PSAR, SUPPLEMENT 6 OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + RESPONSE SPECTRUM

2-029436 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION A-4 - SEISMIC ANALYSIS OF CLASS I ITEMS
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 66 PAGES, 21 FIGURES, 2 TABLES, PAGE A-16 THRU A-21 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE IN GENERAL THE METHODS USED FOR SEISMIC ANALYSIS OF CLASS I ITEMS. PROVIDE MINIMUM INFORMATION FOR THE CONTAINMENT AND A FEW OF THE MOST ESSENTIAL CLASS I SYSTEMS. (A) ANALYTICAL MODEL USED, E.G., LOCATION OF LUMPED MASSES, VALUES OF PARAMETERS USED, AND SUPPORT CONDITIONS. (ANSWER ON PAGES A-15 THRU -21 PLUS 60 PAGES OF TABLES AND FIGURES.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + REACTOR, LMCR + REPORT, SAR + ANALYTICAL TECHNIQUE + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING

2-029437 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION A-5 - ACCOMMODATION OF EARTHQUAKE-INDUCED EXCITATION BY PUMP HANGERS
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 1 PAGE, PAGE A-28 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE HOW VERTICAL FORCES AND MOTIONS, DUE TO EARTHQUAKE INDUCED EXCITATION, ARE ACCOMMODATED BY THE CONSTANT TENSION HANGERS SUPPORTING THE E-M PUMPS. (ANSWER) SPRING CONSTANT CLOSE TO ZERO AND ACTS AS UPWARD LOADS ON PUMPS, COUNTERBALANCING GRAVITY. PERMIT BUILDING MOTION WITHOUT TRANSMITTING TO PUMP.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + SUPPORT STRUCTURE + DAMPING + EARTHQUAKE ENGINEERING + PUMPS

2-029438 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION A-7 - SEISMIC ANALYSIS
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 2 PAGES, PAGE A-29 AND A-30 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EXPAND TABLES 1 AND 2 OF SUPPLEMENT 6 BY CORRECTING THE BUILDING ACCELERATION, AND BY ADDING A COLUMN EXPLAINING, IN A SHORT NOTE FOR EACH LISTED ITEM, THE PRINCIPAL CAUSE OF THE SEISMIC AMPLIFICATION. INDICATE IN A SEPARATE COLUMN THE PERCENTAGE OF SEISMIC STRESSES WITH RESPECT TO THE TOTAL SUM OF COMBINED STRESSES. FOR EACH LISTED CLASS I ITEM, INDICATE THE NATURAL FREQUENCY

CATEGORY 2
SITING OF NUCLEAR FACILITIES

2-029438 *CONTINUED*

OR THE PERIOD OF THE ITEM, INCLUDING ITS IMMEDIATE SUPPORTS. FOR PIPING LISTED IN TABLE 1, INDICATE WHETHER THE SYSTEM FOR WHICH THE PERIOD IS GIVEN, INCLUDES THE SUPPORT. (ANSWER ON PGS A-29 AND -30.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + SUPPORT STRUCTURE + ACCELERATION + EARTHQUAKE ENGINEERING + INTERACTION, FOUNDATION AND STRUCTURE + PIPES AND PIPE FITTINGS

2-029439 ALSO IN CATEGORIES 10 AND 18 DATED 1968

RESPONSE TO QUESTION A-8 - BATTERIES AND RACKS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE A-30 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ARE THE BATTERIES AND BATTERY RACKS LOCATED UNDER THE CONTROL ROOM CLASS I ITEMS. DISCUSS THE CRITERIA AND METHODS OF ANALYSIS USED IN THE DESIGN OF THE BATTERIES, AND THE RACKS HOLDING THE BATTERIES, TO ENSURE THAT THEY WILL FUNCTION PROPERLY UNDER THE MAXIMUM EARTHQUAKE LOADING. (ANSWER) BATTERIES AND RACKS ARE CLASS I ITEMS, CERTIFIED EARTHQUAKE RESISTANT BY MANUFACTURER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + EMERGENCY POWER, ELECTRIC + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + BATTERIES & CHARGERS

2-029440 ALSO IN CATEGORIES 9 AND 18 DATED 1968

RESPONSE TO QUESTION A-9 - SEISMIC EFFECTS ON CRITICAL INSTRUMENTATION

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, PAGE A-30 THRU A-32 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CERTAIN CONTROLS AND INSTRUMENTATION IN THE CONTROL ROOM AND OTHER PARTS OF THE FACILITY ARE CRITICAL TO SAFE SHUTDOWN OF THE FACILITY. PROVIDE A DISCUSSION OF THE CRITERIA AND METHODS OF ANALYSIS EMPLOYED IN THE DESIGN OF THESE CRITICAL CONTROLS AND INSTRUMENTATION FOR SEISMIC EFFECTS INCLUDING SEISMIC LOADINGS, POSSIBLE TILT (I.E., MERCURY SWITCHES AND RELAYS), ETC. (ANSWER ON PGS A-30 THRU -32.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + INSTRUMENT, CONTROL + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + CONTROL PANEL/ROOM

2-029441 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION A-10 - PIPING

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

13 PAGES, 8 FIGURES, PAGE A-32 THRU -44 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ELABORATE ON PIPING STRESS ANALYSIS PARTICULARLY COMBINING HORIZONTAL AND VERTICAL COMPONENT OF EARTHQUAKE MOTION. SUBSTANTIATE STATEMENT THAT DIFFERENTIAL MOTION OF POINTS OF SUPPORT OF PIPING IS ZERO. IS RELATIVE OR OBSOLETE ACCELERATION OF EQUIPMENT FROM BUILDING MOTION ASSUMED. WHAT DESIGN PROVISIONS ENSURE NO DAMAGE TO SERVICE LINES WHICH PASS CLOSE TO WALLS. DISCUSS DISLODGE MENT POTENTIAL OF FLOOR GRATINGS UNDER EARTHQUAKE LOADING AND PROTECTION PROVIDED PIPING TO PREVENT PUNCTURE BY GRATING RODS AT OPENINGS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + STRESS ANALYSIS + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + EARTHQUAKE ENGINEERING + PIPES AND PIPE FITTINGS

2-029442 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION A-11 - PROTECTION OF CLASS I EQUIPMENT FROM TORNADES

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE A-45 AND A-46 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS THE DESIGN PROVISIONS MADE FOR PROTECTION OF CLASS I EQUIPMENT FROM TORNADO WINDS AND TORNADO GENERATED MISSILES. DISCUSS THE TORNADO GENERATED MISSILES IN DETAIL. (ANSWER) GRATING SURROUNDS HEAT EXCHANGERS AND DIESEL GENERATOR TO PROTECT FROM FLYING DEBRIS. SEE ALSO SUPPLEMENT 1 OF PSAR PG 3-62 AND APPENDIX E.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CATEGORY 2
SITING OF NUCLEAR FACILITIES

2-029442 *CONTINUED*
REACTOR, LMCR + MISSILE GENERATION AND PROTECTION + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + DESTRUCTIVE WIND + GENERATOR, DIESEL + HEAT EXCHANGERS

2-029444 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION A-6 - DYNAMIC LOADING OF CLASS I ITEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE A-28 AND -29 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS METHODS USED TO TAKE INTO ACCOUNT THE INFLUENCE OF ROCKING, HORIZONTAL TRANSLATION AND ELASTICITY OF THE SUPPORTING STRUCTURES ON THE DYNAMIC LOADING OF CLASS I ITEMS. DISCUSS THE RESULTS OBTAINED FOR A REPRESENTATIVE CLASS I STRUCTURE AND CLASS I SYSTEM. (ANSWER) SEE QUESTION B-2. PIPING IS DISCUSSED IN QUESTIONS A-7 AND A-10B AND C.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + SUPPORT STRUCTURE + EARTHQUAKE ENGINEERING + INTERACTION, SOIL AND STRUCTURE + PIPES AND PIPE FITTINGS

2-029445 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION B-2 - MATHEMATICAL MODEL FOR SEISMIC DESIGN
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, PAGE B-13 THRU -15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THE MATHEMATICAL MODEL USED FOR THE SEISMIC DESIGN OF THE CONTAINMENT STRUCTURE ASSUMES A VERTICAL FLEXIBLE CANTILEVER. ROCKING AND DIFFERENTIAL HORIZONTAL TRANSLATION OF THE BASE AND THE INTERACTION BETWEEN THE PRIMARY AND SECONDARY CONTAINMENTS APPEAR TO HAVE BEEN NEGLECTED. DISCUSS THE ERROR INTRODUCED INTO THE DESIGN BY THESE ASSUMPTIONS. (ANSWER ON PGS B-13 THRU B-15.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + REACTOR, LMCR + REPORT, SAR + CONTAINMENT STRUCTURE + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + MATHEMATICAL TREATMENT + INTERACTION, SOIL AND STRUCTURE

2-029449 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION C-1 - INTERACTION OF INNER AND OUTER CONTAINMENTS DURING EARTHQUAKE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE C-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE THE LOCAL LOADING GENERATED DURING AN EARTHQUAKE BY THE INTERACTION OF THE INNER AND OUTER CONTAINMENTS. INCLUDE THE ROCKING AND THE DIFFERENTIAL (WITH RESPECT TO THE GROUND) HORIZONTAL TRANSLATION OF THE BUILDING. PROVIDE THE CORRESPONDING MAXIMUM STRESSES AND STRAINS. (ANSWER) SEE ANSWER TO QUESTION B-2.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + INTERACTION, SOIL AND STRUCTURE

2-029450 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION C-2 - LOCAL STRESSES DUE TO EARTHQUAKE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE C-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE THE LOCAL STRESSES, DUE TO AN EARTHQUAKE, AT THE JUNCTION OF THE STEEL SHELL AND HEAVY CONCENTRATED MASSES, SUCH AS LOCKS, HATCHES, AND LARGE PENETRATIONS. FOR EXAMPLE, AN EMERGENCY ESCAPE LOCK PROTRUDES FROM THE CONTAINMENT SHELL AT GROUND LEVEL AND APPEARS TO BE SUPPORTED BY THE SHELL. THIS PROTRUSION IS MASSIVE AND LONG AND MAY REQUIRE ADDITIONAL SUPPORT TO PREVENT IT FROM OVERLOADING THE CONTAINMENT SHELL AT THE POINT OF FASTENING. ADDITIONAL INFORMATION IS REQUESTED AS TO THE DESIGN OF THIS ESCAPE LOCK AND THE PROVISIONS THAT HAVE BEEN OR WILL BE TAKEN TO INSURE THAT IT CANNOT IN ANY WAY JEOPARDIZE THE CONTAINMENT OF THE STRUCTURE UNDER ACCIDENT AND EARTHQUAKE LOADING CONDITIONS. (ANSWER) SEE SECT. 7.3 AND SUPPL. 7 TO FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + STRESS + CONTAINMENT AIR LOCK + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING

CATEGORY 2
SITING OF NUCLEAR FACILITIES

2-029483 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION H-2 - FRICTION FORCES BETWEEN REFLECTORS AND EXTENSION SHAFTS
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 2 PAGES, PAGE H-3 AND-4 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE AND DISCUSS POSSIBILITIES OF HIGH FRICTIONAL FORCES BEING CREATED BETWEEN REFLECTORS AND/OR EXTENSION SHAFTS AND FIXED STRUCTURES DUE TO VARIATIONS IN COEFFICIENT OF FRICTION BETWEEN REFLECTOR PADS AND GUIDE RAILS AND MINIMUM RUNNING CLEARANCES AS A RESULT OF FABRICATION TOLERANCES, INSTALLATION TOLERANCES, AND DISPLACEMENTS DUE TO SEISMIC LOADINGS. (ANSWER) SEE SECT. 7 OF SUPPLEMENT 11 TO FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

GUIDE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + EARTHQUAKE + REFLECTOR + RESPONSE TO NRC QUESTION + CORE COMPONENTS + STRUCTURAL INTEGRITY + ABLATION + DISPLACEMENT DESIGN

2-029511 ALSO IN CATEGORIES 16 AND 18 DATED 1968
RESPONSE TO QUESTION M-2 - ON-SITE METEOROLOGICAL DATA
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 1 PAGE, PAGE M-2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE ON-SITE METEOROLOGICAL DATA DESCRIBED IN TABLE III-5 OF THE FDSAR. (ANSWER) REPORT IN FINAL STAGES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SITE CLIMATOLOGY + METEOROLOGY + SEFOR (RE) + RESPONSE TO NRC QUESTION

2-029512 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION M-3 - LIQUID WASTE STORAGE FIELD
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 4 PAGES, 1 FIGURE, 1 TABLE, PAGE M-2 THRU -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE ELEVATION OF TILE FIELD INTO WHICH LIQUID RADIOACTIVE WASTES WILL BE DISCHARGED. DURING A PEAK PRECIPITATION PERIOD ASSOCIATED WITH PEAK RUNOFF CONDITIONS (MAXIMUM PROBABLE FLOODING CONDITIONS) PROVIDE LEVEL TO WHICH FALL CREEK WOULD RISE AND DISCUSS POTENTIAL FOR FLOODING OF TILE FIELD CARRYING ITS ACCUMULATED RADIOACTIVE FISSION PRODUCTS. (ANSWER ON PGS M-2 THRU -5.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + WASTE STORAGE + WASTE DISPOSAL, LIQUID + SEFOR (RE) + RESPONSE TO NRC QUESTION + FLOOD

2-032817 ALSO IN CATEGORY 18 DATED 1968
ADDITIONAL INFORMATION ON SEISMIC DESIGN
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 50231-6 +. 6 PAGES, 1 FIGURE, 2 TABLES, SEFOR, PSAR, SUPPLEMENT 6, FEB. 14, 1968, DOCKET 50-231

RESULTS OF DYNAMIC AND STATIC SEISMIC ANALYSIS GIVEN. TABLE LISTS LOADING OF ALL MAJOR EQUIPMENT (PUMPS, CRANES, TANK, AND HEAT EXCHANGERS). RESULTS FOR 109-FT STACK ARE - PERIOD 0.71 SEC, MAX. DISPLACEMENT 1.82 IN., AND MAX. MOMENT 31.8 FT. K.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, FAST + REACTOR, LMCR + STACK + STRESS ANALYSIS + REACTOR, PULSED + SEFOR (RE) + EQUIPMENT + REPORT, PSAR + EARTHQUAKE ENGINEERING + INTERACTION, SOIL AND STRUCTURE + STRUCTURAL ANALYSIS, DYNAMIC + REACTOR, LMFB

2-032880 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION 3 - TORNADO - REFUELING CELL OPERATION
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 50231-20 +. 1 PAGES, PAGE 12 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

WHAT SAFETY PRECAUTIONS ARE PLANNED FOR ENSURING SAFE OPERATION IN THE REFUELING CELL WITH REGARD TO POTENTIAL TORNADO DAMAGE. *** (ANSWER) FOR DESIGN PROVISIONS, SEE PSAR, SUPPLEMENT 1, PP. 3-61 THRU 3-66, FDSAR, VOL. 1, PAGE 7-1, PAGE 7-9, PAGE 7-16, AND FDSAR SUPPLEMENT 17, ANSWER TO QUESTIONS A-11, D-2, AND D-8.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CATEGORY 2
SITING OF NUCLEAR FACILITIES

2-032880 *CONTINUED*
REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REFUELING + FUEL STORAGE + STORAGE CONTAINER + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + DAMAGE + DESTRUCTIVE WIND + REPORT, PSAR + REACTOR, LMFBR

2-032915 ALSO IN CATEGORIES 16 AND 18 DATED 1968
AEC QUESTIONS - 11 TOPICS
GENERAL ELECTRIC
50231-21 +. 150 PAGES, 25 FIGURES, TABLES, SEFOR, PSAR, SUPPLEMENT 19, OCTOBER 1968, DOCKET 50-231

SEVENTY-THREE PAGES DEVOTED TO RESPONSES TO 11 AEC QUESTIONS ON CONTROL SYSTEM MODIFICATION, AUXILIARY POWER, REFUELING ACCIDENT, THE EXCURSION DEVICE (FRED) LEAKS, TRANSIENT TEST LOADS, NOZZLE FAILURE, FUEL-ROD PIN-HOLE LEAKS, SURVEILLANCE PROGRAM, THERMAL INSULATION, AND EFFECT OF SODIUM IN THE CORE DURING MHA. APPENDIX (43 PAGES) CONSISTS OF SITE METEOROLOGICAL DATA.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

NRC QUESTION + CONTAINMENT VESSEL LOADING + CONTROL SYSTEM + EMERGENCY POWER, ELECTRIC + FAILURE, EQUIPMENT + REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + ACCIDENT, REFUELING + SITE CLIMATOLOGY + ACCIDENT, HYPOTHETICAL + REACTOR, PULSED + SEFOR (RE) + THERMAL INSULATION + NOZZLE + REPORT, PSAR + MAIN COOLING SYSTEM + SURVEILLANCE PROGRAM + LEAK + REACTOR, LMFBR

2-032917 ALSO IN CATEGORIES 10 AND 18 DATED 1968
RESPONSE TO QUESTION 2 - LOSS OF SITE POWER - TORNADO
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 5 PAGES, 7 FIGURES, PAGES 6 THRU 11 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

WHAT PROVISIONS HAVE BEEN MADE FOR LOSS OF SITE POWER DUE TO A TORNADO PASSING THROUGH OR NEAR THE REACTOR SITE. *** ANSWER ON PAGES 6 THRU 11.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF POWER + ELECTRIC POWER, NORMAL + REACTOR, PULSED + SEFOR (RE) + SAFE FAILURE CRITERION + RESPONSE TO NRC QUESTION + DESTRUCTIVE WIND + REPORT, PSAR + REACTOR, LMFBR

2-033535 ALSO IN CATEGORIES 11 AND 18 DATED 1968
LMFBR PROGRAM PLAN. VOL. 2 - PLANT DESIGN
ARGONNE NATIONAL LAB., LMFBR PROGRAM OFFICE
WASH-1102 +. 197 PAGES, 8 FIGURES, 17 TABLES, AUGUST 1968

DISCUSSES DEVELOPMENT OF PLANT CONCEPTS AND TECHNOLOGY FOR PLANT DESIGN. GIVES DETAILS OF OVERALL PLAN. MAIN TOPICS - (1) PLANT DESIGN DEVELOPMENT STUDIES, (2) GENERAL STUDIES - SITING, ETC., (3) DEMONSTRATION PLANT PROGRAM, AND (4) COMPILATION, ASSESSMENT, AND DISSEMINATION OF DESIGN INFORMATION

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
ECONOMICS + DESIGN STUDY + INFORMATION RETRIEVAL + R AND D PROGRAM + REACTOR, LMFBR + JACOBS

CATEGORY 3
TRANSPORTATION AND HANDLING OF RADIOACTIVE MATERIALS

3-012530 DATED 1959
SUDDATH JC
NUCLEAR AND RADIATION HAZARDS EVALUATION OF SRE FUEL PROCESSING AND STORAGE
OAK RIDGE NATIONAL LABORATORY
ORNL-CF-59-5-90 +. 3 PAGES, MAY 20, 1959

RESULTS ARE PRESENTED OF AN EVALUATION OF NUCLEAR SAFETY AND RADIATION CONTROL RELATED TO THE SHIPMENT, MECHANICAL PROCESSING, AND STORAGE OF SRE-1 FUEL ELEMENTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA., \$1.10 CY, \$0.80 MF

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + SRE (RE) + HAZARDS ANALYSIS + FUEL ELEMENTS

3-012533 DATED 1959
SUDDATH JC
DECAY HEAT IN SHIPPING CONTAINERS
OAK RIDGE NATIONAL LABORATORY
ORNL-CF-59-11-30 +. 6 PAGES, NOVEMBER 6, 1959

A TEST EMPLOYING A SIMULATED SRE FUEL ELEMENT IN THE SHIPPING CARRIER INDICATED THAT THE MAXIMUM SURFACE TEMPERATURE OF THE ACTUAL SRE ELEMENTS WILL NOT EXCEED 532 F WHEN 10 OF THE MOST ACTIVE ELEMENTS ARE IN THE CARRIER. USING MEASURED SURFACE TEMPERATURES AND HEAT FLUXES, THE AIR FILM COEFFICIENT WAS CALCULATED TO BE 2.9 BTU/(HR)(SQ FT)(DEG F).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA., \$1.80 CY, \$1.80 MF

DECAY HEAT + REACTOR, GRAPHITE MODERATED + HEAT TRANSFER + REACTOR, LMCR + SHIPPING CONTAINER + SRE (RE)

3-014469 DATED 1960
KETZLECH N
CRITICALITY STUDY FOR HALLAM FUEL FABRICATION AND SHIPMENT
ATOMICS INTERNATIONAL
NAA-SR-MEMO-5975 +. 25 PAGES, DECEMBER 16, 1960

AN EVALUATION OF THE CRITICALITY PARAMETERS HAS BEEN MADE TO DETERMINE THE SAFE-HANDLING LIMITS OF URANIUM ALLOY ELEMENTS FOR THE HALLAM CORE LOADING. THE FUEL CONTAINS 3.6 PERCENT U235 ENRICHED URANIUM ALLOYED WITH MOLYBDENUM SO THAT THE ALLOY CONTAINS 10 PERCENT MOLYBDENUM BY WEIGHT. THE RESULTING ALLOY DENSITY IS 17.2 GRAMS/CM3. RESULTS INDICATE IT IS SAFE TO HANDLE 60 SLUGS (1.8 KG CONTAINED U235) OF 0.59-IN. DIAM HAVING A MAXIMUM LENGTH OF 12 INCHES. WHERE THERE IS NO CONTROL OF FUEL GEOMETRY (E.G., SAWDUST CHIPS, ETC.), THE MAXIMUM SAFE BATCH SIZE SHOULD CONTAIN NO MORE THAN 0.94 KG CONTAINED U235. WHERE THERE IS NO CONTROL OF SPACING BETWEEN FUEL RODS, THE MAXIMUM SAFE NUMBER IS 29.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICRONEGATIVE

CRITICALITY SAFETY + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + NEUTRON INTERACTION + FUEL ELEMENTS

3-014870 ALSO IN CATEGORY 7 DATED 1966
SCHLAPP FW
STORAGE OF MODERATOR ELEMENTS IN HNPf MODERATOR STORAGE CELLS
ATOMICS INTERNATIONAL
NAA-SR-MEMO-12247 +. 7 PAGES, NOVEMBER 28, 1966

GROUNDWATER WAS ASSUMED TO CONTACT AND START DISSOLVING THE ACTIVATED STAINLESS STEEL PORTIONS OF THE THREE IRRADIATED MODERATOR ELEMENTS NOW STORED IN THE HNPf MODERATOR-STORAGE CELLS. THE SOLUBILITY AND DECOMPOSITION RATES OF THE METAL AND THE CONCENTRATION OF RADIOACTIVITY RELEASABLE TO THE GROUNDWATER WERE ANALYZED. THE RESULTS SHOWED THAT INSIGNIFICANT CONCENTRATIONS OF RADIOACTIVE MATERIALS CAN BE RELEASED, EVEN IF THE CONCRETE CELL STRUCTURE AND THE STEEL LINERS ARE NEGLECTED IN CALCULATING THE DISSOLUTION AND DISPERSAL OF THE MATERIAL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICRONEGATIVE

FISSION PRODUCT RELEASE + FUEL INTEGRITY + REACTOR, GRAPHITE MODERATED + GROUND WATER + HALLAM (LMR) + REACTOR, LMCR + RADIOACTIVITY RELEASE

3-023229 ALSO IN CATEGORY 13 DATED 1968
UNGER WE + CROUSE DJ + WATSON CD + IRVINE AR
LMFBR FUEL PROCESSING STUDIES STATUS REPORT FOR FEBRUARY 1968
OAK RIDGE NATIONAL LABORATORY
ORNL-CF-68-3-13 +. 21 PAGES, 4 FIGURES, 3 TABLES, MARCH 11, 1968

MONTHLY STATUS REPORT OF FAST BREEDER REACTOR AQUEOUS PROCESS DEVELOPMENT IN CHEMICAL TECHNOLOGY DIVISION. STUDIES UNDERWAY OF DISPOSAL OF SODIUM, BEHAVIOR AND REMOVAL OF IODINE, SOLVENT

CATEGORY 3
TRANSPORTATION AND HANDLING OF RADIOACTIVE MATERIALS

3-023229 *CONTINUED*
STABILITY UNDER RADIATION AND CHEMICAL DEGRADATION, REMOVAL OF NOBLE GASES, SHIPPING OF REACTOR FUEL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + IODINE + REACTOR, LMCR + NOBLE GAS + PLUTONIUM + SHIPPING CONTAINER + SODIUM + URANIUM + ORNL + RADIOCHEMICAL PROCESSING + EXPERIMENT + SOLVENT EXTRACTION PROCESS + FUEL REPROCESSING

3-023755 DATED 1967
BERGER S + BONYHADY K + DAVIS B
HNPF SIX-ELEMENT IRRADIATED FUEL SHIPPING CASK
ATUMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-12,547 +. 442 PAGES, FIGURES, TABLES, NOVEMBER 23, 1967

PRESENTS DETAILED ANALYSES AND DRAWINGS OF THE HALLAM SPENT-FUEL SHIPPING CASK. THE CASK IS DESIGNED AS A GENERAL-PURPOSE CASK FOR USE AFTER THE HALLAM FUEL SHIPMENTS ARE COMPLETED BY THE AEC. ALSO DESIGNED TO MEET REQUIREMENTS OF USAEC MANUAL CHAPTER 0529, DATED 8/22/66, WHICH PARALLELS CFR TITLE 10 PART 71.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00 COPY, \$0.65 MICROFICHE

CRITICALITY SAFETY + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + SHIPPING CONTAINER + SHIELDING + HEAT TRANSFER ANALYSIS + STRUCTURAL INTEGRITY + SAFETY ANALYSIS

3-029831 ALSO IN CATEGORIES 5 AND 17 DATED 1968
RATEICK RC + MCHUGH WE + SALING JH
RESULTS AND CONCLUSIONS FROM THE EXAMINATION OF FUEL SUBASSEMBLIES REMOVED FROM THE ENRICO FERMI REACTOR AFTER THE FUEL DAMAGE INCIDENT
POWER REACTOR DEVELOPMENT COMPANY + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + BATTELLE MEMORIAL INSTITUTE
2 PAGES, 3 REFERENCES, ANS TRANS. 11(1), PAGES 100-101, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

ON OCT. 5, 1966, ZR-CLAD U-MO FUEL PINS PARTIALLY MELTED. OFF-SITE EXAM MADE TO DETERMINE IF FAILURE CAUSED BY FLOW BLOCKAGE. CONCLUSIONS - (1) SHIPMENT OF SUBASSEMBLIES WITHOUT PRIOR CLEANING CAN BE DONE AND EXAM ADEQUATELY PERFORMED IN CONVENTIONAL HOT CELL. (2) MELTING WAS CAUSED BY FLOW BLOCKAGE, (3) EVEN THOUGH SS WRAPPER WALLS LOST INTEGRITY IN MELTED SUBASSEMBLIES, GROSS FUEL LOSSES DID NOT OCCUR. (4) REDISTRIBUTION IN TWO MELTED ASSEMBLIES ACCOUNTED FOR 11.5 CENTS REACTIVITY OF TOTAL LOSS OF 22 CENTS. REMAINING 10.5 CENTS THOUGHT TO BE DUE TO RESULT OF DEFORMATION.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + HOT CELL + REACTOR, LMCR + TRANSPORTATION AND HANDLING + INCIDENT + REACTIVITY EFFECT, SLUMP + EXAMINATION + FLOW BLOCKAGE + REACTOR, LMFBR

3-030256 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 6 - SPENT FUEL STORAGE
NORTHERN STATES POWER COMPANY
2 PAGES, PP. 34-35 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

INITIAL OPERATION OF THE IRRADIATED FUEL STORAGE TANK DESCRIBED. ***REFERS TO SAR SUPPLEMENT 13. ORIGINAL DESIGN WAS FOR STORAGE OF 1.5 CORES, BUT PRESENT PLAN IS FOR NO REFUELING SO INITIAL OPERATION WILL BE FOR 30 RODS MAX. STORED IN GAS RATHER THAN SODIUM. CALCULATIONS (PRESENTED) SHOW PEAK TEMPERATURE OF CLAD IS 820 F. EVEN IF 6 RODS WERE PLACED IN ONE CHANNEL, TEMPERATURE WOULD BE ONLY 1225 F.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REPORT, SAR + FUEL STORAGE + HIGH TEMPERATURE + SEFOR (RE) + RESPONSE TO NRC QUESTION + REACTOR, LMFBR + FUEL ELEMENTS

3-033428 ALSO IN CATEGORIES 13 AND 18 DATED 1968
LMFBR PROGRAM PLAN. VOLUME 8 - FUEL RECYCLE
LMFBR PROGRAM OFFICE, ARGONNE NATIONAL LAB.
WASH-1108 +. 350 PAGES, FIGURES, TABLES, AUGUST 1968

CONTENTS INCLUDE CHARACTERISTICS OF FUELS, FABRICATION, COSTS, PROCESSING OF CORE AND BLANKET, AND A THOROUGH DISCUSSION OF THE TASKS, INCLUDING COORDINATION OF RECYCLE DEVELOPMENTS, SYSTEM DEMONSTRATIONS, HANDLING AND SHIPPING, RECOVERY, FUEL PREPARATION AND FABRICATION, AND GENERAL STUDIES. APPENDIXES INCLUDE FLOW SHEETS, SUB-TASKS, ECONOMICS, AND A COLLATION OF PROGRAM TASKS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, BREEDER + ECONOMICS + FABRICATION + REACTOR, FAST + RADIOCHEMICAL PROCESSING + RECOVERY PROCESS + FUEL REPROCESSING + R AND D PROGRAM + REACTOR, LMFBR + FUEL MANAGEMENT + JACOBS + FUEL ELEMENTS

CATEGORY 3
TRANSPORTATION AND HANDLING OF RADIOACTIVE MATERIALS

3-034706 ALSO IN CATEGORIES 12 AND 13 AND 17 DATED 1969
JOHNSON KP + MATTE J
OPERATING EXPERIENCE WITH THE FERMI FUEL HANDLING CASK CAR
POWER REACTOR DEVELOPMENT COMPANY, MONROE, MICHIGAN + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN
15 PAGES, 9 FIGURES, PP. 135-149 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

THE ENRICO FERMI ATOMIC POWER PLANT WAS PROVIDED WITH A CASK CAR FOR TRANSPORTING SPENT FUEL BETWEEN THE REACTOR BUILDING AND THE FUEL AND REPAIR BUILDING. THIS CAR, A MULTI-STATION, SELF-CONTAINED UNIT, EXHIBITED A VARIETY OF PROBLEMS AND DEFECTS IN EARLY OPERATION. ANALYSIS OF THESE PROBLEMS AND THE AVAILABLE SOLUTIONS REQUIRED TO ACHIEVE SATISFACTORY PERFORMANCE UNDER DESIGNED OPERATING CONDITIONS LED TO CONSIDERATION OF REPLACING THE CASK CAR WITH A NEW FACILITY.

AVAILABILITY - K.P. JOHNSON, POWER REACTOR DEVELOPMENT COMPANY, P. O. BOX 725, MONROE, MICHIGAN 48161

REACTOR, BREEDER + FAILURE, EQUIPMENT + REACTOR, FAST + FERMI (LMFBR) + FUEL HANDLING + REACTOR, LMCR + SHIPPING CONTAINER + TRANSPORTATION AND HANDLING + FUEL HANDLING MACHINE + OPERATING EXPERIENCE + REACTOR, LMFBR

3-034707 ALSO IN CATEGORY 12 DATED 1969
WILLIAMS RO + HALLINAN GJ + KIRKBY RC
THE TECHNICAL EVOLUTION OF THE FERMI FUEL TRANSPORT FACILITY
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF. + POWER REACTOR DEVELOPMENT COMPANY, MONROE, MICHIGAN
11 PAGES, 8 FIGURES, 4 REFERENCES, PP. 150-160 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

THE ENRICO FERMI ATOMIC POWER PLANT IS EQUIPPED WITH A FUEL TRANSPORT FACILITY WHICH FUNCTIONS TO WITHDRAW FUEL SUBASSEMBLIES AND SPECIAL CORE COMPONENTS FROM EITHER THE FUEL-STORAGE OR REACTOR TRANSFER ROTOR, AND TO TRANSPORT AND DEPOSIT THE SUBASSEMBLY OR COMPONENT INTO THE OPPOSITE ROTOR. THIS PAPER DISCUSSES THE EVOLUTION OF THE FUEL TRANSPORT FACILITY FROM THE ORIGINAL MULTI-BARRELED MACHINE TO THE PRESENT SINGLE-BARREL MACHINE. THE DECISION TO PURCHASE THE NEW MACHINE WAS MADE AFTER REVIEW OF THE MAINTENANCE REQUIREMENTS NECESSITATED BY DESIGN DEFICIENCIES IN THE ORIGINAL MACHINE.

AVAILABILITY - R.O. WILLIAMS, JR., ATOMICS INTERNATIONAL, P.O. BOX 309, CANOGA PARK, CALIFORNIA 91304

REACTOR, BREEDER + DESIGN CRITERIA + REACTOR, FAST + FERMI (LMFBR) + FUEL HANDLING + REACTOR, LMCR + TRANSPORTATION AND HANDLING + REACTOR, POWER + FUEL HANDLING MACHINE + REMOTE MANIPULATING AND VIEWING + OPERATING EXPERIENCE + REACTOR, LMFBR

3-034711 ALSO IN CATEGORY 13 DATED 1969
MALECHA RF + COLEMAN LF + FINEMAN P + KELECIUS A + SLAWECKI MA
EBR-II INTRASITE SPENT-FUEL SHIPPING CASKS
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
7 PAGES, 3 FIGURES, 1 REFERENCES, PP. 205-211 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

A CHANGE IN THE STATUS OF THE EXPERIMENTAL BREEDER REACTOR II (EBR-II) FROM AN EXPERIMENTAL REACTOR TO A NEUTRON IRRADIATION FACILITY NECESSITATED THE CONSTRUCTION OF TWO DIFFERENT CASKS. THE LEAD-SHIELDED CASKS ARE TO BE USED TO TRANSFER SPENT FUEL TO AN ON-SITE STORAGE AREA. ONE CASK HAS BEEN DESIGNED AND BUILT WHICH PROVIDES AN WATER-COOLED FUEL CAVITY. THE SECOND CASK IS DESIGNED FOR BOTTOM LOADING AND INCORPORATES AN INTEGRAL, TOTALLY ENCLOSED, RECIRCULATING AIR COOLING SYSTEM FOR THE FUEL.

AVAILABILITY - R.F. MALECHA, ARGONNE NATIONAL LABORATORY, 9700 S. CASS AVENUE, ARGONNE, ILLINOIS 60439

REACTOR, BREEDER + DESIGN CRITERIA + REACTOR, FAST + REACTOR, LMCR + SHIPPING CONTAINER + TRANSPORTATION AND HANDLING

3-044436 ALSO IN CATEGORY 1 DATED 1969
SEABORG, GT
LOOKING AHEAD IN NUCLEAR (GLENN SEABORG)
U.S. ATOMIC ENERGY COMMISSION
5 PAGES, MECHANICAL ENGINEERING, PP. 30-34 (AUGUST 1969)

REPORT OF AN ADDRESS AT THE 37TH CONVENTION OF THE EDISON ELECTRIC INSTITUTE WHEREIN SEABORG DISCUSSED THE BROAD ASPECTS OF THE PROSPECTS FOR N-POWER. HE REPORTS ON THE NUMBER OF PRESENT AND PROJECTED POWER REACTORS, THEIR POWER OUTPUT, AND THE PROGRESS IN AND NEED FOR BREEDER REACTORS. REGARDING THE LATTER, HE MENTIONS AEC REPORT -- COST-BENEFIT ANALYSIS OF U.S. BREEDER PROGRAM -- A STUDY BASED ON OPTIMIZATION OF U.S. ELECTRIC ENERGY ECONOMY FROM 1970 TO 2020. THE QUALITY ASSURANCE INFLUENCE ON THE AEC, THE NUCLEAR INDUSTRY, AND THE UTILITIES INTERACTION IS DISCUSSED. NUCLEAR PHOBIA, THERMAL POLLUTION, AND RADIOACTIVE WASTE MANAGEMENT ARE DISCUSSED, AND SEABORG IMPLIES THAT THESE ARE (PROBLEM) AREAS WHERE CONCERN IS REALLY NOT WARRANTED, BEING SOLVED OR SOLVABLE.

*N-POWER FORECAST + PROPONENT + *SPOKESMAN, AEC + *SOCIO/PHILOSOPHICAL CONSIDERATION + REACTOR POWER + REACTOR, BREEDER + REACTOR, LMFBR + REACTOR, HTGR + N-POWER, SAFETY OF + DATA COLLECTION + ECONOMICS + ENERGY PREDICTION + FUEL, NUCLEAR + FUEL, FOSSIL + QUALITY ASSURANCE + WASTE MANAGEMENT + PUBLIC RELATIONS + THERMAL POLLUTION + ATMOSPHERIC POLLUTION + WASTE HEAT MANAGEMENT + BENEFICIAL USE, AGRICULTURE + BENEFICIAL USE, COMMERCIAL + BENEFICIAL USE, RECREATIONAL + ESTHETICS + AGENCY, PROMOTIONAL + INDUSTRY, UTILITY + ENVIRONMENT

CATEGORY 3
TRANSPORTATION AND HANDLING OF RADIOACTIVE MATERIALS

3-044436 *CONTINUED*
+ FISSION PRODUCT RELEASE

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-000899 DATED 1959
GARELIS E
SODIUM-AIR ACCIDENT STUDY FOR A SODIUM COOLED REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + U. S. ATOMIC ENERGY COMMISSION
AECU-4161 +. 22 PAGES, 5 TABLES, 12 REFERENCES, APRIL 6, 1959, CFSTI, \$4.80 CY

THE DESIGN REQUIREMENTS OF A HYPOTHETICAL SODIUM-AIR REACTION ON THE CONTAINMENT BUILDING FOR A SODIUM-COOLED REACTOR WERE INVESTIGATED FOR AN ATOMIC POWER PLANT. THE ACCIDENT ASSUMES A MAJOR LEAK IN THE SODIUM SYSTEM WITH A FAILURE OF THE VENTILATION AND COOLING SYSTEM, WHICH ALLOWS AIR TO ENTER THE AREAS AROUND THE SODIUM SYSTEM (NORMALLY MAINTAINED IN AN INERT ATMOSPHERE). THE RESULTING SODIUM POOL BURNS AT A RATE DEPENDING UPON OXYGEN CONCENTRATION AND GAS TEMPERATURE IN THE CONTAINMENT BUILDING. A SERIES OF EXPERIMENTS WERE CARRIED OUT TO DETERMINE THIS BURNING RATE AND ITS MECHANISM. THE RESULTING GAS TEMPERATURE AND PRESSURE DUE TO THE SODIUM-AIR REACTION WERE DETERMINED BY SOLVING A SYSTEM OF EQUATIONS TO GIVE A MAXIMUM GAS PRESSURE OF ABOUT 26 PSIG AND A CORRESPONDING GAS TEMPERATURE OF 1115 F, WELL BELOW THE CONTAINMENT DESIGN PRESSURE OF 32 PSIG. THE THEORETICAL MAXIMUM PRESSURE AND TEMPERATURE FOR THE CONSTANT-VOLUME, ADIABATIC, SODIUM-AIR REACTION IS ALSO DISCUSSED.

ACCIDENT ANALYSIS + REACTOR, BREEDER + CHEMICAL REACTION + CONTAINMENT DESIGN + DESIGN CRITERIA + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + SODIUM

5-003279 DATED 1963
DAWSON JK + SOWDEN RG
CHEMICAL ASPECTS OF NUCLEAR REACTORS, VOL. 3 - MISCELLANEOUS TOPICS
3 PAGES- CHAPTER 1, LIQUID METALS AS REACTOR COOLANTS - SECTION 1.1.3, PROBLEMS OF SAFETY- 1963, BUTTERWORTHS, LONDON

THE HAZARDS OF THE OXIDATION AND PYROPHORIC REACTION OF SODIUM IN A SODIUM-COOLED REACTOR ARE DISCUSSED (AND REFERENCES GIVEN). THE ACCIDENTAL RELEASE OF FISSION PRODUCTS INTO SODIUM IS TREATED BRIEFLY (AND DETAILED WORK REFERENCED).

REACTOR, LMCR + OXIDATION + PYROPHORIC REACTION + SODIUM

5-003647 DATED 1964
Dwyer DE (BNL) + LYON RN (ORNL)
LIQUID-METAL HEAT TRANSFER
BROOKHAVEN NATIONAL LABORATORY + OAK RIDGE NATIONAL LABORATORY
15 PAGES- UNITED NATIONS THIRD INTERNATIONAL CONFERENCE ON THE PEACEFUL USES OF ATOMIC ENERGY, GENEVA, AUGUST 31-SEPTEMBER 9, 1964- A/CONF. 28/P/225

HEAT TRANSFER EQUATIONS, DEVELOPED FOR LIQUID METALS, INVOLVE THE EDDY DIFFUSIVITY OF MOMENTUM TRANSFER EPSILON M AND THE RATIO PSI OF THE EDDY DIFFUSIVITIES OF HEAT AND MOMENTUM TRANSFER. THE DETERMINATION OF EPSILON M AND PSI AND THEIR EFFECTS ON NUSSELT NUMBER ARE DISCUSSED. THEORY IS COMPARED WITH EXPERIMENT FOR CHANNEL FLOW. THE CALCULATION OF HEAT TRANSFER COEFFICIENTS FOR CROSS FLOW THROUGH ROD BUNDLES AND TUBE BANKS IS CONSIDERED. GEOMETRICAL ASYMMETRY EFFECTS ON HEAT TRANSFER RATES ARE DISCUSSED. FINALLY, BOILING LIQUID-METAL HEAT TRANSFER IS DISCUSSED BRIEFLY.

HEAT TRANSFER, BOILING + HEAT TRANSFER + REACTOR, LMCR

5-004938 ALSO IN CATEGORY 7 DATED 1964
THOMPSON DH + MURPHY G + FISHER RW
ENGINEERING DESIGN CONSIDERATIONS FOR A LIQUID-METAL IN-PILE LOOP
AMES LABORATORY
80 PAGES, 12 TABLES, 65 REFERENCES- JANUARY 1, 1964, IS-760- OTS

THIS PAPER PRESENTS THE RESULTS OF AN ANALYTICAL INVESTIGATION IN THE AREAS OF FISSION-PRODUCT ACTIVITY, HEAT TRANSFER, AND MATERIALS FOR THE PRELIMINARY PLANNING OF A HIGH-TEMPERATURE CIRCULATING-LIQUID-METAL-FUEL IN-PILE LOOP. EQUATIONS FOR THE FISSION-PRODUCT NUMBERS AND ACTIVITIES IN A CIRCULATING FUEL WERE DERIVED FROM EQUATIONS EXISTING IN THE LITERATURE FOR THE GROWTH AND DECAY OF FISSION PRODUCTS RESULTING FROM ONE REACTOR OPERATION AND SHUTDOWN CYCLE. THESE EQUATIONS WERE DERIVED FOR CONTINUOUS LOOP IRRADIATION AND FOR THE MORE REALISTIC LOOP IRRADIATION IN A REACTOR OPERATING ON A PERIODIC CYCLE.

DESIGN CRITERIA + IN PILE EXPERIMENT + REACTOR, LMCR + HAZARDS ANALYSIS

5-005106 ALSO IN CATEGORY 6 DATED 1964
WEAVER LM
SODIUM REACTOR EXPERIMENT REACTIVITY HISTORY PROGRAM
ATOMICS INTERNATIONAL
28 PAGES, 4 FIGURES, 11 REFERENCES- NAA-SR-10010, OCTOBER 15, 1964- OTS

A CALCULATIONAL TECHNIQUE HAS BEEN DEVELOPED FOR USE AS AN ANALYTICAL TOOL FOR SURVEILLANCE OF THE SODIUM REACTOR EXPERIMENT (SRE) NUCLEAR PERFORMANCE. THIS PROGRAM SPECIFICALLY MAKES A COMPARISON OF THE EXCESS CALCULATED REACTIVITY AND THE MEASURED REACTIVITY DETERMINED FROM THE POSITIONS OF THE CALIBRATED SHIM RODS.

REACTIVITY, EXCESS + REACTOR, LMCR + SRE (IRE)

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-005116 DATED 1964
WISE WK + PROCTOR JF + WALKER LP
RESPONSE OF ENRICO FERMI REACTOR TO TNT SIMULATED NUCLEAR ACCIDENTS
U. S. NAVAL ORDNANCE LABORATORY
157 PAGES, 47 FIGURES, 11 TABLES, 19 REFERENCES- NOLTR 62-207, NOVEMBER 13, 1964- OTS

THE MHA HAS BEEN POSTULATED FOR THE PURPOSE OF THIS REPORT NOT TO EXCEED IN VIOLENCE THE DETONATION OF 1000 POUNDS OF TNT AT THE CORE LOCATION. THIS REPORT IS ADDRESSED TO PREDICTING THE STRUCTURAL RESPONSE OF FERMI TO THE MHA, PARTICULARLY THE POSSIBLE JUMP OF THE SHIELD PLUG. AN EXTENSIVE EXPERIMENTAL PROGRAM WAS CONDUCTED IN WHICH SCALED CHARGES OF EXPLOSIVES (SOME CASED IN MOLTEN SODIUM) WERE DETONATED IN IDEALIZED MODELS OF THE FERMI REACTOR. THE RESULTS OF THIS EXPERIMENTAL PROGRAM TOGETHER WITH AVAILABLE THEORY INDICATE THAT THE ACCIDENT WOULD BE CONTAINED WITHIN THE REACTOR SHIELDING COMPLEX, BUT THAT THE 140-TON SHIELD PLUG, IF UNIMPEDED, MIGHT BE PROPELLED AS HIGH AS 102 FEET AND THROUGH THE ROOF OF THE CONTAINMENT BUILDING. IN RESPONSE TO THE FINDINGS OF THIS EXTENSIVE STUDY, THE FERMI REACTOR WILL BE EQUIPPED WITH AN ARRESTER THAT WILL PREVENT A HAZARDOUS PLUG JUMP IN THE UNLIKELY EVENT OF A NUCLEAR ACCIDENT.

REACTOR, BREEDER + CONTAINMENT + CONTAINMENT, HIGH PRESSURE + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + MISSILE GENERATION AND PROTECTION + CONTAINMENT, SHOCK GENERATION AND PROTECTION + IMPACT SHOCK

5-005444 DATED 1964
ARNOLD JL + ARNESON SO + MILLER KJ
URANIUM-10 MOLYBDENUM FUEL ELEMENT IRRADIATION IN SRE
ATOMICS INTERNATIONAL
1 PAGE, 3 REFERENCES- AMERICAN NUCLEAR SOCIETY 1964 WINTER MEETING, SAN FRANCISCO, CALIFORNIA, NOVEMBER 30-DECEMBER 3, 1964- P. 447- SOURCE

AS PART OF THE DEVELOPMENT PROGRAM FOR HALLAM, ATOMICS INTERNATIONAL CONDUCTED A SERIES OF CAPSULE-IRRADIATION TESTS TO DEFINE THE BEHAVIOR OF U-10 WT% MO AT TEMPERATURES AND FISSION RATES REPRESENTATIVE OF THOSE EXPECTED IN THE FIRST CORE LOADING. ALSO, A FULL-SIZE FUEL ASSEMBLY (SU 9) WAS FABRICATED AND IRRADIATED IN SRE FOR THE PRIMARY PURPOSE OF TESTING MECHANICAL ASPECTS OF THE FUEL-ELEMENT DESIGN. RESULTS OF THIS EXPERIMENT SHOW THAT THE TYPE OF SWELLING BEHAVIOR OBSERVED IN SU 9 IS NOT INCONSISTENT WITH EARLIER RESULTS OBTAINED ON THE NAA 47 SERIES OR IN OTHER U-10 WT% MO IRRADIATION PROGRAMS, BUT THAT THE SEVERE SWELLING OBSERVED IN THESE PROGRAMS DOES NOT APPEAR TO OCCUR AT BURNUPS BELOW ABOUT 4000 MWD/MTU.

ALLOY + CONTAINMENT, PRESSURE VENTING + FUEL INTEGRITY + HALLAM (LMR) + REACTOR, LMCR + MOLYBDENUM + URANIUM

5-006684 DATED 1965
GLASGOW LE
SODIUM REACTOR EXPERIMENT
ATOMICS INTERNATIONAL
22 PAGES, 2 FIGURES- APRIL 15, 1965, NAA-SR-11200, PAGES I 1-I 22 OF QUARTERLY TECHNICAL PROGRESS REPORT AEC UNCLASSIFIED PROGRAMS JANUARY-MARCH 1965, CFSTI

THE PENDING REVIEW BY DRL (AEC, WASHINGTON) MAY RESULT IN ADDITIONAL SAFETY ANALYSIS EFFORT- HOWEVER, THE ORIGINAL WORK SCOPE HAS BEEN COMPLETED. ACCUMULATION OF REFERENCE BACKUP MATERIAL RELATED TO SRE-PEP HAS BEEN CONTINUED. A PRELIMINARY SEISMIC ANALYSIS HAS BEEN MADE OF SRE STRUCTURES- NO CONDITION HAS BEEN FOUND THAT WOULD INCREASE OR CHANGE THE PREVIOUSLY ANALYZED MAXIMUM CREDIBLE ACCIDENT.

REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + SEISMOLOGY + SRE (RE)

5-006817 ALSO IN CATEGORIES 11 AND 12 DATED 1965
BOULANGER JR
SNAP-10A-PIPING STRESS ANALYSIS
ATOMICS INTERNATIONAL
30 PAGES- JANUARY 22, 1965, NAA-SR-MEMO-9817- CFSTI

THIS REPORT CONTAINS THE STRESS ANALYSIS OF THE SNAP 10A PIPING SYSTEM. THIS SYSTEM IS MADE UP OF VARIOUS SIZES AND SHAPES OF STAINLESS STEEL TUBING WELDED TOGETHER INTO A NAK CONTAINMENT NETWORK. STAINLESS STEEL IS USED BECAUSE OF ITS STABILITY AT THE OPERATING TEMPERATURES AND ITS COMPATIBILITY WITH THE REMAINDER OF THE SYSTEM. BACKGROUND INFORMATION REGARDING THE CONFIGURATION AND MATERIALS USED, THE LOADS AND THEIR SOURCES AND THE ANALYTICAL VERIFICATIONS ARE PROVIDED. THIS ANALYSIS VERIFIES THE STRUCTURAL INTEGRITY OF THE SYSTEM DURING BOTH THE SNAP 10A THERMAL AND VIBRATION ENVIRONMENTS.

INTEGRITY + REACTOR, LMCR + SNAP 10A (SR) + REACTOR, SPACE + STEEL, STAINLESS + STRESS ANALYSIS + MAIN COOLING SYSTEM + PIPES AND PIPE FITTINGS

5-007188 ALSO IN CATEGORY 18 DATED 1965
SAFETY ASPECTS OF THE EXPERIMENTAL POWER STATION KNK
3 PAGES, 2 FIGURES- NUCLEAR ENGINEERING 10(112)- 335-337 (SEPTEMBER 1965)

THE PROJECT FOR A 20 MEGAWATT ELECTRICAL EXPERIMENTAL POWER STATION WITH A SODIUM-COOLED, ZIRCONIUM HYDRIDE-MODERATED REACTOR (TYPE KNK), IS THE RESULT OF EXTENSIVE DEVELOPMENT WORK UNDERTAKEN BY INTERATOM SINCE THE SPRING OF 1960. WORK ON THE PLANNING OF THE PROJECT IS

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-007188 *CONTINUED*

PRACTICALLY COMPLETE. START ON CONSTRUCTION AT THE KARLSRUHE RESEARCH CENTRE IS ESTIMATED FOR THE END OF THIS YEAR. THIS REPORT DEALS WITH THE LAYOUT OF THE PLANT. SPECIAL ATTENTION IS GIVEN TO THE SAFETY ASPECTS.

ACCIDENT ANALYSIS + CONTROL SYSTEM + REACTOR, LMCR + REACTIVITY EFFECT + SAFETY REVIEW + REACTOR SAFETY SYSTEM + SODIUM + GERMANY + TECHNICAL SPECIFICATIONS + ENGINEERED SAFETY FEATURE

5-007556 ALSO IN CATEGORIES 7 AND 12 DATED 1965

BIERY JC + KENNEDY RM

MATHEMATICAL AND EXPERIMENTAL ANALYSIS OF TEMPERATURE DISTRIBUTIONS IN STORED IRRADIATED LAMPRE CAPSULES
LOS ALAMOS SCIENTIFIC LABORATORY

18 PAGES, 14 FIGURES, 12 REFERENCES- CHEMICAL ENGINEERING PROGRESS SYMPOSIUM SERIES 61(57)- 322-339 (1965)-
HEAT TRANSFER-BOSTON, AMERICAN INSTITUTE OF CHEMICAL ENGINEERS

TEMPERATURE PROFILES IN STORED IRRADIATED LAMPRE (LOS ALAMOS MOLTEN PLUTONIUM REACTOR EXPERIMENT) FUEL CAPSULES WERE SIMULATED BOTH MATHEMATICALLY AND EXPERIMENTALLY. THE MATHEMATICAL ANALYSIS PRODUCED SOLUTION EQUATIONS FROM THE FOUR COUPLED SECOND-ORDER DIFFERENTIAL EQUATIONS BY AN ITERATIVE GENERATION PROCEDURE. THE GENERATED EQUATIONS WERE THEN USED IN AN ITERATIVE COMPUTER PROGRAM WHICH PRODUCED THE CONVERGENT TEMPERATURE PROFILES BY A CALCULATION AND CORRECTION TECHNIQUE. THE ITERATIVELY GENERATED EQUATIONS, COUPLED WITH THE COMPUTER PROGRAM, PRODUCED TEMPERATURES WHICH AGREED VERY SATISFACTORILY WITH THE EXPERIMENTAL SIMULATION RESULTS.

REACTOR, BREEDER + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + FUEL STORAGE + HEAT TRANSFER, CONDUCTION + HEAT GENERATION, INTERNAL + FUEL ELEMENTS

5-009456 ALSO IN CATEGORY 18 DATED 1959

SPIEGLER P

SODIUM-24 SPECIFIC ACTIVITY IN HNPf U-MO AND U-C REACTOR SYSTEMS

ATOMICS INTERNATIONAL

NAA-SR-MEMO-3830 +. 17 PAGES, MAY 21, 1959, CFSTI, \$1.00 CY

THE SPECIFIC ACTIVITY OF THE PRIMARY SODIUM IN THE HNPf WAS DETERMINED FOR SIX POSSIBLE CORE CONFIGURATIONS (3 U-MO AND 3 U-C). THE ANALYSIS WAS PERFORMED USING A NEUTRON ECONOMY METHOD. SINCE THIS METHOD OF CALCULATING THE SPECIFIC ACTIVITY IS DIFFERENT FROM THAT PREVIOUSLY USED, IT WAS FIRST APPLIED TO THE SRE, AND THE RESULTANT SODIUM ACTIVITY WAS COMPARED WITH AVAILABLE EXPERIMENTAL DATA. AGREEMENT WITH EXPERIMENT WAS GOOD, AND THE METHOD WAS THEREFORE APPLIED TO THE HNPf CORES. THE RESULTS OF THE CALCULATIONS ARE SUMMARIZED IN TABLE I.

ACTIVATION + ACTIVITY BUILDUP + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + SODIUM + SRE (RE)

5-009520 DATED 1965

GRASS G + KOTTOWSKI H + MILLIOT B

STUDY OF LIQUID METAL BOILING UNDER FORCED CONVECTION

EUROPEAN ATOMIC ENERGY COMMUNITY, ISPRA, ITALY

ANL-TRANS-257 +. 32 PAGES, OCTOBER, 1965, TO BE PUBLISHED IN NUKLEONIK, JCL, \$3.60 FS, \$1.16 MF

THIS WORK REPRESENTS AN INITIAL INVESTIGATION ON SAFETY PROBLEMS IN FAST REACTORS. A FORCED-CONVECTION BOILING-POTASSIUM LOOP WAS USED. A NEW MEASURING METHOD, TAKING ADVANTAGE OF DIRECT JOULE HEATING AND SIMULTANEOUS ANALYSIS OF THE VOLTAGE CHANGES WAS USED TO STUDY FLOW PATTERNS. FOR THE CONDITIONS OF THE TESTS, STEADY-STATE BOILING OCCURRED ONLY RARELY. BOILING WAS ABRUPT, LEADING TO THE SUDDEN PERIODICAL EMPTYING OF THE TEST SECTION, PRODUCING PRESSURE SHOCKS FOLLOWING VAPOR CONDENSATION. A BYPASS WAS INTRODUCED TO SIMULATE PARALLEL-CHANNEL FLOW AND LED TO MARKED DIFFERENCES IN THE BOILING BEHAVIOR. THE BOILING AND CONDENSATION PHENOMENA WERE INTERPRETED WITH THE USE OF THE NEW MEASURING TECHNIQUE. FURTHER STUDIES ARE REQUIRED TO ESTABLISH MECHANISMS AND CORRELATION.

HEAT TRANSFER, BOILING + REACTOR, FAST + REACTOR, LMCR + OUT OF PILE EXPERIMENT + POTASSIUM + METAL, LIQUID

5-010430 ALSO IN CATEGORY 6 DATED 1965

ARGONNE NATIONAL LABORATORY PUBLICATIONS, JULY 1, 1964 - JUNE 30, 1965

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ANL-7087 +. 328 PAGES, AUGUST 1965 CFSTI, \$7.00 CY, \$1.50 MN

THIS PUBLICATION LIST IS A BIBLIOGRAPHY OF SCIENTIFIC AND TECHNICAL ACCOUNTS ORIGINATED AT ARGONNE AND PUBLISHED DURING FISCAL YEAR 1965. IT INCLUDES ITEMS PUBLISHED AS JOURNAL ARTICLES, TECHNICAL REPORTS, BOOKS, ETC., ALL OF WHICH HAVE BEEN MADE AVAILABLE TO THE PUBLIC. THE LIST IS DIVIDED INTO THREE PARTS - THE FIRST A LIST DIVISION, THE SECOND AN ALPHABETICAL AUTHOR INDEX, AND THE THIRD PART A KWIC SUBJECT INDEX. INCLUDED ARE SECTIONS ON RADIOLOGICAL PHYSICS, REACTOR PHYSICS, REACTOR ENGINEERING, INDUSTRIAL HYGIENE AND SAFETY, FIRE PROTECTION, AND CHEMICAL ENGINEERING.

ACCIDENT ANALYSIS + BIBLIOGRAPHY + REACTOR, BREEDER + BURNOUT HEAT FLUX + CHEMICAL REACTION + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + METAL WATER REACTION + NOISE + REACTIVITY COEFFICIENT + REACTOR DYNAMICS + REACTOR KINETICS + REACTOR STABILITY + REACTOR TRANSIENT + SPACE DEPENDENT DYNAMICS + VOID COEFFICIENT + FFTF (TR) + DOSE MEASUREMENT, EXTERNAL

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-011040 ALSO IN CATEGORY 17 DATED 1966
FERMI POWER COEFFICIENT ANALYSIS
POWER REACTOR DEVELOPMENT COMPANY
2 PAGES, JULY 22, 1966, ATOMIC ENERGY CLEARING HOUSE, 12(32), PAGES 12-13 (AUGUST 8, 1966)

REVIEW OF DATA SHOWS THAT POWER COEFFICIENT IS LINEAR WITH POWER, AND TRANSFER-FUNCTION MEASUREMENTS SHOW REACTOR STRONGLY STABLE. PRINCIPLE CHANGES WERE - MORE-NEGATIVE CORE SODIUM EXPANSION EFFECT AND LESS-NEGATIVE CORE RADIAL EXPANSION EFFECT. NO CAUSE FOR CONCERN EXISTS, AS PREVIOUS SAFETY ANALYSIS USED POWER COEFFICIENTS AS LOW AS MINUS 0.15 CENT PER MEGAWATT, AND PRESENT COEFFICIENT IS MINUS 0.18.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + POWER COEFFICIENT + COMPARISON, THEORY AND EXPERIENCE

5-011299 DATED 1965
WUHLBERG C
ESTIMATES OF THE POSSIBILITY OF HYDROGEN EMBRITTLEMENT OF TANTALUM IN THE FRCTF
LOS ALAMOS SCIENTIFIC LABORATORY, LOS ALAMOS
LA-3508-MS +. 50 PAGES, 11 FIGURES, 12 TABLES, 44 REFERENCES, AUGUST 18, 1965, CFSTI \$3.00 CY

SINCE NAH IS NOT REMOVED BY TRAPPING ON HOT ZIRCONIUM, AN EVALUATION WAS MADE OF THE EXTENT TO WHICH THE CLADDING MIGHT BE HYDRIDED. COLD TRAPPING GIVES A 40-PPM EQUILIBRIUM VALUE IN THE TANTALUM, WELL BELOW THE 800-PPM ROOM-TEMPERATURE EMBRITTLEMENT VALUE. LITTLE IS KNOWN OF EMBRITTLEMENT EFFECTS AT 500 TO 600 C. EXTREME EXTRAPOLATIONS HAD TO BE MADE OF PARTIAL PRESSURES OF HYDROGEN OVER UNSATURATED SODIUM.

CLADDING + HYDROGEN + REACTOR, LMCR + FFTF (TR) + EMBRITTLEMENT + TANTALUM + COOLANT CHEMISTRY

5-011396 ALSO IN CATEGORY 11 DATED 1965
LEMCOE M
STUDY OF INSTANTANEOUS COLLAPSE AND CREEP BUCKLING CHARACTERISTICS OF LARGE SGR PROCESS TUBES
ATOMICS INTERNATIONAL, CANOGA PARK
NAA-SR-11323 +. 47 PAGES, 4 TABLES, REFERENCES, JULY 25, 1965, CFSTI \$3.00 CY, \$0.50 MN

THE INSTANTANEOUS COLLAPSE (LESS THAN 0.2 HR) AND CREEP-BUCKLING CHARACTERISTICS (UP TO 5000 HR) OF TYPE 304 STAINLESS STEEL TUBES (4.00 IN. IN OD, WITH 0.025-IN. WALL), UNDER EXTERNAL RADIAL PRESSURE AND VARYING AXIAL LOAD, WERE DETERMINED AT TEMPERATURES UP TO 1200 F IN AN ARGON ENVIRONMENT. THE LENGTH OF EACH TUBE WAS 6 FT, WHICH IS ESSENTIALLY INFINITE IN TERMS OF BUCKLING. IN ADDITION, TUBES IN COMBINATION WITH FUEL-ELEMENT BUNDLES CORRESPONDING TO THOSE PROPOSED FOR A LARGE SODIUM GRAPHITE REACTOR (LSGR) WERE ALSO INVESTIGATED TO DETERMINE THE EXTENT TO WHICH THE FUEL-ELEMENT BUNDLE INCREASED THE RESISTANCE TO COLLAPSE OF THE PROCESS TUBE. THE RESULTS OF THE STUDY INDICATED THAT STIFFENER RINGS OR THICKER-WALLED TUBES SHOULD NOT BE NECESSARY UNDER NORMAL OPERATING CONDITIONS.

BUCKLING + REACTOR, LMCR + STEEL, STAINLESS + PIPES AND PIPE FITTINGS

5-011673 DATED 1957
KROEGER HK + STEPHENSON JW
THE EFFECT OF SODIUM VAPOR FORMATION ON THE OPERATION OF THE ENRICO FERMI REACTOR. AN ESTIMATE
ASTRA, INC.
AECU-3697 +. 55 PAGES, 18 FIGURES, 4 TABLES, JULY 1957, CFSTI \$9.30 FS, \$3.60 MF

THE RESULTS OF CALCULATIONS CONCERNING TWO-PHASE SODIUM FLOW IN THE FERMI REACTOR ARE PRESENTED. THE LARGE SPECIFIC VOLUME OF SODIUM VAPOR, COUPLED WITH THE RESISTANCE OF THE TOP AXIAL BLANKET SECTION, RESULTED IN LARGE PRESSURE DROPS FOR SMALL VALUES OF NET VAPOR. ONCE THE COOLANT REACHED THE SATURATION TEMPERATURE, THE PRESSURE DROP RESULTING FROM VAPOR FORMATION EXCEEDED THE ADDED PRESSURE DROP FROM FLOW REDUCTION. ALTHOUGH THERE APPEARS TO BE A REGION OF STABLE OPERATION WITH VAPOR FORMATION, THE LOWEST VALUE OF PRESSURE DROP IN THIS REGION IS MUCH GREATER THAN AT THE INITIATING FLOW RATE. IT IS CONCLUDED THAT ANY VAPOR FORMATION IN THE REACTOR PRODUCES AN UNSTABLE CONDITION THAT WOULD LEAD TO BURNOUT.

ACCIDENT ANALYSIS + HEAT TRANSFER, BOILING + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + FLOW BLOCKAGE

5-011674 DATED 1957
ELROD HG
TRANSIENT FUEL AND COOLANT TEMPERATURES IN THE ENRICO FERMI REACTOR CORE DURING A PERIOD OF FLOW REVERSAL
UNITED NUCLEAR CORPORATION
NDA-14-179 +. 35 PAGES, 3 FIGURES, JULY 1957, CFSTI \$6.30 FS, \$3.00 MF

THE TRANSIENT TEMPERATURES OF THE SODIUM COOLANT ARE CALCULATED UP TO ITS BOILING POINT, ASSUMING THAT A LARGE LEAK SUDDENLY DEVELOPED IN THE LOWER PLENUM CHAMBER OF THE REACTOR, WHILE AT THE SAME TIME THE SCRAM SYSTEM FAILED. IT IS ALSO ASSUMED THAT THE COOLANT FLOW HAS STOPPED AT TIME ZERO AND THEN REVERSES UNDER THE INFLUENCE OF GRAVITY. ON THE ASSUMPTION THAT THE LOWER PLENUM PRESSURE IS ATMOSPHERIC, THE ABSOLUTE PRESSURE AT THE CORE EXIT IS 15.2 PSIA, CORRESPONDING TO A BOILING TEMPERATURE OF 1630 F FOR SODIUM. THIS TEMPERATURE IS REACHED IN 1.8 SECONDS. ASYMPTOTIC TEMPERATURE DISTRIBUTIONS ARE VERY NEARLY ATTAINED THROUGHOUT THE CORE.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + THERMAL TRANSIENT

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-011915 DATED 1961
 REED GL + NOYES KC
 SODIUM CONDENSING HEAT TRANSFER. AN EXPERIMENTAL STUDY OF ONE ASPECT OF SODIUM COOLED REACTOR SAFETY
 ATOMICS INTERNATIONAL
 NAA-SK-7325 +. 49 PAGES, 7 TABLES, 18 FIGURES, DECEMBER 15, 1961, CFSTI \$1.25 CY

EXPERIMENTAL DETERMINATIONS OF CONDENSING HEAT-TRANSFER RATES IN A MODEL OF THE REGION BELOW THE LOADING FACE SHIELD OF THE HALLAM REACTOR WERE MADE. THE PRIMARY RESISTANCE TO HEAT TRANSFER WAS FOUND TO BE NONCONDENSABLE GASES. THE GAS (USUALLY HELIUM) WOULD BE PRESENT BECAUSE OF ITS NORMAL USE AS A COVER GAS IN THE CORE TANK. A THEORETICAL EQUATION BASED ON GASEOUS DIFFUSION HAS BEEN DERIVED, AND THE HEAT-TRANSFER RATE IS PREDICTABLE PROVIDED THE QUANTITY OF NONCONDENSABLE GAS IS KNOWN. FURTHER EXPERIMENTS WERE USED TO STUDY THE VENTING BEHAVIOR OF THE GAS SPACE BELOW THE HNPFL LOADING FACE SHIELD DURING SIMULATED ACCIDENT CONDITIONS. THESE EXPERIMENTS PROVIDE INFORMATION REQUIRED TO PREDICT THE QUANTITY OF NONCONDENSABLE GAS LEFT IN THE SYSTEM AFTER A POSTULATED ACCIDENT AND TO PREDICT ASSOCIATED HEAT TRANSFER RATES. TWO VENTING METHODS WERE STUDIED TO DETERMINE THE MOST EFFICIENT EMERGENCY VENTING PROCEDURE. IT WAS FOUND THAT IF THE REGION IMMEDIATELY BELOW THE TOP SHIELD IS VENTED, SUFFICIENT HEAT REMOVAL CAPABILITY EXISTS TO CONDENSE ALL SODIUM VAPOR THAT COULD BE GENERATED BY AFTERGLOW HEAT. THIS WOULD ASSURE SAFETY OF THE SYSTEM BY ELIMINATING THE POSSIBILITY OF ANY PRESSURE BUILDUP AS A RESULT OF THE POSTULATED ACCIDENT.

REACTOR, GRAPHITE MODERATED + HALLAM (LMK) + HEAT TRANSFER + REACTOR, LMCR + SODIUM + CONDENSATION + METAL, LIQUID + HEAT TRANSFER EXPERIMENT + SAFETY PROGRAM

5-014676 ALSO IN CATEGORY 17 DATED 1966
 NUCLEATE BOILING SHOWN TO HAVE SEVERAL REGIMES - LITERATURE SURVEY
 LOS ALAMOS SCIENTIFIC LABORATORY
 LA-3625-MS +. 1 PAGE, QUARTERLY STATUS REPORT ON ADVANCED REACTOR TECHNOLOGY (ART) FOR PERIOD ENDING OCTOBER 31, 1966, PAGE 13, NOVEMBER 1966

A FOUR-MONTH SURVEY OF THE LITERATURE ON BOILING HEAT TRANSFER AND TWO-PHASE FLOW WAS COMPLETED. THE MORE RECENT PUBLICATIONS REVEALED THAT RESEARCHERS HAVE TURNED TO A STUDY OF THE BASIC MECHANISM OF BOILING. THESE STUDIES HAVE SHOWN THAT THE CONVENTIONAL CONCEPT OF NUCLEATE BOILING UPON WHICH ALL PRIOR CORRELATIONS WERE BASED IS NOT CORRECT. THREE AND POSSIBLY FOUR REGIMES OF NUCLEATE BOILING HAVE BEEN DEMONSTRATED, WITH ONLY THE LOW-HEAT-FLUX RANGE CORRESPONDING TO THE IDEAS HELD PREVIOUSLY ABOUT NUCLEATE BOILING. THE NEW VIEWS AND THEIR CONSEQUENCES HAVE BEEN EXPLORED - A SUMMARY OF THE LITERATURE AND SPECIFIC RECOMMENDATIONS FOR FUTURE WORK HAVE BEEN PREPARED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICRONEGATIVE

HEAT TRANSFER, BOILING + REACTOR, LMCR + SODIUM + HEAT TRANSFER EXPERIMENT + NUCLEATE BOILING

5-014917 ALSO IN CATEGORIES 6 AND 17 DATED 1967
 DICKERMAN CE + ROBINSON LE + BLUMENTHAL B + STEWART R
 BEHAVIOR OF TH-20 WT/PERCENT U FAST-REACTOR FUEL UNDER TRANSIENT HEATING TO FAILURE
 ARGONNE NATIONAL LABORATORY
 10 PAGES, 10 FIGURES, 2 TABLES, 17 REFERENCES, NUCLEAR APPLICATIONS 3(1) PAGES 9-17 (JANUARY 1967)

SODIUM-BONDED TH-20 WT % U FBR II FUEL ELEMENTS WERE STUDIED IN TRANSIENT MELTDOWN EXPERIMENTS RANGING UP TO COMPLETE SAMPLE MELTDOWN, CONDUCTED IN TREAT. THE HIGH-MELTING THORIUM MATRIX SUPPORTED THE FUEL DURING TRANSIENT HEATING AT TEMPERATURES SIGNIFICANTLY BEYOND THE SOLIDUS. BECAUSE THE FUEL WAS STILL RELATIVELY INTACT AT THE TEMPERATURES OF CLADDING FAILURE, INTERNAL PRESSURE WAS NOT EFFECTIVE IN EJECTING FUEL UPON FAILURE. THE DOMINANT MECHANISM OF FUEL MOVEMENT WAS GRAVITY.

REACTOR, FAST + FUEL MELTDOWN + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + TREAT (PRR) + THERMAL TRANSIENT

5-016164 ALSO IN CATEGORIES 6 AND 9 DATED 1966
 MACFARLANE DR
 AN ANALYTIC STUDY OF THE TRANSIENT BOILING OF SODIUM IN REACTOR COOLANT CHANNELS
 ARGONNE NATIONAL LABORATORY, ILLINOIS
 ANL-7222 +. 117 PAGES, 48 FIGURES, 11 TABLES, JUNE 1966

TWO TYPES OF TRANSIENTS WERE STUDIED - (1) MILD REACTIVITY ADDITION AND (2) PUMP SHUTOFF WITH FLOW COASTDOWN. NUMERICAL SOLUTIONS OBTAINED FOR THE NONLINEAR FORM OF THE COUPLED FLUID DYNAMIC AND HEAT CONDUCTION PARTIAL DIFFERENTIAL EQUATION BY MEANS OF IMPLICIT FINITE-DIFFERENCE EQUATIONS AND AN ITERATIVE ALGOL COMPUTER PROGRAM. SONIC EFFECTS AND CRITICAL FLOW PHENOMENA WERE NOT TREATED. TWO-PHASE COOLANT DENSITY IS PRESSURE DEPENDENT, AND TEMPORAL- AND SPATIAL-VARYING SLIP VELOCITY ARE INCLUDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTIVITY, EXCESS + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + REACTOR TRANSIENT + SODIUM + VOID COEFFICIENT + METAL, LIQUID + FLOW, TWO PHASE

5-017167 ALSO IN CATEGORY 17 DATED 1967

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-017167 *CONTINUED*

FRIEDLAND AJ

A STUDY OF THE PROBABILITY OF A FERMI SUBASSEMBLY HAVING OVERHEATED BECAUSE OF A MIGRATORY PLUG
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN
CONF-670-713 +. 17 PAGES, 8 FIGURES, 1 TABLE, 5 REFERENCES, ABSTRACT IN ANS TRANSACTIONS, SUPPLEMENT TO
VOLUME 10, PAGE 33, (JULY 1967), PAPER PRESENTED AT THE CONFERENCE ON REACTOR OPERATING EXPERIENCE, REACTOR
OPERATIONS DIVISION OF THE AMERICAN NUCLEAR SOCIETY, ATLANTIC CITY, JULY 23-26, 1967

REACTOR OPERATING HISTORY WAS USED TO ESTIMATE THE PROBABILITY DISTRIBUTION OF PLUG SIZE, AND
THESE TO DETERMINE PROBABILITY OF OVERHEATING. THIS APPROACH COULD BE GENERALLY APPLIED. SMALL-
DIAMETER PLUGS ARE UNLIKELY TO CAUSE MELTDOWN, WHILE LARGE PLUGS WOULD NOT HAVE GONE THROUGH 11
OPERATING PERIODS WITHOUT CAUSING A MELTDOWN. THE MOST PROBABLE PLUG DIAMETER IS 1.5 TIMES THE
GRID-PLATE-HOLE DIAMETER AND GIVES A 0.045 PROBABILITY OF OVERHEATING. THE OVERALL PROBABILITY
OF OVERHEATING WAS CALCULATED AS 15%.

REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + ANALYTICAL MODEL + ACCIDENT, PROBABILITY OF +
COOLANT CHEMISTRY + FLOW BLOCKAGE + MATHEMATICAL TREATMENT + FUEL COOLANT INTERACTION

5-017339

DATED 1963

RUMPF NK

PRESSURE BUILDUP IN THE CELL AND PRIMARY VAULT OF THE FAKET FROM SODIUM FIRES OR SPILLS
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA
NAA-SR-MEMO-8324 +. 14 PAGES, MARCH 18, 1963

TWO DIFFERENT TYPES OF FIRES ARE ANALYZED WHICH MIGHT RESULT FROM THE EXPLOSION OF A SUPERCRITICAL
MASS IN THE CORE WITH THE EQUIVALENT FORCE OF 300 LB TNT. (1) IF A FINE SODIUM SPRAY IS EXPELLED
INTO AN ATMOSPHERE OF AIR, THEN PRESSURE OF 48 PSIG POSSIBLE. THIS ACCIDENT IS CONSIDERED
INCREDIBLE. (2) IF THE TOP SHIELD PLUG COULD BECOME DISLODGED, THE PRESSURE AFTER 3 HOURS IS
4.22 PSIG AND THE MAXIMUM POSSIBLE PRESSURE IS 10 PSIG, WELL WITHIN THE 30-PSIG DESIGN PRESSURE.
THE PRIMARY VAULT, WHICH HAS THE LARGEST FLOOR AREA TO OVERALL SURFACE-AREA RATIO, IS ANALYZED
FOR A SPILL. THE INITIAL PRESSURE INCREASE FOR A SPILL OF 1000 LB OR MORE IS SUCH THAT GAS MUST
BE REMOVED AT 1608 CFM. A RELIEF VALVE MUST BE IN A 4-INCH LINE TO KEEP THE PRESSURE IN THE
VAULT AT LESS THAN 2 PSIG, WHICH IS WITHIN THE DESIGN LIMITS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FIRE + REACTOR, LMCR + SODIUM + FFTF (TR) + PRESSURE, INTERNAL

5-017576

ALSO IN CATEGORIES 6 AND 8

DATED 1967

SAVING JM + SIEGEL R

EXPERIMENTAL AND ANALYTICAL STUDY OF THE TRANSIENT SOLIDIFICATION OF A WARM LIQUID FLOWING OVER A CHILLED FLAT
PLATE

LEWIS RESEARCH CENTER, CLEVELAND, OHIO. NASA

NASA-TN-U-4015 +. 63 PAGES, 12 FIGURES, 26 REFERENCES, JUNE 1967

ANALYTICAL PREDICTIONS AND EXPERIMENTAL DATA ARE PROVIDED FOR THE TRANSIENT GROWTH OF A FROZEN
LAYER THAT FORMS WHEN A WARM LIQUID FLOWS OVER ONE SIDE OF A CHILLED FLAT PLATE. THE PLATE IS
COOLED TO A TEMPERATURE BELOW THE FREEZING POINT OF THE LIQUID BY A COOLANT FLOWING ALONG THE
OTHER SIDE OF THE PLATE. THIS COULD BE ENCOUNTERED, FOR EXAMPLE, IN A HEAT EXCHANGER WHERE WATER
IS BEING COOLED BY A FLUID AT LOW TEMPERATURE. THREE ANALYTICAL PROCEDURES WERE USED AND
COMPARED FOR ACCURACY AND CONVENIENCE IN APPLICATION. ONE IS AN ITERATIVE TECHNIQUE THAT YIELDED
THREE CLOSED-FORM SOLUTIONS, EACH SUCCESSIVE SOLUTION BEING A MORE ACCURATE HIGHER ORDER
APPROXIMATION. NUMERICAL RESULTS FROM THE THIRD APPROXIMATION ARE PRESENTED GRAPHICALLY IN
SUFFICIENT DETAIL SO THAT RAPID ESTIMATES OF FROZEN LAYER GROWTH CAN BE MADE. IN THE EXPERIMENT,
WATER FLOWED OVER ONE SIDE OF A THIN INCONEL PLATE WHILE THE OTHER SIDE OF THE PLATE WAS COOLED
BY FLOWING CHILLED ALCOHOL. TRANSIENT ICE GROWTH WAS INITIATED BY A SUDDEN INCREASE OF THE
ALCOHOL FLOW. THE ICE THICKNESS AND THE PLATE SURFACE TEMPERATURES WERE MEASURED THROUGHOUT THE
TRANSIENT GROWTH PERIOD AND COMPARED FAVORABLE WITH THEORETICALLY PREDICTED VALUES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REACTOR, MOLTEN SALT + FILTER, LIQUID + HEAT TRANSFER, NATURAL CONVECTION + HEAT TRANSFER
ANALYSIS + CRYOGENICS + HEAT EXCHANGERS

5-017577

ALSO IN CATEGORY 6

DATED 1967

FENN DB + ACKER LW + COE HH

STEADY-STATE PERFORMANCE OF A SEVEN-TUBE NAK-COOLED POTASSIUM CONDENSER

LEWIS RESEARCH CENTER, CLEVELAND, OHIO. NASA

NASA-TN-D-4020 +. 39 PAGES, 18 FIGURES, 2 TABLES, 11 REFERENCE, JUNE 1967

POTASSIUM VAPOR WAS CONDENSED INSIDE THE TUBES OF A HORIZONTAL, MULTITUBE HEAT EXCHANGER. THE
TUBES WERE COOLED WITH THE LIQUID SODIUM-POTASSIUM (NAK) FLOWING IN A COUNTER-CURRENT DIRECTION
IN THE SPACE SURROUNDING THE TUBES. STEADY-STATE DATA ARE PRESENTED TO SHOW THE EFFECTS OF NAK
COOLANT INLET TEMPERATURE AND COOLANT FLOW RATE ON THE CONDENSING LENGTH, THE COOLANT TEMPERATURE
PROFILES, AND THE OVERALL PRESSURE DROP OF THE CONDENSER OVER A RANGE OF POTASSIUM INLET
TEMPERATURES FROM 1200 TO 1500 F. A CORRELATION OF THE OVERALL PRESSURE DROP ON THE POTASSIUM
SIDE OF THE CONDENSER WAS OBTAINED BY PLOTTING THE RATIO OF THE MEASURED OVERALL PRESSURE DROP TO
THE VELOCITY HEAT IN THE TUBE INLET AS A FUNCTION OF THE CONDENSING-LENGTH-TO-DIAMETER RATIO.

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-017577 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

HEAT TRANSFER, BUILDING + REACTOR, LMCR + CONDENSATION + METAL, ALKALI + METAL, LIQUID + HEAT TRANSFER, CONDENSATION

5-016205 ALSO IN CATEGORIES 11 AND 12 DATED 1967
HARTY H + BATCH JM + PEDERSEN LT + WHEELER RG + FOX JC
ENGINEERING DEVELOPMENT DEPARTMENT QUARTERLY REPORT, APRIL-JUNE 1966
BATTTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-367 +. 76 PAGES, 18 FIGURES, 2 TABLES, APRIL 1967

PROGRESS REPORT COVERING SUCH TOPICS AS THE FAST FLUX TEST FACILITY, HEAT TRANSFER STUDIES, CROSS-
FLOW MIXING BETWEEN PARALLEL FLOW, BUILDING-STABILITY STUDIES, IRRADIATION DAMAGE TO REACTOR
METALS, UPSTREAM BUILDING BURNOUT, THE CSE, IODINE FORMS IN STEAM CONDENSATE, AND DECONTAMINATION
OF COATINGS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

HEAT TRANSFER, BUILDING + BURNOUT HEAT FLUX + IODINE + REACTOR, LMCR + PLUTONIUM + STEAM + AEROSOL PROPERTIES + CONTAINMENT R AND D + CSE + SAMPLING + HEAT TRANSFER EXPERIMENT

5-018857 ALSO IN CATEGORY 17 DATED 1967
PHILLIPS JL
DOONREAY FAST REACTOR MODIFICATIONS
UNITED KINGDOM ATOMIC ENERGY AUTHORITY, DOONREAY
4 PAGES, 6 FIGURES, NUCLEAR ENGINEERING 12(128) PAGES 40-43 (OCTOBER 18, 1967)

MODIFICATIONS WERE NECESSARY BECAUSE OF COOLANT-FLOW INSTABILITIES IN THE OUTER BREEDER ELEMENTS
AS A RESULT OF OPPOSING CONVECTIVE AND BUOYANCY-INDUCED FLOWS. THERMAL EFFECTS CAUSED EXCESSIVE
FUEL SWELLING IN 1/3 OF THE OUTER BREEDER ELEMENTS TO SUCH AN EXTENT THAT MAJOR MODIFICATIONS
WERE NECESSARY TO DISCHARGE MOST OF THEM. THE MODIFICATIONS, TOOLING, AND METHODS ARE DESCRIBED.

REACTOR, BREEDER + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR, LMCR + MAINTENANCE AND REPAIR + UNITED
KINGDOM + MODIFICATION + FAILURE, DESIGN ERROR + DOONREAY (LMFBR) + FLOW STABILITY + MAINTENANCE, REMOTE

5-019860 ALSO IN CATEGORY 17 DATED 1961
FILLMORE FL
SRE FUEL ELEMENT DAMAGE
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-4488 (SUPPL) +. 49 PAGES, 21 FIGURES, TABLES, 11 REFERENCES, JUNE 30, 1961

THE CAUSE OF THE FUEL-ELEMENT FAILURES WAS ATTRIBUTED TO FORMATION OF LOW-MELTING ALLOY WITH THE
CLADDING AND TO THERMAL CYCLING AND EXPANSION OF THE FUEL SLUGS UNTIL THE CLADDING BURST.
REACTIVITY CHANGES BEFORE AND DURING THE POWER EXCURSION WERE EXPLAINED. A NEGATIVE 0.06%
REACTIVITY EXCURSION AT 1807 HR WAS ATTRIBUTED TO COLLAPSE OF VAPOR VOIDS WHICH HAD FORMED IN
PARTIALLY PLUGGED CHANNELS. AT 1821 HR, A POSITIVE RISE OF 0.04% IN A 3-MIN RAMP WAS ATTRIBUTED
TO INCREASED MODERATOR TEMPERATURE OR VOID FORMATION. AT 1825 HR, THE POSITIVE EXCURSION OF 0.3%
REACTIVITY IN A 5- TO 10-SEC RAMP WAS ATTRIBUTED TO VOID FORMATION IN 10 FUEL CHANNELS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + REACTOR TRANSIENT + VOID COEFFICIENT + SRE (RE) + INCIDENT + TEMPERATURE REACTIVITY EFFECT + FAILURE, CLADDING + FLOW BLOCKAGE + REACTIVITY EFFECT, ANOMALOUS

5-020733 ALSO IN CATEGORY 17 DATED 1966
ANDERSON WF + ANDERSON WJ + HEISLER MP + MCKISSON RL + ZWENG DJ
SCTI SODIUM HEATER FAILURE
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-MEMU-12090 +. 57 PAGES, 18 FIGURES, 2 TABLES, 12 REFERENCES, AUGUST 1, 1966

DURING CHECKOUT, A CIRCUMFERENTIAL TUBE CRACK RELEASED 50 GAL OF SODIUM. ADJACENT TUBES WERE
DISTORTED BY REACTION AND ALSO BY KADCAST. CAUSE WAS A LACK OF MOBILITY DURING THERMAL EXPANSION
BY IMPROPER KADCAST INSULATION, WHICH HAD BEEN POURED AS A MONOLITHIC SLAB INSTEAD OF AS A SERIES
OF BLOCKS. CHLORIDE STRESS CORROSION EVIDENCE DISCUSSED. THE HEATER CAN BE REPAIRED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + THERMAL INSULATION + FAILURE, INSTALLATION ERROR + FAILURE, TUBING + HEAT EXCHANGERS

5-022456 ALSO IN CATEGORY 17 DATED 1967
EBR II MEASUREMENTS OF PROMPT POWER COEFFICIENT FOILED

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-022456 *CONTINUED*
ARGONNE NATIONAL LAB., ARGONNE, ILL.
ANL-7349 +. 7 PAGES, 5 FIGURES, 2 TABLES, PAGE 48-54 OF ARGONNE NATIONAL LABORATORY. REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, NOVEMBER 1967

TRANSFER FUNCTIONS WERE MEASURED. THE WORTH OF THE ROD USED IN ROD-DROP EXPERIMENTS CHANGES ABOVE THE 0.5-MW CALIBRATION LEVEL. DURING THE DROPPING OF THE ROD, IT SEEMS THAT ROD WORTH INCREASES ENOUGH TO CANCEL THE EFFECT OF PROMPT NEGATIVE FEEDBACK FOR 0.3 SEC. TOTAL FEEDBACK IS ABOUT 0.33 CENT. REACTIVITY DIFFERENCES BETWEEN FULL-POWER FLOW AND REDUCED-POWER FLOW CONDITIONS SEEM TO BE STEADILY INCREASING WITH FUEL BURNUP.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

CONTROL ROD WORTH + EBR 1 AND 2 (RE) + FUEL BURNUP + REACTOR, LMCR + POWER COEFFICIENT + TRANSFER FUNCTION + RESPONSE TIME + CONTROL ROD CALIBRATION + REPORT, OPERATIONS ANALYSIS + TEST, PHYSICS

5-026121 DATED 1966
WILLIAMS RF + POLONIK EE
PRESSURE DROP AND DYE TESTS WITH WATER ON THE SEFOR PROTOTYPE FUEL BUNDLE
GENERAL ELECTRIC COMPANY, SAN JOSE, CALIF.
GEAP-5129 +. 53 PAGES, FEBRUARY 1966

COVERS THE MEASUREMENT OF PRESSURE-LOSS CHARACTERISTICS AND OBSERVATIONAL FLOW STUDIES WITH DYE, ON A PROTOTYPE FUEL BUNDLE FOR THE SEFOR REACTOR. WATER WAS USED AS THE COOLANT IN THESE EXPERIMENTS TO SIMULATE THE SODIUM COOLANT IN SEFOR BY REGULATING WATER TEMPERATURES TO MATCH THE REYNOLDS NUMBER OF THE SODIUM. THIS METHOD IS WIDELY USED TO OBTAIN INFORMATION ON FLOW CONDITIONS IN SODIUM SYSTEMS. SEFOR (SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR) IS A 20 MW(T) FAST SPECTRUM REACTOR. SEFOR WILL HAVE CHARACTERISTICS SIMILAR TO THE LARGE, SOFT-SPECTRUM, FAST BREEDER REACTORS FUELED WITH MIXED PUO₂-UO₂. ECONOMIC STUDIES OF THESE LARGE POWER REACTORS INDICATE THE POTENTIAL FOR PRODUCING LOW COST POWER. SEFOR WILL BE USED TO OBTAIN PHYSICS AND ENGINEERING DATA AT FUEL COMPOSITIONS, TEMPERATURES, AND CRYSTALLINE STATES CHARACTERISTIC OF POWER REACTOR OPERATING CONDITIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + SODIUM + METAL, LIQUID + REACTOR, PULSED + SEFOR (RE) + FLOW, TURBULENT + FUEL ELEMENT CLUSTER

5-026286 DATED 1967
KOSKY PG
SOME ASPECTS OF BOILING AND VAPOUR VOIDAGE GROWTH PROBLEMS IN A LIQUID-METAL-COOLED REACTOR
UKAEA, BERKS HARWELL
24 PAGES, 14 FIGURES, 21 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST NUCLEAR REACTORS, SEPTEMBER 1967

PRESENTS SOME THEORETICAL STUDIES OF THE INITIATION OF BOILING OF A LIQUID METAL IN FORCED CONVECTIVE FLOW WHICH SUGGEST THAT THE BURN-OUT HEAT FLUX MAY BE EXCEEDED WHEN THE FIRST VAPOUR VOID IS FORMED. THE HYPOTHESIS IS DISCUSSED IN THE LIGHT OF PUBLISHED EXPERIMENTAL DATA. IT LEADS DIRECTLY TO AN INTEREST IN THE GROWTH RATE AND CONFIGURATION OF VAPOUR VOIDS. PUBLISHED ASYMPTOTIC-GROWTH-RATE EQUATIONS CANNOT BE DIRECTLY APPLIED TO THIS PARTICULAR VOID-GROWTH PROBLEM BECAUSE OF THE ENHANCED RATE OF HEAT TRANSFER TO THE EVAPORATION INTERFACE DUE TO THE VERY HIGH THERMAL CONDUCTIVITY OF LIQUID METAL. A SIMPLE MODEL IS THEREFORE PROPOSED TO TAKE ACCOUNT OF THIS EFFECT AND CALCULATIONS BASED ON THIS MODEL ARE GIVEN FOR SODIUM UNDER VARIOUS CONDITIONS.

AVAILABILITY - P.G. KOSKY, CHEMICAL ENGINEERING AND PROCESS TECHNOLOGY DIVISION, A.E.R.E. HARWELL, ENGLAND

HEAT TRANSFER, BOILING + BURNOUT HEAT FLUX + REACTOR, FAST + REACTOR, LMCR + CONVECTIVE BOILING + METAL, LIQUID + HEAT TRANSFER, CONVECTION

5-026427 DATED 1967
FRIEDLAND, AJ
RELATION BETWEEN MAXIMUM CORE COOLANT TEMPERATURE AND OUTLET THERMOCOUPLE READINGS FOR PLUGGED SUBASSEMBLIES
ATOMIC POWER DEVELOPMENT ASSOCIATES, DETROIT, MICHIGAN
35 PAGES, 9 FIGURES, 3 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST NUCLEAR REACTORS, SEPTEMBER, 1967

COMPUTER CALCULATIONS WERE RUN FOR THE TEMPERATURE DISTRIBUTIONS IN FERMI SUBASSEMBLIES WITH VARIOUS DEGREES OF PLUGGING, FOR POWER LEVELS BELOW THOSE AT WHICH SODIUM BOILING OCCURS. THE MAXIMUM COOLANT TEMPERATURE OCCURRED AT THE CJRE MIDPLANE FOR COMPLETE FLOW BLOCKAGE, AND SHIFTED TO THE TOP OF THE CORE FOR 4% OR MORE OF NOMINAL TWO-LOOP FLOW THROUGH THE SUBASSEMBLY. THE AT TO THE THERMOCOUPLE HOLE INLET WAS CALCULATED TO REACH A MAXIMUM OF ABOUT 10 TIMES THE MIXED AT OF A NORMAL SUBASSEMBLY, AT ABOUT 4-5% OF NOMINAL FLOW.

AVAILABILITY - AJ FRIEDLAND, ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN

ACCIDENT ANALYSIS + FERMI (LMFBR) + ACCIDENT, LOSS OF FLOW + TEMPERATURE GRADIENT + INSTRUMENT, TEMPERATURE + REACTOR STARTUP EXPERIENCE + HEAT TRANSFER ANALYSIS + REACTOR, LMFBR

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-026461 DATED 1966
PULUMIK EE + WILLIAMS RF
COOLANT SURFACE TURBULENCE STUDY - SEFOR REACTOR VESSEL
GENERAL ELECTRIC COMPANY, SAN JOSE
GEAP-5099 +. 53 PAGES, 1966

HALF-SCALE TEST EQUIPMENT WAS BUILT TO SIMULATE THE UPPER PLENUM OF THE SEFOR SODIUM-COOLED FAST REACTOR FOR THE PURPOSE OF EVALUATING SURFACE CONDITIONS AND GAS ENTRAINMENT TO THE COOLANT. WATER WAS USED AS THE FLOW MEDIUM. TESTS WERE CONDUCTED OVER THE FLOW RANGE OF 7 TO 210% OF REACTOR FLOW. BOTH UNIFORM AND COSINE VELOCITY PROFILES IN THE FLOW TO THE UPPER PLENUM WERE STUDIED WITH A FREE SURFACE CONDITION. THE COSINE VELOCITY DISTRIBUTIONS WERE ALSO STUDIED WITH THREE TYPES OF RADIATION SHIELDS IMMERSED AT VARIOUS DEPTHS IN THE UPPER PLENUM WATER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT + REACTOR, FAST + FLOW THEORY AND EXPERIMENTS + REACTOR, LMCR + SODIUM + REACTOR, PULSED + SEFOR (RE) + FLOW, TURBULENT + SURFACE

5-026034 DATED 1967
HUCHKREITER LE + SESOVSKA A
TURBULENT TEMPERATURE FLUCTUATIONS IN FLOWING SODIUM-POTASSIUM
PURDUE UNIVERSITY
2 PAGES, 2 FIGURES, 5 REFERENCES, ANS TRANSACTIONS 10(1), PAGES 353-354 OF THE 1967 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, JUNE 11-15, 1967

AS PART OF A STUDY OF THE MECHANISM OF LIQUID-METAL HEAT TRANSFER APPLICABLE TO SODIUM-COOLED REACTORS, MEASUREMENTS WERE MADE OF TURBULENT TEMPERATURE-FLUCTUATION CHARACTERISTICS IN NAK-56. VERTICAL TRAVERSES WERE MADE WITH AN ESPECIALLY DESIGNED FAST-RESPONSE THERMOCOUPLE ACROSS A 1.0-IN.-I.D. HORIZONTAL TEST SECTION WITH THE FLUID AT ABOUT 200 F HEATED BY A CONSTANT WALL FLUX (12,000 BTU/HR-FT.²). THE TEST SECTION WAS PRECEDED BY A 40-DIAM NONHEATED SECTION FOLLOWED BY A 50-DIAM THERMAL ENTRANCE SECTION.

HEAT TRANSFER + REACTOR, LMCR + NAK + HEAT TRANSFER EXPERIMENT + FLOW, TURBULENT + THERMAL TRANSIENT

5-026035 DATED 1967
MACFARLANE DR
THEORETICAL STUDIES OF THE TRANSIENT BOILING OF SODIUM IN FAST-REACTOR COOLANT CHANNELS
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
1 PAGE, FIGURE, 2 REFERENCES, ANS TRANSACTIONS 10(1), PAGE 354 OF THE 1967 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, JUNE 11-15, 1967

THE WORK DESCRIBED IS PART OF A CONTINUING PROGRAM TO OBTAIN A BETTER UNDERSTANDING OF SODIUM COOLANT VAPORIZATION EFFECTS IN LARGE CERAMIC-FUELED FAST-POWER BREEDER REACTORS. THE APPROACH TAKEN IN THE ANALYSIS WAS TO SEEK NUMERICAL SOLUTIONS ON THE COMPUTER FOR THE COUPLED, NONLINEAR, PARTIAL DIFFERENTIAL EQUATIONS OF FLUID DYNAMICS AND HEAT CONDUCTION WRITTEN FOR A SINGLE REACTOR COOLANT CHANNEL AND FUEL ROD.

CONVECTIVE BOILING + SODIUM + METAL, LIQUID + HOT CHANNEL + REACTOR, LMCR + THERMAL TRANSIENT

5-027043 DATED 1967
YAROSH MM
AN EXPERIMENT EMPLOYING ULTRASONIC ENERGY TO PROMOTE BOILING IN LIQUID METAL
OAK RIDGE NATIONAL LABORATORY, OAK RIDGE, TENNESSEE
30 PAGES, 13 FIGURES, 14 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST NUCLEAR REACTORS, SEPTEMBER, 1967

THE TENDENCY FOR LIQUID METALS SUCH AS NA AND K TO UNDERGO SUBSTANTIAL LIQUID SUPERHEATING AND SUBSEQUENT EXPLOSIVE BOILING HAS BEEN DEMONSTRATED AT OAK RIDGE NATIONAL LABORATORY AND AT OTHER RESEARCH INSTALLATIONS. AN EXPERIMENT WAS CONDUCTED EMPLOYING ULTRASONIC ENERGY EXCITATION TO INITIATE AND MAINTAIN STABLE BOILING IN K UNDER CONDITIONS THAT NORMALLY PRODUCED LIQUID SUPERHEATING AND UNSTABLE, EXPLOSIVE BOILING. UNSTABLE BOILING IS CHARACTERIZED BY LARGE AND ERRATIC TEMPERATURE DIFFERENCES BETWEEN THE TEMPERATURES OF THE LIQUID AND VAPOR. THESE TEMPERATURE DIFFERENCES WERE SIGNIFICANTLY REDUCED OR ELIMINATED UPON EXCITATION OF THE ULTRASONIC PROBE. THE TENDENCY TOWARD LIQUID SUPERHEATING WAS SUBSTANTIALLY ELIMINATED UNDER MANY OPERATING CONDITIONS.

AVAILABILITY - M.M. YAROSH, OAK RIDGE NATIONAL LABORATORY, OAK RIDGE, TENNESSEE

REACTOR, LMCR + SODIUM + POTASSIUM + METAL, LIQUID + BUBBLE NUCLEATION + NUCLEATE BOILING + THERMAL PROPERTY

5-027438 DATED 1968
SEMEKIA R
SODIUM BOILING IN FAST NEUTRON REACTOR CORES
CENTRE D ETUDES NUCLEAIRES DE GRENOBLE, FRANCE
5 PAGES, 16 REFERENCES, BULLETIN D'INFORMATIONS SCIENTIFIQUES ET TECHNIQUES, 122, PAGES 19-23 (JANUARY 1968)

A DESCRIPTION OF THE PHYSICAL PHENOMENA WHICH OCCUR DURING SODIUM BOILING, AND FLASHING OUT OF THE

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-027438 *CONTINUED*

REACTOR CORE, IS ATTEMPTED. THE CHARACTERISTICS OF THE CALCULATION CODES AS WELL AS THE GOALS OF THESE CODES ARE BRIEFLY PRESENTED. THE IMPORTANT RESULTS GAINED FROM THE CURRENT STUDIES ARE OUTLINED.

HEAT TRANSFER, BOILING + COMPUTER PROGRAM + REACTOR, FAST + ACCIDENT, LOSS OF COOLANT + SODIUM + REACTOR, LMFBR

5-027440
FARLANE M

DATED 1968

THEORETICAL STUDIES OF THE RESPONSE OF FAST REACTORS DURING SODIUM BOILING ACCIDENTS
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

18 PAGES, 13 FIGURES, 2 TABLES, 10 REFERENCES, BULLETIN D'INFORMATION SCIENTIFIQUES AND TECHNIQUES, 122, PAGES 35-52 (JANUARY 1968)

THIS PAPER DESCRIBES A CALCULATIONAL METHOD FOR THE TRANSIENT PHENOMENA WHICH CAN TAKE PLACE DURING COOLANT BOILING IN A REACTOR CORE. IT IS THUS POSSIBLE TO STUDY THE WEAK OVERPOWER TRANSIENTS ORIGINATING FROM THE REACTIVITY ADDITION OR THOSE ORIGINATING FROM THE COOLANT FLOW SHUT DOWN. THE MODEL SHOULD ALSO PERMIT THE FORMULATION AND ANALYSIS OF TRANSIENT BOILING EXPERIMENTS WHICH SHOULD SUPPLY STILL MISSING DATA. THE CALCULATIONS APPLY PARTICULARLY TO CERAMICS-FUELED AND SODIUM-COOLED FAST-BREEDER POWER REACTORS.

HEAT TRANSFER, BOILING + REACTOR, FAST + CONVECTIVE BOILING + SODIUM + REACTOR, LMFBR

5-027552

DATED 1967

REYNES L

MECHANICAL AND THERMAL PROBLEMS IN THE STARTUP OF RAPSODIE
7 PAGES, IND. AT., 11(7-8), PAGES 35-41 (1967) (IN FRENCH)

THE USE OF NA AS A COOLANT ALLOWS THE PRODUCTION FUNCTION OF A REACTOR, WHICH UTILIZES THE HIGH TEMPERATURE TO BE UTILIZED. BEYOND ABOUT 500 C THE REQUIRED TEMPERATURE BECOMES THE PRINCIPAL PARAMETER OF THE PROBLEM OF DESIGN AND OPERATION. CERTAIN DIFFICULTIES WHICH COME UNEXPECTEDLY IN THE COURSE OF STARTUP OF RAPSODIE ARE PRESENTED.

FRANCE + HEAT TRANSFER + REACTOR, LMCR + SODIUM + REACTOR STARTUP EXPERIENCE + METAL, LIQUID + RAPSODIE (LMCR)

5-027595

ALSO IN CATEGORIES 9 AND 17

DATED 1968

HARRIGAN PM + MATTEL J + TOTH WL

OPTICAL, LIGHTING, AND PHOTOGRAPHIC EQUIPMENT FOR THE ENRICO FERMI ATOMIC POWER PLANT REPAIR PROGRAM
POWER REACTOR DEVELOPMENT COMPANY + ATOMIC POWER DEVELOPMENT ASSOCIATES

1 PAGE, 1 FIGURE, ANS TRANSACTIONS 11(1), PAGE 358, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

A MELTDOWN OF TWO CORE SUBASSEMBLIES REQUIRED THE USE OF VERY LONG OPTICAL INSTRUMENTS AND LIGHTING EQUIPMENT FOR THE PURPOSE OF INSPECTION OF THE CORE AND REACTOR COMPONENTS. BECAUSE LITTLE INFORMATION WAS AVAILABLE ON THE DESIGN OF SUITABLE VIEWING AND LIGHTING EQUIPMENT FOR USE IN SODIUM REACTOR ENVIRONMENTS, MUCH DEVELOPMENTAL WORK AND TESTING HAD TO BE DONE TO PROVIDE SUITABLE EQUIPMENT. MUCH DEVELOPMENT WAS REQUIRED TO PROVIDE LIGHTING EQUIPMENT WITH ADEQUATE LAMP LIFE AND MANEUVERABILITY. THE USE OF PHOTOGRAPHY THROUGH THE VIEWING INSTRUMENTS WAS NOT SUCCESSFUL INITIALLY, BUT IMPROVEMENTS BOTH IN TECHNIQUE AND EQUIPMENT LED TO REMARKABLY GOOD RESULTS IN THE LATTER PART OF THE PROGRAM.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + INSTRUMENT, IN CORE + REACTOR, LMCR + MAINTENANCE AND REPAIR + INSTRUMENT, SURVEILLANCE + EQUIPMENT DESIGN + REMOTE MANIPULATING AND VIEWING + MAINTENANCE, REMOTE

5-027596

ALSO IN CATEGORIES 9 AND 17

DATED 1968

DUFFY JG + MATTE J + RYAN JD + WARBERG FS

OBJECT RETRIEVAL TOOLS FOR THE ENRICO FERMI ATOMIC POWER PLANT REPAIR PROGRAM

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN

3 PAGES, 2 FIGURES, ANS TRANSACTIONS 11(1), PAGES 358-360, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

THE METHOD OF RETRIEVING FROM THE BOTTOM OF THE FERMI REACTOR AN OBJECT THAT CAUSED A MELTDOWN INVOLVED PROCEDURES AND TOOLS WHICH ARE UNIQUE AND COULD HAVE APPLICATION IN FUTURE PROBLEMS. TOOLS WERE DEVELOPED TO CUT THE OBJECT INTO PIECES SMALL ENOUGH TO BE REMOVED UPWARD THROUGH THE SUPPORT PLATE. SEVERAL GRIPPERS FOR THIS PURPOSE WERE ALSO DEVELOPED. SIMULTANEOUSLY, SPECIAL TOOLS WERE DEVELOPED TO REMOVE THE OBJECT INTACT VIA THE SODIUM INLET PIPE.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + INSTRUMENT, IN CORE + REACTOR, LMCR + MAINTENANCE AND REPAIR + EQUIPMENT DESIGN + REMOTE MANIPULATING AND VIEWING + MAINTENANCE, REMOTE

5-027725

DATED 1966

HIRAKAWA N + KLINKMAN AE

A COMPARISON OF THE METHODS USED IN FAST-REACTOR MELTDOWN CALCULATIONS

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + JAPAN ATOMIC ENERGY COMMISSION

1 PAGE, 1 TABLE, 4 REFERENCES, ANS TRANS. 9(2), PAGE 552 (NOVEMBER 1966)

BECAUSE LITTLE INFORMATION HAS BEEN MADE AVAILABLE ON THE COMPARISON OF VARIOUS METHODS USED IN

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-027725 *CONTINUED*

THE CALCULATION OF THE ENERGY RELEASE IN A FAST-REACTOR MELTDOWN, A STUDY EMPLOYING THE VARIOUS METHODS ON A SIMPLIFIED MODEL OF THE CORE OF THE ENRICO FERMI REACTOR HAS BEEN COMPLETED. FOUR METHODS WERE USED - THE JANKUS METHOD, THE NICHOLSON METHOD, THE AX-1 METHOD, AND THE WEAK EXPLOSION CODE. THE FIRST TWO ARE HAND CALCULATIONS, AND THE LATTER TWO ARE DIGITAL COMPUTER METHODS.

REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMFBR

5-027751

DATED 1966

GRASS G + KUTTOWSKI H + WAKNSING R

BOILING OF LIQUID ALKALI METALS

EUROPEAN ATOMIC ENERGY COMMUNITY, ISPRA (ITALY). JOINT NUCLEAR RESEARCH CENTER
ANL-TRANS-390 +. 37 PAGES, TRANSLATED FROM REPORT, SEPT. 1966

BOILING OF LIQUID METALS AT LOW PRESSURES IS UNSTABLE UNDER STAGNANT CONDITIONS AS WELL AS UNDER CONDITIONS OF NATURAL AND FORCED CONVECTION. THIS UNSTABLE TYPE OF BOILING MAY BE INTERPRETED AS AN EJECTION SEQUENCE OF SINGLE BUBBLES. THE KINETICS INVOLVED IN THIS EJECTION PROCESS ARE SIGNIFICANTLY INFLUENCED BY THE VAPOR PRESSURE IN THE BUBBLE. IN THE WORK HERE DESCRIBED INITIAL SUPERHEAT, INITIAL PRESSURE AND SUBSEQUENT PRESSURE VARIATIONS, WERE FOR VARIOUS EXPERIMENTAL CONDITIONS COMPARED WITH THE KINETICS OF THE SINGLE BUBBLE EJECTIONS. A CALCULATION MODEL WAS SET UP TO CORRELATE QUANTITATIVELY THE PHENOMENA.

AVAILABILITY - SPECIAL LIBRARIES ASSOCIATION TRANSLATION CENTER, JOHN CRERAR LIBRARY, 35 WEST 33RD STREET, CHICAGO, ILLINOIS 60616

CONVECTIVE BOILING + SODIUM + METAL, ALKALI + BUBBLE NUCLEATION + NUCLEATE BOILING + BUBBLE + VOID FRACTION + REACTOR, LMFBR + BOILING

5-027804

DATED 1968

MCKEON JI

PRIMARY PIPING FLEXIBILITY ANALYSIS FOR LIQUID-METAL-COOLED FAST BREEDER REACTORS

THE M. W. KELLUGG COMPANY, RESEARCH AND DEVELOPMENT DEPT., MECHANICAL ENGINEERING DEVELOPMENT DIVISION, PIDGATWAY, JERSEY CITY, N. Y.

15 PAGES, 2 FIGURES, 4 TABLES, 21 REFERENCES, NUCLEAR ENGINEERING AND DESIGN, 7(5), PAGE 427-441, (MAY 1968)

DISCUSSION OF THE FOUR MAIN PROBLEM AREAS IN THE DESIGN AND ANALYSIS OF PRIMARY PIPING FOR LIQUID-METAL-COOLED FAST BREEDER POWER REACTOR SYSTEMS - VALIDITY OF THE STRESS RANGE CONCEPT, LOCAL OVERSTRAIN AND CREEP CONCENTRATIONS, THERMAL RATCHETTING, ELASTIC AND INELASTIC BUCKLING. EXAMINATION OF THE APPLICABILITY OF PRESENT DESIGN METHODS TO RIGID PIPING SYSTEMS (IN WHICH THE THERMAL EXPANSION IS ABSORBED BY THE INHERENT FLEXIBILITY OF PIPING COMPONENTS) AND TO NON-RIGID PIPING SYSTEMS (IN WHICH THE THERMAL EXPANSION IS ACCOMMODATED BY BELLOW JOINTS), AND CRITICISM OF PRESENT DESIGN RULES.

STRESS ANALYSIS + EQUIPMENT DESIGN + EQUIPMENT + REACTOR, LMFBR + PIPES AND PIPE FITTINGS

5-028889

DATED 1967

TRENT US

APPLICATIONS OF GEOMETRIC MODELS FOR THE FFTF HYDRAULIC CORE MOCKUP

BATTELLE-NORTHWEST, RICHLAND, WASH.

BNWL-575 +. 127 PAGES, FIGURES, REFERENCES, NOVEMBER 1967

MODELING THE FFTF CORE TO ESTABLISH HYDRAULIC AND STRUCTURAL BEHAVIOR IS FEASIBLE AND ESSENTIAL FOR OBTAINING RELIABLE, COMPREHENSIVE ENGINEERING DATA TO BOTH GUIDE AND CONFIRM THE REACTOR CORE DESIGN. FROM THE STANDPOINT OF MODELING SIMILITUDE IT IS CONCLUDED THAT ONE, MULTIFLUID, FULL SCALE MULTIPURPOSE MODEL IS FEASIBLE FOR ACCOMPLISHING REQUIRED HYDRAULIC AND STRUCTURAL TEST PROGRAMS. THE FULL SCALE STRUCTURAL MODEL CAPABILITY SHOULD ENCOMPASS THE REQUIRED THERMAL MECHANICAL STUDIES AND SOME HYDRAULIC STUDIES, BUT THE REDUCED SCALE HYDRAULIC STUDIES MODEL NEED ONLY MEET HYDRAULIC TEST REQUIREMENTS. VARIOUS SINGLE PURPOSE FACILITIES SHOULD BE CONSTRUCTED AND TESTED AS REQUIRED BY THE REACTOR DESIGN EFFORT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + HYDRAULIC ANALYSIS + REACTOR, LMFBR + REACTOR, TEST + FFTF (TR) + CORE COMPONENTS + MODEL TESTING

5-029002

ALSO IN CATEGORIES 9 AND 12

DATED 1968

DAVIDSON DF + KOACH PF

AN EXPERIMENTAL CONTINUOUS-INDICATION PLUGGING METER FOR IMPURITY MONITORING IN LIQUID ALKALI METALS

UNITED KINGDOM ATOMIC ENERGY AUTHORITY

TRG REPORT 1640 (R) +. 7 PAGES, FIGURES, TABLES, REFERENCES, 1968

A CONTINUOUS READING PLUGGING METER WAS DEMONSTRATED THAT MONITORED SODIUM SATURATION TEMPERATURE 125 TO 280 C. THE LARGEST CYCLE VARIATION IN THE TEST TRACE WERE ± 5 C AT 125 C. UNDER STABLE CONDITIONS SATURATION TEMPERATURE COULD BE READ TO ± 2 C AT 125 AND ± 1 C AT 280.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-029002 *CONTINUED*
REACTOR, LMCR + SODIUM + UNITED KINGDOM + COOLANT CHEMISTRY + INSTRUMENT, FLOW + INSTRUMENT, SURVEILLANCE + IMPURITY

5-029257 ALSO IN CATEGORY 17 DATED 1967
GAJAC G + REYNES L
EXPERIENCE GAINED FROM THE FINAL CONSTRUCTION PHASE AND THE APPROACH OF POWER OF RAPSODIE
CENTRE D ETUDES NUCLEAIRES DE CADARACHE, FRANCE
22 PAGES, 11 FIGURES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPT. 1967

ANALYZES TWO SODIUM LEAKS (OCTOBER 1966). (1) TOO RAPID PREHEATING OF PIPING WHICH HAD TWO COLD SPOTS CAUSED IT TO BURST. AMONG CAUSES ARE FILLING LINE WITH SODIUM TOO SOON (PREHEATING COULD NOT BE ALLUCATED). LEAK INDICATORS DID NOT WORK. DOUBLE WALLS NOW FILLED ALL OVER SYSTEM. (2) UNTIMELY OPENING OF AN ARGON VALVE PRESSURIZED THE SODIUM, DRIVING IT OUT THROUGH CONTROL-ROD-DRIVE PLUGS NOT BULTED (COMMUNICATION FAILURE). VALVE MADE SLOWER ACTING, SAFETY VALVES MADE LARGER, EQUALIZING OF DRIVES AND COVER GAS PRESSURES DONE. UNCORRECTED, THIS INCIDENT WOULD HAVE BEEN REPEATED YEARLY. ***RAPSODIE INCIDENTS ARE ALL RECORDED AND STUDIED.

AVAILABILITY - G. GAJAC OR L. REYNES, COMMISSARIAT A L'ENERGIE ATOMIQUE - CEN/CADARACHE - DRP

FRANCE + FAILURE, OPERATOR ERROR + REACTOR, LMCR + FAILURE, PIPE + REACTOR, SAFETY RESEARCH + THERMAL MECHANICAL EFFECT + INCIDENT, EQUIPMENT + MODIFICATION + REPORT, OPERATIONS ANALYSIS + FAILURE, DESIGN ERROR + INFORMATION RETRIEVAL + MAIN COOLING SYSTEM + LEAK + PRESSURE TRANSIENT

5-029374 ALSO IN CATEGORY 18 DATED 1955
BRITTAN RG + CERUTTI B + LICHTENBERGER HV + RICE R + THALGOTT FW
HAZARD EVALUATION REPORT ON THE FAST REACTOR ZERO POWER EXPERIMENT (ZPR-III)
ARGONNE NATIONAL LABORATORY, LEMONT, ILLINOIS
ANL-FW1-105 +. 75 PAGES, 19 FIGURES, TABLES, APRIL 1955

THE ASSEMBLY IS SMALL PIECES IN DRAWERS PLACED IN ARRAY OF SQUARE TUBES ON TWO CARRIAGES SO THAT TWO HALF-CRITICAL SECTIONS MAY BE BROUGHT TOGETHER AND CONTROLLED. PART OF PROGRAM TO ESTABLISH FEASIBILITY OF SODIUM COOLED BREEDER. PROGRAM COVERS USE OF U-235. SUMMARIZES SAFETY MECHANISMS, REACTIVITY COEFFICIENTS, AND KINETIC CHARACTERISTICS. THE MOST SEVERE ACCIDENT OCCURS IF A LARGE POSITIVE TEMP COEFFICIENT EXISTS. SAFETY RODS DO NOT OPERATE AND A HIGH RATE OF REACTIVITY INSERTION OCCURS. DETAILED CALCULATIONS GIVEN IN APPENDIX. DIRECT DOSE FROM CLOUD ASSUMING 100% FISSION PRODUCTS ARE IN CLOUD IS GIVEN FOR CLOUD HEIGHTS OF 30 TO 200 METERS AND WIND SPEEDS OF 1/2 AND 5 METERS/SEC.

ACCIDENT, CONSEQUENCES + CRITICAL ASSEMBLY + DOSE + REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + REPORT, SAR + TESTING + CRITICALITY EXPERIMENT + HAZARDS ANALYSIS + REACTOR PHYSICS + ZPR 3 (CAF) + OFF SITE

5-029375 ALSO IN CATEGORY 18 DATED 1959
GERSHUN TL
PRELIMINARY SAFEGUARDS REPORT BASED ON URANIUM-MOLYBDENUM FUEL FOR THE HALLAM NUCLEAR POWER FACILITY
ATOMIC INTERNATIONAL
NAA-SR-3379 +. 100 PAGES, FIGURES, TABLES, FEBRUARY 10, 1959

THE HMPF IS A 240 MWt SODIUM-COOLED GRAPHITE-MODERATED REACTOR. FUEL ELEMENTS ARE 19-ROD CLUSTERS OF U/10% MO ALLOY WITH STAINLESS STEEL CLADDING. CONTROL RODS ARE BORON-NICKEL RINGS ASSEMBLED ON AN SS PULL TUBE. CONTAINMENT IS SUBATMOSPHERIC. RADIOACTIVE COMPONENTS ARE LOCATED IN CONCRETE SHIELDED CELLS. AN INERT GAS ATMOSPHERE IS USED IN ALL AREAS CONTAINING PRIMARY-SYSTEM PIPING AND COMPONENTS. DOSE FOR MAJOR STACK RELEASE AT EXCLUSION BOUNDARY UNDER NEUTRAL CONDITIONS GIVEN (BUT TIME FOR THIS DOSE IS NOT GIVEN.) THE MCA IS ONE IN WHICH CONTROL ROD THIMBLE FAILS IN REGION OF HELIUM ATMOSPHERE AND GAS LEAKS OUT. DOSE RATE AT SITE BOUNDARY IS LESS THAN 0.1 MR/HR.

ACCIDENT ANALYSIS + ACCIDENT, CONSEQUENCES + CONTAINMENT DESIGN + DOSE + FISSION PRODUCT RELEASE + HALLAM (LMR) + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR DESCRIPTION + REACTOR TRANSIENT + INSTRUMENT, CONTROL + HAZARDS ANALYSIS + SYSTEM DESCRIPTION + EQUIPMENT + REPORT, PSAR + ENGINEERED SAFETY FEATURE + AUXILIARY COOLING SYSTEM + OFF SITE

5-029446 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION B-3 - PRESSURE LOADING
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE B-16 AND -17 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

IT IS STATED IN THE APPLICATION THAT PRESSURE LOADING CAN BE APPLIED AS UNIFORM LOAD SIMULTANEOUSLY IN ALL CELLS AND THAT THE COMMUNICATIONS PROVIDED BETWEEN CELLS ARE SUCH THAT THIS ASSUMPTION IS VALID. DISCUSS A TRANSIENT PRESSURE DISTRIBUTION AND ITS INFLUENCE, OR LACK OF INFLUENCE, ON THE DESIGN. INCLUDE A DISCUSSION ON COMMUNICATIONS BETWEEN CELLS. (ANSWER) SEE SECTION XVI OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFILME

ACCIDENT, CONSEQUENCES + CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + PRESSURE, INTERNAL + RESPONSE TO NRC QUESTION

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-029459 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION D-9 - ARGON GAS COOLANT LOOP LINES
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 2 PAGES, PAGE D-12 AND -13 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ARGON GAS COOLANT LOOP PASSES THROUGH AIR ZONE IN ORDER TO REACH FAN AND COOLER LOCATED IN NITROGEN ZONE. SHOULD EITHER LINE DEVELOP LARGE LEAK, (A) HOW LONG WOULD IT TAKE ARGON FAN TO EMPTY ALL GAS INTO OUTER CONTAINMENT AIR ZONE AND REPLACE IT WITH AIR FROM OUTER CONTAINMENT, (B) WHAT WOULD BE COURSE OF SUCH ACCIDENT, AND POTENTIAL DUSES SHOULD IT OCCUR AT TIME WHEN REACTOR VESSEL HEAD AND IRRADIATED FUEL STORAGE TANK SHIELDING COVER WERE OFF, (C) WHAT MEANS TERMINATE REACTION BETWEEN EXPOSED SODIUM AND CONTAINMENT AIR, (D) COULD IT BE TERMINATED BEFORE ALL OXYGEN IN OUTER CONTAINMENT WAS CONSUMED. (ANSWER ON PGS D-12 AND -13).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + REACTOR, LMCR + FAILURE, PIPE + REPORT, SAR + ACCIDENT, HYPOTHETICAL + SEFOR (RE) + RESPONSE TO NRC QUESTION + AUXILIARY COOLING SYSTEM

5-029503 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION L-3 - COOLANT FLOW TIME CONSTANTS
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 1 PAGE, PAGE L-2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

INDICATE ON A SIMPLIFIED FLOW DIAGRAM OF THE NA COOLANT SYSTEM THE TIME DELAYS THROUGH THE PRIMARY AND SECONDARY SYSTEMS AS WELL AS SIGNIFICANT TIME CONSTANTS (E.G., FUEL CLAD, GAP, PELLETS, IHX TUBES, ETC.) FOR FORCED AND NATURAL CONVECTION FLOW RATES. (ANSWER) SEE APPENDIX A OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + COOLING, SHUTDOWN + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + FLOW + RATE + FLOW, CONVECTION

5-029504 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION L-4 - AVOIDING FUEL MELTING
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 2 PAGES, PAGE L-2 AND -3 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CLARIFY STATEMENT ON TOP OF PAGE (16-5) REGARDING AVOIDING FUEL MELTING. DEFINE TEMPERATURE IN TRANSIENT RESPONSE FIGURES AT END OF CHAPTER XVI. WHAT IS SIGNIFICANCE OF THERMAL TEMPERATURE WAVE INDICATED IN FIGURES XVI-5 AND XVI-7, TO REST OF THE SYSTEM (ALSO, REFERENCE P.16-6). (ANSWER) NO MELTING CRITERIA GIVEN IN SUPPL. 3 OF FDSAR. THERMAL TRANSIENTS DISCUSSED IN SECT. 16 OF FDSAR, ACCIDENT TRANSIENTS IN APPENDIX A OF SUPPLEMENT 16 TO FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FUEL MELTDOWN + REACTOR, LMCR + REACTOR TRANSIENT + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + SAFETY ANALYSIS + THERMAL TRANSIENT

5-029505 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION L-5 - LOSS-OF-PUMPING-POWER ACCIDENT
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 2 PAGES, PAGE L-3 AND -4 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

TABLE XVI-13 (PAGE 16-14 PRESENTS ELEVEN LOSS OF PUMPING POWER ACCIDENTS. CASES 2, 3, 4, AND 6 RESULT IN MAXIMUM COOLANT TEMPERATURES IN EXCESS OF (SYSTEM DESIGN TEMPERATURE) 1000 F. STATEMENTS BELOW TABLE IMPLY OVER-DESIGN COOLANT TEMPERATURES PRODUCE NO DETRIMENTAL EFFECTS ON SYSTEM. DETAIL RATIONALE BEHIND THESE FINDINGS, INCLUDING CONSIDERATIONS OF FUEL TEMPERATURE AND THERMAL STRESSES IN CRITICAL LOCATIONS IN PRIMARY SYSTEM AND SECONDARY SODIUM COOLANT SYSTEMS. ALSO PROVIDE TIME-TEMPERATURE GRAPH OF THESE FOUR CASES - CARRYING EACH IN TIME UNTIL NORMAL CONDITIONS ARE RE-ESTABLISHED. DEFINE THE MEANING OF MAXIMUM COOLANT TEMPERATURE AND INDICATE CORRESPONDING MAXIMUM HOT CHANNEL-HOT SPOT TEMPERATURES. *** (ANSWER ON PAGES L-3 AND -4.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + REPORT, SAR + STRESS + THERMAL MECHANICAL EFFECT + HOT SPOT + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + HOT CHANNEL + THERMAL TRANSIENT

5-029506 ALSO IN CATEGORIES 6 AND 18 DATED 1968
RESPONSE TO QUESTION L-6 - ENERGY DENSITIES OF FUEL IN TRANSIENT TESTS
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-029506 *CONTINUED*
2 PAGES, 1 TABLE, PAGE 5-4 AND -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT ARE EXPECTED ENERGY DENSITIES IN CAL/GM OF FUEL IN PLANNED SUB-PROMPT AND PROMPT CRITICAL TRANSIENT TESTS. (ANSWER) TABLE OF ENERGY DENSITIES GIVEN BASED ON TRANSIENT ANALYSIS DESCRIBED IN SECTION SII OF FDSAR AND SUPPLEMENT 10.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REACTOR TRANSIENT + REPORT, SAR + THERMAL CONSIDERATION + TEST, PHYSICS + SEFOR (RE) + HEAT TRANSFER EXPERIMENT + RESPONSE TO NRC QUESTION + FUEL ELEMENTS

5-029507 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION L-7 - CLAD FAILURE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE L-5 AND -6 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT IS CLADDING FAILURE THRESHOLD AND UNDER WHAT CONDITIONS WILL CLADDING FAIL. (ANSWER) SEE SUPPLS. 3 AND 10 OF FDSAR. ONLY OBSERVED EFFECT OF CYCLIC OPERATION THROUGH FUEL MELTING POINT WAS SOME LOCAL CLAD EXPANSION. ENERGY DENSITY OF 400 CAL/GRAM NEEDED FOR CLAD TO FAIL FROM INTERNAL PRESSURE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CLADDING + INTEGRITY + REACTOR, LMCR + REPORT, SAR + FAILURE, CLADDING + SEFOR (RE) + PRESSURE, INTERNAL + RESPONSE TO NRC QUESTION

5-029508 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION L-8 - ACCIDENT ANALYSES FOR LOSS OF EXTERNAL HEAT SINKS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, PAGE L-6 THRU -8 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE BASES, TIME HISTORY, AND RESULTS OF ACCIDENT ANALYSES FOR PARTIAL AND FINALLY COMPLETE LOSS OF EXTERNAL HEAT SINKS AS CLASS I COMPONENTS AND DESIGN MARGINS ARE AFFECTED. INCLUDE - (A) LOSS OF THREE INDEPENDENT FAN-COOLERS WHICH PROVIDE HEAT SINKS FOR INNER CONTAINMENT ATMOSPHERE. (B) LOSS OF SECONDARY SODIUM COOLANT FROM MAIN COOLANT LOOP. (C) COMBINATION OF (A) AND (B.3). (D) LOSS OF DIESEL GENERATOR AND OFF-SITE POWER. (E) COMBINATION OF (A) AND (D). (F) COMBINATION OF (B.3) AND (D). (G) COMBINATION OF (A), (B.3) AND (D). (ANSWER ON PGS L-6 THRU - 8.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT ANALYSIS + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + REPORT, SAR + HEAT SINK + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + AUXILIARY COOLING SYSTEM

5-029509 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION L-9 - CORE BOUNDARY PROBLEMS IN BETHE-TAIT ANALYSIS OF MHA
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE L-8 AND -9 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

HOW ARE THE CORE BOUNDARY PROBLEMS TREATED IN THE BETHE-TAIT ANALYSIS OF THE MHA. ALSO, WHEN THE FRED IS PLACED IN THE CORE, HOW IS ITS BOUNDARY CONSIDERED IN THE FUEL MELTDOWN AND BETHE-TAIT ANALYSES. (ANSWER ON PGS L-8 AND -9.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

BETHE TAIT + FUEL MELTDOWN + REACTOR, LMCR + REPORT, SAR + ACCIDENT, HYPOTHETICAL + SEFOR (RE) + RESPONSE TO NRC QUESTION

5-029606 ALSO IN CATEGORY 18 DATED 1968
PETERS NT
DEVELOPMENT OF THE PRIMARY SYSTEM FOR THE ENRICO FERMI ATOMIC POWER PLANT
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-308 +. 173 PAGES, JUNE 1968

DOCUMENTS THE COURSE OF DEVELOPMENT OF THE PRIMARY HEAT TRANSPORT SYSTEM, STARTING WITH THE EARLY HISTORY OF THE OBJECTIVES AND CONTINUING THROUGH THE EVOLUTION OF THE SYSTEM DESIGN CONCEPTS TO THE FINAL DESIGN. PRESENTS CONSTRUCTION, TESTING, AND OPERATION OF SYSTEM WITH REACTOR POWER UP TO 100 MWT WITH EVALUATION OF THE GENERAL PERFORMANCE AND IMPORTANT FEATURES, INCLUDING RECOMMENDATIONS FOR USE IN DESIGN OF NEW LMFBRS. PRESENTS INTERACTION WITH DESIGN OF OTHER FEATURES OF PLANT (SUCH AS REACTOR CONTAINMENT BUILDING). SYSTEM COMPONENTS AND AUXILIARY

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-029606 *CONTINUED*

FEATURES ARE DESCRIBED ONLY AS RELEVANT TO THE DISCUSSION OF PRIMARY SYSTEM AS AN INTEGRATED, FUNCTIONAL UNIT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + SYSTEM DESCRIPTION + MAIN COOLING SYSTEM + R AND D PROGRAM + REACTOR, LMFBR

5-029722 ALSO IN CATEGORIES 1 AND 18 DATED 1967

BAKER L + EGGEN DT + IVINS RU + SZANLEWICZ SA

US LMFBR SAFETY PROGRAM

ARGONNE NATIONAL LABORATORY + US ATOMIC ENERGY COMMISSION

25 PAGES, 2 FIGURES, 2 TABLES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPTEMBER 1967

DESCRIBES OBJECTIVES AND STATUS OF PLANNING EFFORT FOR SAFETY ANALYSES AND EVALUATIONS IN THREE TECHNICAL AREAS - (1) ANALYSIS OF ACCIDENTS AND ENERGY RELEASE, (2) PREVENTION AND MITIGATION OF THE CONSEQUENCES OF ACCIDENTS, AND (3) ESTABLISHMENT OF SAFETY-RELATED CRITERIA, CODES AND STANDARDS. BASED ON A TABULATION OF POSSIBLE KNOWN OR POTENTIAL OFF-NORMAL OPERATIONS, INSTABILITIES, AND MALFUNCTIONS, DISCIPLINE-ORIENTED REQUIREMENTS WERE DEVELOPED TO SPECIFY THE RESEARCH AND DEVELOPMENT EFFORT. A BRIEF INDICATION OF THE OBJECTIVES AND CURRENT EFFORTS PROCEEDING UNDER EACH TASK ARE GIVEN.

AVAILABILITY - L. BAKER, JR., ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ACCIDENT ANALYSIS + CODES AND STANDARDS + SAFETY PROGRAM + R AND D PROGRAM + ENGINEERED SAFETY FEATURE + REACTOR, LMFBR

5-029831 ALSO IN CATEGORIES 3 AND 17 DATED 1968

RATEICK RC + MCHUGH WE + SALING JH

RESULTS AND CONCLUSIONS FROM THE EXAMINATION OF FUEL SUBASSEMBLIES REMOVED FROM THE ENRICO FERMI REACTOR AFTER THE FUEL DAMAGE INCIDENT

POWER REACTOR DEVELOPMENT COMPANY + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + BATTELLE MEMORIAL INSTITUTE 2 PAGES, 3 REFERENCES, ANS TRANS. 11(1), PAGES 100-101, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

ON OCT. 5, 1966, ZR-CLAD U-MO FUEL PINS PARTIALLY MELTED. OFF-SITE EXAM MADE TO DETERMINE IF FAILURE CAUSED BY FLOW BLOCKAGE. CONCLUSIONS - (1) SHIPMENT OF SUBASSEMBLIES WITHOUT PRIOR CLEANING CAN BE DONE AND EXAM ADEQUATELY PERFORMED IN CONVENTIONAL HOT CELL. (2) MELTING WAS CAUSED BY FLOW BLOCKAGE. (3) EVEN THOUGH SS WRAPPER WALLS LOST INTEGRITY IN MELTED SUBASSEMBLIES, GROSS FUEL LOSSES DID NOT OCCUR. (4) REDISTRIBUTION IN TWO MELTED ASSEMBLIES ACCOUNTED FOR 11.5 CENTS REACTIVITY OF TOTAL LOSS OF 22 CENTS. REMAINING 10.5 CENTS THOUGHT TO BE DUE TO RESULT OF DEFORMATION.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + HOT CELL + REACTOR, LMCR + TRANSPORTATION AND HANDLING + INCIDENT + REACTIVITY EFFECT, SLUMP + EXAMINATION + FLOW BLOCKAGE + REACTOR, LMFBR

5-029984 ALSO IN CATEGORY 17 DATED 1968

COSTELLO RH

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT, APRIL 1968

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.

APDA-CFE-21 +. 46 PAGES, 21 FIGURES, JULY 1968

DESCRIBES 4 WAYS OF REMOVING ZIRCONIUM PLATE FASTENER AND PLATES. NINE SMALL OBJECTS WERE REMOVED FROM VESSEL. A MOCKUP OF FUEL ELEMENTS WAS MADE IN A LOOP TO DETERMINE THAT THE OBJECT REMOVED HAD CAUSED THE FLOW BLOCKAGE. A FLOW BLOCKAGE OF 95 AND 96% WAS CALCULATED TO BE REQUIRED FOR THE MELTDOWN. THE OBSERVED BLOCKAGE WAS 92 AND 30% UNDER REALISTIC CONDITIONS (THE OBJECT HAD BEEN DISTORTED IN HANDLING).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + HYDRAULIC EXPERIMENT + REACTOR, LMCR + MOCKUP + REPORT, OPERATIONS ANALYSIS + REMOTE MANIPULATING AND VIEWING + FLOW BLOCKAGE + REACTOR, LMFBR

5-030253 ALSO IN CATEGORIES 7 AND 18 DATED 1968

RESPONSE TO QUESTION 5 - SOURCE TERMS IN RELEASE CALCULATION

NORTHERN STATES POWER COMPANY

11 PAGES, PP. 23-33 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

FISSION PRODUCT SOURCE TERMS USED IN ACCIDENT CALCULATIONS AND OPERATIONAL CONSIDERATIONS IS DISCUSSED. ***THREE ACCIDENTS (CORE DISASSEMBLY IN MELTDOWN, SECONDARY LOOP RUPTURE, GAS RADWASTE TANK RUPTURE) WERE DISCUSSED IN SAR. ESSENTIAL FEATURES OF THESE ANALYSES ARE REPEATED HERE. (A) 2/3 OF FUEL MELTS BUT 99% OF IODINE REACTS WITH SODIUM, (B) ALTHOUGH SOME PRIMARY SODIUM MIGHT LEAK THROUGH TO THE SECONDARY SYSTEM, IT WOULD SPILL IN LOW OXYGEN AREAS. ONLY 0.0187 CU FT OF PRIMARY SODIUM WAS ESTIMATED RELEASED. (C) ASSUMED ONE TANK RELEASES 15% OF

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-030253 *CONTINUED*
GASEOUS FP FROM CENTER FUEL ELEMENTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

FISSION GAS RELEASE + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REPORT, SAR + WASTE STORAGE + ACCIDENT, HYPOTHETICAL + WASTE TREATMENT, CAS + SEFOR (RE) + RESPONSE TO NRC QUESTION + REACTOR, LMFBR

5-030255 ALSO IN CATEGORIES 10 AND 12 AND 18 DATED 1968
RESPONSE TO QUESTION 4 - EMERGENCY COOLING
NORTHERN STATES POWER COMPANY
2 PAGES, PP. 21-22 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

EMERGENCY COOLING IS THAT FOLLOWING LOSS OF SECONDARY HEAT SINKS AND TOTAL LOSS OF SITE POWER. ***REFERS TO SUPPLEMENT 19, QUESTION 2, WHERE SAME QUESTION ANSWERED. A 50-KW AUXILIARY POWER UNIT HAS BEEN PROVIDED FOR AUX. PC PUMP, AND ALSO POWERS A GAS BLOWER TO COOL THIS PUMPS WINDINGS. LATTER GAS BLOWER SYSTEM IS DESCRIBED HERE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

EMERGENCY POWER, ELECTRIC + ACCIDENT, LOSS OF POWER + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + AUXILIARY COOLING SYSTEM + REACTOR, LMFBR + PUMPS

5-030633 ALSO IN CATEGORY 18 DATED 1967
SEFOR FSAR VOL. 11 - SAFETY ANALYSIS
GENERAL ELECTRIC COMPANY
150 PAGES, FIGURES, TABLES, SEFOR FACILITY DESCRIPTION AND SAFETY ANALYSIS REPORT, VOL. 2, JULY 21, 1967, DOCKET 50-231

CONTAINS SECTIONS ON AUXILIARY SYSTEMS, EXPERIMENTAL PROGRAM AND EQUIPMENT, CONDUCT OF OPERATIONS, ELECTRIC POWER SUPPLY, AND THE SAFETY ANALYSIS. APPENDICES INCLUDE AN ANALOG MODEL OF SEFOR, RESULTS OF MOCKUP TESTS IN ZPR-3, BLAST SHIELDING AND DESIGN, AND RADIOLOGICAL EFFECTS CALCULATIONS. SAFETY ANALYSIS CONSIDERS THE SAFETY FEATURES FOR OPERATION, POSITIVE REACTIVITY INSERTIONS, AND LOSS OF COOLANT FLOW. THE MHA IS A SUDDEN INCREASE IN POWER DENSITY FOLLOWED BY LOSS OF COOLANT FLOW LEADING TO RAPID FUEL MELTING AND COMPACTION OF CORE INTO A SUPERCRITICAL MASS, WHEREAS THE CONTAINMENT DESIGN CRITERIA IS SUCH THAT IT MEETS A SET OF CONDITIONS MORE SEVERE THAN THOSE PREDICTED FOR THE MHA. THE CALCULATIONAL BASIS FOR RADIOLOGICAL EFFECTS ASSUMES CONDITIONS MORE THAN THOSE FOR THE MHA AND CONTAINMENT CAPABILITY LESS EFFECTIVE THAN THOSE OF THE DESIGN CRITERIA.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

COMPUTER, ANALOG + CONTAINMENT DESIGN + DOPPLER COEFFICIENT + DOPPLER EFFECT + DOSE + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + ACCIDENT, MAXIMUM CREDIBLE (MCA) + MOCKUP + REACTOR DESCRIPTION + REPORT, SAR + ANALYTICAL MODEL + ACCIDENT, HYPOTHETICAL + HAZARDS ANALYSIS + ANALOG SIMULATION + ELECTRIC POWER, NORMAL + TEST, PHYSICS + SEFOR (RE) + SHIELDING + SYSTEM DESCRIPTION + EXPERIMENT + SIMULATION + SAFETY ANALYSIS + R AND D PROGRAM + ENGINEERED SAFETY FEATURE + SYSTEM CAPABILITY + OFF SITE + REACTOR, LMFBR

5-031420 ALSO IN CATEGORY 8 DATED 1967
IVINS RD + TEVEBAUGH AD
SECTIONS FROM ANL-7425 CHEMICAL ENGINEERING DIVISION SUMMARY REPORT JULY-DEC 1967. SECTION IV REACTOR SAFETY
ARGONNE NATIONAL LABORATORY
ANL-KCV-4500 +. DEC. 1967, 111P., FIGURES, TABLES

PREPRINT OF SECTION IV, REACTOR SAFETY, OF ANL-7425, CHEMICAL ENGINEERING DIVISION SUMMARY REPORT, JULY-DECEMBER 1967. THE ANL WORK FOR THIS PERIOD WAS CONCENTRATED ON (1) THERMAL (WATER-COOLED) REACTOR STUDIES AND (2) FAST (SODIUM-COOLED) REACTOR STUDIES. ITEM (1) EFFORTS WERE DIRECTED TO ANALYTICAL STUDIES DESCRIBING THE CORE BEHAVIOR DURING A LOSS-OF-COOLANT ACCIDENT, ANALYTICAL AND EXPERIMENTAL STUDIES OF THE BEHAVIOR OF ZIRCALOY-2 CLAD UO₂ FUEL RODS DURING NUCLEAR EXCURSIONS AND OF THE POSSIBILITY OF PRESSURE PULSES DURING EITHER OF THESE TWO ACCIDENTS. ITEM (2) EFFORTS WERE DIRECTED TO EXPERIMENTALLY DETERMINING THE TRANSPORT PROPERTIES OF LIQUID URANIUM DIOXIDE, ANALYTICAL AND EXPERIMENTAL STUDIES OF THE THERMAL AND MECHANICAL INTERACTIONS BETWEEN FUEL, CLADDING, AND COOLANT, AND OF THE SODIUM-AIR REACTION, AND EXPERIMENTAL STUDIES OF FUEL MIGRATION.

HEAT TRANSFER, BOILING + CHEMICAL REACTION + COMPUTER PROGRAM + FUEL MELTDOWN + ACCIDENT, LOSS OF COOLANT + OXIDATION + REACTOR, SAFETY RESEARCH + ACCIDENT, REACTIVITY + SODIUM + TREAT (PKR) + ANALYTICAL MODEL + MOLTEN FUEL + ACCIDENT MODEL + AIR + ANL + IN PILE EXPERIMENT + REACTOR, LWR + URANIUM DIOXIDE + REACTOR, PULSED + HEAT TRANSFER ANALYSIS + HEAT TRANSFER EXPERIMENT + ZIRCALOY + PROPERTY, PHYSICAL + PRESSURE PULSE + PRESSURE TRANSIENT + REACTOR, LMFBR + GALLIUM + THERMAL TRANSIENT + FUEL COOLANT INTERACTION

5-031448 ALSO IN CATEGORY 18 DATED 1968
KIRSCHLER LH + HILTZ RH + RODGERS SJ
THE EFFECT OF HIGH TEMPERATURE SODIUM ON THE MECHANICAL PROPERTIES OF CANDIDATE ALLOYS FOR THE LMFBR PROGRAM
MSA RESEARCH CORPORATION, PA.
MSAR-68-135 +. 100 PAGES, 37 FIGURES, 13 TABLES, SEPTEMBER 1968

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-031448 *CONTINUED*

THREE LOOPS (ONE OPERATING, ONE EVALUATING A CARBON METER), FOUR RETORTS (ALL USED IN SCREENING TESTS). 500-HR HIGH-TEMPERATURE OPERATION OF 30% AND 31% SS COMPLETED, CREEP RUPTURE TESTING BEGUN. PRELIMINARY DATA INDICATE 50% REDUCTION IN 1400 F RUPTURE LIFE, COMPARED WITH 1200 F. A CHANGE IN OXIDATION MECHANISM SUGGESTED.

ALLOY + REACTOR, LMCR + METAL + SODIUM + OUT OF PILE EXPERIMENT + STEEL, STAINLESS + CREEP BEHAVIOR + COOLANT CHEMISTRY + PROPERTY, PHYSICAL + INSTRUMENT, COOLANT QUALITY + R AND D PROGRAM + REACTOR, LMFBR + PIPES AND PIPE FITTINGS

5-031574 ALSO IN CATEGORY 18 DATED 1963

KLICKMAN AE + NIMS JB + MCCARTHY WJ + NICHOLSON RB
SAFETY CONSIDERATIONS IN THE ENRICO FERMI REACTOR DESIGN
COMITATO NAZIONALE ENERGIA NUCLEARE
CONF-193(VOL.2) +. 38 PAGES, PP. 593-630 OF REATTORI VELUCI, FROM THE 8TH NUCLEAR CONGRESS, ROME, JUNE 1963

THE GENERAL APPROACH TAKEN TO GUARD AGAINST SEVERE REACTOR ACCIDENTS IS DESCRIBED. A HYPOTHETICAL MELTDOWN ACCIDENT IS POSTULATED AND ANALYZED FOR ENERGY RELEASE AND CONTAINMENT REQUIREMENTS. PRECAUTIONS TAKEN IN DESIGN OF THE FERMI FAST BREEDER REACTOR TO PREVENT AUTOCATALYTIC EFFECTS AND RESONANCE INSTABILITY ARE DISCUSSED.

REACTOR, BREEDER + DESIGN CRITERIA + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + ACCIDENT, HYPOTHETICAL + ITALY + SAFETY PRINCIPLES AND PHILOSOPHY + ENGINEERED SAFETY FEATURE + REACTOR, LMFBR

5-031909 DATED 1968

JOHANNSEN K
ANALYTICAL INVESTIGATION OF THE EFFECT OF VARIABLE WALL HEAT FLUX ON TURBULENT LIQUID-METAL HEAT TRANSFER COEFFICIENTS IN REACTOR COOLANT CHANNELS
MASSACHUSETTS INSTITUTE OF TECHNOLOGY
4 PAGES, ATOMKERNENERGIE 13, PAGES 21-4 (1968)

THE FORCED-CONVECTION HEAT TRANSFER TO A LIQUID METAL FLOWING TURBULENTLY BETWEEN PARALLEL PLATES WITH A VARIABLE WALL HEAT FLUX IN FLOW DIRECTION WAS INVESTIGATED USING POPPENDIEKS SIMPLIFIED MODEL. SAMPLE CALCULATIONS FOR A SINUSOIDALLY VARYING HEAT FLUX SHOW THE STRONG DEPENDENCE OF THE HEAT TRANSFER COEFFICIENT ON CHANNEL LENGTH. IN SOME CASES, THIS EFFECT COULD POSE PROBLEMS FOR SMALL LIQUID-METAL-COOLED CORES OF HIGH POWER DENSITY.

REACTOR, LMCR + METAL, LIQUID + HEAT TRANSFER, CONVECTION + THEORETICAL INVESTIGATION + MATHEMATICAL TREATMENT + HEAT TRANSFER COEFFICIENT

5-032266 ALSO IN CATEGORY 6 DATED 1968

FLOWERS CD + GERHARDSTEIN LH
ANALOG-HYBRID DYNAMIC SIMULATION OF THE FFTF REACTOR AND HEAT TRANSPORT SYSTEM
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-707 +. 101 PAGES, FIGURES, TABLES, 18 REFERENCES, APRIL 1968

DEVELOPED MATHEMATICAL MODEL FOR INVESTIGATION DYNAMIC RESPONSE AND CONTROL CHARACTERISTICS OF THE FAST FLUX TEST FACILITY REACTOR AND HEAT-REMOVAL SYSTEMS. MODEL WAS DEVELOPED FOR SOLUTION ON THE ANALOG-HYBRID COMPUTER AT PNL. REPORT DISCUSSES MODELING TECHNIQUES USED TO SIMULATE EACH SYSTEM COMPONENT, ALSO GIVES SAMPLE TRANSIENT RESULTS. DISCUSSES PROSPECTIVE USES OF THE SIMULATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFILME

COMPUTER PROGRAM + DESIGN CRITERIA + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + REACTOR KINETICS + REACTOR, TEST + THERMAL ANALYSIS + ANALYTICAL MODEL + FFTF (TR) + ANALOG SIMULATION + HEAT TRANSFER ANALYSIS

5-032526 ALSO IN CATEGORY 9 DATED 1967

VOLK ER + JENS WH + KISHIDA K + NIMS JB
MALFUNCTION DETECTION IN FAST REACTORS
ATOMIC PWR. DEVELOP. ASSOC. INC.
CONF-670916 +. 13 PAGES, PAGES VA2.1-13 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPT. 19-22, 1967

REVIEWS THE METHODS OF MALFUNCTION DETECTION IN FAST REACTORS. EQUIPMENT AND OPERATIONAL PROCEDURES REQUIRED FOR SUCH SYSTEMS ARE OUTLINED. THE PROPOSED MALFUNCTION DETECTION SYSTEM FOR THE FERMI FAST BREEDER REACTOR IS DESCRIBED. CALCULATIONAL PROCEDURES USED IN ON-LINE ANALYSES ARE DETAILED. COMPUTER AND DATA ACQUISITION SYSTEM SPECIFICATIONS ARE TABULATED.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + DATA PROCESSING + SAFETY ANALYSIS + TEST, ON LINE + REACTOR, LMFBR + FAILURE, INCIPIENT

5-032527 ALSO IN CATEGORIES 6 AND 9 DATED 1967

STACHURA SJ + BLAINE RA + BERLAND RF + HAUSKNECHT DF
AIROS-II - REACTOR DYNAMICS SIMULATOR FOR LMFBR DESIGN
ATOMIC INTERNATIONAL, CANOGA PARK, CALIF.

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-032527 *CONTINUED*
CONF-670916 +. 9 PAGES, PAGES IVA5.1-9 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

MODIFICATIONS TO THE DIGITAL KINETICS CODE, AIROS, HAVE BEEN INCORPORATED TO PROVIDE CAPABILITIES FOR TREATING PHASE CHANGES DURING TRANSIENTS SUCH AS COOLANT VOIDING AND FUEL MELTING BASED UPON SIMPLIFIED MODELS. THE GENERALIZED NATURE OF THE FEEDBACK EQUATIONS PERMITS DETAILED TREATMENT OF EITHER NEUTRONIC, THERMAL, OR FLOW BEHAVIOR.

COMPUTER PROGRAM + FUEL MELTDOWN + REACTOR DYNAMICS + VOID COEFFICIENT + ANALYTICAL MODEL + PHASE CHANGE + SIMULATION + REACTOR, LMFBR

5-032542 ALSO IN CATEGORY 18 DATED 1967
WOODFIELD FW
FAST FLUX TEST FACILITY DEVELOPMENT MONTHLY PROGRESS REPORT, OCTOBER 1967
BATTELLE NORTHWEST
BNWL-615 +. 38 PAGES, 3 REFERENCES, NOVEMBER 10, 1967

FUEL FAILURE ACTIVITY IN THE EX-REACTOR PORTION WILL EXCEED CRUD ACTIVITY BY FACTOR OF 10 (DETAILS IN BNWL-565). LIQUID NA SAMPLING TOPICAL REPORT IS BNWL-491. ***TA RESISTANCE HEATER WAS OPERATED AT 10(6TH) BTU/HK-FT² IN 300 F WATER. ***CONCEPTUAL STUDIES ON NOZZLE SEAL CLEARANCE PROGRESS. ***FUEL CLAD IRRADIATION SAMPLES HAVE BEEN SENT TO EBR-***. ***CONSTRUCTION COMPLETED ON POSTIRRADIATION CREEP-RUPTURE TEST DEVICE. ***UNIRRADIATED SERPENTINE DOES NOT LOSE WATER OR HYDROGEN ON CONTACT WITH 930 F SODIUM, ALTHOUGH IT COULD NOT BE KEPT IN CONTACT FOR LONG.

AVAILABILITY - F.W. WOODFIELD, BATTELLE-NORTHWEST, RICHLAND, WASH.

ACTIVITY BUILDUP + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR, LMCR + REACTOR, TEST + TEST, PROOF + SAMPLING + FFTF (TR) + CRUD + THERMAL INSULATION + CREEP + R AND D PROGRAM + COOLANT QUALITY + REACTOR, LMFBR

5-032623 ALSO IN CATEGORY 6 DATED 1967
MACFARLANE DR + MCNEAL N + MENELEY DA + SANATHANAN CK
THEORETICAL STUDIES OF THE RESPONSE OF FAST REACTORS DURING SODIUM BOILING ACCIDENTS
ARGONNE NATIONAL LAB., ILL.
CONF-670916 +. 27 PAGES, 16 REFERENCES, PP. 15.1-26 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

THE PHYSICAL MODELS AND EQUATIONS USED TO DESCRIBE THE KINETIC BEHAVIOR OF A LARGE FAST POWER BREEDER REACTOR UNDER NA BOILING CONDITIONS ARE DESCRIBED. THE MODELS DESCRIBED ARE THE NA VOID MODEL, HEAT TRANSFER MODEL, NEUTRON KINETICS MODEL, AND FEEDBACK REACTIVITY COEFFICIENTS. PARAMETERS FOR A REFERENCE CORE DESIGN ARE TABULATED. CALCULATED DATA SHOWN GRAPHICALLY.

REACTOR DYNAMICS + SODIUM + ANALYTICAL MODEL + REACTOR, LMFBR + BOILING

5-032624 DATED 1967
OYAMA A + AN S + KONDO S
ANALYSIS OF FAST REACTOR CORE MELTDOWN ACCIDENTS
TOKYO UNIVERSITY
CONF-670916 +. 16 PAGES, 15 REFERENCES, PP. III4.1-15 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

ENERGY RELEASE DURING CORE MELTDOWN ACCIDENT IS EXAMINED FOR A TYPICAL LARGE FAST BREEDER REACTOR USING CODE FASTAC, DEPENDENCE OF RELEASED ENERGY UPON SOME OF THE TYPICAL CORE PARAMETERS. CHANGE OF CORE SHAPE (NEGLECTED IN THE SO-CALLED MODIFIED BETHE-TAIT METHOD) AND CALCULATED ENERGY RELEASES SHOW GOOD AGREEMENT WITH VALUES BY OTHERS. THE DEPENDENCE OF ENERGY UPON VOLUME IS LINEAR WHEN THE CHANGE OF THE PROMPT NEUTRON LIFETIME AND THE POWER PEAKING FACTOR ARE TAKEN INTO CONSIDERATION. *** PARTIAL MELTDOWN IS ALSO ANALYZED, VARYING THE SPATIAL DISTRIBUTION OF THE INITIAL TEMPERATURE. THE AVAILABLE ENERGY DOES NOT SHOW A MARKED DECREASE IRRESPECTIVE OF THE INITIAL MELTDOWN AREA, AND THE MELTDOWN IS LIMITED TO THE LOCAL PART OF THE CORE. NO DANGEROUS AUTOCATALYTIC BEHAVIOR IS FOUND EVEN WITH COMPACTION OF THE FUEL.

BETHE TAIT + DOPPLER EFFECT + REACTOR, FAST + COMPARISON, THEORY AND EXPERIENCE + ANALYTICAL MODEL + CORE MELTDOWN + REACTOR, LMFBR

5-032627 ALSO IN CATEGORIES 9 AND 17 DATED 1968
REPORT ON THE FUEL MELTING INCIDENT IN THE ENRICO FERMI ATOMIC POWER PLANT
POWER REACTOR DEVELOPMENT COMPANY
50 PAGES, FIGURES, TABLES, REFERENCES, DECEMBER 15, 1968, Docket 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

DESCRIBES THE OCT. 5 HAPPENINGS (FLOW BLOCKAGE DUE TO ZIRCONIUM PLATE CAUSED PARTIAL MELTDOWN, RELEASING FISSION GAS AND LOSING EXCESS REACTIVITY), RESULTS OF INVESTIGATIONS (ZIRCONIUM FROM CONICAL FLOW GUIDE, FUEL FAILURE DID NOT PROPAGATE, FLOW REDUCTIONS CAME ONLY FROM ZIRCONIUM SEGMENT), PLANT MODIFICATION (ON-LINE COMPUTER WILL ALARM ON DEVIATION FROM CALCULATED VALUE OF REACTIVITY BALANCE, FISSION PRODUCTS, OR SUBASSEMBLY DELTA T AND WILL LATER CAUSE SCRAM). STARTUP TEST PROGRAM OUTLINED. APPENDICES GIVE DETAILS ON FUEL EXAMINATION. *** OCT. 5 OPERATION WAS TO CHECK EARLIER HIGH DELTA T IN ELEMENTS WHICH MELTED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-032627 *CONTINUED*
CHARGE \$2.00

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + INCIDENT + INSTRUMENT, TEMPERATURE + REPORT, OPERATIONS ANALYSIS + INCIDENT, RECOVERY FROM + EXAMINATION + REMOTE MANIPULATING AND VIEWING + FLOW BLOCKAGE + INSTRUMENT, ABNORMAL INDICATION + INSTRUMENT, OPERATING REACTIVITY + COMPUTER CONTROL + REACTOR, LMFBR + FUEL ELEMENTS

5-032629 ALSO IN CATEGORIES 11 AND 17 DATED 1968
EXAMINATION OF MELTED FERMI FUEL
POWER REACTOR DEVELOPMENT COMPANY
55 PAGES, FIGURES, TABLES, APPENDIXES B AND C OF REPORT ON THE FUEL MELTING INCIDENT, DECEMBER 15, 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

AN UNMELTED ELEMENT SHOWED ABOUT 200 PPM HYDROGEN IN CLAD (WELL BELOW 5000 SPALLING LEVEL), HIGH VALUES FROM A PARTICULAR MELT INQUIRY. OTHERWISE NORMAL. ANOTHER ELEMENT THERMOCOUPLE READING HIGHER THAN A MELTED ELEMENT, WAS ONLY BADLY BOWED. *** MOLTEN FUEL ADHERED TO STEEL CLAD BUT DID NOT PENETRATE. HYDROGEN CONTENT REACHED 4000 PPM WITHOUT CRACKING. ONLY Xe-KR FOUND IN COVER GAS. CS, SR, I, BA-LA FOUND IN SODIUM COOLANT. WHILE THERE WAS A COMMON HOLE IN THE WRAPPER CANS BETWEEN MELTED ELEMENT, NO EVIDENCE OF INTERCHANGE OF FUEL. FLOW TESTS AND HEAT TRANSFER ANALYSIS SHOWED ABOUT 3% NOMINAL FLOW IN TWO MELTED ELEMENTS, 30 AND 7% FLOW IN DAMAGED ADJACENT ELEMENTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (108 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + SODIUM + EMBRITTLEMENT + HYDRIDE + REPORT, OPERATIONS ANALYSIS + HEAT TRANSFER ANALYSIS + ZIRCALOY + FISSION PRODUCT RETENTION + EXAMINATION + FLOW BLOCKAGE + STRUCTURAL INTEGRITY + REACTOR, LMFBR

5-032815 ALSO IN CATEGORY 18 DATED 1968
CORE DESIGN AND ANALYSIS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-10 +. 175 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 10, FEB. 29, 1968, DOCKET 50-231

CONTAINS DETAILED DESCRIPTION OF MECHANICAL DESIGN AND ANALYSIS OF CORE. EXPANDED VERSION OF INFORMATION IN SECT. IV OF PSAR. THE MECHANICAL ANALYSIS IS DESCRIBED IN SECTION I, DESIGN IN SECTION II. SECTION III CONTAINS ADDITIONAL PHYSICS INFORMATION WHICH SUPPLEMENTS PSAR. A SUMMARY OF CODES USED IN DESIGN OF CORE IS INCLUDED. SUMMARIES OF THE MORE SIGNIFICANT RESULTS OF R AND D EFFORT HAVE EITHER BEEN TRANSMITTED AS SEPARATE SUPPLEMENTS (SUPPLEMENTS 1, 3, AND 4), ARE SUMMARIZED IN APPENDIX OF THIS SUPPLEMENT OR ARE REFERRED TO.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (108 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, FAST + HYDRAULIC ANALYSIS + REACTOR, LMCR + THERMAL ANALYSIS + REACTOR, PULSED + SEFOR (RE) + HEAT TRANSFER ANALYSIS + REPORT, PSAR + REACTOR, LMFBR + FUEL ELEMENTS

5-032819 ALSO IN CATEGORIES 17 AND 18 DATED 1967
FLUID FLOW AND VIBRATION TESTS ON 1/2 SCALE VESSEL MODEL
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-4 +. 60 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 4, DEC. 5, 1967, DOCKET 50-231

WATER, AT TEMPERATURES UP TO 265 F, WAS USED TO SIMULATE COOLANT CONDITIONS IN THE SODIUM-COOLED REACTOR. FLOWS WERE VARIED FROM 10 TO 100 PERCENT OF RATED REYNOLDS NUMBER IN THE REACTOR. DATA WAS OBTAINED ON RELATIVE CORE FLOW DISTRIBUTION, FLOW PATTERNS, PRESSURE LOSSES, AND VIBRATION CHARACTERISTICS OF THE CANTILEVER-MOUNTED CORE SHROUD. FLUID FLOW RESULTS PROVIDED INFORMATION NECESSARY FOR VERIFICATION OF FINAL DESIGN. VIBRATION RESULTS INDICATED NO SERIOUS EFFECTS OF AMPLITUDE OR STRESS IN THE FULL-SCALE REACTOR.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (108 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, FAST + FLOW THEORY AND EXPERIMENTS + REACTOR, LMCR + PRESSURE VESSELS + TEST, PRESSURE VESSEL + SODIUM + TESTING + REACTOR COOLANT + METAL, LIQUID + REACTOR, PULSED + SEFOR (RE) + PRESSURE DROP + FLOW DISTRIBUTION + VIBRATION + REPORT, PSAR + MODEL TESTING + FLOW + REACTOR, LMFBR

5-032836 ALSO IN CATEGORIES 9 AND 11 DATED 1969
RYDEN J
NONDESTRUCTIVE TESTING OF SMALL DIAMETER, STAINLESS STEEL FUEL CLAD TUBING
BATTELLE-NORTHWEST, RICHLAND
BNWL-SA-2275 + CONF-690311-1 +. 20 PAGES, FROM NATIONAL SPRING CONFERENCE OF THE AMERICAN SOCIETY FOR NONDESTRUCTIVE TESTING, LOS ANGELES, 1969

DISCUSSES THE TECHNIQUES USED BY THE PACIFIC NORTHWEST LAB. TO INSPECT FFTF FUEL CLADDING FOR DEFECTS. DESCRIBES THE THREE DIFFERENT TESTERS USED TO INSPECT THIS TUBING. THEY ARE A SINUSOIDAL EDDY CURRENT TESTER, AN ULTRASONIC PULSE-ECHO TESTER, AND AN ULTRASONIC TESTER THAT USES A SEND-RECEIVE SYSTEM OF TRANSDUCERS. MICROGRAPHS, STRIP CHARTS, AND STATISTICAL DATA OBTAINED FROM THE INSPECTION OF 20,000 FEET OF TUBING ARE PRESENTED.

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-032836 *CONTINUED*
AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + TESTING + STEEL, STAINLESS + FFTF (TR) + INSTRUMENT
CALIBRATION + TEST, NONDESTRUCTIVE + R AND D PROGRAM + TUBING

5-032853 ALSO IN CATEGORY 12 DATED 1968
PADILLA A
ANALYSIS OF COOLANT MIXING IN THE FFTF FUEL BUNDLE
BATTELLE NORTHWEST, RICHLAND, WASHINGTON
BNWL-SA-1688 + CONF-660832-3 +. 32 PAGES, 17 FIGURES, 1 TABLE, 8 REFERENCES, MARCH 14, 1968, FROM 10TH
NATIONAL HEAT TRANSFER CONFERENCE, PHILADELPHIA, PA.

PRESENTS SOME OF THE COOLANT-MIXING STUDIES IN THE FUEL BUNDLE WHICH WERE CARRIED OUT DURING THE
CONCEPTUAL STAGE IN THE DESIGN OF THE LTR. THE PURPOSE OF THESE STUDIES WAS TO DETERMINE THE
TEMPERATURE AND FLOW DISTRIBUTION IN THE FUEL BUNDLE, PROVIDE A TECHNICAL BASIS FOR
INSTRUMENTATION OF THE DRIVER SUBASSEMBLY, AND IDENTIFY THERMAL-HYDRAULIC PROBLEM AREAS IN THE
DESIGN OF THE DRIVER ASSEMBLY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + TEMPERATURE GRADIENT + FLOW DISTRIBUTION +
EXPERIMENT + COOLANT QUALITY

5-032876 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 1 - GUILLOTINE PIPE BREAK
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 4 PAGES, 4 FIG., 2 TAB., PAGE 1 THRU 8 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DEVELOP IN MORE DETAIL THE CONSEQUENCES OF THE GUILLOTINE-TYPE BREAK IN THE MAIN PRIMARY COOLANT
SYSTEM DISCUSSED IN SECTION 16.3.3.2.3.3 OF THE FDSAR. PROVIDE BASIS FOR CONCLUSION THAT
CIRCULATION WOULD BE RESUMED THROUGH THE REACTOR VESSEL AND AUXILIARY COOLANT LOOP FOLLOWING THIS
ACCIDENT. (ANSWER ON PAGES 1 THRU 8).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + SODIUM + REACTOR, PULSED
+ SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

5-032679 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 2 - MAIN OUTLET THERMAL SLEEVE FAILURE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 2 PAGES, 1 FIG., PAGE 9 THRU 11 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

IF THE FASTENER OF THE SLEEVE IN THE REACTOR VESSEL MAIN PRIMARY OUTLET NOZZLE SHOULD FAIL, THE
SLEEVE COULD MOVE AND CAUSE AN INCREASE IN THE AMOUNT OF COOLANT FLOW BYPASSING THE CORE.
DESCRIBE THE EVENTS FOLLOWING MOVEMENT OF THIS SLEEVE. INCLUDE THE METHOD OF DETECTION, MEANS OF
SCRAMMING THE REACTOR AND THE TEMPERATURE TIME HISTORY OF THE CORE COOLANT. (ANSWER ON PAGES 9
THRU 11).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + REACTOR, PULSED + SEFOR (RE) + NOZZLE + RESPONSE TO
NRC QUESTION + REPORT, PSAR + DISPLACEMENT + FLOW, BYPASS + COOLING + REACTOR, LMFBR

5-032882 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 5 - BLOCKAGE OF MAIN OUTLET NOZZLE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 2 PAGES, 1 FIG., PAGE 16 AND 17 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DISCUSS THE EFFECT OF BLOCKAGE OF THE UPFLOW PASSAGE LEADING TO THE REACTOR VESSEL MAIN COOLANT
FLOW EXIT NOZZLE. INCLUDE THE TIME HISTORY OF THE CORE COOLANT TEMPERATURE. ***ANSWER ON PAGES
16 AND 17.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + REACTOR, PULSED + SEFOR (RE)
+ RESPONSE TO NRC QUESTION + FLOW BLOCKAGE + REPORT, PSAR + REACTOR, LMFBR

5-032883 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 6 - SPENT FUEL STORAGE TANK SHIELD PLUG DROP
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-032883 *CONTINUED*
50231-20 +. 1 PAGE, PAGE 18 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

ANALYZE THE CONSEQUENCES OF DROPPING A SHIELD PLUG ON THE SPENT FUEL STORAGE TANK. ***ANSWER ON PAGE 18.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REFUELING + FUEL STORAGE + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

5-032888 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION 11 - DYNAMIC LOADS FROM FRED TESTS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 6 PAGES, 1 TAB., PAGES 35 THRU 40 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DISCUSS DYNAMIC LOADINGS IMPOSED BY MAXIMUM PLANNED TRANSIENT ON THE CORE SUPPORT STRUCTURE AND REACTOR VESSEL. ***ANSWER ON PAGES 35 THRU 40.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + CONTAINMENT VESSEL LOADING + REACTOR, FAST + REACTOR, LMCR + PRESSURE VESSELS + REACTOR TRANSIENT + STRESS + THERMAL MECHANICAL EFFECT + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + CORE COMPONENTS + REACTOR, LMFBR

5-032893 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 16 - VERTICAL STROKE OF TEST DEVICE POSITIONER
GENERAL ELECTRIC
50231-20 +. 1 PAGE, PG. 52 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

DISCUSS HOW STROKE OF EXCURSION DEVICE POSITIONER IS INTERLOCKED WITH DEVICE MOUNTED ON IT IN ORDER TO AVOID INADVERTENTLY LOWERING THE POISON ROD MORE THAN PLANNED, THEREBY INCREASING THE LIKELIHOOD OF GREATER REACTIVITY CHANGES THAN ANTICIPATED, AND DAMAGE TO THE DRY WELL THIMBLE. *** (ANSWER) LOWEST POSSIBLE POINT OF POISON ROD IS SAME WITH EITHER OSCILLATOR OR FRED INSTALLED. IN OPERATION WITH OSCILLATOR DEVICE, THE POSITIONER USES 20 IN. OR LONG GUIDE BARS WITH A RING MOUNTED ON THE TOP. FAILURE OF UPPER LIMIT SWITCH WOULD MEAN DEVICE WOULD BE DRIVEN UP AGAINST RING, AND MOTOR WOULD STALL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REACTIVITY + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + MEASUREMENT, REACTIVITY + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + INSTRUMENTS, MISC.

5-032895 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 18 - CONTROL OF POISON SLUGS USED WITH TEST DEVICES
GENERAL ELECTRIC
50231-20 +. 1 PAGE, PG. 55 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

DESCRIBE THE METHOD TO ENSURE THAT PROPER POISON ROD FOR OSCILLATOR AND FRED EXPERIMENTS WILL ALWAYS BE INSTALLED PRIOR TO ACTUAL EXPERIMENT. *** (ANSWER) WITH EXCEPTION OF ROD DIAMETER, OSCILLATOR AND FRED POISON RODS ARE COMPLETELY DIFFERENT. ATTACHMENT OF OSCILLATOR ROD TO MECHANISM IS ACCOMPLISHED BY A DIFFERENT GRAPPLING MEANS THAN FRED POISON ROD WHICH IS ATTACHED TO THE PISTON. PROCEDURAL CONTROL USED TO INSURE PROPER FRED POISON ROD IS INSTALLED. REACTIVITY WORTH OF POISON ROD WILL BE VERIFIED AS IT IS LOWERED INTO CORE WITH POSITIONER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT + ADMINISTRATIVE CONTROL + REACTOR, FAST + FAILURE, OPERATOR ERROR + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + CONTROL RODS

5-032903 ALSO IN CATEGORIES 12 AND 18 DATED 1969
SIMULTANEOUS FAILURES - LOCA AND SAFEGUARD
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231 +. 7 PAGES, 1 FIGURE, SEFOR, PSAR, SUPPLEMENT 23, FEB. 20, 1969, DOCKET 50-231.

DESCRIBES MODIFICATIONS MADE TO MITIGATE CONSEQUENCES OF TWO SIMULTANEOUS FAILURES, I.E., A FAILURE IN THE PRIMARY COOLANT SYSTEM FOLLOWED BY A FAILURE OF ONE OF THE SYSTEMS OR COMPONENTS (ENGINEERED SAFEGUARDS FEATURES). GIVES SUMMARY OF PROTECTION AND INHERENT DESIGN FEATURES OF THESE SAFEGUARDS THAT ASSURE OPERABILITY IN THE ENVIRONMENT IN WHICH THEY ARE REQUIRED TO OPERATE FOLLOWING A PRIMARY SYSTEM FAILURE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-032903 *CONTINUED*
REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + MODIFICATION + REACTOR, PULSED + SEFOR (RE) +
REPORT, PSAR + FAILURE, SIMULTANEOUS + SYSTEM OPERABILITY IN ACCIDENT + ENGINEERED SAFETY FEATURE + REACTOR,
LMFBR

5-032904 ALSO IN CATEGORIES 12 AND 18 DATED 1969
RESPONSE TO AEC QUESTION
GENERAL ELECTRIC
50231-26 +. 7 PAGES, SEFOR, PSAR, SUPPLEMENT 22, JANUARY 24, 1969, DOCKET 50-231

SUPPLEMENTARY INFORMATION GIVEN TO PREVIOUSLY ANSWERED QUESTIONS CONCERNING THE CONSEQUENCES OF
COOLANT SYSTEM LEAKS AND RESULTING SYSTEM CHARACTERISTICS (SEE COMMENTS 1 IN SUPPLEMENT 18 AND
PAGE 53 OF SUPPLEMENT 19), SPECIFIC TOPICS AND SMALL LEAKS IN MAIN COOLANT SYSTEM, AND
EVALUATION OF EQUIPMENT NEEDED TO RECOVER FROM A GUILLOTINE PIPE FAILURE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

NRC QUESTION + ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + INCIDENT,
RECOVERY FROM + REACTOR, PULSED + SEFOR (RE) + EQUIPMENT + REPORT, PSAR + REACTOR, LMFBR

5-032905 ALSO IN CATEGORIES 11 AND 18 DATED 1969
RESPONSE TO QUESTION 1 - SMALL LEAKS IN MAIN COOLANT SYSTEM
GENERAL ELECTRIC
50231-26 +. 3 PAGES, PAGES 1 THRU 3 OF SEFOR, PSAR, SUPPLEMENT 22, JAN. 24, 1969, DOCKET 50-231

DISCUSS CONSEQUENCES AND SYSTEM RESPONSE TO SMALL LEAKS IN THE MAIN COOLANT SYSTEM. *** ANSWER
ON PAGES 1 THRU 3.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + REACTOR, PULSED + SEFOR
(RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + MAIN COOLING SYSTEM + LEAK + REACTOR, LMFBR + PIPES AND PIPE
FITTINGS

5-032906 ALSO IN CATEGORIES 12 AND 18 DATED 1969
RESPONSE TO QUESTION 2 - RECOVERY EQUIPMENT FOR LOCA PIPE FAILURE
GENERAL ELECTRIC
50231-26 +. 4 PAGES, PG. 4 THRU 7 OF SEFOR, PSAR, SUPPLEMENT 22, JAN. 24, 1969, DOCKET 50-231

DISCUSS EQUIPMENT NEEDED TO RECOVER FROM A GUILLOTINE PIPE FAILURE *** ANSWER ON PAGES 4 THRU 7.
SEE ALSO COMMENT 1 IN SUPPLEMENT 18.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + INCIDENT, RECOVERY FROM + REACTOR, PULSED + SEFOR
(RE) + RESPONSE TO NRC QUESTION + EQUIPMENT + REPORT, PSAR + REACTOR, LMFBR

5-032920 ALSO IN CATEGORIES 9 AND 18 DATED 1968
RESPONSE TO QUESTION 5 - COOLANT LEAK EFFECT ON CONTROL
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 4 PAGES, 1 FIGURE, PAGES 52 THRU 56 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

DISCUSS POSSIBILITY OF SODIUM LEAK IN PRIMARY COOLANT SYSTEM THAT MIGHT INTERFERE WITH OPERATION
OF REFLECTOR CONTROL SYSTEM. *** ANSWER ON PAGES 52 THRU 57.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + CONTROL SYSTEM + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + REACTOR
CONTROL + REACTOR, PULSED + SEFOR (RE) + REFLECTOR + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR
+ CONTROL RUDS

5-032922 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 7 - COOLING CAPABILITY WITH OUTLET NOZZLE FAILURE
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 2 PAGES, PAGES 63 AND 64 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL INFORMATION ON COOLING CAPABILITIES FOR FAILURE OF FASTENER OF THERMAL SLEEVE
IN VESSEL MAIN PRIMARY OUTLET NOZZLE. *** ANSWER ON PAGES 63 AND 64.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + FAILURE, PRESSURE VESSEL + REACTOR, PULSED + SEFOR (RE) + NOZZLE + RESPONSE TO

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-032922 *CONTINUED*
NRC QUESTION + REPORT, PSAR + SYSTEM OPERABILITY IN ACCIDENT + MAIN COOLING SYSTEM + SYSTEM CAPABILITY + REACTOR, LMFBR

5-032926 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION 11 - SODIUM IN CORE - EFFECT ON MHA
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 1 PAGE, PAGE 73 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

DISCUSS EFFECT OF SODIUM IN CORE REGION ON MHA ANALYSIS. *** ANSWER ON PAGE 73.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + REACTOR STABILITY + SODIUM + ACCIDENT, HYPOTHETICAL + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + FUEL ELEMENTS

5-033056 ALSO IN CATEGORIES 8 AND 12 DATED 1967
POPLAVSKY VM + ARISTARKOV NN + BAGDASAROV YE + BLAGOVOLIN SM + VYBLOV NF + PINKHASIK MS
SAFETY PROBLEMS WITH SINGLE WALL SODIUM/WATER HEAT EXCHANGERS FOR FAST REACTORS
UKAEA, DOUNREAY EXPERIMENTAL REACTOR ESTABLISHMENT
CONF-670916-4 +. 26 PAGES, 4 FIGURES, 6 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPTEMBER 19-22, 1967

SUMMARY OF RUSSIAN WORK REGARDING SODIUM-WATER REACTIONS DUE TO A TUBE RUPTURE IN A SINGLE-WALL STEAM GENERATOR. DISCUSSION INCLUDED EXPERIMENTAL WORK AND OTHER FACTORS REGARDING THIS REACTION INFLUENCING THE DESIGN OF THE BN-350 AND THE BOR STEAM GENERATORS. FACTORS CONSIDERED WERE SHOCK, HYDRODYNAMIC PHENOMENA IN THE SODIUM LOOP INCLUDING STRUCTURAL LOADING, THERMAL EFFECTS, EFFICIENCY OF THE REACTION PRODUCT EVACUATION SYSTEM, AND DETERIORATION OF THE STRUCTURAL COMPONENTS.

CHEMICAL REACTION + METAL WATER REACTION + SODIUM + STEAM + WATER + PRESSURE RELIEF + USSR + METAL, ALKALI + METAL, LIQUID + EQUIPMENT DESIGN + STEAM GENERATOR + PRESSURE PULSE + REACTOR, LMFBR + HEAT EXCHANGERS

5-033281 DATED 1968
AMENDOLA A
A STATISTICAL METHOD FOR EVALUATION OF HOT CHANNEL FACTORS IN REACTOR DESIGN
KERNFORSCHUNGSZENTRUM, KARSRUHE
EURFNR-604 + KFK-843 + EUR-3979 +. 44 PAGES, 10 FIGURES, 18 REFERENCES, JULY 1968

PREVIOUS METHODS OF HOT-CHANNEL ANALYSIS ASSUMED THAT ALL COOLING CHANNELS WERE INDEPENDENT. THERE ARE TWO TYPES OF UNCERTAINTIES, LOCAL AND GLOBAL. THE GLOBAL UNCERTAINTIES INTRODUCE EVIDENTLY A CORRELATION AMONG TEMPERATURES OF THE COOLANT IN THE CHANNELS THEY EFFECT. GIVES A QUANTITATIVE ASSESSMENT OF THE DIFFERENT EFFECTS OF THESE TWO TYPES OF UNCERTAINTIES AND PROPOSES METHOD OF EVALUATING THE HOT CHANNEL FACTOR, TAKING INTO ACCOUNT THIS CORRELATION WITH REFERENCE TO SODIUM COOLED FAST REACTORS, BUT IS QUITE GENERAL AND MAY READILY BE APPLIED TO OTHER REACTOR TYPES.

AVAILABILITY - U. S. DEPOSITORY LIBRARIES IN THE U. S. AND OVERSEAS

EURATOM + ANALYTICAL TECHNIQUE + HOT CHANNEL ANALYSIS + GERMANY + HOT CHANNEL FACTOR + STATISTICAL ANALYSIS + REACTOR, LMFBR

5-033333 ALSO IN CATEGORIES 6 AND 18 DATED 1967
PETERSON RE + SIMPSON DE + LITTLE WW
SELECTED SAFETY CONSIDERATIONS IN DESIGN OF THE FAST FLUX TEST FACILITY
BATTTELLE MEMORIAL INSTITUTE
BNWL-SA-1236 +. CONF-670916 +. 10 FIGURES, 4 TABLES, 9 REFERENCES, JUNE 1967, PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS

REVIEW OF SKEWED-TUBE DESIGN. LOW FERTILE CONTENT GIVES LOW DOPPLER COEFFICIENT. FEW CORE LOOPS, HIGH-WORTH RODS REQUIRE LOOPS OUTSIDE CORE. REFLECTOR INSTEAD OF BREEDING BLANKET MAXIMIZES NEUTRON FLUX. SUMMARIZES RESULTS OF VARIOUS PRELIMINARY TRANSIENTS OR ACCIDENTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, TEST + FFTF (TR) + R AND D PROGRAM + REACTOR, LMFBR

5-033427 ALSO IN CATEGORIES 6 AND 18 DATED 1969
GOLDEN GH
ELEMENTARY NEUTRONICS CONSIDERATIONS IN LMFBR DESIGN
ARGONNE NATIONAL LABORATORY
ANL-7532 +. 68 PAGES, 12 FIGURES, 10 TABLES, 35 REFERENCES, MARCH 1969

THE SUBJECT MATERIAL PROGRESSES FROM BASIC PHYSICS CONCEPTS TO THE DESIGN OF REAL LMFBRs - FLUX SPECTRA AND CROSS-SECTION VARIATION WITH ENERGY IN FAST REACTORS, ASPECTS OF MULTIGROUP

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-033427 *CONTINUED*

CALCULATIONS IN DIFFUSION THEORY, THE SIGNIFICANCE OF BREEDING AND CONVERSION RATIOS, SPECIFIC POWER, AND DOUBLING TIME, RADIAL CORE ZONING, REFUELING (I.E. TIME-DEPENDENT BEHAVIOR), A DISCUSSION OF REACTIVITY EFFECTS AS RELATED TO SAFETY, WITH SPECIFIC REFERENCE TO SODIUM VOID, DOPPLER, AND EXPANSION EFFECTS, AND CURRENT DESIGN APPROACHES. PRESENTATION IS APPLICABLE TO DESIGN OF ANY FAST BREEDER, REGARDLESS OF COOLANT. HENCE, PERTINENT COMMENTS WERE MADE REGARDING GAS-COOLED FAST-BREEDER REACTORS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT ANALYSIS + REACTOR, BREEDER + DOPPLER EFFECT + REACTOR, FAST + REACTIVITY COEFFICIENT + REACTIVITY EFFECT + REACTOR DYNAMICS + REACTOR KINETICS + VOID COEFFICIENT + ANL + DESIGN STUDY + REACTOR PHYSICS + R AND D PROGRAM + REACTOR, HTGR + REACTOR, LMFBR

5-033542 ALSO IN CATEGORY 18 DATED 1968
KATO WY + KUSCH GK + DATES LR + TILL CE + VAN DOORNINCK J + CHEEVER CL
FINAL SAFETY ANALYSIS REPORT ON THE USE OF PLUTONIUM IN ZPR-6 AND -9
ARGONNE NATIONAL LAB.
ANL-7442 +. 303 PAGES, FIGURES, TABLES, REFERENCES, JULY 1, 1968

THE ZPR-6 AND -9 ARE SPLIT-TABLE CRITICAL FACILITIES USED FOR STUDY OF FAST-REACTOR PHYSICS SINCE JULY 1963. MODIFICATIONS WERE MADE SO THAT PLUTONIUM MAY BE USED AS FUEL TO PROVIDE EXPERIMENTAL PHYSICS DATA FOR THE LMFBR PROGRAM. THE DESIGN-BASIS ACCIDENT RESULTS IN MELTING AND BURNING 60 KG OF PU AND VAPORIZATION OF 22 KG OF SODIUM. OFF-SITE DOSE FROM NOBLE GASES AND RADIOIODINE ARE - 190 REMS AT 225 METERS AND 13 REMS AT 1500 METERS.

CRITICAL ASSEMBLY + PLUTONIUM + REPORT, SAR + ZPR 9 (CAF) + ZPR 6 (CAF) + HAZARDOUS ANALYSIS + R AND D PROGRAM + REACTOR, LMFBR

5-033944 ALSO IN CATEGORY 6 DATED 1969
LONG JK
ANALYSIS OF THE STATIC POWER COEFFICIENT OF EBR-II WITH REDUCED COOLANT FLOW
ARGONNE NATIONAL LABORATORY
21 PAGES, NUCLEAR APPLICATION 6, PAGES 116-26 (FEB. 1969)

A STUDY OF EBR-II OPERATING BEHAVIOR, WHICH BEGAN WITH A REEXAMINATION OF THE THERMAL ANALYSIS OF FUEL PINS UNDER VARYING RATES OF COOLANT FLOW, INCORPORATES THE RECENT INTERPRETATIONS OF THE BOWING BEHAVIOR OF THE SUBASSEMBLIES. THE NONLINEAR BOWING EFFECTS ARE COMBINED WITH THE LINEAR EXPANSION EFFECTS IN EQUATIONS THAT EXPRESS THE POWER-REACTIVITY DECREMENT AS A FUNCTION OF BOTH POWER AND FLOW. COMPARISON WITH EXPERIMENTS INDICATES AGREEMENT TO WITHIN A FEW HOURS. DIFFERENTIATION OF THE EQUATIONS LEADS TO EXPRESSIONS FOR POWER AND FLOW COEFFICIENTS OF REACTIVITY. THE REDUCED FLOW DATA ARE USED TO DETERMINE AN EFFECTIVE AVERAGE THERMAL EXPANSION COEFFICIENT FOR THE FUEL.

REACTIVITY EFFECT, EXPANSION + REACTOR, FAST + FUEL ELEMENT BOWING + REACTOR, LMCR + POWER COEFFICIENT + REACTIVITY COEFFICIENT + REACTOR, TEST + COOLANT COEFFICIENT + FUEL EXPANSION COEFFICIENT

5-033977 ALSO IN CATEGORIES 11 AND 18 DATED 1967
DENIELOU G + ROZENHOLC M + STORRER F + VAUTREY + VELLENEUVE J
PHENIX - MAIN SAFETY CRITERIA - CONSEQUENCES FOR THE REACTOR DESIGN
CEA-CEN/CADARACHE/DRP, CEA-CEN/SACLAY/DRP, EURATOM-CEN/CADARACHE/DRP
12 PAGES, 3 FIGURES, 4 REFERENCES, PAPER PRESENTED AT THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

DISCUSSES INFLUENCE OF SAFETY CONSIDERATIONS ON THE DESIGN OF PHENIX PROTOTYPE. DISCUSSES CORE CULLING FOR VARIOUS ACCIDENTS, CONTAINMENT OF FISSION PRODUCTS, AND MEANS FOR DETECTING LOCAL REACTIVITY AND TEMP. MEASUREMENTS.

AVAILABILITY - F. STORRER, EURATOM-CEN/CADARACHE/DRP

DESIGN CRITERIA + FRANCE + SODIUM + DESIGN STUDY + SAFETY PRINCIPLES AND PHILOSOPHY + REACTOR, LMFBR

5-033984 DATED 1967
PEPPLER W + SCHLECHTENDAHL EG + SCHULTHEISS GF + SMIDT D
SODIUM BOILING AND FAST REACTOR SAFETY
KERNFORSCHUNGSZENTRUM, KARLSRUHE, WEST GERMANY
EURFNR-353 + KFK-612 +. 24 PAGES, JUNE 1967

FOR FAST SODIUM-COOLED REACTOR SAFETY ANALYSIS, LIQUID-METAL SUPERHEAT AND COOLANT FLASHING IS VERY IMPORTANT. INVESTIGATION OF THE EJECTION MECHANISM WITH THE DIGITAL CODE BLOW SHOWS GOOD AGREEMENT WITH EXPERIMENTAL RESULTS. PRESENT KNOWLEDGE ON SODIUM SUPERHEAT IS REVIEWED AND DISCUSSED IN SOME DETAIL, ESPECIALLY TO DEMONSTRATE THE GREAT DIFFERENCES IN EXPERIMENTAL DATA, AND OUTLINES THE GERMAN FAST BREEDER. ALSO RECONDENSATION EFFECTS WITH THEIR HIGH PRESSURE PEAKS ARE INVESTIGATED. THEORETICAL AND EXPERIMENTAL RESULTS ARE PRESENTED.

AVAILABILITY - USAEC DEPOSITORY LIBRARIES FOR THE U.S. AND OVERSEAS

COMPUTER, DIGITAL + REACTOR, FAST + HYDRAULIC ANALYSIS + REACTOR, LMCR + SODIUM + COMPARISON, THEORY AND

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-033984 *CONTINUED*
EXPERIENCE + THERMAL ANALYSIS + THERMODYNAMICS + CONDENSATION + GERMANY + METAL, LIQUID + R AND D PROGRAM + REACTOR, LMFBR + BOILING

5-034496 DATED 1967
BUBBLE GROWTH MEASUREMENTS IN UNIFORMLY SUPERHEATED LIQUIDS
ATOMIC ENERGY RESEARCH ESTABLISHMENT, HARWELL (ENGLAND)
AERE-R-5553 +. 28 PAGES, 13 FIGURES, 1 TABLE, 18 REFERENCES, AUGUST 1967

BUBBLE GROWTH MEASUREMENTS WERE MADE IN UNIFORMLY, SUPERHEATED WATER AT PRESSURES FROM 1/2 TO 1.2 ATMOSPHERES WHICH SUPPLEMENT THE DATA OF DERGABEDIAN AND OF HOOPER AND ABDELMESSIH. THE DATA IN THE ASYMPTOTIC PHASE OF SPHERICAL GROWTH SHOW GOOD AGREEMENT WITH THE THEORY OF PLESSET AND ZWICK. OVER THE COMPLETE GROWTH AN EXTENDED THEORY IS FOUND TO BE IN GOOD AGREEMENT WITH THE DATA. VAPOUR SLUG GROWTH, WHEN SPHERICAL VAPOUR BUBBLES ELONGATE DUE TO THE RESTRAINING WALLS OF THE APPARATUS, WAS ALSO INVESTIGATED. THE RATE OF GROWTH OF THESE SLUGS IS GREATLY INCREASED BY A THIN FILM OF EVAPORATING SUPERHEATED LIQUID LEFT ON THE WALLS OF THE APPARATUS.

AVAILABILITY - SCIENTIFIC ADMINISTRATION OFFICE, ATOMIC ENERGY RESEARCH ESTABLISHMENT, HARWELL, DIDCOT, BERKSHIRE, ENGLAND

HEAT TRANSFER, BOILING + HEAT TRANSFER + REACTOR, LMCR + METAL, LIQUID + FLOW, TWO PHASE + HEAT TRANSFER EXPERIMENT + NUCLEATE BOILING + REACTOR, LMFBR + BOILING

5-034497 DATED 1967
TURBULENT VELOCITY DISTRIBUTION IN A ROD BUNDLE
GENERAL ELECTRIC CO., SUNNYVALE, CALIFORNIA
GEAP-5411 +. 35 PAGES, 19 FIGURES, 4 REFERENCES, OCTOBER 1967

A METHOD TO DETERMINE THE TWO-DIMENSIONAL VELOCITY DISTRIBUTION FOR TURBULENT, CONSTANT PROPERTY SINGLE-PHASE FLOW THROUGH AN INFINITE TRIANGULAR ROD BUNDLE IS PRESENTED. TURBULENT MOMENTUM TRANSPORT IS DIRECTLY CALCULATED VIA A MIXING LENGTH THEORY BASED ON ACTUAL FLOW GEOMETRY. THE VELOCITY SOLUTION VECTOR IS DETERMINED BY ITERATION BETWEEN THE MOMENTUM EQUATION AND THE DEFINITION OF TURBULENT MOMENTUM DIFFUSIVITY. CORRELATION OF THE RESULTS SHOWS THE METHOD TO BE CONSISTENT WITH EXPERIMENTAL DATA AND PREFERABLE TO OTHER ANALYTIC METHODS INVESTIGATED. MORE DETAILED EXPERIMENTAL DATA ARE NECESSARY TO DEMONSTRATE A GENERAL VALIDITY OF THE METHOD. A DIGITAL COMPUTER CODE WAS WRITTEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROICHE

COMPUTER PROGRAM + REACTOR, LMCR + HYDRODYNAMIC ANALYSIS + METAL, LIQUID + FLOW, TURBULENT + FLOW DISTRIBUTION + SUL-GEL PROCESS + REACTOR, LMFBR

5-034635 DATED 1968
MACFARLANE DK + MCNEAL NA + MENELEY DA + SANATHANAN CK
SODIUM BOILING TRANSIENTS IN OXIDE-FUELED FAST REACTORS
ARGONNE NATIONAL LABORATORY, 9700 S. CASS AVE., ARGONNE, ILLINOIS
16 PAGES, 22 FIGURES, 5 TABLES, 11 REFERENCES, NUCLEAR ENGINEERING AND DESIGN, VOL. 7, PAGES 411-426 (MAY 1968)

METHODS AND RESULTS OF THEORETICAL INVESTIGATION OF DYNAMIC RESPONSE OF LARGE OXIDE-FUELED FAST REACTOR CORES DURING TRANSIENT OVER-POWER OR UNDER-FLOW SITUATIONS WHICH RESULT IN BULK BOILING OF COOLANT ARE PRESENTED. VALIDITY OF ANALYSIS IS RESTRICTED TO REGIME BEFORE CLADDING MELTS OR FAILS. MODEL FOR PREDICTING COOLANT BOILING (SODIUM VOIDING) BEHAVIOR HAS BEEN INCORPORATED INTO CLOSED LOOP CALCULATION WHICH INCLUDES REACTOR KINETICS WITH SODIUM VOIDING AND DOPPLER REACTIVITY FEEDBACKS. RESULTS OF CALCULATIONS FOR REACTIVITY RAMP FOR TWO CORES WITH DIFFERENT GEOMETRY ARE PRESENTED. RESULTS OF CALCULATIONS FOR A FLOW COASTDOWN TRANSIENT ARE ALSO PRESENTED. SOME LIMITATIONS OF MODEL AND METHOD ARE DISCUSSED BRIEFLY.

HEAT TRANSFER, BOILING + DOPPLER COEFFICIENT + REACTIVITY, EXCESS + REACTOR, FAST + FLOW THEORY AND EXPERIMENTS + REACTOR TRANSIENT + SODIUM + SODIUM COEFFICIENT + VOID COEFFICIENT + ANALYTICAL MODEL + MATHEMATICAL TREATMENT + REACTOR, LMFBR + BOILING

5-034694 DATED 1968
KASHCHEEV VM + NOMOFILOV EV
TURBULENCE SCALE IN CHANNELS FORMED BY CLOSELY PACKED RODS
ENERGY-PHYSICS INSTITUTE, OBNISK, U.S.S.R.
11 PAGES, 4 FIGURES, 10 REFERENCES, INTERNATIONAL JOURNAL OF HEAT AND MASS TRANSFER, 11(2), PAGES 267-277 (FEBRUARY 1968)

ACCORDING TO OBUKHOVS METHOD THE FORMULA WAS OBTAINED TO CALCULATE TURBULENCE SCALE DISTRIBUTION IN CHANNELS FORMED BY CLOSELY PACKED RODS ARRANGED IN-LINE OR STAGGERED, THE FUNCTIONS CONFORMALLY REPRESENTING THE CROSS-SECTIONS OF THE ABOVE CHANNELS TO THE INSIDE OF THE UNIT CIRCLE, WHICH ARE NECESSARY FOR DETERMINING TURBULENCE SCALE, ARE ALSO PRESENTED.

REACTOR, LMCR + HYDRODYNAMIC ANALYSIS + FLOW, TURBULENT + REACTOR, LMFBR

5-035345 DATED 1968

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-035345 *CONTINUED*
ZINIUK MA + WILBER HA
TRANSIENT EFFECTS OF LOSS OF ONE COOLANT CIRCUIT ON THE ENRICO FERMI REACTOR AND ASSOCIATED HEAT TRANSPORT SYSTEM

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN
APDA-NIS-20 +. 35 PAGES, 6 FIGURES, 4 TABLES, 4 REFERENCES, SEPTEMBER 1968

DESCRIBES A SINGLE-CIRCUIT SHUTDOWN TEST OF THE ENRICO FERMI REACTOR AT A POWER LEVEL OF 20 MW. THE TEST CONSISTED OF TWO SINGLE-CIRCUIT SHUTDOWNS, THE FIRST BEING CONDUCTED WITH THE REACTOR AND THE FEEDWATER FLOW CONTROL SYSTEMS IN MANUAL CONTROL, AND THE SECOND WITH THE REACTOR SYSTEM IN AUTOMATIC CONTROL. RESULTS IN THE FIRST PART OF THE TEST SHOW A MAXIMUM NEGATIVE DN/DT (RATE OF CHANGE OF NEUTRON FLUX) OF 0.75 MW/MIN, DUE TO THE CORE TEMPERATURE COEFFICIENT WITH STABILIZATION AT 13.4 MW, REPRESENTING 67% OF THE INITIAL POWER LEVEL. POWER STABILITY WAS REACHED IN A NONOSCILLATORY DECAY IN ABOUT 12 MIN. THE TEMPERATURE OF THE SODIUM OUTLET OF THE REACTOR EXHIBITED AN INCREASE OF ABOUT 4F, WITH SIMILAR TRANSIENTS OCCURRING IN THE TEMPERATURES OF THE SODIUM INLET AND OUTLET OF THE STEAM GENERATOR AND THE SODIUM INLETS OF THE LOOPS REMAINING IN OPERATION. TEMPERATURE TRANSIENTS OBSERVED DURING THE SECOND PART OF THE TEST WERE SUBSTANTIALLY LIKE THOSE OF THE FIRST PART. INASMUCH AS THE TEST POWER LEVEL WAS LOW, THE SPECIFIC RATES OF CHANGE OF POWER AND THE THERMAL TRANSIENTS WHICH WERE EXPERIENCED AT 20 MW WERE WELL WITHIN ESTABLISHED SAFETY LIMITS.

AVAILABILITY - M.A. ZINIUK/H.A. WILBER, ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., 1911 FIRST STREET, DETROIT, MICHIGAN 48226

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + ACCIDENT, LOSS OF FLOW + SODIUM + MAIN COOLING SYSTEM + REACTOR, LMFBR

5-035349 ALSO IN CATEGORY 6 DATED 1969
LUNG JK
ANALYSIS OF THE STATIC POWER COEFFICIENT OF EBR-II WITH REDUCED COOLANT FLOW
ARGONNE NATIONAL LABORATORY
11 PAGES, 6 FIGURES, 2 TABLES, 13 REFERENCES, NUCLEAR APPLICATIONS, 6(3), PAGES 116-26 (FEBRUARY 1969)

THIS STUDY OF EBR-II OPERATING BEHAVIOR, WHICH BEGAN WITH A REEXAMINATION OF THE THERMAL ANALYSIS OF FUEL PINS UNDER VARYING RATES OF COOLANT FLOW, INCORPORATES THE RECENT INTERPRETATIONS OF THE BOWING BEHAVIOR OF THE SUBASSEMBLIES. THE NONLINEAR BOWING EFFECTS ARE COMBINED WITH THE LINEAR EXPANSION EFFECTS IN EQUATIONS THAT EXPRESS THE POWER-REACTIVITY DECREMENT AS A FUNCTION OF BOTH POWER AND FLOW. COMPARISON WITH EXPERIMENTS INDICATES AGREEMENT TO WITHIN A FEW INHOURS. DIFFERENTIATION OF THE EQUATIONS LEADS TO EXPRESSIONS FOR POWER AND FLOW COEFFICIENTS OF REACTIVITY. THE REDUCED FLOW DATA ARE USED TO DETERMINE AN EFFECTIVE AVERAGE THERMAL EXPANSION COEFFICIENT FOR THE FUEL.

FUEL ELEMENT BOWING + REACTOR, LMCR + REACTIVITY EFFECT + REACTOR DYNAMICS + SODIUM + REACTOR, TEST + REACTOR COOLANT + ANL + REACTOR, LMFBR

5-035822 DATED 1968
AMENDOLA A
A STATISTICAL METHOD FOR EVALUATION OF HOT CHANNEL FACTORS IN REACTOR DESIGN
KERNFORSCHUNGSZENTRUM, KARLSRUHE
KFK-843 +. 44 PAGES, 10 FIGURES, 18 REFERENCES, JULY 1968

PREVIOUS METHODS OF HOT-CHANNEL ANALYSIS ASSUMED ALL THE COOLING CHANNELS IN A CORE TO BE INDEPENDENT. TWO TYPES OF UNCERTAINTIES EXIST, - LOCAL UNCERTAINTIES (WHICH MAY VARY FROM POINT TO POINT WITHIN A REACTOR), AND GLOBAL UNCERTAINTIES (WHICH DO NOT VARY WITHIN A REACTOR, OR AT LEAST WITHIN PARTS OF A REACTOR). A QUANTITATIVE ASSESSMENT OF THE DIFFERENT EFFECTS OF THESE TWO TYPES OF UNCERTAINTIES IS GIVEN AND A NEW METHOD OF EVALUATING THE HOT-CHANNEL FACTOR IS PROPOSED. AN EXAMPLE IS GIVEN FOR THE REACTOR N2-2 AND A COMPARISON WITH THE PREVIOUS METHOD.

AVAILABILITY - AEC DEPOSITORY LIBRARIES IN THE U.S. AND OVERSEAS

HEAT TRANSFER + REACTOR, LMCR + COMPARISON, THEORY AND EXPERIENCE + HOT CHANNEL ANALYSIS + STATISTICAL ANALYSIS + MATHEMATICAL TREATMENT

5-035885 DATED 1968
STURIALE LA
SURVEY OF STATE-OF-THE-ART OF INTERMEDIATE HEAT EXCHANGERS. 1000-MWE LMFBR FOLLOW-ON STUDY
WESTINGHOUSE ELECTRIC CORP., MADISON PA. ADVANCED REACTORS DIV.
WARD-2000-23 +. 49 PAGES, 18 FIGURES, 3 TABLES, 18 REFERENCES, MARCH 1968

THIS STATE-OF-THE-ART SURVEY SUMMARIZES INFORMATION AVAILABLE ON INTERMEDIATE HEAT EXCHANGERS (IHX) USED IN LIQUID-METAL (SODIUM) REACTOR SYSTEMS AND IS INTENDED TO PROVIDE THE FRAME OF REFERENCE NECESSARY TO AID SELECTION OF AN IHX CONFIGURATION. THE CONCLUSIONS DRAWN PROVIDE THE BASIS FOR SELECTION OF THE IHX CONFIGURATION AS THE REFERENCE CONCEPT FOR THE 1000-MWE FOLLOW-ON STUDY. THIS STUDY IS LIMITED TO A QUALITATIVE EVALUATION OF SIGNIFICANT DESIGN FEATURES EMPLOYED IN RECENT INTERMEDIATE HEAT EXCHANGERS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + REACTOR, FAST + SODIUM + DESIGN STUDY + METAL, LIQUID + REVIEW + EQUIPMENT DESIGN + SURVEY + OPERATING EXPERIENCE + REACTOR, LMFBR + HEAT EXCHANGERS

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-036209 ALSO IN CATEGORY 9 DATED 1969
ANDERSON TT + GRATE TA
ACOUSTIC BOILING DETECTION IN REACTOR VESSELS
ARGONNE NATIONAL LABORATORY
4 PAGES, 10 REFERENCES, IEEE TRANSACTIONS ON NUCLEAR SCIENCE, NS-16(1), PAGES 260-3 (FEBRUARY 1969)

REACTOR COOLANT WHICH BOILS IN A FAST REACTOR OR HIGH-FLUX RESEARCH REACTOR, ENDANGERING REACTOR SAFETY, CAN BE DETECTED ACOUSTICALLY. AS A CONTRIBUTION TO DESIGN OF ACOUSTIC BOILING DETECTORS, THE TANK WALL OF A REACTOR IS SHOWN THEORETICALLY TO AFFECT STANDING WAVES IN THAT TANK. EXPERIMENTAL RESONANCE FREQUENCIES OF A 3-FT-DIAMETER TANK DEMONSTRATE THAT FOR LOW-PRESSURE REACTOR TANKS, THE PRESSURE AT STANDING-WAVE FREQUENCIES IS NEARLY ZERO AT THE TANK WALL, IN CONTRAST TO THE RESULTS FOR A PERFECTLY RIGID WALL.

AVAILABILITY - T.A. GRATE, ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

REACTOR, FAST + REACTOR, LMCR + NOISE + COMPARISON, THEORY AND EXPERIENCE + REACTOR, TEST + INSTRUMENT, COMPONENT + BOILING + ACOUSTICS

5-036236 ALSO IN CATEGORY 18 DATED 1967
SUFER GA + CHERRY BH
DESIGN AND SAFETY OPTIMIZATIONS IN THE USE OF PLUTONIUM-URANIUM CARBIDE FUEL IN LARGE FAST BREEDER REACTORS
COMMISSARIAT A L'ENERGIE ATOMIQUE, PARIS
12 PAGES, PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPTEMBER 19-22, 1967.
(SEE CONF-670916)

OPERATING CHARACTERISTICS OF A 1000-MW(E) NA-COOLED PUC-UC REACTOR, CONSISTING OF NEUTRONICALLY UNCOUPLED THREE-MODULE CORE, ARE DESCRIBED AND DISCUSSED. ITS SAFETY CHARACTERISTICS ARE ANALYZED AND COMPARED TO A SINGLE-MODULE CORE WITH A L/D EQUAL TO 1. THE INFLUENCE OF NA VS HE AS THE THERMAL BOND BETWEEN FUEL AND CLADDING ON THE SAFETY CHARACTERISTICS OF BOTH DESIGNS IS EVALUATED. FUEL CYCLE COST ESTIMATES FOR MODULAR AND L/D EQUAL TO 1 DESIGNS ARE COMPARED AND DISCUSSED.

CLADDING + REACTOR, FAST + THERMAL CONSIDERATION + DESIGN STUDY + SAFETY PRINCIPLES AND PHILOSOPHY + ECONOMIC STUDY + COMPARISON, REACTOR CHARACTERISTICS + REACTOR, LMFBR + FUEL ELEMENTS

5-036238 ALSO IN CATEGORY 9 DATED 1968
EVALUATION ON INCONEL 718 FOR APPLICATION IN THE FAST TEST REACTOR
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-930 +. 31 PAGES, 7 FIGURES, 6 TABLES, 28 REFERENCES, DECEMBER 1968

INCONEL 718, A PRECIPITATION HARDENABLE NICKEL-BASE ALLOY, HAS RECENTLY BEEN CONSIDERED FOR USE IN THE FAST TEST REACTOR (FTR). PROPERTIES OF INTEREST WHICH QUALIFY THIS ALLOY FOR REACTOR USE ARE EXCELLENT WEAR RESISTANCE, FATIGUE RESISTANCE, EASE OF FABRICATION, SUPERIOR HIGH TEMPERATURE MECHANICAL PROPERTIES, AND STRUCTURAL STABILITY. THREE DISTINCT AREAS OF POSSIBLE APPLICATION HAVE THUS FAR BEEN IDENTIFIED - FLOW DUCT CONTACT PADS, SAFETY ROD ACCELERATOR SPRINGS, AND FUEL-PIN GRID-SPACER ASSEMBLIES.

REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + STRESS ANALYSIS + FFTF (TR) + CORROSION + CONTROL ROD SCRAM MECHANISM + MATERIAL + PROPERTY, PHYSICAL + INCONEL + CORE COMPONENTS + ACCELERATION + STRUCTURAL ANALYSIS, DYNAMIC + PROPERTY, MECHANICAL

5-036396 DATED 1969
HENRY RE
CRITICAL FLOW OF STEAM-WATER MIXTURES
NASA, LEWIS RESEARCH CENTER
2 PAGES, 2 FIGURES, 4 REFERENCES, AHS TRANSACTIONS, 12(1), PAGES 353-4 (JUNE 19, 1969)

TO INVESTIGATE THE NATURE OF ONE-COMPONENT CRITICAL FLOW IN THE LOW-QUALITY RANGE WITH EXHAUST PRESSURES GREATER THAN 150 PSIA, A BLOWDOWN FACILITY WAS EMPLOYED THAT USED STEAM-WATER AS THE WORKING MEDIUM. THE RESULTS AGREE WITH THE RELATIONSHIP THAT THE RATIO OF THE CRITICAL FLOW RATE TO THE HOMOGENEOUS EQUILIBRIUM CRITICAL FLOW RATE VARIES INVERSELY AS 20 TIMES THE THERMODYNAMIC EQUILIBRIUM MIXTURE QUALITY WHICH WAS PREVIOUSLY DEVELOPED BY HENRY FOR LOWER-PRESSURE CRITICAL FLOW RATES.

AVAILABILITY - ROBERT E. HENRY, NATIONAL AERONAUTICS AND SPACE ADMINISTRATION, LEWIS RESEARCH CENTER, 21000 BROOKPARK ROAD, CLEVELAND, OHIO 44135

REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + REACTOR, PWR + REACTOR, LWR + FLOW, TWO PHASE + FLOW, CRITICAL + REACTOR, LMFBR

5-036397 DATED 1969
FAUSKE HK + QUINN DJ + JEANS WC
CRITICAL FLOW DATA OF SODIUM LIQUID-VAPOR MIXTURES
ARGONNE NATIONAL LABORATORY
2 PAGES, 1 FIGURE, 4 REFERENCES, AHS TRANSACTIONS, 12(1), PAGES 354-5 (JUNE 19, 1969)

EXPERIMENTAL DATA ON CRITICAL FLOW OF SODIUM LIQUID-VAPOR MIXTURES WERE OBTAINED AND COMPARED WITH RELATIONSHIPS DERIVED FOR TWO-PHASE FLOW OF NONMETALLIC FLUIDS. THE HOMOGENEOUS EQUILIBRIUM

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-036397 *CONTINUED*

MODEL FAILS TO PREDICT THE DATA, BUT THE SLIP EQUILIBRIUM MODEL SHOWS GOOD AGREEMENT WITH VALUES MEASURED WITH THESE MIXTURES OF LIQUID SODIUM AND ITS VAPOR. THUS, THESE TESTS VERIFY EARLIER CONCLUSIONS IN REGARD TO THE SIMILARITY BETWEEN NONMETALLIC FLUIDS AND SODIUM, NAMELY, (A) NO MAJOR DIFFERENCES EXIST, AND (B) THE THEORY DEVELOPED FOR PREDICTING MAXIMUM DISCHARGES OF NONMETALLIC FLUIDS CAN BE USED TO PREDICT METALLIC SYSTEMS.

AVAILABILITY - HANS K. FAUSKE, ARGONNE NATIONAL LABORATORY, 9700 S. CASS AVENUE, ARGONNE, ILLINOIS 60439

REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + SODIUM + METAL, LIQUID + FLOW, TWO PHASE + FLOW, CRITICAL + REACTOR, LMFR

5-036403

DATED 1969

CRONENBERG AW + FAUSKE HK + BANKOFF SG

SIMPLIFIED MODEL FOR SODIUM COOLANT EXPULSION AND REENTRY

NORTHWESTERN UNIVERSITY + ARGONNE NATIONAL LABORATORY

1 PAGE, 1 FIGURE, 5 REFERENCES, ANS TRANSACTIONS, 12(1), PAGE 364 (JUNE 19, 1969)

A SIMPLE EXPULSION MODEL, WHICH SATISFIED ALL CONSERVATION CONDITIONS, HAS BEEN DEVELOPED TO PREDICT THE EXPULSION AND REENTRY BEHAVIOR OF SODIUM COOLANT UPON OVERHEATING OF A REACTOR CHANNEL. THIS MODEL, TAKING INTO ACCOUNT WALL FILM IN THE VAPOR REGION, PREDICTS FASTER EXPULSION THAN THE PRESENT COCURRENT TWO-PHASE FLOW MODELS, WHICH DO NOT FULLY TAKE INTO ACCOUNT MOMENTUM CONSIDERATIONS AND SELF-PRESSURIZATION DURING EXPULSION. LIKEWISE, THE BLOW CODE, ALSO A SLUG MODEL, ASSUMES WALL FILM AS THE ONLY VAPOR SOURCE, AND HENCE GIVES A SLOWER INITIAL VOIDING RATE THAN THIS MODEL, WHICH CONSIDERS VAPORIZATION FROM BOTH THE LIQUID SLUG SURFACES AND THE WALL FILM.

AVAILABILITY - H.K. FAUSKE, ARGONNE NATIONAL LABORATORY, 9700 S CASS AVENUE, ARGONNE, ILLINOIS 83401

COMPUTER PROGRAM + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + ACCIDENT, LOSS OF FLOW + SODIUM + REACTOR COOLANT + METAL, LIQUID + REACTOR, LMFR

5-036408

ALSO IN CATEGORIES 8 AND 11

DATED 1966

EXPLOSION CONTAINMENT IN THE PROTOTYPE FAST REACTOR SINGLE TANK SYSTEM

SAFEX/P.09 +. 30 PAGES, 25 FIGURES, 3 TABLES, 3 REFERENCES, JANUARY 1966

DESCRIBES RESULTS OF SECOND TEST ON 1/16TH-SCALE MODEL, CHECKING EFFECTS OF DETONATION OF TNT. RESULTS OF ROUND 11 AGREED WITH THOSE OF ROUND 10. ROUND-11 TEST SHOWED LIMITED DAMAGE.

AVAILABILITY - UNITED KINGDOM ATOMIC ENERGY AUTHORITY, 11 CHARLES 11 STREET, LONDON, S.W. 1, ENGLAND

REACTOR, BREEDER + CONTAINMENT + REACTOR, FAST + REACTOR, LMCR + UNITED KINGDOM + EXPLOSIVE, CONVENTIONAL + PFR (LMFR) + MODEL TESTING

5-036597

ALSO IN CATEGORIES 6 AND 18

DATED 1967

SIMPSON DE + LITTLE WW + PETERSON RE

SELECTED SAFETY CONSIDERATIONS IN DESIGN OF THE FAST FLUX TEST FACILITY

BAITELLE-NORTHWEST, WASHINGTON

37 PAGES, 10 FIGURES, 9 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 1967

DESCRIBES FFTF. ACCIDENT-SEVERITY CRITERIA DISCUSSED IN TERMS OF MAX. TRANSIENT FUEL TEMPERATURES AND EXTENT OF MELTING. THE MAJOR SAFETY CONSIDERATIONS AT THE CONCEPTUAL STAGE ARE OUTLINED. TOPICS - FFTF DESCRIPTION, SAFETY PHILOSOPHY AND CRITERIA, TRANSIENT AND ACCIDENT ANALYSES, DESIGN-BASIS ACCIDENT, DESIGN CONSIDERATIONS, AND EFFECTS OF SAFETY ON TESTING CONDITIONS.

AVAILABILITY - D.E. SIMPSON + W.W. LITTLE + R.E. PETERSON, PACIFIC NORTHWEST LABORATORY, P.O. BOX 999, RICHLAND, WASHINGTON

ACCIDENT ANALYSIS + DESIGN CRITERIA + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + SAFETY PRINCIPLES AND PHILOSOPHY + R AND D PROGRAM + REACTOR, LMFR

5-036802

ALSO IN CATEGORIES 8 AND 18

DATED 1969

HIGH-TEMPERATURE LIQUID-METAL TECHNOLOGY REVIEW

BROOKHAVEN NATIONAL LABORATORY

80 PAGES, 11 FIGURES, 1 TABLE, APRIL 1969

SUMMARIZES PROGRESS IN VARIOUS R AND D PROGRAMS ON MATERIAL DEVELOPMENT, CORROSION, HEAT TRANSFER, FLUID DYNAMICS, INSTRUMENTATION, PHYSICAL PROPERTIES, CHEMICAL ANALYSIS PERTINENT TO FAST-BREEDER DEVELOPMENTAL PROGRAM. PROGRAMS REVIEWED IN CATEGORY OF SAFETY INCLUDE IN-CORE FLOWMETER DEVELOPMENT, EFFECT OF FLOWING SODIUM ON CREEP RATE OF FUEL JACKET (VANADIUM ALLOYS), DEVELOPMENT OF HIGH-TEMP.-BOILING DETECTOR, SODIUM-FIRE FISSION-PRODUCT RELEASE, BOILING STUDIES, SODIUM CHEMISTRY, AND NOISE-ANALYSES INSTRUMENTATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CHEMICAL REACTION + FIRE + FISSION PRODUCT RELEASE + SODIUM + NOISE ANALYSIS + CORROSION + METAL, LIQUID + REVIEW + R AND D PROGRAM + REACTOR, LMFR + BOILING + INSTRUMENTS, MISC.

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-036957 ALSO IN CATEGORIES 8 AND 11 DATED 1969
UKAEA SAFETY RESEARCH BULLETIN ISSUE NUMBER 4
UKAEA, AUTHORITY HEALTH & SAFETY BRANCH, RISLEY
UKAEA SAFETY RESEARCH BULLETIN ISSUE NO. 4 +. 50 PAGES, 62 REFERENCES, 1969

PROGRESS REVIEW OF UKAEA WORK IN NUCLEAR REACTOR SAFETY. SUMMARIZED ARE STUDIES PERTAINING TO THE SAFETY OF WATER COOLED REACTORS, LIQUID METAL FAST BREEDER REACTORS, AND THE FRACTURE OF CONCRETE AND STEEL PRESSURE VESSELS. INCLUDED IN THE WATER REACTOR STUDIES ARE COOLANT DEPRESSURIZATION, NATURAL CONVECTION HEAT TRANSFER, HEAT TRANSFER DURING DRYOUT CONDITIONS, BUBBLE GROWTH, ALUMINUM-WATER INTERACTION, AND RELIABILITY OF EMERGENCY CORE-COOLING SYSTEMS. INCLUDED IN THE LIQUID METAL FAST BREEDER REACTOR STUDIES ARE HEAT TRANSFER AND PRESSURE GENERATION FROM CORE DEBRIS FOLLOWING A MELTDOWN, CONTAINMENT OF RADIOACTIVITY, CONSEQUENCES OF A SODIUM FIRE, AND PRESSURE EFFECTS DUE TO A SODIUM FIRE.

AVAILABILITY - HER MAJESTY'S STATIONERY OFFICE, P.O. BOX 569, LONDON S.E.1, ENGLAND

ACCIDENT ANALYSIS + PRESSURE VESSELS + THERMAL ANALYSIS + WATER + CONCRETE, PRESTRESSED + CORE MELTDOWN + REACTOR, LWR + METAL, LIQUID + HEAT TRANSFER ANALYSIS + BLOWDOWN + BUBBLE + EMERGENCY COOLING + PRESSURE TRANSIENT + REACTOR, LMFBR + UKAEA + JACOBS

5-037462 ALSO IN CATEGORY 18 DATED 1968
BUKAU F
A PROTECTIVE GAS SYSTEM OF HIGH PURITY FOR THE APPLICATION OF HIGH-MELTING REACTIVE METALS AT HIGH TEMPERATURES
GESELLSCHAFT FÜR KERNFORSCHUNG MBH, KARLSRUHE
KFK-927 + 5 PAGES, DECEMBER, 1969

SODIUM IS THE REACTOR COOLANT OF PREFERENCE WHICH OFFERS IN REGARD TO HEAT FLOW AND OPERATING TEMPERATURE AN APPRECIABLE POTENTIAL FOR DEVELOPMENT. TO OBTAIN INFORMATION ON THE TECHNICAL LIMITATIONS OF THE USE OF SODIUM, A RESEARCH PROGRAM WAS INSTITUTED, AN EXPERIMENTAL APPARATUS HAS BEEN PLANNED AND ITS CONSTRUCTION ALMOST COMPLETED. THE CRITICAL HEAT FLUX (CHF) MUST BE REGARDED AS AN ABSOLUTE LIMIT OF THE HEAT FLUX. THIS IS DEPENDENT ON VARIOUS PARAMETERS. IN THE CASE OF SODIUM FORCED FLOW OF THE CHF IN THE REGION OF PRESSURE OF TECHNICAL INTEREST IS SO HIGH THAT UP TO THE PRESENT TIME WITH THE USUAL HEATING METHODS IT COULD NOT BE SIMULATED EXPERIMENTALLY.

AVAILABILITY - USAEC DEPOSITORY LIBRARIES IN THE U.S. AND OVERSEAS

REACTOR, LMCR + SODIUM + TESTING + THERMAL EXPERIMENT + COVER GAS + THERMAL CONSIDERATION + HIGH TEMPERATURE + MATERIAL + HEAT TRANSFER, CONDUCTION + R AND D PROGRAM + HEAT FLUX, CRITICAL

5-037463 DATED 1969
FETTE P + GROSSGARTEN HD
SIMULATION OF COOLANT EXPULSION IN NA-2 FUEL ELEMENTS
GESELLSCHAFT FÜR KERNFORSCHUNG MBH., KARLSRUHE
KFK-925 + EUR-41690, 3 PAGES, JANUARY, 1969

IN REGARD TO THE SAFETY OF FAST SODIUM-COOLED BREEDER REACTORS THE ANALYSIS OF BOILING OF THE COOLANT IS OF GREAT IMPORTANCE. EARLIER THEORETICAL AND EXPERIMENTAL INVESTIGATIONS FOLLOWED THE MODEL OF A SINGLE-BUBBLE EJECTION IN ONE COOLING CHANNEL. IN THE EXPERIMENTS WITH WATER (3) THE EJECTION PROCESS WAS RECORDED WITH A HIGH FREQUENCY FILM CAMERA. APPRECIABLE PHENOMENOLOGICAL CHARACTERISTICS OF THE EJECTION OF THE COOLANT COULD BE INTERPRETED IN THIS WAY. IN THE EXPERIMENTS DESCRIBED THE EJECTION OF BUBBLES IN A BUNDLE OF TUBES WITH ORIGINAL FUEL ELEMENT DIMENSIONS IS FOLLOWED OPTICALLY.

EXPERIMENT + SIMULATION + SAFETY ANALYSIS + MODEL TESTING + REACTOR, LMFBR + BOILING + FUEL ELEMENTS

5-037838 ALSO IN CATEGORIES 12 AND 17 DATED 1968
HIGH-TEMPERATURE LIQUID-METAL TECHNOLOGY REVIEW
BROOKHAVEN NATIONAL LABORATORY
BNL-50115(PR-32) +. 88 PAGES, 8 FIGURES, REFERENCES, APRIL 1968

THE PURPOSE OF THE REVIEW IS TO PROVIDE UP-TO-DATE INFORMATION ON THE VARIOUS RESEARCH AND DEVELOPMENT PROGRAMS IN THE UNITED STATES IN THE FIELD OF HIGH-TEMPERATURE LIQUID-METAL TECHNOLOGY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + REACTOR, BREEDER + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + THERMODYNAMICS + REACTOR COOLANT + CORROSION + METAL, LIQUID + MASS TRANSFER + PROPERTY, PHYSICAL + REVIEW + R AND D PROGRAM

5-037850 ALSO IN CATEGORIES 6 AND 12 DATED 1967
DESCAMPS C + EGLEME M + LEE J + MICHEL A
ANALYSIS OF THERMAL AND HYDRAULIC BEHAVIOR OF SODIUM DURING A PROMPT POWER INSERTION IN A FUEL ASSEMBLY
COMMISSARIAT A L'ENERGIE ATOMIQUE, PARIS
CONF-670916 +. 17 PAGES, 14 REFERENCES, SEPT. 1967

DURING THE STUDY AND THE DESIGN OF AN IN-PILE NA LOOP FOR FUEL ELEMENT RUPTURE TEST, THE FUEL

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-037850 *CONTINUED*
DISPERSION AND SUBSEQUENT NA BEHAVIOR AFTER A PROMPT POWER INSERTION HAD TO BE EVALUATED. THE FUEL HEATING, THE CLADDING RUPTURE, THE FUEL DISPERSION IN THE NA AND THE SUDDEN NA BOILING WERE STUDIED. THE EXPERIMENTAL MEANS ARE DESCRIBED WHICH COULD CONTROL THE PROPOSED THEORETICAL MODEL.

AVAILABILITY - C. DESCAMPS, ET. AL., COMMISSARIAT A L'ENERGIE ATOMIQUE, PARIS, FRANCE

FAILURE, FUEL ELEMENT + HYDRAULIC ANALYSIS + IN PILE EXPERIMENT + REACTOR, LMCR + ACCIDENT, REACTIVITY + SODIUM + THERMAL ANALYSIS + ANALYTICAL MODEL + ACCIDENT MODEL + FAILURE, CLADDING + BOILING

5-037957 DATED 1967
EIFLER W + NIJSING R + AIROIA J
DESCRIPTION OF IBM 360 COMPUTER PROGRAM FOR THE CALCULATION OF LIQUID COOLED 7-ROD CLUSTER FUEL ELEMENTS
EUROPEAN ATOMIC ENERGY COMMUNITY, ISPRA (ITALY)
EUR-3733 +. 32 PAGES, OCTOBER 24, 1967

A CALCULATION MODEL TAKING ACCOUNT OF THE VARIOUS HEAT TRANSFER PHENOMENA OCCURRING IN CLUSTER TYPE FINNED FUEL ELEMENTS WAS DESCRIBED IN NUCLEAR ENGINEERING AND DESIGN, 4 (1966), P. 253-275. THE NUMERICAL EXAMPLES GIVEN THERE WERE CALCULATED EMPLOYING AN IBM 360 COMPUTER PROGRAM. THE PRESENT DOCUMENT PRESENTS A DESCRIPTION OF THE ABOVE PROGRAM. FIRST, THE GENERAL FEATURES OF THIS PROGRAM ARE OUTLINED. SUBSEQUENTLY THE SINGLE PARTS ARE DESCRIBED. IN THE DESCRIPTION, REFERENCE IS MADE TO THE CONSIDERATIONS OF THE ABOVE MENTIONED ARTICLE. A COMPLETE LISTING OF THE PROGRAM HAS BEEN GIVEN AS AN APPENDIX.

AVAILABILITY - USAEC DEPOSITORY LIBRARIES IN THE U.S. AND OVERSEAS

COMPUTER PROGRAM + HEAT TRANSFER + REACTOR, LMCR + ANALYTICAL MODEL + REACTOR, LWR + R AND D PROGRAM + FUEL ELEMENT CLUSTER

5-038234 DATED 1967
HOLTZ RE + SINGER RM
ON THE SUPERHEATING OF SODIUM AT LOW HEAT FLUXES
ARGONNE NATIONAL LABORATORY
ANL-7383 +. 33 PAGES, 7 FIGURES, NOVEMBER 1967

EXPERIMENTAL DATA ON THE SUPERHEAT NECESSARY TO INITIATE BOILING OF SODIUM WERE COMPARED WITH A THEORETICAL NUCLEATION MODEL. THE SODIUM WAS CONTAINED IN A VERTICAL PIPE AND WAS HEATED USING A HIGH-TEMPERATURE RESISTANCE WIRE WHICH LIMITED THE HEAT FLUX TO APPROXIMATELY 6 W/CM² (19,000 BTU/HR-FT). THE SYSTEM PRESSURE, HEAT FLUX, AND PRESSURE-TEMPERATURE HISTORY ALL SIGNIFICANTLY AFFECTED THE INCIPENT SUPERHEATS. IT WAS ALSO CONCLUDED THAT PRESSURE PULSES OCCURRED DURING VAPOR COLLAPSE, EVEN THOUGH THE LIQUID SODIUM WAS INITIALLY SUPERHEATED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFILME

DESIGN CRITERIA + REACTOR, FAST + HEAT TRANSFER + SODIUM + SUPERHEAT + ANALYTICAL MODEL + METAL, LIQUID + FLOW, TWO PHASE + FLOW STABILITY + REACTOR, LMFBF + BOILING

5-038356 DATED 1967
BORISHANSKI VM + ZHINKINA VB + FIRSOVA EV
EFFECT OF OXIDES ON HEAT TRANSFER DURING CROSS FLOW OF SODIUM THROUGH A STAGGERED TUBE BANK
BROOKHAVEN NATIONAL LAB
BNL-IR-184 +. 7 PAGES, FIGURES, 1967

PRESENTS THE RESULTS OF AN EXPERIMENTAL STUDY OF HEAT TRANSFER DURING TRANSVERSE OF FLOW OF SODIUM THROUGH A TEN-ROW STAGGERED TUBE BUNDLE AS A FUNCTION OF THE OXYGEN CONCENTRATION IN THE SODIUM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFILME

HEAT TRANSFER + REACTOR, LMCR + OXIDE + SODIUM + OUT OF PILE EXPERIMENT + FLOW, CROSS + COOLANT QUALITY

5-040402 ALSO IN CATEGORIES 6 AND 17 DATED 1969
HORI M + HOSLER A
A STUDY OF GAS ENTRAINMENT AND ABSORPTION-DESORPTION IN THE ENRICO FERMI REACTOR AND THEIR EFFECT ON CORE PERFORMANCE
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-LA-6 +. 50 PAGES, FIGURES, TABLES, 19 REFERENCES, JULY 1969, DOCKET 50-16, TYPE--LMFBF, MFG--APDA, AE--COMMONWEALTH ASSOC.

THE POSSIBLE PRESENCE OF GAS IN THE PRIMARY COOLANT AND ITS EFFECT ON CORE HEAT TRANSFER ARE DISCUSSED. GAS COULD BE SLOWLY INTRODUCED INTO THE PRIMARY SYSTEM BY GAS BUBBLE ENTRAINMENT AND/OR DIFFERENTIAL TEMPERATURE SOLUBILITY BETWEEN THE HOT AND COLD PORTIONS OF THE PRIMARY SYSTEM. A FEW POTENTIAL SITES FOR THE ACCUMULATION OF THIS GAS HAVE BEEN FOUND AND THE MAXIMUM ACCUMULATED VOLUMES AND RATES OF GAS RELEASE FROM THESE SITES DETERMINED. THE EFFECT OF A SINGLE LARGE BUBBLE AND ALSO THE CONTINUOUS PRESENCE OF GAS ON THE CORE HEAT TRANSFER HAVE BEEN STUDIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CATEGORY 5
HEAT TRANSFER AND THERMAL TRANSIENTS

5-040402 *CONTINUED*

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + HEAT TRANSFER + REACTOR, LMCR + REACTOR
STABILITY + COOLANT CHEMISTRY + INCIDENT + BUBBLE + REACTIVITY EFFECT, VOID + GAS + REACTOR, LMFBR

5-106947 ALSO IN CATEGORIES 7 AND 18 DATED 1965

OKRENT D

PROCEEDINGS OF THE CONFERENCE ON SAFETY, FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS - OCTOBER 11-14,
1965

ARGONNE NATIONAL LABORATORY, ILL.

ANL-7120 +. 910 PPS, OCT. 11-14, 1965

PRESENTS THE TEXT OF 79 PAPERS PRESENTED AT THE CONFERENCE. TOPICS INCLUDE: PHYSICS
UNCERTAINTIES AND SAFETY STUDIES; FAST REACTOR CONCEPTS; SAFETY OF LARGE FAST POWER REACTORS;
FUELS AND FUEL BEHAVIOR; REACTIVITY COEFFICIENTS; MAXIMUM ACCIDENT AND CONTAINMENT; FISSION
PRODUCTS, REACTIONS AND INTERACTIONS; AND SODIUM BOILING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + *REACTOR, FAST + *REACTOR, LMFBR + DESIGN STUDY + REACTOR, GCFR + FUEL ELEMENTS + REACTIVITY
COEFFICIENT + CONTAINMENT + PRESSURE TRANSIENT + FISSION PRODUCT RETENTION + BOILING + SODIUM + ACCIDENT
ANALYSIS

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-003219
GURINSKY UN

DATED 1963

BROOKHAVEN NATIONAL LABORATORY PROGRESS REPORT, NUCLEAR ENGINEERING DEPARTMENT, METALLURGY DIVISION
BROOKHAVEN NATIONAL LABORATORY
28 PAGES, 26 FIGURES, 9 TABLES, MAY 1 - AUGUST 31, 1963, BNL 823 (S-65)- PP. 66-93

INCREASING NA COOLANT VOLUME FRACTION IN AXIAL FLOW REACTORS CAUSED IMPROVEMENT IN NA TEMPERATURE COEFFICIENT. NA TEMPERATURE COEFFICIENTS WERE RE-EVALUATED AND WERE FOUND STILL TO BE NEGATIVE FOR RADIAL FLOW REACTORS. THE DOPPLER COEFFICIENTS FOR AXIAL FLOW REACTORS WERE FOUND TO BE NEGATIVE BUT DECREASING IN MAGNITUDE WITH INCREASING TEMPERATURE AND DECREASING REACTOR SIZE.

DOPPLER COEFFICIENT + REACTOR, LMCR + SODIUM

6-003407
KEATEN RW + PEARSON EN

DATED 1963

REACTIVITY CALCULATIONS AND MEASUREMENTS AT THE SRE
ATOMICS INTERNATIONAL

33 PAGES, 5 TABLES, 10 FIGURES, 21 REFERENCES- ANS TOPICAL MEETING NUCLEAR PERFORMANCE OF POWER-REACTOR CORES, SAN FRANCISCO, CALIFORNIA, SEPTEMBER 26-27, 1963- TID-7672- PP. 123-156

ORIGINAL CALCULATIONS USING STANDARD TWO-GROUP TECHNIQUES ARE COMPARED TO MEASURED VALUES OF CRITICAL MASS, SODIUM WORTH, STATISTICAL WEIGHT, FLUX DISTRIBUTIONS, AND TEMPERATURE COEFFICIENTS. A DISCREPANCY LARGER THAN THE EXPECTED CALCULATIONAL ERROR WAS REVEALED. REVISED PHYSICS CALCULATIONS HAVE UTILIZED THORIUM AND U-235 RESONANCE INTERVALS CALCULATED BY A MONTE CARLO CODE, AND THE RESULTS AGREE VERY WELL WITH THE MEASURED VALUES.

COMPUTER, ANALOG + COMPUTER, DIGITAL + REACTOR, LMCR + SODIUM COEFFICIENT + TEMPERATURE COEFFICIENT + SRE (RE)

6-003408
FILLMORE FL

DATED 1963

APPLICATION OF MONTE CARLO CALCULATIONS TO SGR REACTORS
ATOMICS INTERNATIONAL

15 PAGES, 8 TABLES, 7 FIGURES, 9 REFERENCES- ANS TOPICAL MEETING NUCLEAR PERFORMANCE OF POWER-REACTOR CORES, SAN FRANCISCO, CALIFORNIA, SEPTEMBER 26-27, 1963- TID-7672- PP. 157-172

DYNAMIC INSTABILITY IN THE SRE WAS CAUSED BY REACTIVITY CHANGE ASSOCIATED WITH BOWING OF THE FUEL RODS DUE TO THERMAL STRESS. IT WAS DECIDED TO MAKE A STUDY USING THE MONTE CARLO METHOD. INCREASE IN REACTIVITY OF ABOUT 1% COULD BE CAUSED BY THIS EFFECT.

COMPUTER, DIGITAL + FUEL ELEMENT BOWING + REACTOR, LMCR + SRE (RE)

6-003454

DATED 1963

CARTER JC + SPARKS DW + TESSIER JF
THE STEADY-STATE AND TRANSIENT KINETICS OF THE EBR-I, MK-III
ARGONNE NATIONAL LABORATORY

13 PAGES, 11 FIGURES, 3 REFERENCES- PROCEEDINGS OF A SYMPOSIUM ON REACTOR KINETICS AND CONTROL HELD AT THE UNIVERSITY OF ARIZONA, MARCH 25-27, 1963- APRIL 1964- TID-7662, PP. 405-418

DYNAMIC BEHAVIOR OF THE EBR-I, MK-III SHOWS THAT THE FEEDBACK FUNCTION CONTAINS A SIGNIFICANT NONLINEARITY. IT IS POSTULATED THAT THE OBSERVED NONLINEARITY ARISES ON FREE THERMAL EXPANSION BY FRICTIONAL FORCES. A MATHEMATICAL MODEL OF THE RESTRAINED THERMAL EXPANSION IS PRESENTED. STABILITY OF THE REACTOR UNDER POWER-EQUILIBRIUM CONDITIONS IS ASSESSED THROUGH LINEAR EXTRAPOLATION OF TEST DATA AND CONSIDERATION OF THE DESCRIBING FUNCTION FOR THE NONLINEAR THERMAL EXPANSION. THE INFLUENCE OF POSITIVE REACTIVITY ON STABILITY IS EXAMINED THROUGH USE OF A ROOT-LOCUS FAMILY.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + EQUATION, NONLINEAR + REACTOR STABILITY + REACTOR TRANSIENT

6-003460

DATED 1963

KEATEN RW + GUNBY AL + CARLSON WJ
A MATHEMATICAL MODEL DESCRIBING THE STATIC AND DYNAMIC INSTABILITY IN THE SRE CORE II
ATOMICS INTERNATIONAL

19 PAGES, 13 FIGURES, 9 REFERENCES- PROCEEDINGS OF A SYMPOSIUM ON REACTOR KINETICS AND CONTROL HELD AT THE UNIVERSITY OF ARIZONA, MARCH 25-27, 1963- APRIL 1964, TID-7662, PP. 528-547

MATERIAL DESCRIBED IN TOPICAL REPORTS.

FUEL ELEMENT BOWING + REACTOR, LMCR + TRANSFER FUNCTION + SRE (RE)

6-003469

DATED 1964

GUNBY AL + KALLFELZ JM
MEASUREMENT OF THE REACTIVITY ABSORBED BY SODIUM IN THE SRE
ATOMICS INTERNATIONAL

23 PAGES, 6 TABLES, 5 FIGURES, 9 REFERENCES- JULY 15, 1964, NAA-SR-9509

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-003469 *CONTINUED*

THE REACTIVITY ABSORBED BY SODIUM IN THE SODIUM REACTOR EXPERIMENT CORE II WAS MEASURED TO BE \$7.50. SUBCRITICAL MULTIPLICATION TECHNIQUES WERE USED IN CONJUNCTION WITH A DRAINING EXPERIMENT. SUBSEQUENT CALCULATIONS WERE SUFFICIENTLY ACCURATE TO OBIATE THE NEED FOR FUTURE DRAINING EXPERIMENTS.

REACTOR, LMCR + VOID COEFFICIENT + SRE (RE)

6-003588 ALSO IN CATEGORY 9 DATED 1964

PELLET MR
CORE II PHYSICS TESTS ON THE SODIUM REACTOR EXPERIMENT
ATOMICS INTERNATIONAL
27 PAGES, 13 FIGURES- AUGUST 1, 1964, NAA-SR-9510

EXPERIMENTS WERE PERFORMED DURING THE OPERATION OF THE SRE-CORE II. TRANSFER FUNCTION WAS MEASURED WITH A PILE OSCILLATOR. MEASUREMENTS AT-POWER WERE FIRST PERFORMED WHILE THE CORE HAD A POSITIVE PROMPT POWER COEFFICIENT. AFTER FUEL CLUSTERS CONSTRAINED, TRANSFER FUNCTION AT 10, 15, AND 20 MW DEMONSTRATED ABSENCE OF POSITIVE COEFFICIENT. COMBINED FUEL TEMPERATURE COEFFICIENT WAS -0.12 CENTS/DEGREE F. INLET TEMPERATURE COEFFICIENT WAS DETERMINED TO BE -0.23CENTS/DEGREE F. XENON PARAMETERS PREVIOUSLY DETERMINED HAVE BEEN VERIFIED. PRESENT SHIM ROD WORTH IS 94% OF THAT INITIALLY DETERMINED. INTEGRATED ISOTHERMAL TEMPERATURE COEFFICIENT WAS 72.7 CENTS AS COMPARED TO 81.9 CENTS PREVIOUSLY.

CONTROL ROD WORTH + FUEL ELEMENT BOWING + REACTOR, LMCR + OSCILLATOR, REACTIVITY + REACTIVITY COEFFICIENT + XENON + SRE (RE)

6-003589 ALSO IN CATEGORIES 9 AND 17 DATED 1964

DARLEY DK + JENKINS OG + LARSON CL + MCCANN PH + MILLS RM + STIENS RK + WALTERS RJ
HALLAM NUCLEAR POWER FACILITY - REACTOR OPERATIONS ANALYSIS PROGRAM SEMIANNUAL PROGRESS REPORT NO. 2, MARCH 1, 1963 - AUGUST 31, 1963
ATOMICS INTERNATIONAL
162 PAGES, 23 TABLES, 51 FIGURES- MARCH 1, 1964, NAA-SR-9265

SECONDARY-TO-PRIMARY SODIUM LEAK WAS IN INTERMEDIATE HEAT EXCHANGER. IN SECONDARY SODIUM PUMP NO. 1, THERE WAS INTERFERENCE BETWEEN THE IMPELLER AND DIFFUSER. CONDENSATE LEAK IN STEAM DUMP WAS TRACED TO A FAULTY GASKET. SODIUM FLOWMETER CALIBRATION LED TO OVERSHOTS IN POWER. THE THREE SECONDARY PUMPS WERE BINDING. ONE CONTROL ROD THIMBLE HAD A VERTICAL CRACK APPROXIMATELY 6-1/2 IN. LONG. FURTHER REACTIVITY CHECKS AND CONTROL ROD CALIBRATIONS WERE MADE. THREE FAILURES OF ZIRCALOY-2 THIMBLES LED TO REPLACE ALL COMPONENTS. CODE HAS BEEN DEVELOPED FOR THE COMPONENT RELIABILITY PROGRAM.

CONTAINMENT, PRESSURE VENTING + CONTROL ROD BURNUP + FAILURE, EQUIPMENT + HALLAM (LMR) + REACTOR, LMCR + FAILURE, PIPE + RELEASE RATE

6-003600 ALSO IN CATEGORY 17 DATED 1962

HALL DB
PLUTONIUM AS A FUEL FOR THE LOS ALAMOS MOLTEN PLUTONIUM REACTOR EXPERIMENT
LOS ALAMOS SCIENTIFIC LABORATORY
14 PAGES, 10 FIGURES, 6 REFERENCES- DECEMBER 1962, HW-75007- PP. 19.1-19.14- PROCEEDINGS PLUTONIUM AS A POWER REACTOR FUEL, AMERICAN NUCLEAR SOCIETY TOPICAL MEETING, RICHLAND, WASHINGTON, SEPTEMBER 13-14, 1962

DESCRIPTION AND OPERATING EXPERIENCE WITH LAMPRE ARE PRESENTED.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR DESCRIPTION + PRESSURE DROP

6-003657 DATED 1964

THALGOIT FW + LONG JK + DAVEY WG (ANL, IDAHO FALLS) + KATO WY (ANL, ILLINOIS) + CARPENTER SG (ATOMICS INTERNATIONAL) + MOREWITZ HA (AI) + BEST GH (LOS ALAMOS SCIENTIFIC LABORATORY)
FAST CRITICAL EXPERIMENTS AND THEIR ANALYSIS
ARGONNE NATIONAL LABORATORY, ATOMICS INTERNATIONAL, LOS ALAMOS SCIENTIFIC LABORATORY
15 PAGES- UNITED NATIONS THIRD INTERNATIONAL CONFERENCE ON THE PEACEFUL USES OF ATOMIC ENERGY, GENEVA, AUGUST 31-SEPTEMBER 9, 1964- A/CONF. 28/P/265

A REVIEW OF PU- AND TH-CYCLE FAST CRITICAL ASSEMBLIES IS PRESENTED. OPERATIONS OF ZPR-III, ZPR-VI, THE EPITHERMAL CRITICAL EXPERIMENTAL LABORATORY FACILITY, AND LAMPRE-I ARE REPORTED.

REACTOR, BREEDER + CRITICAL ASSEMBLY + REACTOR, FAST + REACTOR, LMCR + ZPR 6 (CAF) + PRESSURE DROP + LAMPRE I (RE)

6-003901 DATED 1963

DICKERMAN C + SOWA E + ROBINSON L + GOPINATH D
MELTDOWN STUDIES
ARGONNE NATIONAL LABORATORY
14 PAGES, 9 FIGURES, 18 REFERENCES- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 99-112- OTS

TREAT EXPERIMENTS ON- EBR-II ELEMENTS IN STAGNANT SODIUM, IRRADIATED EBR-II AND FERMI A SAMPLES,

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-003901 *CONTINUED*

URANIUM SULFIDE ELEMENTS. SAFETY ANALYSIS OF PUMPING POWER FAILURE, BLOCKED ORIFICE, AND DROPPED FUEL SUBASSEMBLY (SEE ANL 6844).

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + CRITICALITY SAFETY + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + FUEL HANDLING + FUEL MELTDOWN + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR, SAFETY RESEARCH + TREAT (PRR) + URANIUM SULFIDE + REACTOR, PULSED + FLOW BLOCKAGE + SAFETY ANALYSIS

6-003903

DATED 1963

HUEBUTTER PH

STUDY OF FAST REACTOR MELTDOWN ACCIDENTS USING SIMULANT MATERIALS

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

47 PAGES, 19 FIGURES, 1 TABLE- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 123-169- OTS

REGARDING MELTDOWN ACCIDENTS IN A SODIUM-COOLED FAST REACTOR FUELED WITH URANIUM-ALLOY PIN-TYPE ELEMENTS, WE PERFORMED DYNAMIC MOCKUP TESTS WITH LEAD AND WATER SUBSTITUTING FOR URANIUM ALLOY AND SODIUM, RESPECTIVELY, AND ELECTRICAL RESISTANCE HEATING TO SIMULATE FISSION HEATING IN ORDER TO ANSWER THESE QUESTIONS- UNDER WHAT CONDITIONS WOULD A FAST REACTOR BE VULNERABLE TO A MELTDOWN ACCIDENT. WHERE WOULD THE FUEL GO IN THE EVENT OF A MELTDOWN. HOW FAST WOULD THE FUEL REDISTRIBUTE ITSELF. THE COMPLETE REPORT IS APDA-155.

REACTOR, FAST + FUEL MELTDOWN + REACTOR, LMCR

6-003904

ALSO IN CATEGORY 8

DATED 1963

WOLFE B + FRIEDMAN N + RILEY D

INFLUENCE OF THE DOPPLER EFFECT ON THE MELTDOWN ACCIDENT

GENERAL ELECTRIC COMPANY, SAN JOSE

24 PAGES, 6 FIGURES, 1 TABLE, 9 REFERENCES- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 171-192- OTS

BY MEANS OF A DIGITAL COMPUTER, THE INFLUENCE OF THE DOPPLER EFFECT IN THE CORE-DISASSEMBLY PROCESS FOLLOWING A MELTDOWN ACCIDENT IS EXAMINED WITH A BETHE-TAIT MODEL IN WHICH THE DOPPLER EFFECT, AS WELL AS CORE DISASSEMBLY, IS CONSIDERED. FOR THE WORST HYPOTHETICAL ACCIDENT DESCRIBED IN THE FERMI HAZARDS REPORT, MAXIMUM EXPLOSIVE ENERGY RELEASES ARE REDUCED FROM ABOUT 650 LB OF HIGH EXPLOSIVE ENERGY WITH NO DOPPLER EFFECT TO LESS THAN 20 LB IN THE CASE OF A DOPPLER COEFFICIENT.

BETHE TAIT + REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + COMPUTER, DIGITAL + DOPPLER COEFFICIENT + EQUATION OF STATE + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + EXPLOSION + ACCIDENT, CORE DISRUPTIVE

6-003910

DATED 1963

PERKS MA + LORD RM

EFFECTS OF AXIAL AND RADIAL BLANKET DESIGN ON BREEDING AND ECONOMICS

UNITED KINGDOM ATOMIC ENERGY AUTHORITY, RISLEY

27 PAGES, 6 FIGURES, 11 TABLES, 2 REFERENCES- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 367-393- OTS

THIS PAPER IS NOT DIRECTLY CONCERNED WITH SAFETY BUT CONTAINS INCIDENTALLY SOME CALCULATIONS ON THE SODIUM VOID COEFFICIENT, OXIDE-FUELLED FAST REACTORS.

COMPUTER, DIGITAL + REACTOR, FAST + REACTOR, LMCR + VOID COEFFICIENT

6-003911

DATED 1963

CAMPISE AV + STOKER DJ

UNCERTAINTIES IN FAST POWER REACTOR DOPPLER COEFFICIENTS AND THEIR EFFECTS ON ECONOMICS

ATOMICS INTERNATIONAL

21 PAGES, 5 TABLES, 5 FIGURES, 8 REFERENCES- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 395-415- OTS

PAPER CONSIDERS INFLUENCE ON REFUELING INTERVALS OF A HYPOTHETICAL REQUIREMENT THAT THE AVAILABLE REACTIVITY IN A FAST REACTOR CORE BE LIMITED TO THAT WHICH CAN BE OVERRIDDEN BY THE DOPPLER EFFECT. REFUELING INTERVAL IS UNCERTAIN BY FACTOR OF 3 TO 5 BECAUSE OF UNCERTAINTY IN DOPPLER COEFFICIENT. COMPARISON OF U-PU SYSTEM AND TH-U-233 SYSTEM, OXIDES AND CARBIDES. EXCEPT FOR ON-STREAM REFUELING, THE CRITERION IS SAID TO BE IMPRACTICAL.

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR

6-003912

DATED 1963

CREAGAN RJ

FAST NEUTRON BREEDER ECONOMICS

WESTINGHOUSE ELECTRIC CORPORATION, PITTSBURGH

15 PAGES, 1 TABLE, 11 FIGURES- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING,

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-003912 *CONTINUED*

ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 395-415- OTS

THIS PAPER IS ONLY INDIRECTLY CONCERNED WITH SAFETY. SOME GENERAL REMARKS REGARDING LIMITS ON EXCESS REACTIVITY ARE ON PAGE 429, THE EBR-I MELTDOWN ON PAGE 430, AND CONTAINMENT ON PAGE 431.

REACTOR, BREEDER + CONTAINMENT + EBR 1 AND 2 (RE) + ECONOMICS + REACTIVITY, EXCESS + REACTOR, FAST + REACTOR, LMCR

6-003913

DATED 1963

SOFER GA + KAZI AH + BOBKER J

ECONOMICS AND SAFETY ASPECTS OF LARGE CERAMIC U-TH FAST BREEDER REACTORS

UNITED NUCLEAR CORPORATION

20 PAGES, 7 TABLES, 2 FIGURES, 10 REFERENCES- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 435-454- OTS

SPECIAL PROBLEMS OF A FAST REACTOR USING THE THORIUM-U-233 CYCLE- LOSS IN REACTIVITY DUE TO BURNUP OF FISSIONABLE MATERIAL AND OCCURRING BEFORE PA-233 DECAYS- CONTROL OF EXCESS REACTIVITY CAUSED BY PA-233 DECAY AFTER SHUTDOWN- FUEL-HANDLING PROBLEMS RESULTING FROM U-232 AND ITS DAUGHTERS- SODIUM VOID COEFFICIENT (WHICH IS, FORTUNATELY, LESS POSITIVE THAN IN (PU-U) OXIDE BREEDERS)- DELAYED NEUTRON FRACTION- SMALLER DOPPLER COEFFICIENT.

ALLOY + DELAYED NEUTRON + DOPPLER COEFFICIENT + REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + THORIUM + VOID COEFFICIENT

6-003914

DATED 1963

SPINRAD BI + LINK LE + WEILLS JT

A STUDY OF THE FUTURE ECONOMIC PROSPECTS FOR FAST REACTORS

ARGONNE NATIONAL LABORATORY

17 PAGES, 2 FIGURES- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 455-471- OTS

THIS PAPER IS ONLY INDIRECTLY RELATED TO SAFETY. IT CONTAINS SOME MATERIAL ON THE SODIUM VOID COEFFICIENT.

REACTOR, FAST + REACTOR, LMCR + VOID COEFFICIENT

6-003915

DATED 1963

ZALESKI CP

SOME IDEAS ON THE REACTIVITY COEFFICIENTS IN THE FAST LARGE REACTORS

EURATOM CEA (CADARACHE), FRANCE

21 PAGES, 12 FIGURES, 7 TABLES, 5 REFERENCES- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 473-793- OTS

DISCUSSES, IN THE USUAL FASHION, THE REACTIVITY EFFECTS OF FUEL DISPLACEMENT, DOPPLER EFFECT, SODIUM VOIDS, AND REACTOR STABILITY, RAMP-ADDITION OF REACTIVITY, DECREASE OF COOLANT FLOW, MAXIMUM HYPOTHETICAL ACCIDENT. SEE ALSO FURTHER DISCUSSION OF THIS PAPER, ANL-6792, PAGE 565.

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR STABILITY + ACCIDENT, CONTROL ROD WITHDRAWAL + THERMAL MECHANICAL EFFECT + VOID COEFFICIENT

6-003916

DATED 1963

COHEN K + GREEBLER P + MCNELLY MJ + MURPHY PM + SHERER D + ZEBROSKI EL

REACTOR SAFETY AND CONSIDERATIONS OF FUEL-CYCLE ECONOMICS FOR FAST REACTORS

GENERAL ELECTRIC COMPANY, SAN JOSE

16 PAGES, 6 TABLES, 12 FIGURES, 19 REFERENCES- DECEMBER 1963- ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 495-511- OTS

UPDATES GEAP-4080 WITH RESPECT TO SODIUM VOID COEFFICIENT. DOPPLER COEFFICIENT GIVES DESIGN CRITERIA FOR REACTIVITY COEFFICIENTS. COMPARISON OF HARD- AND SOFT-SPECTRUM CURES, LOCAL VOID COEFFICIENT AND ITS RELATION TO THE PROPAGATION OF SODIUM VOIDS DURING AN ACCIDENT. DISCUSSION TOUCHES ON COMPACTING OF FUEL CAUSING INCREASE IN REACTIVITY.

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + URANIUM OXIDE + VOID COEFFICIENT

6-003922

DATED 1963

LEVINE MM + ARONSUN A

NUCLEAR PARAMETER STUDIES OF THE FAST SETTLED BED REACTOR

BROOKHAVEN NATIONAL LABORATORY

9 PAGES, 4 TABLES, 7 FIGURES- DECEMBER 1963- ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 595-603- OTS

SIGN OF SODIUM VOID OR TEMPERATURE COEFFICIENT DEPENDS ON DESIGN. DOPPLER EFFECT SMALL. LARGER SODIUM FRACTION GIVES MORE FAVORABLE COEFFICIENTS. REACTIVITY EFFECTS OF CHANGE IN BED DENSITY

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-003322 *CONTINUED*
SMALL.

REACTOR, FAST + REACTOR, LMCR + REACTIVITY EFFECT + VOID COEFFICIENT

6-004026 DATED 1964
STACHURA SJ + SILBERBERG M + CORDY RN
URANIUM CARBIDE TRANSIENT HEATING EXPERIMENTS - PHASE I
ATOMICS INTERNATIONAL
96 PAGES, 9 TABLES, 59 FIGURES, 10 REFERENCES- AUGUST 1, 1964, NAA-SR-9508- JTS

TWO CAPSULES CONTAINING MUCK-UP HALLAM URANIUM CARBIDE FUEL RODS WERE SUBJECTED TO NEUTRON BURSTS IN TREAT. THE FIRST CAPSULE RECEIVED TWO PRELIMINARY BURSTS AND A FINAL BURST WHICH BROUGHT FUEL TO 2800 DEGREE F. NO APPRECIABLE STRUCTURAL CHANGES. THE SECOND CAPSULE WAS ABOVE 4000 DEGREE F. NO GROSS DEFORMATION OF FUEL OR CLADDING OCCURRED. BOND BUILDING DURING THE FINAL TRANSIENT.

FUEL INTEGRITY + HALLAM (LMR) + REACTOR, LMCR + REACTOR, SAFETY RESEARCH + TREAT (PRR) + URANIUM CARBIDE + REACTOR, PULSED + THERMAL TRANSIENT + FUEL ELEMENTS

6-004027 DATED 1964
STECK RB (COMPILED BY)
LIQUID METAL FAST BREEDER REACTOR DESIGN STUDY, FINAL REPORT
WESTINGHOUSE ELECTRIC CORPORATION, PITTSBURGH
541 PAGES, 133 FIGURES, 43 TABLES, JANUARY 1964, WCAP-3251-1- OTS

THE SAFETY OF THE REACTOR IS ASSURED BY A NEGATIVE DOPPLER COEFFICIENT, A CONTROLLED EXPANSION FUEL ASSEMBLY WHICH PROVIDES A NEGATIVE COEFFICIENT OF REACTIVITY, AND THE MODULAR DESIGN WHICH PROVIDES NEUTRONICS SIMILAR TO A SMALL REACTOR. THE FUEL IS SEPARATED INTO MULTIPLE COMPARTMENTS TO MINIMIZE THE EFFECT OF FUEL MOVEMENT.

REACTOR, BREEDER + DOPPLER EFFECT + REACTOR, FAST + REACTOR, LMCR + REACTIVITY COEFFICIENT + URANIUM CARBIDE + DESIGN STUDY

6-004479 DATED 1963
UTTER JM
ENERGY RELEASE IN AN HNPF UC FUEL SLUG IN TREAT, AND NUCLEAR EFFECTS OF THE TEST ON THE TREAT CORE
ATOMICS INTERNATIONAL, DIVISION OF NORTH AMERICAN AVIATION, INC.
11 PAGES, 7 REFERENCES- APRIL 30, 1963, NAA-SR-MEMO-8456- JTS

CALCULATIONS OF THE ENERGY RELEASE PER UNIT VOLUME OF THE FUEL SLUG PER TOTAL ENERGY RELEASE IN THE TREAT CORE ARE DESCRIBED. NUCLEAR EFFECTS OF THE EXPERIMENT ON TREAT PERFORMANCE ARE MINOR EVEN IN EXTREME CIRCUMSTANCES. THE ONLY NOTABLE EFFECT IS THE GROSS REACTIVITY CHANGE OF -2 % FROM REPLACING THE CENTRAL DUMMY ELEMENT OF THE REACTOR WITH THE EXPERIMENT.

CONTAINMENT, PRESSURE VENTING + HALLAM (LMR) + REACTOR, LMCR + REACTOR, TEST + TREAT (PRR) + URANIUM CARBIDE

6-005093 DATED 1964
PECKINPAUGH CL
DESIGN OF AN EXPERIMENTAL BOWABLE FUEL ELEMENT FOR THE SRE
ATOMICS INTERNATIONAL
45 PAGES, 1 TABLE, 11 FIGURES, 4 REFERENCES- OCTOBER 15, 1964, NAA-SR-10152- OTS

AN EXPERIMENTAL BOWABLE FUEL ELEMENT WAS DEVELOPED TO STUDY TEMPERATURE OSCILLATIONS IN THE SECOND CORE LOADING OF THE SRE. THE BOWABLE ELEMENT FEATURED TWO SETS OF WIRE SPRING MECHANISMS WHICH WERE REMOTELY OPERATED DURING AT-POWER REACTOR OPERATION. ONE SET BOWED THE FIVE FUEL RODS OUTWARD TO THE COOLANT CHANNEL WALL WHILE THE SECOND SET EXTENDED BETWEEN THE FUEL RODS TO THE CHANNEL WALL.

FUEL ELEMENT BOWING + REACTOR, LMCR + SRE (RE)

6-005094 DATED 1964
ARONCHICK MP
NUCLEAR STARTUP EXPERIMENTS FOR HNPF
ATOMICS INTERNATIONAL
26 PAGES, 2 TABLES, 3 FIGURES, 6 REFERENCES- OCTOBER 15, 1964, NAA-SR-10078- OTS

THE IMPORTANT RESULTS OF THE NUCLEAR STARTUP EXPERIMENTS AT THE HALLAM NUCLEAR POWER FACILITY (HNPF) ARE PRESENTED AND THEN COMPARED WITH THE THEORETICAL PREDICTIONS AS FOLLOWS- (1) THE DRY CRITICAL LOADING WAS 30.6 ELEMENTS, COMPARED WITH AN INITIAL PREDICTION OF 23 AND AN UPDATED PREDICTION OF 32. THE CORRESPONDING WET CRITICAL VALUES WERE 52, 37, AND 50, RESPECTIVELY. (2) THE EXPERIMENTALLY DETERMINED EXCESS REACTIVITY FOR 140 ELEMENTS AT AN ISOTHERMAL TEMPERATURE OF 330 DEGREE F WAS 7.5% DELTA K/K, COMPARED WITH A PREDICTED 6.9% DELTA K/K. (3) THE EXPERIMENTAL SODIUM WORTH WAS 4.1% DELTA K/K, COMPARED WITH PREDICTIONS OF 4.8% DELTA K/K. (4) TOTAL ROD WORTH WAS MEASURED AS 12.5% DELTA K/K, COMPARED WITH PREDICTIONS OF 12.7% DELTA K/K. OTHER QUANTITIES COMPARED AND DISCUSSED ARE POWER COEFFICIENTS AND XENON WORTH.

CONTAINMENT, PRESSURE VENTING + CONTROL ROD WORTH + REACTIVITY, EXCESS + HALLAM (LMR) + REACTOR, LMCR +

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-005094 *CONTINUED*
REACTIVITY COEFFICIENT + XENON

6-005104 DATED 1964
PECKINPAUGH CL
TEMPERATURE OSCILLATIONS IN THE SRE
ATOMICS INTERNATIONAL
37 PAGES, 13 FIGURES, 10 REFERENCES- NAA-SR-10151, OCTOBER 15, 1964- OTS

WITH THE SECOND U-TH LOADING OF THE SRE, AT 50% OF FULL POWER, FUEL ELEMENT TEMPERATURE OSCILLATIONS OF 100 DEGREE F OCCURRED. THESE WERE REDUCED BY SPREADING APART THE ROD BUNDLE. THEY WERE DUE TO MOTION OF THE ENTIRE ROD CLUSTER. DESIGN CRITERIA FOR NONOSCILLATING ELEMENT- (1) MINIMIZE MOVEMENT AND CENTER THE ELEMENT WITHIN THE COOLANT CHANNEL, (2) PROVIDE MIXING OF SODIUM IN THE CHANNEL, AND (3) MATCH REGIONAL POWER GENERATION TO FLOW RATE WITHIN THE CLUSTER.

REACTOR, LMCR + REACTOR STABILITY + THERMAL ANALYSIS + THERMAL MECHANICAL EFFECT + HYDRODYNAMIC ANALYSIS + SRE (RE)

6-005106 ALSO IN CATEGORY 5 DATED 1964
WEAVER LM
SODIUM REACTOR EXPERIMENT REACTIVITY HISTORY PROGRAM
ATOMICS INTERNATIONAL
28 PAGES, 4 FIGURES, 11 REFERENCES- NAA-SR-10010, OCTOBER 15, 1964- OTS

A CALCULATIONAL TECHNIQUE HAS BEEN DEVELOPED FOR USE AS AN ANALYTICAL TOOL FOR SURVEILLANCE OF THE SODIUM REACTOR EXPERIMENT (SRE) NUCLEAR PERFORMANCE. THIS PROGRAM SPECIFICALLY MAKES A COMPARISON OF THE EXCESS CALCULATED REACTIVITY AND THE MEASURED REACTIVITY DETERMINED FROM THE POSITIONS OF THE CALIBRATED SHIM RODS.

REACTIVITY, EXCESS + REACTOR, LMCR + SRE (RE)

6-005122 DATED 1962
TOBIAS ML
REACTOR STABILITY EXPERIENCE
OAK RIDGE NATIONAL LABORATORY
4 PAGES, 4 REFERENCES- NUCLEAR SAFETY, 3(3)- 16-18 AND 24 (MARCH 1962)

OLD SURVEY OF STABILITY OF EBR-I, SRE, AND HRT.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, HOMOGENEOUS + HRT (RE) + REACTOR, LMCR + REACTOR STABILITY + SRE (RE)

6-005759 DATED 1964
LURIE H + NOYES RL
BOILING STUDIES FOR SODIUM REACTOR SAFETY - PART II - POOL BOILING AND INITIAL FORCES CONVECTION TESTS AND ANALYSES
ATOMICS INTERNATIONAL
83 PAGES, 25 FIGURES, 6 TABLES, 19 REFERENCES, OCTOBER 15, 1964, NAA- SR-9477- CFSTI

PRESENTED ARE RECENT RESULTS OBTAINED ON THE POOL-BOILING CHARACTERISTICS AND THE CRITICAL HEAT FLUX OF SATURATED SODIUM BETWEEN 0.5 AND 8 PSIA AND ALSO THE FORCED CONVECTION BOILING CHARACTERISTICS AND CRITICAL HEAT FLUX FOR PRESSURES OF 2 AND 5 PSIA, FLOW RATES OF 0.75 TO 3 GPM, AND AVERAGE SUBCOOLINGS OF 5 TO 116 DEGREES F. CRITICAL-HEAT-FLUX MEASUREMENTS FOR POOL BOILING ON A CYLINDRICAL HEATER RANGED FROM 300,000 TO 1,000,000 BTU/HR-FT², AND INCREASED WITH THE ABSOLUTE PRESSURE TO THE 0.17 POWER. A MODIFIED, GENERAL, NONDIMENSIONAL CORRELATION WHICH INCORPORATES THESE DATA IS PRESENTED. THE FORCED-CONVECTION DATA OBTAINED AGREE QUITE WELL WITH ESTABLISHED CORRELATIONS. CONVECTIVE HEAT TRANSFER DOMINATES UP TO THE CRITICAL HEAT FLUX. CRITICAL HEAT FLUXES UP TO 1.5 X 10⁶ BTU/HR-FT² HAVE BEEN OBTAINED. BURNOUT MARGIN OF SIX WAS CALCULATED FOR A SIMULATED SRE-PEP REACTOR. THE THEORETICAL BASIS OF A DIGITAL COMPUTER CODE TRANS-FUGUE TO BE USED FOR THE TRANSIENT ANALYSIS OF SINGLE CHANNEL SODIUM BOILING, IS PRESENTED.

BURNOUT HEAT FLUX + REACTOR, LMCR + THERMAL ANALYSIS

6-005807 ALSO IN CATEGORIES 12 AND 17 DATED 1965
EBR-II
ARGONNE NATIONAL LABORATORY
6 PAGES, 2 TABLES- MARCH 1965, ANL-7028, SECTION I, PP. 1-6 OF ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT- CFSTI

EBR-II REINVESTIGATION OF THE FAILURE OF THE DIESEL GENERATOR. CONTROL ROD CALIBRATION, SMALL LEAKS IN SEAL OF PERSONNEL AIR LOCK, OTHER MINOR FAILURES.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + CONTROL ROD WORTH + EBR 1 AND 2 (RE) + FAILURE + REACTOR, FAST + REACTOR, LMCR + CONTAINMENT AIR LOCK + GENERATOR, DIESEL

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-005810
NUCLEAR SAFETY
ARGONNE NATIONAL LABORATORY
6 PAGES, 1 TABLE- MARCH 1965, ANL-7028, SECTION IV, PP. 79-84 OF ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT- CFSTI

DATED 1965

EXPERIMENTAL SET-UP FOR STUDIES OF SODIUM EXPULSION FROM SIMULATED REACTOR CHANNEL. CALCULATION OF SODIUM VOIDING AND OF PRESSURE PULSES. TREAT TESTS OF IRRADIATED PU-U-OXIDE AND OF PBF FUEL SAMPLES. AMONG FUEL SAMPLES EXPOSED TO THE KIWI-TNT TEST, ALL OF THE URANIUM ALLOY PINS AND TWO OF THE ALUMINUM ALLOY PLATES MELTED AND APPARENTLY UNDERWENT SUBSTANTIAL METAL-WATER REACTION. ONLY ONE OF EACH TYPE OF METAL CLAD UO₂-CORE SPECIMENS WAS FRAGMENTED.

FAILURE, FUEL ELEMENT + REACTOR, LMCR + METAL WATER REACTION + REACTOR TRANSIENT + ROVER PROGRAM + THERMAL ANALYSIS + TREAT (PKR) + PLUTONIUM OXIDE + PBF (S-RR) + KIWI (S-RR)

6-006227
PITKANEN PH
ARGONNE TRANSIENT REACTOR TEST FACILITY
OAK RIDGE NATIONAL LABORATORY
6 PAGES, 5 TABLES, 19 REFERENCES- NUCLEAR SAFETY, 5(3)- 234-239 (SPRING 1964)- GPO

DATED 1964

THIS REPORT CONTAINS A NOW SUPERSEDED REVIEW OF TREAT TESTS UP TO SEPTEMBER 1963 ON SLIGHTLY ENRICHED URANIUM OXIDE, STAINLESS STEEL CLAD PELLETS- INERT-GAS-BONDED URANIUM OXIDE FUELED EBR-II ELEMENTS (UNIRRADIATED AND IRRADIATED)- PREIRRADIATED SODIUM-BONDED EBR-II ELEMENTS- SINTERED URANIUM SULFIDE PELLETS- ENRICO FERMI TYPE FUEL ELEMENTS- AND THEORETICAL ANALYSIS.

ACCIDENT ANALYSIS + REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + REACTOR, LMCR + REACTOR, TEST + TREAT (PKR) + URANIUM SULFIDE

6-006312
ADAMS RM + GLASSNER A
SODIUM AIR REACTION + FARET CRITICALS
ARGONNE NATIONAL LABORATORY
5 PAGES, 3 FIGURES, 1 TABLE, 2 REFERENCES- PAGES 11-15 FROM REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 19, 1965, ANL-7045- CFSTI

DATED 1965

REGARDING EBR-II, CONTROL ROD CALIBRATION, DELAYED NEUTRON KINETICS, POWER COEFFICIENT, REACTIVITY GAIN BY FUEL REPLACEMENT. (FEW NUMERICAL VALUES GIVEN.) REGARDING FARET, FAIRLY DETAILED DISCUSSION OF TEMPERATURES AND PRESSURES CAUSED BY SODIUM FIRES. DATA ON REACTIVITY COEFFICIENTS, REACTIVITY WORTH AND PROMPT-NEUTRON LIFETIME.

REACTOR, BREEDER + CHEMICAL REACTION + CONTAINMENT, HIGH PRESSURE + CONTROL ROD WORTH + DELAYED NEUTRON + EBR 1 AND 2 (RE) + FIRE + REACTOR, LMCR + POWER COEFFICIENT + REACTIVITY COEFFICIENT + SODIUM + REACTOR, TEST + AIR + FARET (TR) + PROMPT NEUTRON LIFETIME + CONTROL ROD CALIBRATION

6-006325
BREIZY CE
SODIUM-COOLED REACTORS PROGRAM - FAST CERAMIC REACTOR DEVELOPMENT PROGRAM
GENERAL ELECTRIC COMPANY, SAN JOSE
1 PAGE, REFERENCES- JULY, 1964, GEAP-4640, SECTION 7 (PP. 7.3 AND 7.4)- CFSTI

DATED 1964

COMPUTER CODE FOR CALCULATION OF TRANSIENT TWO-PHASE SODIUM FLOW IN A HEATED CHANNEL FOR UNDERSTANDING OF SODIUM EXPULSION EVENTS. IT WAS NECESSARY TO INCLUDE PROVISIONS FOR CALCULATING THE SUPERHEATED VAPOR STATE, AS WELL AS THE SUBCOOLED LIQUID AND TWO-PHASE STATES. ALTERNATE CORE CONFIGURATIONS ARE UNDER INVESTIGATION AS A MEANS OF MINIMIZING THE COOLANT VOID REACTIVITY EFFECT.

COMPUTER, DIGITAL + REACTOR, FAST + REACTOR, LMCR + THERMAL ANALYSIS + VOID COEFFICIENT

6-006331
PECKINPAUGH CL
TEMPERATURE DISTRIBUTIONS IN THE SRE CORE - II FUEL ELEMENT
ATOMICS INTERNATIONAL
21 PAGES, 9 FIGURES, 6 REFERENCES- FEBRUARY 1, 1965, NAA-SR-10153- CFSTI

DATED 1965

MEASUREMENTS WITH AN EXPERIMENTAL BOWABLE FUEL CLUSTER SHOWED LARGE STEADY-STATE TEMPERATURE DIFFERENCES WITHIN THE COOLANT CHANNEL. THE DIFFERENCE WAS CAUSED BY (THERMAL BOWING) MISALIGNMENT OF THE FUEL ROD CLUSTER AND UNBALANCE OF POWER GENERATION AND FLOW RATE AND BY HIGHER VELOCITY SODIUM BYPASSING THE ROD BUNDLE. FUTURE FUEL ELEMENT DESIGNS MAY MINIMIZE THESE TEMPERATURE DIFFERENCES BY (1) MAINTAINING THE FUEL ROD BUNDLE CONCENTRIC IN THE COOLANT CHANNEL, (2) MATCHING THE REGIONAL POWER GENERATION WITH FLOW RATE, AND (3) INCREASING MIXING BY MINIMIZING THE CLEARANCE BETWEEN THE WIRE WRAPS AND THE CHANNEL WALL.

FUEL ELEMENT BOWING + REACTOR, LMCR + THERMAL ANALYSIS + SRE (RE)

6-006357
ALSO IN CATEGORIES 9 AND 17
DATED 1964

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-006357 *CONTINUED*

DARLEY DK + JENKINS OG + LARSON CL + LEW DE + MILLS RM + PEARSON EN + RUBINSTEIN HJ + STIENS RR + TESZLER O + WALTERS RJ + WEAVER LM

NUCLEAR ANALYSIS

ATOMICS INTERNATIONAL

45 PAGES, 25 FIGURES, 7 TABLES, 3 REFERENCES- PAGES 9-53, SECTION III, FROM HALLAM NUCLEAR POWER FACILITY REACTOR OPERATIONS ANALYSIS PROGRAM SEMIANNUAL PROGRESS REPORT NO. 3, SEPTEMBER 1, 1963 - FEBRUARY 29, 1964- SEPTEMBER 15, 1964, NAA-SR-9799, CFSTI

REPLACEMENT OF ZIRCALOY-II CONTROL ROD THIMBLES WITH STAINLESS STEEL THIMBLES AND ADDITION OF 10 (UC) FUEL ELEMENTS. MODERATOR ELEMENT FAILURES LED TO SODIUM FLOODING OF THE GRAPHITE MODERATOR. EFFECT OF FUEL BURNUP IS NOT AS GREAT AS CURRENTLY USED. FUEL ELEMENT VARIABLE ORIFICES BECOMING STUCK. 20 REACTOR SCRAMS HAVE BEEN ANALYZED WITH RESPECT TO CAUSE, CONSEQUENCES, AND MEANS OF REDUCING SCRAM FREQUENCY.

FAILURE, COMPONENT + CONTAINMENT, PRESSURE VENTING + CONTROL SYSTEM + HALLAM (LMR) + REACTOR, LMCR + SCRAM, REAL + URANIUM CARBIDE + REPORT, OPERATIONS ANALYSIS

6-006561 ALSO IN CATEGORIES 9 AND 17 DATED 1965

EBR-II

ARGONNE NATIONAL LABORATORY

6 PAGES, 1 FIGURE, 2 TABLES- JUNE 25, 1965, ANL-7046- PP. 1-6, SECTION A, OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 1965- CFSTI

MEASUREMENT OF POWER COEFFICIENT- FISSION PRODUCT MONITOR- LEAK RATE TESTS, CONTROL ROD MALFUNCTION, REACTIVITY MONITORING DEVICES, FEEDBACK OF AXIAL FUEL EXPANSION.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + REACTIVITY EFFECT, EXPANSION + REACTOR, FAST + FISSION PRODUCT RELEASE + REACTOR, LMCR

6-006673 ALSO IN CATEGORY 17 DATED 1965

NUCLEAR SAFETY

ARGONNE NATIONAL LABORATORY

6 PAGES, 1 TABLE, 2 FIGURES- JULY 30, 1965, ANL-7071- PAGES 60-66 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JUNE 1965, CFSTI

COOLANT EXPULSION BY VAPORIZED COOLANT STUDIED WITH THEORETICALLY RESPECT TO SCALING LAWS. DISCUSSION OF PRESSURE TRANSDUCERS FOR EXPERIMENT ON THIS EXPULSION. TEST OF SAMPLES STUDIED IN TREAT. ADMINISTRATIVE REMARKS ON LARGE TREAT LOOP. FLOW MEASUREMENT IN TREAT INTEGRAL SODIUM LOOPS. REACTIVITY CALCULATIONS OF TREAT WITH EBR-II SUBASSEMBLY. APPARATUS FOR HEAT TRANSFER FROM HOT SPHERE MOVING THROUGH WATER AND PRELIMINARY RESULTS.

REACTOR, BREEDER + COMPUTER PROGRAM + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + REACTIVITY, EXCESS + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + REACTOR, TEST + THERMAL ANALYSIS + TREAT (PRR) + INSTRUMENT, FLOW

6-006974 ALSO IN CATEGORY 1 DATED 1965

AMERICAN NUCLEAR SOCIETY FAST REACTOR TECHNOLOGY NATIONAL TOPICAL MEETING

363 PAGES, FIGURES, TABLES, REFERENCES- APRIL 28, 1965, ANS-100 (CONF-650417)- ANS NATIONAL TOPICAL MEETING ON FAST REACTOR TECHNOLOGY, DETROIT, MICHIGAN, APRIL 26-28, 1965

TWENTY-EIGHT PAPERS IN SESSIONS ON REACTOR OPERATING EXPERIENCE, ADVANCED CONCEPTS, FUEL TECHNOLOGY, PHYSICS AND SHIELDING, SAFETY PROGRAMS AND CORE DESIGN, AND MECHANICAL COMPONENTS AND SODIUM TECHNOLOGY.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + RADIATION DAMAGE + REACTIVITY EFFECT + REACTOR KINETICS + REACTOR STABILITY + POWER DISTRIBUTION + OPERATING EXPERIENCE SUMMARY + DESIGN STUDY + DOWNEY (LMFBR) + SEFOR (RE) + SHIELDING + MAIN COOLING SYSTEM + FUEL ELEMENTS

6-007108 ALSO IN CATEGORY 17 DATED 1965

PLUTONIUM RECYCLE REACTORS

ARGONNE NATIONAL LABORATORY

4 PAGES, 1 REFERENCE- JUNE 25, 1965, ANL-7046- PP. 63-66 OF REACTOR DEVELOPMENT PROGRESS REPORT, MAY 1965- CFSTI

EBR II WAS OPERATED WITH ONE SINGLE UNCLAD FUEL PIN TO TEST FISSION PRODUCT MONITOR CALIBRATION. ANALYSIS OF ROD DROP MEASUREMENTS AT 45 MW SHOWS AT LEAST 16% OF THE REACTIVITY (APPROXIMATELY $.75 \times 10^{-4}$ DELTA K/K) IS RECOVERED IN LESS THAN 0.5 SEC, PROBABLY ASSOCIATED WITH AXIAL FUEL EXPANSION. METALLOGRAPHIC EXAMINATION OF CRACKED 304 SS CLADDING, WHICH HAD BEEN IN EBR FOR 10 YEARS, REVEALED EACH GRAIN OF STEEL SURROUNDED BY BRITTLE CARBIDES, WHICH HAD PRECIPITATED FROM THE SOLID SOLUTION STAINLESS STEEL DURING THE 1150 DEGREE F STRESS RELIEF HEATING. FIVE SURFACE CRACKS WERE FOUND ON OTHER TESTS.

REACTOR, BWR + REACTOR, BREEDER + EBR 1 AND 2 (RE) + EBR (BWR) + REACTOR, FAST + FUEL ELEMENT BOWING + REACTOR, LMCR + PRESSURE VESSELS + REACTIVITY EFFECT + ANL + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + MATERIAL

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-007247 DATED 1965
EGGEN DT + MOREWITZ HA
FEASIBILITY STUDY OF A 1000-MWE SODIUM-COOLED FAST REACTOR. VOLUME III - METHODS DEVELOPMENT
ATOMIC INTERNATIONAL
184 PAGES, TABLES, FIGURES, REFERENCES- JULY 25, 1965, NAA-SR-11378 (VOL. III)- CFSTI

TOPICS OF POTENTIAL SAFETY INTEREST ARE- REACTIVITY COEFFICIENTS, TWO-DIMENSIONAL SYNTHESIS OF
SODIUM VOID EFFECTS, THERMAL AND HYDRAULICS PERFORMANCE CALCULATIONS, TRANSIENT CALCULATIONAL
METHODS, ACCIDENT ANALYSIS, DESCRIPTION OF AIREX 3C, TREAT MELTDOWN STUDIES, AND WEAK-EXPLOSION
PROGRAM.

REACTOR, FAST + FUEL MELTDOWN + REACTOR, GRAPHITE MODERATED + HYDRAULIC ANALYSIS + REACTOR, LMCR + REACTIVITY
COEFFICIENT + REACTOR TRANSIENT + REACTOR, TEST + THERMAL ANALYSIS + TREAT (PRR) + VOID COEFFICIENT +
DESTRUCTIVE TRANSIENT + DESIGN STUDY + REACTIVITY EFFECT, VOID

6-007435 ALSO IN CATEGORIES 11 AND 17 DATED 1965
EBR-II
ARGONNE NATIONAL LABORATORY
7 PAGES, 5 TABLES- AUGUST 24, 1965, ANL-7082, PP 1-7 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT JULY 1965,
CFSTI, \$3.00 CY, \$0.75 MN

REGARDING EBR-II 10- SUBASSEMBLIES CONTAINING SPECIAL SURVEILLANCE FUEL PINS SWELLED FURTHER THAN
NORMAL PINS AND WERE REPLACED. NORMAL PRODUCTION PINS ALSO SWELLED SOMEWHAT. OXIDES OF BISMUTH,
TIN AND SODIUM FORMED IN THE SEAL TROUGH AND INTERFERED WITH ROTATION OF THE PLUGS AND THE
TEMPERATURE DISTRIBUTION OF THE TROUGHS. PUMP M-1 FROZE TWICE, APPARENTLY BECAUSE OF SODIUM
COMBUSTION IN SHAFT. CONCRETE OF PUMP SHIELD PLUGS GIVES OFF WATER. MINOR CONTROL ROD DRIVE
TROUBLE. CONTAINMENT TESTING. CONTROL ROD WORTH AND POWER COEFFICIENT.

REACTOR, BREEDER + FAILURE, COMPLAINT + CONTROL ROD WORTH + EBR 1 AND 2 (RE) + REACTOR, FAST + FAILURE, FUEL
ELEMENT + REACTOR, LMCR + POWER COEFFICIENT + TEST, LEAK RATE + OPERATING EXPERIENCE SUMMARY + PUMPS +
CONTROL ROD DRIVE MECHANISMS

6-007436 ALSO IN CATEGORIES 9 AND 11 DATED 1965
FARET
ARGONNE NATIONAL LABORATORY
17 PAGES, 4 TABLES- AUGUST 24, 1965, ANL-7082, PP 7-17 OF REACTOR DEVELOPMENT PROGRAM PROGRESS
REPORT JULY 1965, CFSTI, \$3.00 CY, \$0.75 MN

FOR FARET, TESTS OF CONTROL ROD DRIVES, LEAK TEST OF VAULT HATCH SEAL, MEASUREMENTS OF CRITICAL
CONTROL ROD WORTH. DANGER COEFFICIENTS OF COMPONENTS FROM OTHER ZONES OF SAME ASSEMBLIES.

REACTOR, BREEDER + CONTROL ROD WORTH + REACTOR, FAST + REACTOR, LMCR + TEST, LEAK RATE + FFTF (TR) + DANGER
COEFFICIENT + FUEL HANDLING MACHINE + CONTROL ROD DRIVE MECHANISMS

6-007441 ALSO IN CATEGORY 9 DATED 1965
FARET
ARGONNE NATIONAL LABORATORY
7 PAGES, 1 FIGURE, 1 TABLE- SEPTEMBER 21, 1965, ANL-7090, PP 7-14 OF REACTOR DEVELOPMENT PROGRAM PROGRESS
REPORT, AUGUST 1965, CFSTI, \$3.00 CY

FOR FARET- TESTS OF CONTROL ROD DRIVES, DANGER COEFFICIENT OF PU, U-235, U-238 AND B-10, NA,
STAINLESS STEEL, DEPLETED U, U3O8, GRAPHITE, ZR.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + FFTF (TR) + DANGER COEFFICIENT + CONTROL ROD DRIVE
MECHANISMS

6-007444 ALSO IN CATEGORIES 8 AND 17 DATED 1965
REACTOR KINETICS
ARGONNE NATIONAL LABORATORY
9 PAGES, 3 FIGURES- SEPTEMBER 21, 1965, ANL-7090, PP 69-75 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT,
AUGUST 1965, CFSTI, \$3.00 CY

EBR-II- MARK 1 FUEL PIN FAILURE IN TREAT (FUEL VIGOROUSLY EXPELLED BY SODIUM VAPOR PRESSURE).
INSTRUMENTATION FOR COOLANT EXPULSION STUDIES. DURING ACCIDENTS, CRITICAL FLOW MAY EXIST AT THE
DISCHARGE END OF COOLANT CHEMICALS, RESULTING IN LARGE PRESSURE GRADIENTS. A PHYSICALLY
MEANINGFUL WEIGHTING FUNCTION WAS DERIVED FOR SPACE-TIME REACTOR DYNAMICS. IRRADIATION OF METAL-
WATER REACTION SAMPLES FOR TREAT. HFIR FUEL ELEMENT SAMPLES MELTED IN TREAT TRANSIENTS,
RESULTING IN SOME METAL-WATER REACTIONS.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR,
GRAPHITE MODERATED + HYDRAULIC ANALYSIS + REACTOR, LMCR + SODIUM + SPACE DEPENDENT DYNAMICS + REACTOR, TEST +
TREAT (PRR) + HFIR (FTR) + PRESSURE, INTERNAL

6-007670 ALSO IN CATEGORIES 12 AND 17 DATED 1965
LIQUID-METAL-COOLED REACTORS
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-007670 *CONTINUED*
87 PAGES, TABLES, FIGURES- ANL-7105, OCTOBER 20, 1965, PP. 1-35 OF ARGONNE NATIONAL LABORATORY REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, SEPTEMBER, 1965

EBR-II - FUEL IRRADIATED PAST 0.8 ATOM PERCENT BURNUP SHOWS AN ACCELERATING RATE OF VOLUME INCREASE WITH BURNUP. FUEL VOLUME INCREASES OVER 8 PERCENT WILL REQUIRE PLASTIC STRAIN OF CLADDING. FUEL WILL BE EXAMINED AFTER EACH OPERATING CYCLE. CP-5 URANIUM DIOXIDE FUEL WAS OPERATED BRIEFLY WITH A CENTERLINE TEMPERATURE OF 2400 DEGREES CENTIGRADE WITH CP-5 AT MAXIMUM (4.2 MW) POWER. CLADDING WAS SEPARATED FROM FUEL WITH TANTALUM RODS IN ARGON ATMOSPHERE. TUNGSTEN - RHENIUM THERMOCOUPLES MEASURED THE CENTERLINE TEMPERATURE.

CRITICAL ASSEMBLY + DOPPLER EFFECT + EBR 1 AND 2 (RE) + FUEL BURNUP + FUEL INTEGRITY + IN PILE EXPERIMENT + INSTRUMENT, IN CORE + REACTOR, LMCR + MEASUREMENT + FFIF (TR) + IN PILE EXPERIMENT + IRRADIATION TESTING + ZPPR (CAF)

6-007695 ALSO IN CATEGORIES 1 AND 9 AND 17 DATED 1965
OKRENT D
A LOOK AT FAST REACTOR SAFETY
ARGONNE NATIONAL LABORATORY
25 PAGES, 1 FIGURE, 2 TABLES, 131 REFERENCES- NUCLEAR SAFETY 6(4)- 317-342 (SUMMER, 1965)

DESIGN HISTORY (DOUNREAY REACTOR, EBR-II, ENRICO FERMI, PATTERNS AND TRENDS, SAFETY-RELATED DESIGN AND OPERATING EXPERIENCE). DIFFERENCES IN NUCLEAR SAFETY FOR LARGE-SODIUM-COOLED FAST REACTORS (DELAYED-NEUTRON FRACTION, DOPPLER COEFFICIENT, SODIUM COEFFICIENT AND VOID EFFECT, STABILITY, MAXIMUM ACCIDENT CALCULATIONS). LARGE-SODIUM-COOLED-FAST REACTOR DESIGNS AND ACCIDENT STUDIES, NON-NUCLEAR SAFETY CONSIDERATIONS, COOLANTS OTHER THAN SODIUM (STEAM, GAS, OTHER METAL COOLANTS), SAFETY PHILOSOPHY AND SAFETY RESEARCH, REFERENCES.

REACTOR, BREEDER + DELAYED NEUTRON + DOPPLER COEFFICIENT + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LARGE + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR STABILITY + SODIUM + SODIUM COEFFICIENT + STEAM + VOID COEFFICIENT + OPERATING EXPERIENCE SUMMARY + HELIUM + DOUNREAY (LMFBR) + SAFETY PRINCIPLES AND PHILOSOPHY

6-007848 ALSO IN CATEGORY 17 DATED 1965
SMITH RR + DOE CB + HAROLDSEN RD + MCGINNIS FD + NOVICK M
TERMINAL REPORT FOR THE MARK IV (PLUTONIUM) LOADING IN EBR-I
ARGONNE NATIONAL LABORATORY
94 PAGES, 45 FIGURES, 20 TABLES, 34 REFERENCES- APRIL 1965, ANL-6865- CFSTI, \$3.00 CY

DESCRIBES DESIGN OF VARIOUS SYSTEMS, CRITICAL LOADING AND OPERATIONS, BREEDING GAIN MEASUREMENTS, STABILITY STUDIES, AND FISSION PRODUCT (IN COVER GAS) MONITOR DEVELOPMENT. A REACTIVITY ANOMALY WAS ASCRIBED TO VARIATION OF SUB-ASSEMBLY SPACING WITH TEMPERATURE.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + MEASUREMENT + PLUTONIUM + REACTOR STABILITY + TRANSFER FUNCTION + OPERATING EXPERIENCE SUMMARY

6-008090 DATED 1965
AMUNDSON PI + GASIDLO JM
DOPPLER COEFFICIENT MEASUREMENTS FOR URANIUM-238 IN FAST-REACTOR SPECTRA
ARGONNE NATIONAL LABORATORY
2 PAGES, 2 TABLES, 11 REFERENCES- NUCLEAR SCIENCE AND ENGINEERING 23(4)- 392-393 (DECEMBER 1965)

MEASUREMENT OF THE URANIUM-238 DOPPLER COEFFICIENT IN ZPR-III ASSEMBLY 45A, A ZONED REACTOR CONTAINING A CENTRAL REGION WHOSE COMPOSITION REPRESENTS A LARGE SODIUM-COOLED FAST POWER-BREEDER REACTOR HAVING PLUTONIUM-239 AND URANIUM-238 MONOCARBIDE FUEL. THE RELATIVE WORTH OF THE DOPPLER ELEMENT WAS MEASURED WITH RESPECT TO A REFERENCE MATERIAL WHEN THE DOPPLER ELEMENTS WERE AT 500 DEGREE K AND AT 1100 DEGREE K. AN AUTOROD OSCILLATION METHOD WAS USED FOR DETERMINATION OF THE REACTIVITY CHANGE.

CRITICAL ASSEMBLY + DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + MEASUREMENT + URANIUM

6-008211 ALSO IN CATEGORY 9 DATED 1964
SHAW PF
HNPF CONTROL SYSTEM RESPONSE DURING TRANSIENT CONDITIONS
ATOMICS INTERNATIONAL
36 PAGES, 7 FIGURES, 7 TABLES, 1 REFERENCE- JUNE 15, 1964, NAA-SR-9584- CFSTI, \$0.75 CY

THIS REPORT DOCUMENTS THE RESULTS OF POWER RAMPS IMPOSED ON THE HNPF DURING THE MONTH OF JANUARY 1964. ANALYSIS AND EVALUATION OF THE DATA ACQUIRED DURING THESE TESTS VERIFIES THE ADEQUACY OF PERFORMANCE OF THE HNPF PLANT CONTROL SYSTEM. IT WAS DEMONSTRATED, WITHIN THE CONDITIONS OF THE TEST, THAT THE HNPF IS CAPABLE OF AUTOMATICALLY FOLLOWING STATION LOAD CHANGE RATES IN EXCESS OF THE 5 MWE/MIN. TEST OBJECTIVE. DATA FROM THE RAMP TESTS AND OTHER PERTINENT INFORMATION ARE INCLUDED IN THIS REPORT TO PERMIT THE READER TO MAKE AN INDEPENDENT EVALUATION OF CONTROL SYSTEM RESPONSE TO THE IMPOSED POWER RAMPS.

CONTROL SYSTEM + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + TEST, PLANT RESPONSE + REACTOR POWER + TEST, SYSTEM OPERABILITY

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-009709 DATED 1965
LOEWENSTEIN WB + KIRN F + SMITH RR
EBR-II PHYSICS. REACTOR PHYSICS DIVISION ANNUAL REPORT, JULY 1, 1964 TO JUNE 30, 1965
ARGONNE NATIONAL LABORATORY
ANL-7110 +. 2 PAGES, 7 REFERENCES, DECEMBER 1965, SECTION III-12, PP. 151-152, CFSTI, \$7.22 CY, \$2.00 MN.

THE REACTOR IS STABLE AT ALL POWER LEVELS OF OPERATION. THE POWER COEFFICIENT IS ESSENTIALLY CONSTANT. AN UNEXPECTED OBSERVATION PERTAINS TO SMALL 10-CPS OSCILLATIONS. THE REACTIVITY OSCILLATOR ROD FAILED.

REACTOR, BREEDER + FAILURE, COMPONENT + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + OSCILLATOR, REACTIVITY + POWER COEFFICIENT + REACTOR STABILITY

6-009711 ALSO IN CATEGORY 7 DATED 1965
DICKERMAN CE + ROBINSON LE + AUGUST C + STEWART R + STEPHANY W + WILLIS L + CARLANDER R + MUELLER M
PHYSICS DIVISION ANNUAL REPORT, JULY 1, 1964 TO JUNE 30, 1965
ARGONNE NATIONAL LABORATORY
ANL-7110 +. 9 PAGES, FIGURES, TABLES, REFERENCES, DECEMBER 1965, SECTIONS III-15 THROUGH III-19, PP. 157-165, CFSTI, \$7.22 CY, \$2.00 MN.

IN TREAT TRANSPARENT MELTDOWN CAPSULES, THE FOLLOWING SAMPLES WERE SUBJECTED TO TRANSIENTS FOR THE PURPOSES INDICATED - U-5 W/O FISSION ALLOY SAMPLES PREIRRADIATED TO 1.3 AT. PERCENT BURNUP, MEASUREMENT OF EXPANSION. EBR-II FUEL PIN PREIRRADIATED TO 0.6 AT. PERCENT, STUDY OF FAILURE MECHANISM. TH-U METALLIC FUEL ELEMENTS OF THE EBR-II MARK I TYPE, STUDY OF FAILURE MECHANISM (SEE ALSO TRANS. AMERICAN NUCLEAR SOCIETY, 8(1), P. 309 (1965). EBR-II TYPE US FUELED, W-RE ALLOY CLAD ELEMENTS, STUDY OF FAILURE, IF ANY (SEE ALSO NUCLEAR APP. 1, P. 168 (1965)). IN THE INTEGRAL SODIUM LOOP EBR-II PINS WERE SUBJECTED TO 3 TRANSIENTS WITH SODIUM FLOW AND THREE TRANSIENTS WITHOUT FLOW. FUEL-CLADDING EUTECTIC WAS FORMED BY MOLTEN FUEL PENETRATING FROM THE CENTER OF THE ROD TO THE SURFACE.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FUEL BURNUP + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, GRAPHITE MODERATED + IN PILE EXPERIMENT + REACTOR, LMCR + REACTOR, SAFETY RESEARCH + REACTOR, RESEARCH + REACTOR, TEST + THORIUM + TREAT (PRR) + URANIUM + URANIUM SULFIDE + HIGH TEMPERATURE + IRRADIATION TESTING + REACTOR, PULSED + FAILURE, CLADDING + PRESSURE, INTERNAL + FUEL ELEMENTS

6-009822 DATED 1966
AKCASU AZ + NOBLE LD
A NONLINEAR STUDY OF REACTORS WITH A LINEAR FEEDBACK
UNIVERSITY OF MICHIGAN
11 PAGES, 4 FIGURES, 1 TABLE, NUCLEAR SCIENCE AND ENGINEERING, 25(1), PP. 47-57 (MAY 1966)

SOLUTIONS OF THE POINT KINETIC EQUATIONS WITH DELAYED NEUTRONS FOR REACTOR SYSTEMS WITH ARBITRARY LINEAR FEEDBACK ARE INVESTIGATED. SOLUTIONS THAT ARE LAPLACE-TRANSFORMABLE ARE BOUNDED FOR ALL INITIAL PERTURBATIONS REGARDLESS OF WHETHER THE SYSTEM IS LINEARLY STABLE, PROVIDED THE LAPLACE TRANSFORM OF THE FEEDBACK KERNEL HAS NO ZEROS ON THE POSITIVE REAL AXIS. THERE ARE ALSO NONTRANSFORMABLE SOLUTIONS THAT POSSESS A FINITE ESCAPE TIME. THE GENERAL CONCLUSIONS ARE VERIFIED BY CONSIDERING SOME SPECIFIC FEEDBACK MODELS FOR WHICH EXACT SOLUTIONS ARE AVAILABLE. NUMERICAL SOLUTIONS FOR REACTOR SYSTEMS WITH MORE REALISTIC FEEDBACK MODELS, SUCH AS ONE USED TO DESCRIBE EBR-I, ARE OBTAINED BY A DIGITAL COMPUTER.

REACTOR, BREEDER + COMPUTER, DIGITAL + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + EQUATION, NONLINEAR + REACTOR STABILITY + TRANSFER FUNCTION + DYNAMICS, NONLINEAR + THEORETICAL INVESTIGATION

6-010057 DATED 1966
FAST REACTOR SAFETY
ARGONNE NATIONAL LABORATORY
ANL-7132 +. 2 PAGES, JANUARY 25, 1966, PAGES 75-76 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, DECEMBER, 1965, CFSTI, \$3.00

COOLANT (WATER) EXPULSION STUDIES SHOW WATER-HAMMER PRESSURE GREATER THAN INITIAL PRESSURE PULSES, WHICH INCREASE WITH INCREASING TEMPERATURE. CONSTRUCTION PROGRESS ON SODIUM SUPERHEAT EXPERIMENT. CRITICAL FLOW STUDIES SHOWED THAT (1) THE FLOWRATES WERE CONSIDERABLY HIGHER THAN THOSE PREDICTED ON THE BASIS OF PREVIOUS WORK, (2) THE CRITICAL FLOWRATE DID NOT EXHIBIT A MAXIMUM AS A FUNCTION OF QUALITY, AS PREDICTED BY LEVY AND MOODY MODELS, (3) THE ASSUMPTION OF THERMAL EQUILIBRIUM DOES NOT SEEM VALID.

HYDRAULIC ANALYSIS + REACTOR, LMCR + SODIUM + THERMAL ANALYSIS + FLOW, TWO PHASE

6-010058 DATED 1966
EQUATION OF STATE FOR FAST REACTOR ACCIDENTS. TRANSIENT-PRESSURE RESPONSE OF TRANSDUCERS AND MARK-I INTEGRAL LOOPS
ARGONNE NATIONAL LABORATORY
ANL-7132 +. 4 PAGES, JANUARY 25, 1966, PAGES 76-79 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, DECEMBER, 1965, CFSTI, \$3.00

EFFECT OF POROSITY OF FUEL ELEMENTS ON THEIR EQUATION OF STATE IS OF INTEREST FOR FAST-REACTOR SAFETY STUDIES AS IT REDUCES THE PRESSURE AND INITIAL DISASSEMBLY RATE. THIS EFFECT IS DISCUSSED. TRANSIENT PRESSURE RESPONSE AND OSCILLATIONS OBSERVED IN MARK-I INTEGRAL LOOPS.

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-010058 *CONTINUED*
THEORETICAL ANALYSIS OF TEMPERATURE DISTRIBUTION IN A SEVEN-PIN EBR-II FUEL SUBASSEMBLY FOR TREAT TESTS.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + EQUATION OF STATE + REACTOR, FAST + REACTOR, GRAPHITE MODERATED + IN PILE EXPERIMENT + REACTOR, LMCR + REACTOR, TEST + TREAT (PRR) + URANIUM + IN PILE EXPERIMENT

6-010127

DATED 1966

MELTDOWN EXPERIMENTS WITH EBR-II PIN CLUSTERS

ARGONNE NATIONAL LABORATORY

ANL-7132 +. 2 PAGES, JANUARY 25, 1966, PP. 78-79 OF ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, DECEMBER 1965, CFSTI, \$3.00 CY

THEORY AND EXPERIMENTS ON LOOP MELTDOWN EXPERIMENTS IN TREAT WITH SEVEN-PIN EBR-II TYPE OF FUEL CLUSTERS.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + THERMAL ANALYSIS + FUEL ELEMENTS

6-010139

ALSO IN CATEGORY 9

DATED 1964

KIRN FS + LOEWENSTEIN WB

EBR-II WET CRITICAL EXPERIMENTS

ARGONNE NATIONAL LABORATORY

ANL-6804 +. 46 PAGES, 15 TABLES, 25 FIGURES, OCTOBER 1964, CFSTI, \$2.00 CY

WET-CRITICAL EXPERIMENTS WERE PERFORMED IN THE EBR-II WITH SODIUM COOLANT IN THE REACTOR SYSTEM. MEASUREMENTS INCLUDED CONTROL-ROD WORTH, NEUTRON-SOURCE STRENGTH, NUCLEAR-INSTRUMENT RESPONSE, CRITICAL SIZE, ISOTHERMAL TEMPERATURE COEFFICIENT, AND REACTIVITY WORTH OF THE SODIUM COOLANT. THE MEASUREMENTS AGREED WELL WITH CALCULATED VALUES AND WITH VALUES EXTRAPOLATED FROM THE DRY CRITICAL EXPERIMENTS AND THE ZERO-POWER MOCKUP.

REACTOR, BREEDER + CONTROL ROD WORTH + EBR 1 AND 2 (RE) + REACTOR, FAST + INSTRUMENT, NUCLEAR + REACTOR, LMCR + SOURCE, NEUTRON + TEMPERATURE COEFFICIENT + CRITICALITY EXPERIMENT + CONTROL ROD CALIBRATION + MEASUREMENT, REACTIVITY + REACTOR STARTUP EXPERIENCE

6-010430

ALSO IN CATEGORY 5

DATED 1965

ARGONNE NATIONAL LABORATORY PUBLICATIONS, JULY 1, 1964 - JUNE 30, 1965

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ANL-7087 +. 328 PAGES, AUGUST 1965 CFSTI, \$7.00 CY, \$1.50 MN

THIS PUBLICATION LIST IS A BIBLIOGRAPHY OF SCIENTIFIC AND TECHNICAL ACCOUNTS ORIGINATED AT ARGONNE AND PUBLISHED DURING FISCAL YEAR 1965. IT INCLUDES ITEMS PUBLISHED AS JOURNAL ARTICLES, TECHNICAL REPORTS, BOOKS, ETC., ALL OF WHICH HAVE BEEN MADE AVAILABLE TO THE PUBLIC. THE LIST IS DIVIDED INTO THREE PARTS - THE FIRST A LIST DIVISION, THE SECOND AN ALPHABETICAL AUTHOR INDEX, AND THE THIRD PART A KWIC SUBJECT INDEX. INCLUDED ARE SECTIONS ON RADIOLOGICAL PHYSICS, REACTOR PHYSICS, REACTOR ENGINEERING, INDUSTRIAL HYGIENE AND SAFETY, FIRE PROTECTION, AND CHEMICAL ENGINEERING.

ACCIDENT ANALYSIS + BIBLIOGRAPHY + REACTOR, BREEDER + BURNOUT HEAT FLUX + CHEMICAL REACTION + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + METAL WATER REACTION + NOISE + REACTIVITY COEFFICIENT + REACTOR DYNAMICS + REACTOR KINETICS + REACTOR STABILITY + REACTOR TRANSIENT + SPACE DEPENDENT DYNAMICS + VOID COEFFICIENT + FFTF (TR) + DOSE MEASUREMENT, EXTERNAL

6-010509

ALSO IN CATEGORY 9

DATED 1964

ANALOG MODELS FOR HNPFC CONTROL AND PROTECTION STUDIES

ATOMICS INTERNATIONAL, CANOGA PARK

NAA-SR-8579 +. 69 PAGES, 16 FIGURES, 7 TABLES, 10 REFERENCES, JUNE 1964, CFSTI, \$1.50 CY

PROBLEM - DEVELOP A MODEL FOR THE DYNAMIC BEHAVIOR OF THE HNPFC. APPROACH - AN ANALOG COMPUTER MODEL WAS DEVELOPED FOR USE IN STUDIES OF THE CONTROL AND SAFETY SYSTEMS. THE MODEL INCLUDES - NEUTRON KINETICS, AFTERHEAT, CORE HEAT-TRANSFER FOR AVERAGE CHANNEL AND HOT CHANNEL, PRIMARY-COOLANT HYDRAULICS FOR FORCED AND CONVECTION FLOWS, ROD DRIVES, CONTROL SYSTEM, AND PROTECTIVE SYSTEM. THE SIMULATIONS OF THE VARIOUS SECTIONS ARE DESCRIBED IN DETAIL.

COMPUTER, ANALOG + COMPUTER PROGRAM + CONTROL SYSTEM + HALLAM (LMR) + REACTOR, LMCR + REACTOR DYNAMICS + REACTOR SAFETY SYSTEM + ANALYTICAL MODEL + MATHEMATICAL STUDY

6-010609

DATED 1966

CARPENTER SG + OTTER JM + PASCHALL RK + ROYDEN HN

TUNGSTEN RESONANCE INTEGRALS AND DOPPLER COEFFICIENTS. SECOND QUARTERLY REPORT, OCTOBER - DECEMBER 1965

ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA

NASA-CR-54888 + AI-66-16 +. 29 PAGES, TABLES, REFERENCES, FEBRUARY 11, 1966, CFSTI, \$2.00 CY, \$0.50 MN

CALCULATIONS ON DOPPLER COEFFICIENT AND FUEL EXPANSION REACTIVITY EFFECT. MEASUREMENTS IN SODIUM GRAPHITE REACTOR CRITICAL ASSEMBLY OF NATURAL-TUNGSTEN SLUGS AT VARIOUS TEMPERATURES, AND PRELIMINARY DOPPLER MEASUREMENTS.

DOPPLER EFFECT + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + SRE (RE) + TUNGSTEN

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-010959 DATED 1966
CARMICHAEL BM
FAST TRANSIENTS IN LIQUID PLUTONIUM
LOS ALAMOS SCIENTIFIC LABORATORY
2 PAGES, 1 TABLE, 4 REFERENCES, ANS TRANSACTIONS, 9(1), PAGES 308-309 (JUNE, 1966), AMERICAN NUCLEAR SOCIETY
ANNUAL MEETING, DENVER, COLORADO, JUNE 20-23, 1966

FAST TRANSIENTS IN CLEAN FUEL WERE STUDIED BY USING A CODE, TRN, WHICH IS BEING DEVELOPED SPECIFICALLY FOR APPLICATIONS TO LIQUID-PLUTONIUM CAPSULES. THE CODE CALCULATES DISTRIBUTED VALUES FOR THE POWER, TEMPERATURE, PRESSURE, AND MOTION OF THE FUEL IN CAPSULES FOR GIVEN REACTIVITY INPUT FUNCTIONS. THE HYDRODYNAMIC-DIFFERENCE EQUATIONS ARE MODELED AFTER THOSE GIVEN BY CHEZEM AND STRATTON.

REACTOR, BREEDER + COMPUTER, DIGITAL + REACTOR, FAST + REACTOR, LMCR + EXCURSION, LARGE + MPBE (MOLTEN PLUTONIUM BURNUP EXPERIMENT) + LAMPRE 1 (RE)

6-010978 ALSO IN CATEGORY 9 DATED 1966
REACTOR PHYSICS EFFORTS REQUIRED IN SUPPORT OF THE FAST BREEDER DEVELOPMENT PROGRAM
DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY, U.S. ATOMIC ENERGY COMMISSION, WASHINGTON, D. C.
WASH-1066 +. 41 PAGES, 3 TABLES, JANUARY 1966, CFSTI, \$3.00 CY, \$0.50 MN

SCATTERED THROUGHOUT THE REPORT ARE REFERENCES TO SODIUM COEFFICIENT, DOPPLER COEFFICIENT, CONTROL SYSTEM, AND DYNAMICS OF LARGE SYSTEMS.

REACTOR, BREEDER + CONTROL SYSTEM + CRITICAL ASSEMBLY + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LARGE + REACTOR, LMCR + REACTOR DYNAMICS + SODIUM COEFFICIENT + SPACE DEPENDENT DYNAMICS + ZPR 9 (CAF) + ZPR 6 (CAF) + SEFOR (RE) + ZPPR (CAF)

6-011153 DATED 1964
WEAVER LM
SRE REACTIVITY — CALCULATION TECHNIQUE
ATOMICS INTERNATIONAL, CANOGA PARK
NAA-SR-MEMO-7936 +. 56 PAGES, APRIL 10, 1964, CFSTI \$3.00 CY

PROBLEM — DEVELOP A CALCULATIONAL TECHNIQUE TO PROVIDE SURVEILLANCE OF THE SRE NUCLEAR PERFORMANCE. RESULTS — A FORTRAN COMPUTER PROGRAM WAS DEVELOPED TO COMPARE THE EXCESS REACTIVITY CALCULATED FROM REACTOR POWER AND TEMPERATURES WITH THE MEASURED REACTIVITY OBTAINED FROM THE POSITIONS OF THE SHIM RODS. VARIATIONS AND CHANGES IN REACTIVITY DIFFERENCES ARE INVESTIGATED ON THE BASIS OF CHANGES IN OPERATING PARAMETERS. THIS REACTIVITY-NORMALCY PROGRAM WAS SUCCESSFULLY USED FOR SRE CORE-II NUCLEAR ANALYSIS.

COMPUTER, DIGITAL + REACTIVITY, EXCESS + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + TEMPERATURE COEFFICIENT + SRE (RE)

6-011162 DATED 1964
HALFEN FJ
SRE-PEP TRANSIENT EXCURSION ANALYSIS
ATOMICS INTERNATIONAL, DIVISION OF NORTH AMERICAN AVIATION, INC., CANOGA PARK, CALIFORNIA
NAA-SR-MEMO-9741 +. 35 PAGES, MARCH 30, 1964, CFSTI, \$2.00 CY

PROBLEM — PREPARE A MATHEMATICAL MODEL FOR THE SRE-PEP CORE NEUTRON AND THERMAL KINETICS, LOOP HYDRAULICS, AND PROTECTIVE SYSTEM. USE THIS MODEL TO INVESTIGATE THE TRANSIENT BEHAVIOR OF THIS SYSTEM DURING VARIOUS INCIDENTS FOR INCLUSION IN THE SAFEGUARDS REPORT. SPECIFICALLY, ANALYZE THE SRE CORE III (PEP) CONFIGURATION FOR THE FOLLOWING INCIDENTS — (1) UNCONTROLLED ROD WITHDRAWAL FROM SHUTDOWN, (2) UNCONTROLLED ROD WITHDRAWAL FROM FULL POWER, (3) LOSS OF MAIN COOLANT FLOW AT FULL POWER. RESULTS — THE ANALYSIS SHOWED THAT THE POSTULATED PROTECTIVE SYSTEM WILL PROVIDE BOTH PRIMARY AND BACKUP PROTECTION AGAINST EXCESSIVELY HIGH TEMPERATURES RESULTING FROM THE INCIDENTS STUDIED.

ACCIDENT ANALYSIS + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + ACCIDENT, CONTROL ROD WITHDRAWAL + SRE (RE)

6-011249 DATED 1966
KEATEN RW + PHELPS RD + GUNBY AL
NUCLEAR PARAMETERS OF THE SRE CORE II
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-9430 +. 37 PAGES, 12 TABLES, 15 REFERENCES, AUGUST 1, 1964, CFSTI \$0.75

VARIOUS NUMBERS GIVEN AS WELL AS FORMULAS USED FOR CALCULATING THEM.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + REACTIVITY COEFFICIENT + SRE (RE) + CROSS SECTION

6-011424 ALSO IN CATEGORIES 9 AND 12 DATED 1966
HORST KM
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. THIRD QUARTERLY REPORT, OCTOBER 1964 — JANUARY 1965

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-011424 *CONTINUED*
GENERAL ELECTRIC, SAN JOSE
GEAP-4799 +. 150 PAGES, FEBRUARY 1965, CFSTI \$4.00 CY, \$1.00 MN

THIS REPORT IS ONE OF A SERIES OF SUCH REPORTS ON THE FOLLOWING SUBJECTS - PLANNING, EXPERIMENTAL PROGRAM IN ZPR-III, FUEL DEVELOPMENT, INSTRUMENTATION, CORE DYNAMICS, REACTIVITY CONTROL, FUEL HANDLING, AND CORE DESIGN

REACTOR, BREEDER + CRITICAL ASSEMBLY + DOPPLER COEFFICIENT + REACTOR, FAST + FUEL HANDLING + HEAT TRANSFER + REACTOR, LMCR + INSTRUMENT, TEMPERATURE + TEST, PHYSICS + SEFOR (RE) + RESEARCH CONTRACT + INSTRUMENTS, MISC. + FUEL ELEMENTS

6-011516 DATED 1965
CAPPEL HH
SGR TEMPERATURE TRANSIENTS
ATOMICS INTERNATIONAL, CANOGA PARK
NAA-SR-MEMO-11458 +. 56 PAGES, TABLES, JUNE 16, 1965, CFSTI \$3.00 CY, \$0.50 MN

DESCRIBES THE ANALYTICAL METHODS AND RESULTS OF A DESIGN STUDY OF THE TEMPERATURE TRANSIENTS (RESULTING FROM A MISMATCH OF POWER AND FLOW) FOR A 200-MW(E) SODIUM-GRAPHITE REACTOR. TWO DIGITAL COMPUTER CODES WERE DEVELOPED - ONE FOR THE PRIMARY SYSTEM, THE OTHER FOR THE LOW-PRESSURE SYSTEM OF THE PRIMARY LOOP. THE SEVERITY OF THE TRANSIENTS IS ATTRIBUTED TO THE SPECIFIC ROD POWER AND THE FUEL-CHANNEL-COOLANT TRANSPORT TIME.

COMPUTER PROGRAM + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + ACCIDENT, HYPOTHETICAL + THERMAL TRANSIENT

6-011996 DATED 1965
NIMS JB + THIE JA + SMITH RR + KLINKMAN AE + NICHOLSON RB
FAST REACTOR STABILITY AND SAFETY
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT + ARGONNE NATIONAL LABORATORY, IDAHO FALLS + UNIVERSITY OF WISCONSIN, MADISON
ANS-100 +. 12 PAGES, 3 FIGURES, 22 REFERENCES, DOCKET NO. 50-19, PAGES 194-211 OF FAST REACTOR TECHNOLOGY, AMERICAN NUCLEAR SOCIETY NATIONAL TOPICAL MEETING, DETROIT, MICHIGAN, APRIL 26-28, 1965

BROAD REVIEW OF CURRENT STATUS OF STABILITY AND SAFETY OF FAST REACTORS IN AREAS OF OPERATIONAL (REACTIVITY) STABILITY, HYPOTHETICAL ACCIDENT ANALYSIS, AND LOSS-OF-FLOW ACCIDENTS. COMPARES 1000-MW(E) DESIGN STUDIES WITH PRESENT REACTORS. SUGGESTS THAT CERTAIN STANDARD (BETHE-TAIT) MODELS ARE NO LONGER APPLICABLE AND THAT SHAPE CHANGES TO MINIMIZE THE SODIUM COEFFICIENT MAY BE AN UNSALABLE LUXURY.

BETHE TAIT + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + REACTOR STABILITY + ACCIDENT, HYPOTHETICAL

6-014800 ALSO IN CATEGORY 18 DATED 1965
PFLASTERER GR + CALDAROLA L
SEFOR EXPERIMENTAL PROGRAM PLANNING. VOLUME II. DESCRIPTIONS OF PLANNED TESTS
GENERAL ELECTRIC, SAN JOSE, ADVANCED PRODUCTS OPERATION
GEAP-5092 (VOL. 2) +. 116 PAGES, AUGUST 1965

VOL. 1 CONTAINS FUNCTIONAL REQUIREMENTS (FOR MAJOR EQUIPMENT ITEMS), BASED ON INFORMATION IN VOL. 2. VOL. 2 DESCRIBES THE TESTS, ANALYSES, AND REQUIRED MEASUREMENTS. TESTS INCLUDE (1) CRITICAL, (2) STATIC, (3) FREQUENCY RESPONSE, (4) REACTIVITY OSCILLATOR, (5) SUPERCRITICAL TRANSIENTS, AND (6) SUPER-PROMPT-CRITICAL TRANSIENTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + DESIGN CRITERIA + REACTOR, LMCR + OSCILLATOR, REACTIVITY + REACTOR KINETICS + REACTOR TRANSIENT + NOISE ANALYSIS + TEST, PLANT RESPONSE + SEFOR (RE)

6-014917 ALSO IN CATEGORIES 5 AND 17 DATED 1967
DICKERMAN CE + RUBINSON LE + BLUMENTHAL B + STEWART R
BEHAVIOR OF TH-20 WT/PERCENT U FAST-REACTOR FUEL UNDER TRANSIENT HEATING TO FAILURE
ARGONNE NATIONAL LABORATORY
10 PAGES, 10 FIGURES, 2 TABLES, 17 REFERENCES, NUCLEAR APPLICATIONS 3(1) PAGES 9-17 (JANUARY 1967)

SODIUM-BONDED TH-20 WT % U EBR II FUEL ELEMENTS WERE STUDIED IN TRANSIENT MELTDOWN EXPERIMENTS RANGING UP TO COMPLETE SAMPLE MELTDOWN, CONDUCTED IN TREAT. THE HIGH-MELTING THORIUM MATRIX SUPPORTED THE FUEL DURING TRANSIENT HEATING AT TEMPERATURES SIGNIFICANTLY BEYOND THE SOLIDUS. BECAUSE THE FUEL WAS STILL RELATIVELY INTACT AT THE TEMPERATURES OF CLADDING FAILURE, INTERNAL PRESSURE WAS NOT EFFECTIVE IN EJECTING FUEL UPON FAILURE. THE DOMINANT MECHANISM OF FUEL MOVEMENT WAS GRAVITY.

REACTOR, FAST + FUEL MELTDOWN + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + TREAT (PRR) + THERMAL TRANSIENT

6-016164 ALSO IN CATEGORIES 5 AND 9 DATED 1966
MACFARLANE DR

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-016164 *CONTINUED*

AN ANALYTIC STUDY OF THE TRANSIENT BOILING OF SODIUM IN REACTOR COOLANT CHANNELS
ARGONNE NATIONAL LABORATORY, ILLINOIS
ANL-7222 +. 117 PAGES, 48 FIGURES, 11 TABLES, JUNE 1966

TWO TYPES OF TRANSIENTS WERE STUDIED - (1) MILD REACTIVITY ADDITION AND (2) PUMP SHUTOFF WITH FLOW COASTDOWN. NUMERICAL SOLUTIONS OBTAINED FOR THE NONLINEAR FORM OF THE COUPLED FLUID DYNAMIC AND HEAT CONDUCTION PARTIAL DIFFERENTIAL EQUATION BY MEANS OF IMPLICIT FINITE-DIFFERENCE EQUATIONS AND AN ITERATIVE ALGOL COMPUTER PROGRAM. SONIC EFFECTS AND CRITICAL FLOW PHENOMENA WERE NOT TREATED. TWO-PHASE COOLANT DENSITY IS PRESSURE DEPENDENT, AND TEMPORAL- AND SPATIAL-VARYING SLIP VELOCITY ARE INCLUDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTIVITY, EXCESS + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + REACTOR TRANSIENT + SODIUM + VOID COEFFICIENT + METAL, LIQUID + FLOW, TWO PHASE

6-016735

DATED 1966

ZPR ASSEMBLY

ARGONNE NATIONAL LAB., ILL.

ANL-7210 +. 48 PAGES, 50 FIGURES, 27 TABLES, 76 REFERENCES, SECTION III-7 THRU III-15, PAGES 129-176 OF ANL REACTOR PHYSICS DIVISION ANNUAL REPORT, JULY 1, 1965 TO JUNE 30, 1966, DECEMBER 1966

DOPPLER-EFFECT MEASUREMENTS IN A ZONED, FAST, URANIUM CARBIDE CORE ZPR-6, ASSEMBLY NO. 42, COMPARISON OF MEASUREMENTS IN A LONG CORE (ZPR-6 ASSEMBLY 5) AND IN A SMALL APPROPRIATELY ZONED CORE (ZPR-6 ASSEMBLY 42) OF SUCH PARAMETERS AS SODIUM VOID COEFFICIENT, DOPPLER COEFFICIENT AND REACTIVITY WORTHS OF SAMPLES, MEASUREMENTS OF REACTIVITY REMOVAL RATES FOR ZPR-6 ASSEMBLY 42, CALCULATION OF CONTROL WORTHS AND DOPPLER EFFECTS IN ZPR-3 ASSEMBLY 48, COMPARISON OF DOPPLER COEFFICIENT AND SODIUM VOID COEFFICIENT OF METAL, OXIDE AND CARBIDE EBR-II DRIVER FUELS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + CRITICAL ASSEMBLY + DOPPLER EFFECT + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + PLUTONIUM + ACCIDENT, REACTIVITY + SODIUM COEFFICIENT + URANIUM CARBIDE + VOID COEFFICIENT + ZPR 6 (CAF) + DANGER COEFFICIENT

6-016736

DATED 1966

FARET AND ZPR-9

ARGONNE NATIONAL LAB., ILL.

ANL-7210 +. 29 PAGES, 9 FIGURES, 24 TABLES, 26 REFERENCES, SECTION III-17 THRU III-21, PAGES 180-208 OF ANL REACTOR PHYSICS DIVISION ANNUAL REPORT, JULY 1, 1965 TO JUNE 30, 1966, DECEMBER 1966

CONTROL-ROD WORTH, WORTH OF FUEL AND STRUCTURAL MATERIALS IN FARET CRITICAL ASSEMBLIES, CONTROL-MATERIAL WORTHS, ROSSI ALPHA AND PROMPT NEUTRON LIFETIME IN ZPR-9, A MOCKUP FOR THE ARGONNE FAST TUNGSTEN ROCKET REACTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + CONTROL ROD WORTH + REACTOR, FAST + REACTOR, LMCR + FFTF (TR) + PROMPT NEUTRON LIFETIME + TUNGSTEN + ROSSI ALPHA + SPACECRAFT

6-017008

DATED 1966

SEGAL BM + HORNE RE

MEASUREMENT OF SODIUM WORTH IN THE ENRICO FERMI REACTOR

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT

APDA-NTS-7 +. 40 PAGES, FIGURES, TABLES, SEPTEMBER 1966

BETWEEN NOVEMBER 18 AND DECEMBER 10, 1963 THE AXIALLY AVERAGED (OR COLUMNAR) CORE WORTH OF SODIUM WAS OBTAINED AT SEVEN CORE RADII BY MEASURING THE REACTIVITY OF SEVEN PAIRS OF CORE LOADINGS, EACH PAIR CONTAINING FOUR SPECIAL SUBASSEMBLIES IN SEPARATED LOCATIONS AT EQUAL RADII FROM THE CORE CENTER. THE SPECIAL SUBASSEMBLIES WERE VOIDED OF SODIUM IN THE FIRST LOADING OF EACH PAIR AND FILLED FOR THE SECOND. THE COLUMNAR CORE WORTH OF SODIUM WAS POSITIVE, VARYING FROM 1.82 CENTS/KG TO 2.19 CENTS/KG. THE EXISTENCE OF NEGATIVE SODIUM TEMPERATURE AND VOID COEFFICIENT OF REACTIVITY IS THUS CONFIRMED. GOOD AGREEMENT WITH THE VALUE FOR TOTAL CORE SODIUM WORTH OF APPROXIMATELY 1.9 CENTS/KG ESTIMATED FROM THE ZPR-III FERMI CRITICAL EXPERIMENT AND MULTIGROUP DIFFUSION THEORY.

AVAILABILITY - MICROCARD EDITIONS, INC., ACCOUNTING AND SHIPPING DEPT., WEST SALEM, WISCONSIN 54669

REACTOR, BREEDER + CRITICAL ASSEMBLY + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MEASUREMENT + SODIUM COEFFICIENT + MEASUREMENT, REACTIVITY + ZPR 3 (CAF)

6-017131

ALSO IN CATEGORY 17

DATED 1967

WOODRUFF RW

EFFECT OF DISSOLVED HELIUM IN SRE

ATOMICS INTERNATIONAL, SAN DIEGO, CALIFORNIA

CONF-670-713 +. 2 PAGES, 4 REFERENCES, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 10, PAGE 5 AND 6, (JULY 1967),

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-017131 *CONTINUED*
CONFERENCE ON REACTOR OPERATING EXPERIENCE, REACTOR OPERATIONS DIVISION OF THE AMERICAN NUCLEAR SOCIETY,
ATLANTIC CITY, JULY 23-26, 1967

SOLUBILITY OF INERT COVER GASES (HE OR AR) WILL BE A DESIGN PROBLEM IN LIQUID-METAL-COOLED REACTORS. SRE TRANSIENTS (20 F AND 25% ON NEUTRON FLUX) OCCURRED ABOUT DAILY AND LASTED 1 SEC - THE CORE TRANSIENT TIME. COVER-GAS BUBBLES WERE THOUGHT TO ACCUMULATE AT A HEAT-EXCHANGER OUTLET UNTIL ENOUGH VOLUME CHOKED FLOW AND SWEEPED OUT A LARGE BUBBLE. SMALL BUBBLES WERE NOT TRAPPED BUT PASSED CONTINUOUSLY THROUGH THE CORE, PROBABLY ACCOUNTING FOR THE NOISE-SPECTRUM DEVIATION FROM A WHITE SPECTRUM. BLEED LINES AT TRAPS, AND DE-ENTRAINMENT AT REACTOR INLETS SHOULD BE USED. ARGON WOULD HAVE SIGNIFICANTLY REDUCED THE VOLUME OF THE GAS TRANSPORTED.

REACTOR, LMCR + NOISE + SRE (RE) + HELIUM + REACTOR, ERDA OWNED + REACTIVITY EFFECT, ANOMALOUS + COOLANT QUALITY

6-017132 ALSO IN CATEGORY 9 DATED 1967
KLICKMAN AE + ZAMAN SU
THE STABILITY OF THE ENRICO FERMI REACTOR UNDER AUTOMATIC CONTROL
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
CONF-670-713 +. 2 PAGES, 2 FIGURES, AHS TRANSACTIONS, SUPPLEMENT TO VOLUME 10, PAGE 6 AND 7, (JULY 1967),
CONFERENCE ON REACTOR OPERATING EXPERIENCE, REACTOR OPERATIONS DIVISION OF THE AMERICAN NUCLEAR SOCIETY,
ATLANTIC CITY, JULY 23-26, 1967

TRANSFER-FUNCTION MEASUREMENTS SHOWED A RESONANCE WHEN UNDER AUTOCONTROL (REG.-ROD VELOCITY MADE PROPORTIONAL TO TIME DERIVATIVE OF NEUTRON FLUX). NYQUIST ANALYSIS SHOWED THAT THIS WAS NOT A SIGN OF INSTABILITY (BECAUSE OF PHASE-ANGLE CONSIDERATIONS), THE MARGIN BEING 68 DEG AT 67 MW AND INCREASING WITH POWER. ANOTHER NYQUIST ANALYSIS (ONE GROUP OF DELAYED NEUTRONS AND NO FEEDBACK OTHER THAN AUTOCONTROL) SHOWED A BROAD RANGE OF STABLE VALUES, BUT A RANGE OF REALISTIC VALUES FOR WHICH INSTABILITY COULD OCCUR.

CONTROL SYSTEM + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + REACTOR STABILITY + TRANSFER FUNCTION

6-017522 DATED 1966
REACTOR SAFETY
BROOKHAVEN NATIONAL LABORATORY
BNL-50,023 +. 2 PAGES, FIGURES, PAGES 10-11 OF BROOKHAVEN NATIONAL LABORATORY ANNUAL REPORT, DECEMBER 31, 1960

STUDIES INDICATE THAT STABILIZATION OF LARGE, FAST, SODIUM-COOLED REACTORS MIGHT BE ACCOMPLISHED BY SPLITTING THE CORE AXIALLY INTO 3 REGIONS, AND SEPARATING THEM WITH A FOOT-THICK PLENUM OF SODIUM. ON EXPANSION OF THE SODIUM, THE REACTIVITY DECREASES BECAUSE OF INCREASED NEUTRON LEAKAGE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REACTOR STABILITY + SODIUM + REACTOR COOLANT + BNL + COUPLED CORES + TEMPERATURE REACTIVITY EFFECT

6-017565 DATED 1966
ENGELMANN P + BOHME K + BRUHN G + DURAND-SMET R + FUELL WK + HABERMANN FWA + SCHRODER R + WINTZER D
INITIAL EXPERIMENTS IN THE KARLSRUHE FAST CRITICAL FACILITY, SNEAK
KERNFORSCHUNGSZENTRUM, KARLSRUHE, GERMANY
KFK-472 +. 31 PAGES, 19 FIGURES, 11 TABLES, 17 REFERENCES, OCTOBER 1966

PROBLEMS OF SIMULATING CORES IN SNEAK, TO INVESTIGATE FEATURES OF SODIUM- AND STEAM-COOLED PLUTONIUM OXIDE REACTORS, INCLUDE PU FUEL SHORTAGE, THE LOW PU-240 CONTENT OF THE FUEL, LACK OF FISSION PRODUCTS, THE HETEROGENEITY OF THE AVAILABLE FUEL (PLATE TYPE), AND THE DIFFERENCE BETWEEN TEMPERATURES OF THE EXPERIMENT AND A POWER REACTOR. PREPARATIONS NECESSARY TO DETERMINE THE SODIUM-VOID REACTIVITY EFFECTS WITH SUFFICIENT ACCURACY WERE DISCUSSED. THE PROGRAM FOR MEASURING THE STEAM DENSITY COEFFICIENT IS GIVEN WITH DISCUSSIONS OF THE NECESSARY CORRECTIONS. THE DOPPLER-MEASUREMENTS PROGRAM PLANNED FOR THE SNEAK ASSEMBLIES IS OUTLINED. RESULTS ARE GIVEN OF A STUDY TO SIMULATE, WITH STABLE ISOTOPES, THE FISSION-PRODUCT INVENTORY OF A TYPICAL STEAM-COOLED FAST REACTOR OF 300 MW WITH A BURNUP OF 2300 MWD PER TON.

AVAILABILITY - MICROCARD EDITIONS, INC., ACCOUNTING AND SHIPPING DEPT., WEST SALEM, WISCONSIN 54669

CRITICAL ASSEMBLY + DOPPLER EFFECT + REACTOR, FAST + FUEL BURNUP + REACTOR, LMCR + MOCKUP + VOID COEFFICIENT + REACTOR, POWER + MEASUREMENT, REACTIVITY

6-017576 ALSO IN CATEGORIES 5 AND 8 DATED 1967
SAVINO JM + SIEGEL R
EXPERIMENTAL AND ANALYTICAL STUDY OF THE TRANSIENT SOLIDIFICATION OF A WARM LIQUID FLOWING OVER A CHILLED FLAT PLATE
LEWIS RESEARCH CENTER, CLEVELAND, OHIO. NASA
NASA-TN-D-4015 +. 63 PAGES, 12 FIGURES, 26 REFERENCES, JUNE 1967

ANALYTICAL PREDICTIONS AND EXPERIMENTAL DATA ARE PROVIDED FOR THE TRANSIENT GROWTH OF A FROZEN LAYER THAT FORMS WHEN A WARM LIQUID FLOWS OVER ONE SIDE OF A CHILLED FLAT PLATE. THE PLATE IS

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-017576 *CONTINUED*

COOLED TO A TEMPERATURE BELOW THE FREEZING POINT OF THE LIQUID BY A COOLANT FLOWING ALONG THE OTHER SIDE OF THE PLATE. THIS COULD BE ENCOUNTERED, FOR EXAMPLE, IN A HEAT EXCHANGER WHERE WATER IS BEING COOLED BY A FLUID AT LOW TEMPERATURE. THREE ANALYTICAL PROCEDURES WERE USED AND COMPARED FOR ACCURACY AND CONVENIENCE IN APPLICATION. ONE IS AN ITERATIVE TECHNIQUE THAT YIELDED THREE CLOSED-FORM SOLUTIONS, EACH SUCCESSIVE SOLUTION BEING A MORE ACCURATE HIGHER ORDER APPROXIMATION. NUMERICAL RESULTS FROM THE THIRD APPROXIMATION ARE PRESENTED GRAPHICALLY IN SUFFICIENT DETAIL SO THAT RAPID ESTIMATES OF FROZEN LAYER GROWTH CAN BE MADE. IN THE EXPERIMENT, WATER FLOWED OVER ONE SIDE OF A THIN INCONEL PLATE WHILE THE OTHER SIDE OF THE PLATE WAS COOLED BY FLOWING CHILLED ALCOHOL. TRANSIENT ICE GROWTH WAS INITIATED BY A SUDDEN INCREASE OF THE ALCOHOL FLOW. THE ICE THICKNESS AND THE PLATE SURFACE TEMPERATURES WERE MEASURED THROUGHOUT THE TRANSIENT GROWTH PERIOD AND COMPARED FAVORABLE WITH THEORETICALLY PREDICTED VALUES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REACTOR, MOLTEN SALT + FILTER, LIQUID + HEAT TRANSFER, NATURAL CONVECTION + HEAT TRANSFER ANALYSIS + CRYOGENICS + HEAT EXCHANGERS

6-017577

ALSO IN CATEGORY 5

DATED 1967

FENN DB + ACKER LW + COE HH

STEADY-STATE PERFORMANCE OF A SEVEN-TUBE NAK-COOLED POTASSIUM CONDENSER

LEWIS RESEARCH CENTER, CLEVELAND, OHIO. NASA

NASA-TN-D-4020 +. 39 PAGES, 18 FIGURES, 2 TABLES, 11 REFERENCE, JJUNE 1967

POTASSIUM VAPOR WAS CONDENSED INSIDE THE TUBES OF A HORIZONTAL, MULTITUBE HEAT EXCHANGER. THE TUBES WERE COOLED WITH THE LIQUID SODIUM-POTASSIUM (NAK) FLOWING IN A COUNTER-CURRENT DIRECTION IN THE SPACE SURROUNDING THE TUBES. STEADY-STATE DATA ARE PRESENTED TO SHOW THE EFFECTS OF NAK COOLANT INLET TEMPERATURE AND COOLANT FLOW RATE ON THE CONDENSING LENGTH, THE COOLANT TEMPERATURE PROFILES, AND THE OVERALL PRESSURE DROP OF THE CONDENSER OVER A RANGE OF POTASSIUM INLET TEMPERATURES FROM 1200 TO 1500 F. A CORRELATION OF THE OVERALL PRESSURE DROP ON THE POTASSIUM SIDE OF THE CONDENSER WAS OBTAINED BY PLOTTING THE RATIO OF THE MEASURED OVERALL PRESSURE DROP TO THE VELOCITY HEAT IN THE TUBE INLET AS A FUNCTION OF THE CONDENSING-LENGTH-TO-DIAMETER RATIO.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

HEAT TRANSFER, BOILING + REACTOR, LMCR + CONDENSATION + METAL, ALKALI + METAL, LIQUID + HEAT TRANSFER, CONDENSATION

6-018195

ALSO IN CATEGORIES 7 AND 8

DATED 1967

DICKERMAN CE + ROBINSON LE + SOWA ES + MONAWECK JH + AUGUST C

TREAT MELTDOWN OF IRRADIATED METALLIC FAST-REACTOR FUEL IN AN INERT ATMOSPHERE

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

13 PAGES, 10 FIGURES, 3 TABLES, 29 REFERENCES, NUCLEAR APPLICATIONS 3(8)- 461-473 (AUG. 1967)

BEHAVIOR OF TWO TYPES OF IRRADIATED METALLIC FAST-REACTOR FUEL PINS UNDER TRANSIENT HEATING IN AN INERT-GAS ATMOSPHERE IN THE TREAT REACTOR WAS STUDIED UNDER CONDITIONS RANGING UP TO COMPLETE MELTDOWN. IRRADIATION-INDUCED CHANGES IN FAILURE THRESHOLDS, MODES OF FAILURE, AND POST-FAILURE FUEL MOTION WERE SIGNIFICANT BUT NOT ENOUGH TO MAKE GROSS CHANGES IN THE COURSE OF AN ACCIDENT. THE AMOUNT OF SWELLING OBSERVED IN THE POSTEXPERIMENTAL FUEL RESIDUE WAS LARGE ENOUGH TO HAVE PLAYED A SIGNIFICANT ROLE IN DETERMINING THE COURSE OF A MELTDOWN ACCIDENT.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + SODIUM + TESTING + TREAT (PRR) + VOID COEFFICIENT + FAILURE, CLADDING

6-018946

ALSO IN CATEGORY 17

DATED 1967

KLICKMAN AE + CALLEN RC

ANOMALOUS REACTIVITY EFFECTS IN THE FERMI INCIDENT

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + POWER REACTOR DEVELOPMENT COMPANY

CONF-670 602 +. 1 PAGE, ANS TRANSACTIONS 10(1) PAGE 334, 1967 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, SAN DIEGO, CALIF., JUNE 11-15, 1967

REVIEWS THE REACTIVITY EFFECTS THAT OCCURRED DURING THE FUEL MELTDOWN OF OCT. 1966. WHEN REACTOR POWER REACHED 5.5 MW, REACTIVITY INSERTION WAS AT THE RATE OF 0.185 CENT/MIN FOR 27 MIN UNTIL POWER REACHED 18 MW. THIS WAS ATTRIBUTED TO OVERHEATING OF THE CORE. AT 18 MW, THE RATE ABRUPTLY CHANGED TO 1.17 CENTS/MIN FOR 18 MIN UNTIL THE POWER WAS 30 MW. SUPERIMPOSED ON THIS LATTER TREND WERE THREE TRANSIENTS. THIS TREND AND THE TRANSIENTS WERE ATTRIBUTED TO FUEL MOTION. THE TOTAL REACTIVITY LOSSES WERE ATTRIBUTED TO FUEL MOTION. THE TOTAL REACTIVITY LOSS DETERMINED BY THE SHIM-AND REGULATING-ROD HISTORIES WAS 26 CENTS.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + REACTOR, LMCR + POWER COEFFICIENT + INCIDENT + REACTIVITY EFFECT, ANOMALOUS

6-019757

ALSO IN CATEGORY 17

DATED 1959

LUNDHOLM JG + JARRETT AA

REACTIVITY ANOMALIES IN SRE

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-4488 +. 33 PAGES, FIGURES, TABLES, PAGES III-1 TO 18 AND IV-D-1 TO 15 FROM SRE FUEL ELEMENT DAMAGE,

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-019757 *CONTINUED*
NOVEMBER 30, 1959

DESCRIBES REACTIVITY EFFECTS RELATED TO FUEL-ELEMENT-DAMAGE INCIDENT. BETWEEN RUNS, FUEL ELEMENTS WERE WASHED TO REMOVE SURFACE DEPOSITS FROM TETRALIN. DURING RUN 9, A REACTIVITY OF 1/2% WAS ATTRIBUTED TO APPROXIMATIONS IN CALCULATIONS OF XE CORRECTION. IN RUN 11, A LOSS OF 1/4% WAS ATTRIBUTED TO THIMBLE CHANGES. AND, NEAR THE END OF RUN, A GAIN OF 1/3% WAS ATTRIBUTED TO PRELIMINARY CALCULATIONS. 3 DAYS LATER, THE REACTIVITY WAS NORMAL. FOR RUN 13, SHIM-ROD MOTION FOR 6 HR INDICATED A 0.3% INCREASE AND THEN A 0.1% INCREASE FOR THE NEXT 3 DAYS, ATTRIBUTED TO A TETRALIN (SUMP COOLANT) LEAK INTO THE PRIMARY SYSTEM. IN RUN 14, A LOSS OF 1.2% DURING THE FIRST 4 DAYS WAS LATER ATTRIBUTED TO A PARTED FUEL CLUSTER DROPPING FROM THE FUEL REGION AND TO GRADUAL FLOODING OF MODERATOR CANS WITH SODIUM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, FUEL ELEMENT + REACTOR, LMCR + COMPARISON, THEORY AND EXPERIENCE + XENON + SRE (RE) + FAILURE, ADMINISTRATIVE CONTROL + REACTIVITY EFFECT, ANOMALOUS

6-021244 DATED 1966
GANDINI A + SALVATORES M
COMPARISON BETWEEN THE ELMOE AND THE ITERATIVE METHODS TO GENERATE TRANSPORT AND ELASTIC REMOVAL CROSS SECTIONS FOR FAST REACTOR EVALUATIONS
COMITATO NAZIONALE PER L'ENERGIA NUCLEARE, ROME, ITALY
RT-FIMA-(66)-3 +. 12 PAGES, TABLES, REFERENCES, 1966

ELASTIC REMOVAL AND TRANSPORT CROSS SECTIONS WERE CALCULATED FOR THE CASE OF DILUTE-PD02-FUELED, NA-COOLED FAST REACTOR USING THE ELMOE CODE AND AN ITERATIVE METHOD. COMPARISON OF THE RESULTS OBTAINED BY THE TWO METHODS INDICATES THAT THE ITERATIVE METHOD IS APPLICABLE TO TYPICAL PROBLEMS OF INTEREST.

AVAILABILITY - MICROCARD EDITIONS, INC., ACCOUNTING AND SHIPPING DEPT., WEST SALEM, WISCONSIN 54669

REACTOR, LMCR + SODIUM + TRANSPORT THEORY + MATHEMATICAL STUDY + CROSS SECTION + PLUTONIUM DIOXIDE

6-021245 ALSO IN CATEGORIES 7 AND 12 DATED 1966
REPORT OF WORK IN THE 4TH QUARTER OF 1966
COMMISSARIAL A L-ENERGIE ATOMIQUE, CADARACHE, FRANCE
EURFNR-261 +. 89 PAGES, 1966

THIS REPORT IS ONE OF A SERIES OF REPORTS ON WORK PERFORMED UNDER THE UNITED STATES-EURATOM FAST REACTOR EXCHANGE PROGRAM. TOPICS COVERED ARE - CRITICALITY STUDIES, METALLURGY, RADIATION EFFECTS, RADIATION SHIELDING AND PROTECTION, REACTOR CONTROL, REACTOR ENGINEERING, REACTOR FUELS, AND REACTOR SAFETY.

AVAILABILITY - MICROCARD EDITIONS, INC., ACCOUNTING AND SHIPPING DEPT., WEST SALEM, WISCONSIN 54669

REACTOR, LMCR + ACCIDENT, REACTIVITY + REACTOR DYNAMICS + SODIUM + REACTOR CONTROL + CRITICALITY EXPERIMENT + FISSION PRODUCT RETENTION

6-023162 DATED 1966
AN EVALUATION OF THE ATOMICS INTERNATIONAL 1000 MWE FAST BREEDER REACTOR
CHICAGO OPERATIONS OFFICE, USAEC
COO-265 +. 58 PAGES, 4 FIGURES, 17 TABLES, JULY 1966

THE ATOMICS INTERNATIONAL DESIGN FOR AN LMFBR WAS EVALUATED AND COMPARED WITH AN EARLIER EVALUATION OF FOUR DESIGNS FROM OTHER FIRMS. THE CORE-REGION SODIUM-VOID EFFECT IS KEPT BELOW \$1.00 IN THIS DESIGN BY THE USE OF COUPLED MODULAR CORES. THE BLANKET DOPPLER COEFFICIENT IS HIGH - MINUS 0.016 T BK/DT. THE EVALUATORS CALCULATED SOMEWHAT DIFFERENT VALUES FOR THE IMPORTANT NUCLEAR PARAMETERS BECAUSE OF DIFFERENCES IN THE CROSS SECTIONS USED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + VOID COEFFICIENT + COUPLED CORES + DESIGN STUDY

6-023870 ALSO IN CATEGORIES 8 AND 18 DATED 1966
DEFELICE J + HYLAND M
FAST TEST REACTOR HYPOTHETICAL MELTDOWN ACCIDENT. PROGRESS REPORT. OCTOBER 1965 - DECEMBER 1965
BATTELLE NORTHWEST, RICHLAND, WASH.
COO-1430-3 +. 65 PAGES, 3 FIGURES, 17 TABLES, 10 REFERENCES, JAN. 21, 1966

A STUDY OF THE ENERGY YIELD FROM A MELTDOWN IS IN PROGRESS. FUEL-ZONING EFFECTS WERE EXPLORED IN DETAIL WITH AX-1 AND WEAK-EXPLOSION COMPUTER CODES. AX-1 INDICATES A SMALLER ENERGY RELEASE DUE TO FUEL ZONING THAN EXPECTED, AND INSENSITIVENESS TO DEGREE OF ZONING. WEAK-EXPLOSION RESULTS IN A LARGE ENERGY RELEASE INCREASE WITH ZONING, HOWEVER ITS LACK OF TREATMENT OF SURFACE EFFECTS UNDOUBTEDLY OVERESTIMATES ZONING EFFECTS. WORK CONTINUES.

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-023870 *CONTINUED*
AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00
COPY, \$0.65 MICROFICHE

COMPUTER PROGRAM + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REACTIVITY + REACTOR, TEST + FFTF (TR) + CORE
MELTDOWN + ACCIDENT, LOSS OF COOLANT

6-025357 DATED 1965
REACTOR STABILITY AND OPTIMAL CONTROL
7 PAGES, FIGURES, REFERENCES, ANS TRANS. 8(2), PAGES 475-481, 1965 WINTER MEETING OF THE AMERICAN NUCLEAR
SOCIETY, WASHINGTON, D. C., NOVEMBER 15-18, 1965

CONVENTION SESSION PROGRAM ENTITLED REACTOR STABILITY AND OPTIMAL CONTROL CONTAINS SHORT ACCOUNTS
OF PAPERS ON (1) CONDITIONS FOR PROBABILISTIC BOUNDEDNESS OF REACTORS WITH RANDOM MACROSCOPIC
CROSS SECTIONS, (2) TWO CRITERIA FOR LAGRANGE STABILITY, (3) EXPERIMENTAL INTERPRETATION OF A
CRITERION FOR NONLINEAR STABILITY, (4) A LIAPUNOV FUNCTION FOR THE PROMPT-JUMP APPROXIMATION IN
REACTOR DYNAMICS, (5) AUTONOMOUS POWER OSCILLATIONS IN REACTORS, (6) EBR-2 DYNAMICS, (7) INPUT-
OUTPUT APPROACH TO OPTIMAL CONTROL, AND (8) LIMITATIONS TO THE APPLICABILITY OF THE RESTRICTED
SPACE-TIME OPTIMAL XENON SHUTDOWN PROBLEM.

EBR 1 AND 2 (RE) + REACTOR, FAST + LIAPUNOV'S FUNCTION + REACTOR, LMCR + REACTOR DYNAMICS + REACTOR STABILITY +
SPACE DEPENDENT DYNAMICS + REACTOR, TEST + REACTOR CONTROL + REACTOR, ERDA OWNED + OSCILLATION

6-025361 DATED 1967
BERMAN D + WILKINSON CD
WEIGHTED DOPPLER ANALYSIS CODE - WEDOP
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIFORNIA
GEAP-5543 +. 66 PAGES, AUGUST 1967

DESCRIBES COMPUTER CODE WEDOP, DEVELOPED TO CALCULATE THE TOTAL REACTOR DOPPLER REACTIVITY EFFECT
IN A FAST REACTOR FOR (1) A CHANGE BETWEEN TWO EQUILIBRIUM REACTOR CONDITIONS OR (2) DURING THE
EARLY PORTION OF A FAST POWER TRANSIENT. DETAILED FUEL TEMPERATURE DISTRIBUTION AND LOCAL
DOPPLER REACTIVITY IS CALCULATED, AND LATTER IS WEIGHTED WITH AN APPROPRIATE FUNCTION TO OBTAIN
TOTAL DOPPLER REACTIVITY. CODE HAS BEEN USED TO PREDICT DOPPLER EFFECTS ASSOCIATED WITH CERTAIN
EXPERIMENTS TO BE CONDUCTED AT SEFOR AND WILL BE USED IN ANALYZING DATA FROM FUTURE SEFOR
EXPERIMENTAL PROGRAM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

COMPUTER, DIGITAL + DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE)

6-025690 DATED 1967
SHIGEHRO AN
KINETICS OF SODIUM-COOLED FAST REACTOR
UNIVERSITY OF TOKYO
9 PAGES, 16 FIGURES, 2 TABLES, 30 REFERENCES, NIPPON GENSIRYSKU GAKKAISHI VOL. 9, PAGES 202-11 (APRIL 1967)
IN JAPANESE

IN THE PROGRAM NOW UNDER WAY FOR DEVELOPING A SODIUM-COOLED FAST REACTOR, RESEARCH ON KINETICS IS
RECOGNIZED TO BE VERY IMPORTANT FOR ESTABLISHING THE DESIGN AND SAFETY CRITERIA. THIS REVIEW
DESCRIBES THE KINETICS PROBLEMS TO BE SOLVED IN IMPORTANT DOMAINS SUCH AS - (1) REACTIVITY
COEFFICIENTS, (2) EXAMPLES OF EXPERIMENT AND ANALYSIS THEREOF FOR A SMALL FAST REACTOR, (3)
THEORETICAL ANALYSIS OF STABILITY AND ACCIDENTS IN A LARGE FAST REACTOR, (4) KINETICS OF A
MODULAR FAST REACTOR.

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + REACTIVITY COEFFICIENT + REACTOR DYNAMICS + REACTOR
KINETICS + REACTOR STABILITY + SODIUM COEFFICIENT + TRANSFER FUNCTION + REVIEW

6-025692 DATED 1967
WEHMEYER DB + MAENE NG + WOLFE GE + YEH CS
PHYSICS CALCULATIONS FOR THE FERMI MIXED-OXIDE CORE
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + CENTRE D ETUDE DE L ENERGIE NUCLEAIRE, MOL, BELGIUM + COMBUSTION
ENGINEERING, INC.
2 PAGES, 1 TABLE, 1 REFERENCE, ANS TRANS. 10(2), PAGES 532-533, 1967 WINTER MEETING OF THE AMERICAN NUCLEAR
SOCIETY, CHICAGO, ILLINOIS, NOVEMBER 5-9, 1967

CALCULATIONS FOR A MIXED PLUTONIUM-URANIUM OXIDE CORE FOR FERMI INCLUDES SODIUM AND DOPPLER
COEFFICIENTS.

DOPPLER COEFFICIENT + FERMI (LMFBR) + REACTOR, LMCR + SODIUM COEFFICIENT + URANIUM DIOXIDE + PLUTONIUM DIOXIDE
+ REACTOR, LMFBR

6-025876 DATED 1965
YIFTAH S + GITTER L + ILAMED Y
SODIUM VOID AND DOPPLER COEFFICIENTS FOR FAST POWER REACTORS
ISRAEL ATOMIC ENERGY COMMISSION, SOREQ NUCLEAR RESEARCH CENTER

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-025876 *CONTINUED*
IA-1057 +. 51 PAGES, FIGURES, TABLES, REFERENCES, AUGUST 1965

CALCULATIONS WERE MADE IN ORDER TO STUDY THE SODIUM VOID AND DOPPLER EFFECTS IN A SERIES OF FAST REACTORS WHICH DIFFER IN VOLUME OF CORE, TYPE OF PLUTONIUM FUEL, AND COMPOSITION OF FUEL. THE SODIUM-VOID AND DOPPLER COEFFICIENTS WERE OBTAINED FROM CRITICALITY FACTORS COMPUTED FOR 486 SYSTEMS WITH THE AID OF MULTIGROUP DIFFUSION THEORY FOR SPHERICAL SYSTEMS.

AVAILABILITY - AEC DEPOSITORY LIBRARIES IN THE U.S. AND OVERSEAS

DOPPLER COEFFICIENT REACTOR, FAST + REACTOR, LMCR + PLUTONIUM + SODIUM COEFFICIENT + VOID COEFFICIENT

6-025946 DATED 1967
JANSEN A
THE LONG TERM BEHAVIOR OF FAST POWER REACTORS WITH PU-RECYCLING
GESELLSCHAFT FÜR KERNFORSCHUNG, KARLSRUHE, GERMANY

KFK630 + FUR-3674E + CONF-671043-7 + EURFNR-399 +. 22 PAGES, 4 FIGURES, 3 TABLES, 5 REFERENCES, OCTOBER 1967, FROM SYMPOSIUM ON FAST REACTOR PHYSICS AND RELATED SAFETY PROBLEMS, KARLSRUHE, GERMANY.

STUDIED THE LONG-TERM BEHAVIOR OF FAST POWER REACTORS DURING MANY RELOADING CYCLES, INCLUDING THE TIME DEPENDENCE OF THE ISOTOPIC COMPOSITION OF PLUTONIUM, THE URANIUM AND PLUTONIUM CONTENT OF THE FUEL, THE BREEDING GAIN, ETC. AFTER STARTING, THE ONLY FISSIONABLE MATERIAL CONSUMED IS THAT BRED IN THE SAME REACTOR. LOADING AND UNLOADING THE CORE AND BLANKET ARE TREATED AS DISCONTINUOUS PROCESSES, AND EXTERNAL FUEL MANAGEMENT AND TIMES FOR REPROCESSING AND REFABRICATION ARE CONSIDERED. NUMERICAL RESULTS ARE GIVEN FOR SODIUM-COOLED REACTOR NA1.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + PLUTONIUM + REACTOR, POWER + FUEL REPROCESSING

6-026783 DATED 1966
TAUBE M
REACTIVITY CHANGES IN SALT-BOILING FAST REACTORS OF WARS AND SAWA TYPES
NATIONAL INSTITUTE OF RADIOLOGICAL SCIENCES, JAPAN
4 PAGES, FIGURES, 10 REFERENCES, NUKLEONIKA, 12, PAGES 231-234 (1967)

DISCUSSES A SPECIFIC EFFECT OF BUBBLE DILUTION IN THE FUSED CHLORIDES IN SALT-BOILING REACTORS. THE MEAN VOLUME EXPANSION COEFFICIENT IN THE HETEROGENEOUS BOILING REACTOR WARS WITH MERCURY AS COOLANT IS ABOUT 0.002/DEG(C), AND IN THE HOMOGENEOUS BOILING REACTOR SAWA WITH ALUMINUM TRICHLORIDE AS COOLANT ABOUT .025/DEG(C). A FAST 10 PER CENT INCREASE IN POWER DECREASES K-EFF ABOUT 10 PER CENT IN SAWA AND 0.8 PER CENT IN WARS.

REACTIVITY EFFECT, EXPANSION + REACTOR, FAST + REACTOR, LMCR + REACTOR, MOLTEN SALT + REACTIVITY COEFFICIENT + REACTIVITY EFFECT + VOID COEFFICIENT

6-027073 DATED 1967
SOFER GA + CHERRY BH
DESIGN AND SAFETY OPTIMIZATIONS IN THE USE OF PLUTONIUM-URANIUM CARBIDE FUEL IN LARGE FAST BREEDER REACTORS
UNITED NUCLEAR CORPORATION, ELMSFORD, NEW YORK
33 PAGES, 9 FIGURES, 4 TABLES, 10 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST NUCLEAR REACTORS, SEPTEMBER, 1967

OPERATING CHARACTERISTICS OF A 1000-MW(E) SODIUM-COOLED PLUTONIUM-URANIUM CARBIDE REACTOR HAVING NEUTRONICALLY UNCOUPLED THREE-MODULE CORE ARE GIVEN. COUPLED-KINETICS/HEAT-TRANSFER CODE SCARE WAS USED FOR SAFETY ANALYSES OF RAMP REACTIVITY INSERTIONS. IT IS CONCLUDED THAT A NEGATIVE TEMPERATURE COEFFICIENT IS VERY IMPORTANT FOR (U-PU)C FUELED DESIGNS, AND THAT FUEL RODS USING HELIUM FOR THERMAL BINDING BETWEEN FUEL AND CLADDING HAVE BETTER SAFETY CHARACTERISTICS THAN SODIUM-BONDED RODS.

AVAILABILITY - G.A.SOFER, UNITED NUCLEAR CORPORATION, RESEARCH AND ENGINEERING CENTER, ELMSFORD, N.Y.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + REACTOR KINETICS + TEMPERATURE COEFFICIENT + URANIUM CARBIDE + SAFETY ANALYSIS + FUEL ELEMENTS

6-027774 ALSO IN CATEGORIES 9 AND 17 DATED 1968
ZINIUK MA + WILBER HA
SCRAM TESTS TO DETERMINE SHUTDOWN TRANSIENT CHARACTERISTICS OF THE ENRICO FERMI REACTOR AND ASSOCIATED HEAT TRANSPORT SYSTEM
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-NTS-15 +. 45 PAGES, 8 FIGURES, 7 TABLES, 4 REFERENCES, APRIL 1968

REACTOR OUTLET TEMPERATURE FLUCTUATION AFTER 67 MWH OPERATION BEGAN 15 SEC AFTER SCRAM AND LASTED TIL 220 SEC. THE RATE WAS 44 F/SEC, OVER A CHANGE OF 76 F. TIME CONSTANTS AND TRANSPORT LAGS WERE DETERMINED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

✓

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-027774 *CONTINUED*
FERMI (LMFBR) + SCRAM, REAL + RESPONSE TIME + INSTRUMENT, TEMPERATURE + REPORT, OPERATIONS ANALYSIS + TEST, INSTRUMENT RESPONSE + MAIN COOLING SYSTEM + REACTOR, LMFBR + THERMAL TRANSIENT

6-027775 ALSO IN CATEGORY 17 DATED 1967
MUELLER RE + WILBER HA
POWER COEFFICIENT MEASUREMENTS FOR THE ENRICO FERMI REACTOR. ENRICO FERMI ATOMIC POWER PLANT NUCLEAR TEST SERIES
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-NTS-9 +. 75 PAGES, 10 FIGURES, 12 TABLES, 22 REFERENCES, NOV. 1967

TESTS WERE MADE OF CONTROL ROD POSITION RESPONSE TO POWER INCREASE/DECREASE, WITH 2 OR 3 LOOP FLOW, FOR POWERS OF 1.3, 2.0, 13.3, 20, 56.7 AND 100 MWTH. COEFFICIENTS ARE ABOUT 28% SMALLER THAN PREDICTED. FLOW DEPENDENCE SHOWS VERY LITTLE FEEDBACK INDEPENDENT OF FLOW. THE 20-MWTH 3-LOOP COEFFICIENT VALUE IS STILL INCONSISTANT. RELIABLE INLET TEMPERATURE DATA HAVE ELIMINATED THE HYSTERESIS EFFECT BETWEEN POWER INCREASE/DECREASE. CHANGES IN ANALYSIS WERE - INCREASED VALUE OF COMPONENTS DEPENDENT ON CORE SODIUM WORTH, DECREASE IN CORE FUEL PIN AXIAL EXPANSION COEFFICIENT, DECREASE IN RADIAL EXPANSION COEFFICIENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTIVITY EFFECT, EXPANSION + FERMI (LMFBR) + POWER COEFFICIENT + COMPARISON, THEORY AND EXPERIENCE + REPORT, OPERATIONS ANALYSIS + TEST, PHYSICS + FLOW STABILITY + REACTOR, LMFBR

6-028163 ALSO IN CATEGORIES 1 AND 9 DATED 1968
GREEBLER P + HENLEY EJ
ADVANCES IN NUCLEAR SCIENCE AND TECHNOLOGY
GENERAL ELECTRIC COMPANY + UNIVERSITY OF HOUSTON
391 PAGES, VOLUME 4, 1968, ACADEMIC PRESS, NEW YORK AND LONDON

THIS VOLUME IS CONCERNED WITH ADVANCED REACTOR CONCEPTS. TWO OF THE CHAPTERS DEAL WITH THE TECHNOLOGY OF, RESPECTIVELY, SODIUM-COOLED FAST BREEDER POWER REACTORS AND GAS-COOLED POWER REACTORS, THE LATTER INCLUDING BOTH THERMAL CONVERTERS AND FAST BREEDERS. COGNIZANCE IS TAKEN OF THE CENTRAL ROLE OF REACTOR SAFETY IN THE DEVELOPMENT OF FAST BREEDER REACTORS IN TWO CHAPTERS DEALING WITH, RESPECTIVELY, THE DOPPLER EFFECT IN FAST REACTORS AND THE ANALYSIS OF MAXIMUM ACCIDENTS. THREE CHAPTERS COVER RECENT DEVELOPMENTS IN AREAS OF MORE GENERAL APPLICABILITY, NAMELY, OPTIMAL CONTROL THEORY, MODERN PERTURBATION THEORY, AND INDUSTRIAL APPLICATION OF IONIZING RADIATIONS. MAIN HEADINGS FOR THE 7 CHAPTERS ARE - GAS-COOLED REACTOR TECHNOLOGY, SAFETY AND ECONOMIC CHARACTERISTICS OF A 1000-MWE FAST SODIUM-COOLED REACTOR DESIGN, AND THE DOPPLER EFFECT IN FAST REACTORS.

BETHE TAIT + DOPPLER EFFECT + ECONOMICS + REACTOR, FAST + REACTOR, GCR + REACTOR, LMCR + REACTOR DYNAMICS + REACTOR SAFETY SYSTEM + REACTOR CONTROL + THEORETICAL INVESTIGATION + MATHEMATICAL TREATMENT + REACTOR, LMFBR

6-028562 ALSO IN CATEGORY 17 DATED 1967
WINKLEBLACK RK
RECENT EBR-II OPERATING EXPERIENCE
ARGONNE NATIONAL LABORATORY
15 PAGES, 4 FIGURES, PAPER PRESENTED AT THE 1967 WINTER MEETING OF THE AMERICAN NUCLEAR SOCIETY, CHICAGO, ILLINOIS, NOVEMBER 5-9, 1967

DISCUSSES A FUEL FAILURE AND FISSION PRODUCT RELEASE TO COOLANT AND A SIGNIFICANT DECREASE IN THE POWER COEFFICIENT FOLLOWING A CHANGE IN CORE CONFIGURATION. DESCRIBES ANALYTICAL TECHNIQUE USED TO LOCATE THE FAULTY FUEL. NEED FOR A FAILED FUEL LOCATING METHOD IS DISCUSSED, AND USE OF A TAG MATERIAL IS CONSIDERED. DECREASE IN POWER COEFFICIENT AFTER CORE ENLARGED FROM 73 TO 81 ASSEMBLIES IS ATTRIBUTED TO CHANGE IN CORE AND REFLECTOR SIZE AND COMPOSITION, AND AN INCREASED BOWING OF SUBASSEMBLIES CAUSED BY INSTALLATION OF STAINLESS STEEL SUBASSEMBLIES.

ACTIVITY BUILDUP + EBR 1 AND 2 (RE) + REACTOR, FAST + FUEL ELEMENT BOWING + REACTOR, LMCR + POWER COEFFICIENT + REACTOR, RESEARCH + REACTOR, TEST + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + REACTOR, ERDA OWNED + FAILURE, CLADDING + REACTIVITY EFFECT, ANOMALOUS + MAIN COOLING SYSTEM + OPERATING EXPERIENCE

6-028801 ALSO IN CATEGORY 17 DATED 1967
KLICKMAN AE + HORNE RE + WILBER HA
OSCILLATOR TESTS IN THE ENRICO FERMI REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-NTS-11 +. 145 PAGES, 71 FIGURES, 24 TABLES, 10 REFERENCES, AUGUST 1967

ROTATING RODS WHICH WOULD VARY REACTIVITY BY 6 OR BY 2 CENTS BETWEEN 0.001 AND 10 CPS WERE USED, TOGETHER WITH A SYNCHRONOUS AND A NULL BALANCE TRANSFER FUNCTION ANALYZER, AS WELL AS SIMPLE RECORDERS. NEGLIGIBLE DIFFERENCES EXIST FOR SPACE DEPENDENT CALCULATIONS. SYNCHRONOUS ANALYZER SEEMED BEST, BUT TRUE FEEDBACK OBSERVABLE ONLY AT HIGHER POWERS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + OSCILLATOR, REACTIVITY + REACTOR STABILITY + SPACE DEPENDENT DYNAMICS + TRANSFER FUNCTION + TEST, PHYSICS + REACTOR, LMFBR

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-028802 ALSO IN CATEGORY 17 DATED 1967
SEELY GT + BALL GL + BRANNAN CE + HORNE RE + MUELLER RE + SEGAL BM + WILBER HA
MEASUREMENT OF SOME REACTIVITY EFFECTS IN THE ENRICO FERMI REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-NTS-12 +. 68 PAGES, 18 FIGURES, 13 TABLES, 3 REFERENCES, NOVEMBER 1967

REACTIVITY WORTH OF PERMANENT NEUTRON SOURCE (23 CENTS AT 10 WATTS, AND 0.23 CENTS AT 1 KW), OF
RETRACTABLE NEUTRON SOURCE (8.3 CENTS), OF COOLANT FLOW RATE (NO GAS ENTRAINMENT AND NEGLIGIBLE
STRUCTURAL DEFORMATION) AND PRESSURE COEFFICIENT (NONE AT 15 AND 25 PSIA), AND OF TEMPORARY
INSTRUMENT THIMBLE (1.41 CENTS) WERE MADE AND EVALUATED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

PRESSURE, EXTERNAL + FERMI (LMFBR) + INSTRUMENT, IN CORE + REACTIVITY EFFECT + SOURCE, NEUTRON + COOLANT
COEFFICIENT + TEST, PHYSICS + FLOW + REACTOR, LMFBR

6-029301 ALSO IN CATEGORY 18 DATED 1967
SAFETY PROBLEMS RELATED TO THE ASSEMBLING OF MOCK-UPS IN MASURCA
22 PAGES, 16 FIGURES, FROM INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPT. 1967

PUNCTUAL ADIABATIC MODEL WITH SIX GROUPS OF DELAYED NEUTRONS, PROGRAMMED FOR 7094 COMPUTER WITH
BRITTON LINEARIZATION METHOD TO ALLOW SAME PRECISION WITH PROMPT AND DELAYED CRITICALITY.
STARTUP ACCIDENT WITH DELAYED SCRAM GIVES MAXIMUM ALLOWABLE \$0.2/SEC, AND \$5/SEC WITH INHERENT
SAFETY. STUDY OF LOADING ERRORS LEADS TO GRAPPLE SPEED LIMIT OF 1 M/MIN. DOSE RATES IN CELL FOR
500 W OPERATION CALCULATED.

CRITICAL ASSEMBLY + ACCIDENT, CONTROL ROD WITHDRAWAL + SAFETY ANALYSIS + REACTOR, LMFBR

6-029521 ALSO IN CATEGORY 9 DATED 1968
PFLASTERER GR
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 16, FEBRUARY-APRIL 1968
GENERAL ELECTRIC COMPANY
GEAP-5629 +. 32 PAGES, REFERENCES, MAY 1968

THIS REPORT IS ONE OF A SERIES OF SUCH REPORTS CONCERNED WITH SUMMARIZING THE WORK PERFORMED ON
THE RESEARCH AND DEVELOPMENT PROGRAM FOR THE SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR (SEFOR).
THE SEFOR PROJECT CONSISTS OF TWO MAJOR PARTS - THE DESIGN AND CONSTRUCTION OF THE FACILITY AND
THE RELATED RESEARCH DEVELOPMENT PROGRAM. SEFOR IS PARTICULARLY DESIGNED FOR THE SYSTEMATIC
DETERMINATION OF THE DOPPLER COEFFICIENT OF REACTIVITY AT TEMPERATURES UP TO THE VICINITY OF FUEL
MELTING. THIS REPORT SUMMARIZES THE TASKS SCHEDULED FOR ACTIVITY AT THIS TIME. REMAINING TASKS
WILL BE INCLUDED IN FUTURE REPORTS AS WORK PROGRESSES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + DESIGN STUDY + REACTOR, PULSED + SEFOR (RE) + REVIEW + R
AND D PROGRAM

6-029466 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION E-1 - REMOVAL OF BEO FROM CORE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE E-1 AND -2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS MORE FULLY THE ANTICIPATED EFFECTS THAT REMOVAL OF BEO WILL HAVE ON REACTIVITY EFFECTS AND
MAGNITUDE OF THE DOPPLER COEFFICIENT. IS EXPERIMENTAL INFORMATION EQUIVALENT TO THAT USED IN
PREDICTING PHYSICS PARAMETERS FOR THE INITIAL CORE LOADING ALSO AVAILABLE FOR PREDICTING THESE
PARAMETERS WITH BEO ABSENT. (ANSWER) REPLACING BEO WITH NICKEL RESULTS IN REDUCING DOPPLER
COEFF. 11 PER CENT. SEE SECT. 3.3.5 OF FDSAR SUPPLEMENT 10.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

BERYLLIUM + DOPPLER COEFFICIENT + REACTOR, LMCR + OXIDE + REACTIVITY EFFECT + REPORT, SAR + NICKEL + SEFOR
(RE) + RESPONSE TO NRC QUESTION + REACTOR PHYSICS

6-029901 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION L-1 - REACTIVITY OF INSTRUMENTED FUEL ASSEMBLY
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE L-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT IS REACTIVITY OF AN INSTRUMENTED FUEL ASSEMBLY (IFA) AS COMPARED TO A SINGLE FUEL ELEMENT.
DISCUSS NORMAL AND SPECIAL CONDITIONS UNDER WHICH AN IFA WOULD BE LOADED INTO OR REMOVED FROM
CORE AT SEFOR. CONSIDER IN THIS DISCUSSION DESIGN CRITERION WHICH LIMITS REACTIVITY WORTH OF ANY
ITEM BEING ROUTINELY HANDLED NEAR CORE TO LESS THAN 1\$. (ANSWER) SEE SECT. 16.2.4.3 OF FDSAR.
NO PLANS TO HANDLE IFA ROUTINELY.

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-029501 *CONTINUED*
AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + FUEL HANDLING + REACTOR, LMCR + REACTIVITY EFFECT + REPORT, SAR + INSTRUMENT, COMPONENT +
SEFOR (RE) + RESPONSE TO NRC QUESTION + FUEL ELEMENTS

6-029506 ALSO IN CATEGORIES 5 AND 18 DATED 1968
RESPONSE TO QUESTION L-6 - ENERGY DENSITIES OF FUEL IN TRANSIENT TESTS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, 1 TABLE, PAGE 5-4 AND -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT ARE EXPECTED ENERGY DENSITIES IN CAL/GM OF FUEL IN PLANNED SUB-PROMPT AND PROMPT CRITICAL
TRANSIENT TESTS. (ANSWER) TABLE OF ENERGY DENSITIES GIVEN BASED ON TRANSIENT ANALYSIS DESCRIBED
IN SECTION SII OF FDSAR AND SUPPLEMENT 10.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REACTOR TRANSIENT + REPORT, SAR + THERMAL CONSIDERATION + TEST, PHYSICS + SEFOR (RE) + HEAT
TRANSFER EXPERIMENT + RESPONSE TO NRC QUESTION + FUEL ELEMENTS

6-030252 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 3 - CRITICALITY ACCIDENTS
NORTHERN STATES POWER COMPANY
2 PAGES, PP. 18-20 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

PROTECTION AGAINST CRITICALITY ACCIDENTS DURING STARTUP OR PLANT SHUTDOWN DESCRIBED. ***(A)
PERIOD TRIPS BACKED UP BY POWER TRIPS, ARE PROVIDED FOR POWER LEVELS ABOVE 100 WATTS. SHORT
PERIODS CANNOT OCCUR AT POWER BECAUSE OF THE DOPPLER EFFECT. (B) WHENEVER CONTAINMENT OR REACTOR
COOLING IS INOPERABLE, CONTROL ROD DRIVE POWER SUPPLY BREAKERS WILL BE OPENED. 5 OF THE 10
REFLECTOR SEGMENTS WOULD HAVE TO BE RAISED FOR CRITICALITY, DUE TO THE SHUTDOWN MARGIN.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM
CHARGE \$2.00)

ACCIDENT, CONTROL ROD WITHDRAWAL + REPORT, SAR + SHUTDOWN MARGIN + ACCIDENT, CRITICALITY + SEFOR (RE) +
RESPONSE TO NRC QUESTION + INSTRUMENT, INTERMEDIATE RANGE + REACTOR, LMFBR

6-030351 ALSO IN CATEGORY 9 DATED 1968
HYNDMAN RM + NICHOLSON RB
THE EBR-II FEEDBACK FUNCTION
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
ANL-7476 +. JULY 1968, 37 FIGURES, 4 REFERENCES, 56P.

CONSIDERABLE EFFORT HAS BEEN MADE TO ASSESS THE SYSTEMS STABILITY OF EBR-II BECAUSE OF BOTH THE
GENERAL NEED TO GAIN INFORMATION ON FAST-REACTOR SYSTEMS AND THE PARTICULAR NEED TO MAINTAIN SAFE
OPERATION OF EBR-II, WHICH PLAYS A CRUCIAL ROLE AS AN IRRADIATION FACILITY IN THE LMFBR PROGRAM.
THIS REPORT IS THE RESULT OF AN ATTEMPT TO DEDUCE AS MUCH INFORMATION AS POSSIBLE ON THE EBR-II
FEEDBACK FUNCTION IN RUN 26 FROM A SERIES OF OSCILLATOR AND ROD-DROP EXPERIMENTS PERFORMED IN
OCT. AND NOV. 1967.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + REACTOR STABILITY + REACTOR, TEST + TRANSFER FUNCTION +
REACTOR, ERDA OWNED + TEST, DROP + CONTROL RODS

6-031349 DATED 1963
ZALESKI CP
COEFFICIENT OF REACTIVITY FOR LARGE POWER REACTORS
COMITATO NAZIONALE ENERGIA NUCLEARE
CONF-193-(VOL.2) +. 35 PAGES, PAGES 55-89 OF 1 REATTORI VELOCI. ROME, COMITATO NAZIONALE ENERGIA NUCLEARE,
1967, FROM 8TH NUCLEAR CONGRESS, ROME, JUNE 1963, IN FRENCH

THREE REACTIVITY EFFECTS EVALUATED FOR FAST-NA COOLED REACTORS ARE THE DOPPLER, THE FUEL DILUTION,
AND NA VOID EFFECT. METHODS USED FOR THE CALCULATION OF THE DOPPLER AND NA VOID EFFECT ARE
GIVEN. THE EFFECTS ARE EVALUATED FOR LARGE (2500 LITERS) AND SMALL (40 LITERS) CORES. DESIGN
CHARACTERISTICS OF THE FOUR REACTOR MODELS USED FOR REFERENCE STUDIES ARE TABULATED. ISOTHERMAL
COEFFICIENTS OF REACTIVITY ARE LISTED. REACTIVITY TRANSFER FUNCTIONS ARE SHOWN GRAPHICALLY.
POWER AND REACTIVITY BEHAVIOR IN THE EVENT OF NA EXPULSION ARE EVALUATED.

DOPPLER COEFFICIENT + TRANSFER FUNCTION + VOID COEFFICIENT + MATHEMATICAL TREATMENT + REACTOR, LMFBR

6-031354 DATED 1968
ZINIUK MA + WILBER HA

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-031354 *CONTINUED*

SETBACK TESTS TO DETERMINE THERMAL TRANSIENTS AND CONTROL PERFORMANCE - ENRICO FERMI REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., MICH.
APDA-NTS-21 +. 33 PAGES, 5 FIGURES, 3 TABLES, 4 REFERENCES, NOVEMBER 1968

GIVES RESULTS OF SETBACK TESTS CONDUCTED ON THE FERMI REACTOR AT THERMAL POWER LEVELS OF 50 AND 75 MW. SETBACKS AT EACH POWER LEVEL WERE INITIATED MANUALLY, AND BY TEMPERATURE AND POWER, TO DETERMINE THERMAL TRANSIENTS, SETBACK-SYSTEM OPERATION, AND PERFORMANCE OF REACTOR CONTROLS. RECORDINGS OBTAINED IN THE TESTS FURNISHED VALUES OF THE THERMAL TRANSIENTS AND THE MAGNITUDE OF THE POWER OF THE SETBACKS FOR CONDITIONS OF THE TEST. SATISFACTORY CONTROL OF THE REACTOR IN SETBACK WAS VERIFIED.

AVAILABILITY - M.A. ZINIOK, ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + REACTOR TRANSIENT + REACTOR CONTROL + TEST, PLANT RESPONSE + THERMAL TRANSIENT

6-031435

DATED 1963

ZALESKI CP

COEFFICIENT OF REACTIVITY FOR LARGE POWER REACTORS

COMITATO NAZIONALE ENERGIA NUCLEARE

CUNF-193(VOL.2) +. 35 PAGES, PAGES 55-89 OF I REATTORI VELOCI., ROME, COMITATO NAZIONALE ENERGIA NUCLEARE, 1967, FROM 8TH NUCLEAR CONGRESS, ROME, JUNE 1963, (IN FRENCH)

THREE REACTIVITY EFFECTS EVALUATED FOR FAST-NA COOLED REACTORS ARE THE DOPPLER, THE FUEL DILUTION, AND NA VOID EFFECT. METHODS USED FOR THE CALCULATION OF THE DOPPLER AND NA VOID EFFECT ARE GIVEN. THE EFFECTS ARE EVALUATED FOR LARGE (2500 LITERS) AND SMALL (40 LITERS) CORES. DESIGN CHARACTERISTICS OF THE FOUR REACTOR MODELS USED FOR REFERENCE STUDIES ARE TABULATED. ISOTHERMAL COEFFICIENTS OF REACTIVITY ARE LISTED. REACTIVITY TRANSFER FUNCTIONS ARE SHOWN GRAPHICALLY. POWER AND REACTIVITY BEHAVIOR IN THE EVENT OF NA EXPULSION ARE EVALUATED.

DOPPLER COEFFICIENT + TRANSFER FUNCTION + VOID COEFFICIENT + MATHEMATICAL TREATMENT + REACTOR, LMFBR

6-032266

ALSO IN CATEGORY 5

DATED 1968

FLOWERS CL + GERHARDSTEIN LH

ANALOG-HYBRID DYNAMIC SIMULATION OF THE FFTF REACTOR AND HEAT TRANSPORT SYSTEM

BATTILLE-NORTHWEST, RICHLAND, WASHINGTON

BNWL-707 +. 101 PAGES, FIGURES, TABLES, 18 REFERENCES, APRIL 1968

DEVELOPED MATHEMATICAL MODEL FOR INVESTIGATION DYNAMIC RESPONSE AND CONTROL CHARACTERISTICS OF THE FAST FLUX TEST FACILITY REACTOR AND HEAT-REMOVAL SYSTEMS. MODEL WAS DEVELOPED FOR SOLUTION ON THE ANALOG-HYBRID COMPUTER AT PNL. REPORT DISCUSSES MODELING TECHNIQUES USED TO SIMULATE EACH SYSTEM COMPONENT. ALSO GIVES SAMPLE TRANSIENT RESULTS. DISCUSSES PROSPECTIVE USES OF THE SIMULATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

COMPUTER PROGRAM + DESIGN CRITERIA + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + REACTOR KINETICS + REACTOR, TEST + THERMAL ANALYSIS + ANALYTICAL MODEL + FFTF (TR) + ANALOG SIMULATION + HEAT TRANSFER ANALYSIS

6-032391

ALSO IN CATEGORY 12

DATED 1969

MAGNUSON DW

CRITICALITY CALCULATIONS FOR LIGHT WATER REACTOR AND LIQUID METAL FAST BREEDER REACTOR FUELS

UNION CARBIDE CORPORATION, Y-12 PLANT, TENN.

Y-OR-9 +. 50 PAGES, 11 FIGURES, 8 TABLES, 6 REFERENCES, MARCH 4, 1969

CRITICALITY CALCULATIONS WERE PERFORMED OVER THE RANGE OF MODERATION TO SHOW CRITICAL SIZES AND MASSES FOR DISCHARGE FUEL COMPOSITIONS ANTICIPATED IN LWRS AND LMFBRs. THE STUDIES WERE DIRECTED TOWARD THE GENERATION OF CALCULATED DATA FOR CRITICALITY CONTROL IN PRELIMINARY EQUIPMENT DESIGNS FOR FUEL-RECYCLE PROCESSING PLANTS.

AVAILABILITY - D.W. MAGNUSON, OAK RIDGE Y-12 PLANT, P.O. BOX Y, OAK RIDGE, TENN. 37830

CRITICALITY SAFETY + DESIGN CRITERIA + PLUTONIUM + REACTOR, LWR + MATHEMATICAL TREATMENT + URANIUM-238 + REACTOR, LMFBR

6-032527

ALSO IN CATEGORIES 5 AND 9

DATED 1967

STACHURA SJ + BLAINE RA + BERLAND RF + HAUSKNECHT DF

AIROS-II - REACTOR DYNAMICS SIMULATOR FOR LMFBR DESIGN

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

CUNF-670916 +. 9 PAGES, PAGES IVAS.1-9 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

MODIFICATIONS TO THE DIGITAL KINETICS CODE, AIROS, HAVE BEEN INCORPORATED TO PROVIDE CAPABILITIES FOR TREATING PHASE CHANGES DURING TRANSIENTS SUCH AS COOLANT VOIDING AND FUEL MELTING BASED UPON SIMPLIFIED MODELS. THE GENERALIZED NATURE OF THE FEEDBACK EQUATIONS PERMITS DETAILED TREATMENT OF EITHER NEUTRONIC, THERMAL, OR FLOW BEHAVIOR.

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-032527 *CONTINUED*
COMPUTER PROGRAM + FUEL MELTDOWN + REACTOR DYNAMICS + VOID COEFFICIENT + ANALYTICAL MODEL + PHASE CHANGE +
SIMULATION + REACTOR, LMFBK

6-032623 ALSO IN CATEGORY 5 DATED 1967
MACFARLANE DR + MCNEAL N + MENELEY DA + SANATHANAN CK
THEORETICAL STUDIES OF THE RESPONSE OF FAST REACTORS DURING SODIUM BOILING ACCIDENTS
ARGONNE NATIONAL LAB., ILL.
CONF-670916 +. 27 PAGES, 16 REFERENCES, PP. 15, 1-26 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE
SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

THE PHYSICAL MODELS AND EQUATIONS USED TO DESCRIBE THE KINETIC BEHAVIOR OF A LARGE FAST POWER
BREEDER REACTOR UNDER NA BOILING CONDITIONS ARE DESCRIBED. THE MODELS DESCRIBED ARE THE NA VOID
MODEL, HEAT TRANSFER MODEL, NEUTRON KINETICS MODEL, AND FEEDBACK REACTIVITY COEFFICIENTS.
PARAMETERS FOR A REFERENCE CORE DESIGN ARE TABULATED. CALCULATED DATA SHOWN GRAPHICALLY.

REACTOR DYNAMICS + SODIUM + ANALYTICAL MODEL + REACTOR, LMFBK + BOILING

6-032656 ALSO IN CATEGORY 18 DATED 1967
BILLURIS G + PFLASTERER R + WOLFE B
THE SEFOR EXPERIMENTAL PROGRAM AND ITS INFLUENCE ON REACTOR DESIGN
GENERAL ELECTRIC CO., SUNNYVALE, CALIF.
CONF-670916 +. 10 PAGES, PP. VII, 1-9 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST
REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

DESCRIBES THE PROPOSED SEFOR EXPERIMENTAL PROGRAM, THE APPROACH TO ANALYSIS OF THE EXPERIMENTAL
DATA, AND THE VARIOUS FEATURES INCORPORATED INTO SEFOR TO OBTAIN HIGH QUALITY EXPERIMENTAL
RESULTS, AND THE APPROACH PLANNED TO ENSURE REACTOR SAFETY DURING THE TRANSIENT TESTS.

REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR DYNAMICS + TESTING + REACTOR, PULSED + SEFOR
(RE) + R AND D PROGRAM + REACTOR, LMFBK

6-032797 DATED 1968
BRAESS D + THURNAY K
THEORETICAL STUDY OF HYPOTHETICAL SERIOUS ACCIDENTS IN FAST POWER REACTORS
KERNFORSCHUNGSZENTRUM, KARLSRUHE, WEST GERMANY
EURFNR-579 + KFK-813 +. 18 PAGES, TRANSLATION, JULY 1968

IN THE DESIGN OF FAST REACTORS, THE SAFETY CONSIDERATIONS ARE GOVERNED BY THE INVESTIGATION OF
SECOND EXCURSIONS. THESE EXCURSIONS MIGHT OCCUR WHEN THERE IS AN ASSUMED FAILURE OF THE COOLING
SYSTEM AND, AT THE SAME TIME, A BREAKDOWN OF ALL SAFETY DEVICES. SECOND EXCURSIONS ARE MOSTLY
STUDIED WITH THE BETHE-TAIT METHOD. THIS METHOD IS DEVELOPED FURTHER, AND THE EFFECT OF DOPPLER
FEEDBACK AND OF DELAYED NEUTRONS IS STUDIED INTIMATELY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT ANALYSIS + BETHE TAIT + DELAYED NEUTRON + DOPPLER COEFFICIENT + ACCIDENT, HYPOTHETICAL + EXCURSION,
LARGE + GERMANY + MATHEMATICAL TREATMENT + REACTOR, LMFBK + ACCIDENT, CORE DISRUPTIVE

6-032821 ALSO IN CATEGORY 18 DATED 1967
MOCKUP EXPERIMENTS IN ZPR-3
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-1 +. 40 PAGES, FIGURES, TABLES, 24 REFERENCES, SEFOR, PSAR, SUPPLEMENT 1, DEC. 5, 1967, DOCKET 50-231

ANALYSES OF A SERIES OF CRITICAL EXPERIMENTS AND APPLICATION OF RESULTS TO THE SEFOR DESIGN ARE
DISCUSSED. CRITICAL-MASS VALUES WERE DETERMINED FOR 1-, 2-, AND 3-SEGMENT FUEL DESIGNS TO HELP
ESTABLISH THE PLUTONIUM ATOM FRACTION. REACTIVITY EFFECTS OF AXIAL FUEL EXPANSION WERE MEASURED
WHICH LED TO SELECTION OF THE 2-SEGMENT DESIGN FOR SEFOR. MEASUREMENTS OF THE REACTIVITY WORTH
OF THE RADIAL REFLECTOR ESTABLISHED THE ADEQUACY OF THE REFLECTOR CONTROL SYSTEM.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM
CHARGE \$2.00)

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + MOCKUP + REACTIVITY COEFFICIENT + CRITICALITY EXPERIMENT
+ MEASUREMENT, REACTIVITY + REACTOR, PULSED + SEFOR (RE) + REPORT, PSAR + REACTOR PHYSICS + CRITICAL MASS +
REACTOR, LMFBK

6-032845 ALSO IN CATEGORY 17 DATED 1968
CHAPELET M + EDELINE JC + LHAUBET G
EXPERIMENTAL STUDY OF THE RAPSDIE PROTECTIONS
COMMISSARIAT A L ENERGIE ATOMIQUE, FONTENAY-AUX-ROSES, FRANCE
CEA-R-3620 +. 192 PAGES, 49 FIGURES, 14 TABLES, OCTOBER 1968

GIVES DETAILS OF PROTECTION TESTS MADE BY THE PROTECTION SUB-COMMISSION DURING THE STARTING-UP
TRIALS ON THE REACTOR RAPSDIE. THE MAIN TYPES OF TESTS REPORTED CONCERN THE FLUX MEASUREMENTS
IN THE CORE, THE REFLECTORS, THE SODIUM AND THE NEAR BIOLOGICAL PROTECTIONS, THE RADIO-PROTECTION

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-032845 *CONTINUED*

MEASUREMENTS IN THE PREMISES, AND THE ACTIVATION OF THE PRINCIPAL FLUIDS. IN THE LAST PART, RESULTS ARE ANALYZED AND COMPARED WITH FORECAST CALCULATIONS.

AVAILABILITY - US DEPOSITORY LIBRARIES FOR THE U. S. AND OVERSEAS

REACTOR, FAST + FLUX DISTRIBUTION + FRANCE + COMPARISON, THEORY AND EXPERIENCE + REACTOR, TEST + REACTOR STARTUP TESTING + REACTOR, LMFBR

6-032889 ALSO IN CATEGORIES 9 AND 18 DATED 1968

RESPONSE TO QUESTION 12 - FRED OPERATING VARIABLES

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-20 +. 3 PAGES, 1 FIGURE, PAGE 41 THRU 43 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

PROVIDE INFORMATION ON OPERATIONAL VARIABLES TO BE USED DURING THE FAST REACTIVITY EXCURSION DEVICE EXPERIMENTS AND INDICATE WHICH VALUES HAVE BEEN OBTAINED BY MEANS OTHER THAN DIRECT MEASUREMENT. *** ANSWER ON PAGES 41 THRU 43.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + MEASUREMENT, REACTIVITY + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + INSTRUMENTS, MISC.

6-032890 ALSO IN CATEGORIES 9 AND 18 DATED 1968

RESPONSE TO QUESTION 13 - INSTALLATION AND REMOVAL OF TEST DEVICES

GENERAL ELECTRIC

50231-20 +. 4 PAGES, 1 FIGURE, PG. 44 THRU 48 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1969, DOCKET 50-231

FROM INFORMATION SUBMITTED, IT APPEARS THAT DUE TO HEIGHT AND OTHER LIMITATIONS, THE EXPERIMENTAL EQUIPMENT FOR EXCURSION MEASUREMENT MUST BE PARTIALLY ASSEMBLED AND DISASSEMBLED IN PLACE. DESCRIBE SEQUENTIALLY THE STEPS INVOLVED AND THE CHECKS THAT WILL BE MADE IN REMOVING AND INSTALLING THE DEVICES. *** ANSWER ON PAGES 44 THRU 48

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + EXCURSION, LARGE + MEASUREMENT, REACTIVITY + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + INSTRUMENTS, MISC.

6-032892 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION 15 - PREOPERATIONAL CHECKS OF TEST DEVICES

GENERAL ELECTRIC, SAN JOSE

50231-20 +. 1 PAGE, PAGE 51 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

WHAT PRECAUTIONS WILL BE TAKEN TO ENSURE THAT ASSEMBLY STEPS HAVE BEEN PROPERLY CARRIED OUT AND EQUIPMENT WILL OPERATE PROPERLY PRIOR TO THE ACTUAL EXCURSION EXPERIMENT. *** (ANSWER) THE FAST REACTIVITY EXCURSION DEVICE CAN BE TEST-FIRED FROM OUTSIDE CORE USING POSITIONER TO SET INITIAL POSITION 30 IN. ABOVE CORE MIDPLANE, PROVIDING CAPABILITY OF RUNNING COMPLETE TEST WITHOUT AFFECTING REACTOR. AN OPERATIONAL CHECKOUT WILL BE PERFORMED TO CHECK PROPER OPERATION OF SWITCHES AND LATCHES BEFORE EACH TEST. SIMILAR TESTS WILL BE MADE WITH OSCILLATOR ROD TO ENSURE PROPER OPERATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

6-032894 ALSO IN CATEGORIES 9 AND 11 AND 18 DATED 1968

RESPONSE TO QUESTION 17 - DESIGN OF PRESSURE VESSELS IN FRED

GENERAL ELECTRIC

50231-10 +. 2 PAGES, 1 FIGURE, PG. 53 AND 54 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

DESCRIBE BLOWOUT PATCH AND ACCUMULATOR ON THE EXCURSION DEVICE. *** ANSWER ON PAGES 53 AND 54.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + INSTRUMENTS, MISC.

6-032896 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION 19 - BOLT PRELOAD FOR FRED EXPANSION CHAMBER

GENERAL ELECTRIC

50231-20 +. 1 PAGE, PG. 56 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-032896 *CONTINUED*

DISCUSS SAFETY FACTOR OF THE SIX STUDS ON THE EXCURSION DEVICE WHEN THE CHAMBER EXPERIENCES ITS MAXIMUM PRESSURE PEAK. *** (ANSWER) THESE STUDS ARE DESIGNED SO THEY WOULD YIELD IF DEVICE WERE OVERPRESSURIZED AND BLOWOUT PATCH FAILED TO OPERATE PROPERLY. THEY ARE SIZED TO BEGIN ELONGATING AT APPROXIMATELY 1500 PSI AND RESULT IN PRESSURE RELIEF. NORMAL PRESSURE IN EXPANSION CHAMBER WILL BE A MAXIMUM OF 200 PSI.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TENSILE PROPERTY + TESTING + FASTENER + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + SAFETY MARGIN + REACTOR, LMFBR

6-032919 ALSO IN CATEGORIES 9 AND 18 DATED 1968

RESPONSE TO QUESTION 4 - DESCRIPTION OF EXCURSION TESTING DEVICE

GENERAL ELECTRIC, SAN JOSE

50231-21 +. 35 PAGES, 13 FIGURES, 1 TABLE, PAGES 16 THRU 51 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL INFORMATION ON THE EXCURSION TESTING DEVICE (FRED) INCLUDING DETAILS OF ITS DESIGN AND HOW IT WILL BE UTILIZED IN THE SEFOR EXPERIMENTAL PROGRAM. *** ANSWER ON PAGES 18 THRU 51.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + EQUIPMENT + REPORT, PSAR + REACTOR, LMFBR

6-032921 ALSO IN CATEGORIES 11 AND 18 DATED 1968

RESPONSE TO QUESTION 6 - CORE COMPONENT LOADINGS DURING TRANSIENT TESTS

GENERAL ELECTRIC, SAN JOSE

50231-21 +. 5 PAGES, 1 FIGURE, 1 TABLE, PAGES 57 THRU 62 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL INFORMATION ON DYNAMIC LOADINGS IMPOSED ON CORE SUPPORT STRUCTURE DURING PROMPT TRANSIENT TEST PROGRAM. *** ANSWER ON PAGES 57 THRU 62.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

6-033333 ALSO IN CATEGORIES 5 AND 18 DATED 1967

PETERSON RE + SIMPSON DE + LITTLE WW

SELECTED SAFETY CONSIDERATIONS IN DESIGN OF THE FAST FLUX TEST FACILITY

BATTELLE MEMORIAL INSTITUTE

BNWL-SA-1236 +. CONF-670916 +. 10 FIGURES, 4 TABLES, 9 REFERENCES, JUNE 1967, PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS

REVIEW OF SKEWED-TUBE DESIGN. LOW FERTILE CONTENT GIVES LOW DOPPLER COEFFICIENT. FEW CORE LOOPS, HIGH-WORTH RODS REQUIRE LOOPS OUTSIDE CORE. REFLECTOR INSTEAD OF BREEDING BLANKET MAXIMIZES NEUTRON FLUX. SUMMARIZES RESULTS OF VARIOUS PRELIMINARY TRANSIENTS OR ACCIDENTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, TEST + FFTF (TR) + R AND D PROGRAM + REACTOR, LMFBR

6-033427 ALSO IN CATEGORIES 5 AND 18 DATED 1969

GOLDEN GH

ELEMENTARY NEUTRONICS CONSIDERATIONS IN LMFBR DESIGN

ARGONNE NATIONAL LABORATORY

ANL-7552 +. 68 PAGES, 12 FIGURES, 10 TABLES, 35 REFERENCES, MARCH 1969

THE SUBJECT MATERIAL PROGRESSES FROM BASIC PHYSICS CONCEPTS TO THE DESIGN OF REAL LMFBRs - FLUX SPECTRA AND CROSS-SECTION VARIATION WITH ENERGY IN FAST REACTORS, ASPECTS OF MULTIGROUP CALCULATIONS IN DIFFUSION THEORY, THE SIGNIFICANCE OF BREEDING AND CONVERSION RATIOS, SPECIFIC POWER, AND DOUBLING TIME, RADIAL CORE ZONING, REFUELING (I.E. TIME-DEPENDENT BEHAVIOR), A DISCUSSION OF REACTIVITY EFFECTS AS RELATED TO SAFETY, WITH SPECIFIC REFERENCE TO SODIUM VOID, DOPPLER, AND EXPANSION EFFECTS, AND CURRENT DESIGN APPROACHES. PRESENTATION IS APPLICABLE TO DESIGN OF ANY FAST BREEDER, REGARDLESS OF COOLANT. HENCE, PERTINENT COMMENTS WERE MADE REGARDING GAS-COOLED FAST-BREEDER REACTORS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-033427 *CONTINUED*

ACCIDENT ANALYSIS + REACTOR, BREEDER + DOPPLER EFFECT + REACTOR, FAST + REACTIVITY COEFFICIENT + REACTIVITY EFFECT + REACTOR DYNAMICS + REACTOR KINETICS + VOID COEFFICIENT + ANL + DESIGN STUDY + REACTOR PHYSICS + R AND D PROGRAM + REACTOR, HTGR + REACTOR, LMFBR

6-033515

DATED 1968

KYUS MW + ADKINS CR + MURLEY TE

AN EXAMINATION OF METHODS FOR CALCULATING THE DOPPLER COEFFICIENT IN FAST BREEDER REACTORS

WESTINGHOUSE ELECTRIC CORPORATION, MADISON, PENNSYLVANIA

NBS-SPEC.-UBL.-299 (VOL.1) + CONF-680307-(VOL. 1) +. 8 PAGES, PAGES 337-44 OF NEUTRON CROSS SECTIONS AND TECHNOLOGY, NATIONAL BUREAU OF STANDARDS, 1968

THE UNRESOLVED RESONANCE REGION IS TREATED RIGOROUSLY THROUGH THE GENERATION OF INDIVIDUAL PSEUDO RESONANCES USING RANDOM SAMPLING TECHNIQUES WITHIN THE FRAMEWORK OF A SINGLE LEVEL BREIT-WIGNER FORMALISM. A PROCEDURE IS USED WHICH ENSURES THAT PARAMETERS YIELD RESONANCE INTEGRALS CONSISTENT WITH BROAD GROUP MEASURED DATA. THE RESONANCE STRUCTURE OF EACH ISOTOPE IS THEN COMPLETE OVER THE ENERGY RANGE OF INTEREST FOR DOPPLER CALCULATIONS. THIS RESONANCE DATA IS USED IN CONJUNCTION WITH AN ULTRAFINE GROUP INTEGRAL TRANSPORT THEORY CODE. THE VALIDITY OF THE USUAL APPROXIMATE METHODS IS EXAMINED BY COMPARING WITH THIS MORE REFINED METHOD OF CALCULATION. A NUMERICAL COMPARISON IS MADE FOR A PLUTONIUM-URANIUM CARBIDE-FUELED, SODIUM-COOLED FAST BREEDER REACTOR.

DOPPLER COEFFICIENT + RESONANCE OVERLAP + NEUTRON INTERACTION + REACTOR PHYSICS + R AND D PROGRAM + REACTOR, LMFBR

6-033944

ALSO IN CATEGORY 5

DATED 1969

LONG JK

ANALYSIS OF THE STATIC POWER COEFFICIENT OF EBR-II WITH REDUCED COOLANT FLOW

ARGONNE NATIONAL LABORATORY

21 PAGES, NUCLEAR APPLICATION 6, PAGES 116-26 (FEB. 1969)

A STUDY OF EBR-II OPERATING BEHAVIOR, WHICH BEGAN WITH A REEXAMINATION OF THE THERMAL ANALYSIS OF FUEL PINS UNDER VARYING RATES OF COOLANT FLOW, INCORPORATES THE RECENT INTERPRETATIONS OF THE BOWING BEHAVIOR OF THE SUBASSEMBLIES. THE NONLINEAR BOWING EFFECTS ARE COMBINED WITH THE LINEAR EXPANSION EFFECTS IN EQUATIONS THAT EXPRESS THE POWER-REACTIVITY DECREMENT AS A FUNCTION OF BOTH POWER AND FLOW. COMPARISON WITH EXPERIMENTS INDICATES AGREEMENT TO WITHIN A FEW INCHES. DIFFERENTIATION OF THE EQUATIONS LEADS TO EXPRESSIONS FOR POWER AND FLOW COEFFICIENTS OF REACTIVITY. THE REDUCED FLOW DATA ARE USED TO DETERMINE AN EFFECTIVE AVERAGE THERMAL EXPANSION COEFFICIENT FOR THE FUEL.

REACTIVITY EFFECT, EXPANSION + REACTOR, FAST + FUEL ELEMENT BOWING + REACTOR, LMCR + POWER COEFFICIENT + REACTIVITY COEFFICIENT + REACTOR, TEST + COOLANT COEFFICIENT + FUEL EXPANSION COEFFICIENT

6-033964

ALSO IN CATEGORY 18

DATED 1967

HAGAN JW + WORTH GA + PADILLA A

PARAMETRIC SCRAM TRANSIENT ANALYSES FOR THE FFTF FAST TEST REACTOR

BATTELLE-NORTHWEST, RICHLAND, WASHINGTON

BNWL-657 +. 27 PAGES, NOVEMBER 1967

PARAMETRIC ANALYSES OF THE TRANSIENT CHARACTERISTICS FOR VARIOUS ASSUMED RAMP REACTIVITY INPUTS WHICH ARE TERMINATED BY TRIP OF REACTOR SAFETY SYSTEM HAVE BEEN PERFORMED AND RESULTS OF STUDIES ARE DESCRIBED. IN EACH CASE REACTOR WAS ASSUMED AT OPERATING CONDITIONS AT INITIATION OF RAMP. THESE RESULTS WILL BE USED IN EVALUATION OF TRANSIENT SENSITIVITIES TO VARIOUS DESIGN PARAMETERS FOR ESTABLISHING DESIGN TRADE-OFFS AND SYSTEM LIMITS. SYSTEM VARIABLES INCLUDED IN STUDY WERE, MAGNITUDE OF RAMP REACTIVITY INSERTION, SYSTEM RESPONSE TIME, SAFETY ROD ACCELERATION RATE, AND MAGNITUDE OF DOPPLER COEFFICIENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + REACTIVITY EFFECT + REACTOR STABILITY + REACTOR TRANSIENT + REACTOR, TEST + FFTF (TR) + REACTOR CONTROL + RESPONSE TIME + THEORETICAL INVESTIGATION + REACTOR PHYSICS + R AND D PROGRAM + REACTOR, LMFBR

6-033983

ALSO IN CATEGORY 18

DATED 1968

ROY DH

FAST BREEDER REACTOR STATIC PHYSICS METHODS DEVELOPMENT AND ANALYSIS PROJECT. QUARTERLY TECHNICAL REPORT, APRIL-JUNE 1968

BABCOCK AND WILCOX COMPANY, LYNCHBURG, VA.

BAW-3867-4 +. 47 PAGES, 4 FIGURES, 21 TABLES, SEPTEMBER 1968

ONE OF A SERIES OF SIMILAR REPORTS ON B AND W PROGRAM OF ANALYTICAL METHODS DEVELOPMENT APPLICABLE TO THE PHYSICS DESIGN OF FAST-BREEDER REACTORS. TOPICS - CROSS SECTION PREPARATION AND CODE DEVELOPMENT, BURNUP STUDIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

COMPUTER PROGRAM + REACTOR, FAST + REACTOR, LMCR + ANALYTICAL TECHNIQUE + REACTOR PHYSICS + R AND D PROGRAM +

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-033983 *CONTINUED*
REACTOR, LMFBR

6-034390 ALSO IN CATEGORY 9 DATED 1966
RANUALL RL + PEKKUL PJ
APPLICATIONS OF ANALOG TIME AND FREQUENCY CORRELATION COMPUTERS TO REACTOR-SYSTEM
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA
24 PAGES, 14 FIGURES, 14 REFERENCES, PAGES 357-80 OF NEUTRON NOISE, WAVES, AND PULSE PROPAGATION, MAY 1967,
PROCEEDINGS OF A SYMPOSIUM AT THE UNIVERSITY OF FLORIDA, GAINESVILLE, FEBRUARY 14-16, 1966

ANALYSIS TECHNIQUES OF ANALOG-CROSS SPECTRAL DENSITY AND ANALOG CROSS CORRELATION ARE APPLIED TO
PRACTICAL PROBLEMS IN REACTOR DESIGN AND PERFORMANCE EVALUATION. CROSS-SPECTRAL-DENSITY
TECHNIQUES AND EQUIPMENT ARE DESCRIBED FOR EXPERIMENTAL DETERMINATION OF A NUMBER OF FEEDBACK
COEFFICIENTS VITAL TO REACTOR STABILITY AND CONTROL. CROSS-CORRELATION TECHNIQUES, TRANSDUCER
DEVELOPMENTS, AND CORRELATION EQUIPMENT ARE DESCRIBED FOR MEASUREMENT OF LOCALIZED IN-CORE
COOLANT VELOCITY AND FUEL-ELEMENT STRUCTURAL VIBRATIONS.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + INCIDENT, CONSEQUENCE + REACTOR STABILITY + SNAP 8 (SR) +
REACTOR, SPACE + TESTING + ANALYTICAL MODEL + DATA PROCESSING + REACTOR CONTROL + SRE (RE) + NOISE ANALYSIS +
ANALOG SIMULATION + FLOW

6-034669 ALSO IN CATEGORIES 11 AND 18 DATED 1969
ROSSIN AD + GLASSNER A
REACTOR DEVELOPMENT PROGRESS REPORT
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
ANL-7553 +. 123 PAGES, FIGURES, TABLES, REFERENCES, FEB. 1969

ONE OF A SERIES OF SIMILAR REPORTS. TOPICS - LMFBR DEVELOPMENT STUDIES, FAST REACTORS, GENERAL
DEVELOPMENT STUDIES, NUCLEAR SAFETY, FUEL-MELTDOWN STUDIES WITH TREAT, CORE STRUCTURAL SAFETY,
AND CONTAINMENT RESPONSE TO HIGH-ENERGY EXCURSION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL MELTDOWN + REACTOR, SAFETY RESEARCH + REACTOR, RESEARCH + TREAT (PRR) + CONTAINMENT
INTEGRITY + EXCURSION, LARGE + REACTOR, PULSED + PRESSURE, INTERNAL + CORE COMPONENTS + STRUCTURAL INTEGRITY
+ R AND D PROGRAM + REACTOR, LMFBR

6-034751 DATED 1968
MACFARLANE DR + MACNEAL N + MENELEY DA + SANATHANAN CI
THEORETICAL STUDIES OF THE RESPONSE OF FAST REACTORS DURING SODIUM BOILING ACCIDENTS
ARGONNE NATIONAL LAB., ILL.
18 PAGES, BULL. INFORM. SCI. TECH. (PARIS); NO. 122, PAGES 35-52 (JAN. 1968) (IN FRENCH)

A CALCULATIONAL METHOD FOR THE TRANSIENT PHENOMENA WHICH CAN TAKE PLACE DURING COOLANT BOILING IN
A REACTOR CORE IS DESCRIBED. IT IS THUS POSSIBLE TO STUDY THE WEAK OVERPOWER TRANSIENTS
ORIGINATING FROM THE COOLANT FLOW SHUTDOWN. THE MODEL SHOULD ALSO PERMIT THE FORMULATION AND
ANALYSIS OF TRANSIENT BOILING EXPERIMENTS WHICH SHOULD SUPPLY STILL MISSING DATA. THE
CALCULATIONS APPLY PARTICULARLY TO CERAMICS FUELED AND SODIUM COOLED FAST BREEDER POWER REACTORS.

REACTOR, FAST + ACCIDENT, LOSS OF COOLANT + SODIUM COEFFICIENT + COOLANT COEFFICIENT + THEORETICAL
INVESTIGATION + REACTOR, LMFBR

6-035039 ALSO IN CATEGORY 18 DATED 1969
STEWART JE
PARAMETRIC EVALUATION OF THE INFLUENCE OF 239PU AND FISSION PRODUCT CAPTURE CROSS SECTIONS ON LMFBR PERFORMANCE
BABCOCK AND WILCOX, LYNCHBURG, VA.
BAW-3867-1 +. 69 PAGES, TABLES, REFERENCES, MARCH, 1969

THIS STUDY INVESTIGATED LARGE UNCERTAINTIES, OVER THE ENTIRE SPECTRUM, IN THE CAPTURE CROSS
SECTIONS OF PU-239 AND LUMPED PLUTONIUM FISSION FRAGMENTS. EMPHASIS WAS PLACED ON DETERMINING
THE EFFECTS OF THE UNCERTAINTIES ON THE DEPLETION AND FUEL MANAGEMENT CHARACTERISTICS OF TWO
LMFBR CORE SIZES. COMPARISONS WERE MADE FOR THE APPROACH TO EQUILIBRIUM AND PARTICULARLY FOR THE
EQUILIBRIUM CYCLE ITSELF.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

FUEL BURNUP + MEASUREMENT + PLUTONIUM + NEUTRON + CROSS SECTION + REACTOR PHYSICS + R AND D PROGRAM + REACTOR,
LMFBR + FUEL MANAGEMENT

6-035349 ALSO IN CATEGORY 5 DATED 1969
LUNG JK
ANALYSIS OF THE STATIC POWER COEFFICIENT OF EBR-II WITH REDUCED COOLANT FLOW
ARGONNE NATIONAL LABORATORY
11 PAGES, 6 FIGURES, 2 TABLES, 13 REFERENCES, NUCLEAR APPLICATIONS, 6(3), PAGES 116-26 (FEBRUARY 1969)

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-035349 *CONTINUED*

THIS STUDY OF EBR-II OPERATING BEHAVIOR, WHICH BEGAN WITH A REEXAMINATION OF THE THERMAL ANALYSIS OF FUEL PINS UNDER VARYING RATES OF COOLANT FLOW, INCORPORATES THE RECENT INTERPRETATIONS OF THE BOWING BEHAVIOR OF THE SUBASSEMBLIES. THE NONLINEAR BOWING EFFECTS ARE COMBINED WITH THE LINEAR EXPANSION EFFECTS IN EQUATIONS THAT EXPRESS THE POWER-REACTIVITY DECREMENT AS A FUNCTION OF BOTH POWER AND FLOW. COMPARISON WITH EXPERIMENTS INDICATES AGREEMENT TO WITHIN A FEW HOURS. DIFFERENTIATION OF THE EQUATIONS LEADS TO EXPRESSIONS FOR POWER AND FLOW COEFFICIENTS OF REACTIVITY. THE REDUCED FLOW DATA ARE USED TO DETERMINE AN EFFECTIVE AVERAGE THERMAL EXPANSION COEFFICIENT FOR THE FUEL.

FUEL ELEMENT BOWING + REACTOR, LMCR + REACTIVITY EFFECT + REACTOR DYNAMICS + SODIUM + REACTOR, TEST + REACTOR COOLANT + ANL + REACTOR, LMFR

6-036060

ALSO IN CATEGORY 18

DATED 1969

HANSEN NP

SEFOR CORE CLAMPING SYSTEM

GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.

2 PAGES, 1 FIGURE, ANS TRANSACTIONS, 12(1), PAGES 321-22 (JUNE 1969)

OBJECTIVE WAS TO MAKE AN ACCURATE DOPPLER-COEFFICIENT MEASUREMENT DURING TRANSIENT CONDITIONS. REACTIVITY EFFECTS CAUSED BY FUEL-ROD THERMAL BOWING MUST BE CONTROLLED. A CORE-CLAMPING SYSTEM WAS NEEDED TO PREVENT MOVEMENT OF RODS THAT WOULD CREATE REACTIVITY EFFECTS. THE CLAMPING SYSTEM DESIGNED TO SATISFY NUMEROUS REQUIREMENTS UTILIZES TORSION RODS TO PROVIDE CLAMPING FORCE AND FLEXIBILITY. THIRTY-SIX TORSION RODS SPACED AROUND THE CHANNELS DISTRIBUTE RADIAL FORCE TOWARD CENTER OF CORE. CLAMPING SYSTEM SUCCESSFULLY MET OBJECTIVES.

AVAILABILITY - N.P. HANSEN, GENERAL ELECTRIC COMPANY, 310 DEGUIGNE DRIVE, SUNNYVALE, CALIFORNIA 94086

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + MEASUREMENT, REACTIVITY + REACTOR, PULSED + SEFOR (RE) + CORE COMPONENTS

6-036078

ALSO IN CATEGORIES 9 AND 17

DATED 1969

SMITH RR + BUMP TR + CUSHMAN RA + HYNDMAN RW + KIRN FS + LOEWENSTEIN WB + LONG KJ + MADELL JT + PERSIANI PJ + WALLIN WR

THE EFFECTS OF AN OVER-COOLED STAINLESS STEEL REFLECTOR ON THE EBR II POWER COEFFICIENT

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ANL-7544 +. 115 PAGES, 60 FIGURES, 20 TABLES, 42 REFERENCES, MAY, 1969

DURING A ROUTINE STARTUP FOR EBR-II RUN 25 ON APRIL 21, 1967, SIGNIFICANT CHANGES WERE NOTED IN THE BASIC BEHAVIOR OF THE POWER COEFFICIENT. THE MAGNITUDE OF THE INTEGRAL POWER COEFFICIENT DECREASED BY A FACTOR OF APPROXIMATELY TWO WHILE THE MAGNITUDE OF THE DIFFERENTIAL POWER COEFFICIENT IN THE MIDPOWER REGION DECREASED BY A FACTOR OF APPROXIMATELY FOUR. CONSIDERABLE ANALYTICAL EFFORT SHOWED THE CHANGE IN THE POWER COEFFICIENT TO BE IDENTIFIED WITH THERMOMECHANICAL DEFORMATION EFFECTS IN A STAINLESS STEEL REFLECTOR THAT WAS INSTALLED BEFORE THE STARTUP.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

POWER COEFFICIENT + REACTOR STABILITY + STEEL, STAINLESS + DEFORMATION + MEASUREMENT, REACTIVITY + REFLECTOR + SYSTEM DESCRIPTION + REACTIVITY EFFECT, ANOMALOUS + REACTOR PHYSICS + OPERATING EXPERIENCE + REACTOR, LMFR

6-036597

ALSO IN CATEGORIES 5 AND 18

DATED 1967

SIMPSON DE + LITTLE WW + PETERSON RE

SELECTED SAFETY CONSIDERATIONS IN DESIGN OF THE FAST FLUX TEST FACILITY

BATTELLE-NORTHWEST, WASHINGTON

37 PAGES, 10 FIGURES, 9 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 1967

DESCRIBES FFTF. ACCIDENT-SEVERITY CRITERIA DISCUSSED IN TERMS OF MAX. TRANSIENT FUEL TEMPERATURES AND EXTENT OF MELTING. THE MAJOR SAFETY CONSIDERATIONS AT THE CONCEPTUAL STAGE ARE OUTLINED. TOPICS - FFTF DESCRIPTION, SAFETY PHILOSOPHY AND CRITERIA, TRANSIENT AND ACCIDENT ANALYSES, DESIGN-BASIS ACCIDENT, DESIGN CONSIDERATIONS, AND EFFECTS OF SAFETY ON TESTING CONDITIONS.

AVAILABILITY - D.E. SIMPSON + W.W. LITTLE + R.E. PETERSON, PACIFIC NORTHWEST LABORATORY, P.O. BOX 999, RICHLAND, WASHINGTON

ACCIDENT ANALYSIS + DESIGN CRITERIA + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + SAFETY PRINCIPLES AND PHILOSOPHY + R AND D PROGRAM + REACTOR, LMFR

6-036600

ALSO IN CATEGORY 18

DATED 1968

ANL - CHEMICAL ENGINEERING DIVISION RESEARCH HIGHLIGHTS

ARGONNE NATIONAL LABORATORY

ANL-7550 +. 118 PAGES, FIGURES, TABLES, DECEMBER 1968

ONE OF A SERIES OF SIMILAR REPORTS. TOPICS - COMPACT PYROCHEMICAL PROCESS, SALT TRANSPORT PROCESS FOR FAST BREEDER FUELS, FLUORIDE VOLATILITY PROCESS (CONCEPTUAL PROCESS FOR FBR FUELS), MATERIALS CHEMISTRY AND THERMODYNAMICS, XENON TAGGING IN EBR-2 FUEL ELEMENTS, CHEMISTRY OF LIQUID METALS, ENERGY CONVERSION (CELLS WITH LITHIUM ANODES AND SODIUM ANODES), CROSS SECTION MEASUREMENT, AND

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-036600 *CONTINUED*
FLUIDIZED-BED COMBUSTION OF FOSSIL FUELS. ADDENDUM LISTS ABOUT 200 REPORTS PUBLISHED IN 1968 ON THE ABOVE SUBJECTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + REACTOR, BREEDER + CHEMICAL REACTION + ENERGY SOURCE + REACTOR, FAST + MEASUREMENT + ANL + CHEMICAL KINETICS + METAL, LIQUID + DIRECT ENERGY CONVERSION DEVICES + CROSS SECTION + FUEL REPROCESSING + R AND D PROGRAM + PYROCHEMICAL + REACTOR, LMFBR

6-037182 ALSO IN CATEGORY 11 DATED 1967
1000 MWE LIQUID METAL FAST BREEDER REACTOR FOLLOW-ON STUDY - REPORT OF TASKS II AND III. A REFERENCE CONCEPTUAL DESIGN OF A 1000 MWE LMFBR. VOLUME II. STATIC DESIGN AND PERFORMANCE ANALYSES
COMBUSTION ENGINEERING, INC., WINDSOR, CONNECTICUT
CEND-337(VOL.2) +. 121 PAGES, FIGURES, TABLES, REFERENCES, 1967

THE 1000 MWE LIQUID METAL FAST BREEDER REACTOR FOLLOW-ON STUDY IS AN EXTENSION OF 1000 MWE DESIGN STUDIES PERFORMED PREVIOUSLY FOR THE ATOMIC ENERGY COMMISSION. (1) THE OBJECTIVE OF THE FOLLOW-ON PROGRAM IS TO PROVIDE INFORMATION WHICH WILL BE OF ASSISTANCE TO THE AEC IN THE FURTHERANCE OF THE LIQUID METAL FAST BREEDER REACTOR (LMFBR) PROGRAM PLAN. THE SCOPE OF THE WORK FOR COMBUSTION ENGINEERING IS IN THE AREA OF THE HEAT GENERATION SYSTEM OF THE NUCLEAR STEAM SUPPLY SYSTEM. THE OBJECTIVE OF THIS STUDY OF THE HEAT GENERATION SYSTEM HAS BEEN TO IDENTIFY THE RESEARCH AND DEVELOPMENT PROGRAMS WHICH ARE REQUIRED TO ACHIEVE THE SELECTED DESIGN BY 1980, OR AS SPECIFIED BY THE CONTRACTOR FOR A YEAR BETWEEN 1975 AND 1985.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + DESIGN STUDY + REFUELING + SYSTEM DESCRIPTION + REACTOR, LMFBR

6-037747 DATED 1969
MAGNUSON DW
LIMITING CRITICAL CONCENTRATIONS OF PLUTONIUM-URANIUM SOLUTIONS AND MIXTURES
UNION CARBIDE CORPORATION
Y-DR-17 +. 5 PAGES, 1 FIGURE, TABLES, MAY 20, 1969

LIMITING CRITICAL CONCENTRATIONS OF AQUEOUS SOLUTIONS OF PLUTONIUM AND URANIUM NITRATES AND OF MIXTURES OF THEIR CRYSTALLINE HYDRATES WERE CALCULATED FOR VARIOUS PLUTONIUM-TO-URANIUM RATIOS, USING HANSEN-ROACH CROSS SECTIONS AND S4 APPROXIMATION IN CODE ANISN. CRITICAL CONCENTRATION FOR PLUTONIUM OR LMFBR PLUTONIUM WAS FOUND NOT TO VARY STRONGLY IN THE RANGE OF U/PU ATOM RATIO FROM 0 TO 20.

AVAILABILITY - D.W. MAGNUSON, UNION CARBIDE NUCLEAR DIVISION, Y-12 PLANT, OAK RIDGE, TENNESSEE 37830

CRITICALITY SAFETY + PLUTONIUM + URANIUM + THEORETICAL INVESTIGATION + REACTOR, LMFBR

6-037850 ALSO IN CATEGORIES 5 AND 12 DATED 1967
DESCAMPS C + EGLME M + LEE J + MICHEL A
ANALYSIS OF THERMAL AND HYDRAULIC BEHAVIOR OF SODIUM DURING A PROMPT POWER INSERTION IN A FUEL ASSEMBLY
COMMISSARIAT A L'ENERGIE ATOMIQUE, PARIS
CONF-670916 +. 17 PAGES, 14 REFERENCES, SEPT. 1967

DURING THE STUDY AND THE DESIGN OF AN IN-PILE NA LOOP FOR FUEL ELEMENT RUPTURE TEST, THE FUEL DISPERSION AND SUBSEQUENT NA BEHAVIOR AFTER A PROMPT POWER INSERTION HAD TO BE EVALUATED. THE FUEL HEATING, THE CLADDING RUPTURE, THE FUEL DISPERSION IN THE NA AND THE SUDDEN NA BOILING WERE STUDIED. THE EXPERIMENTAL MEANS ARE DESCRIBED WHICH COULD CONTROL THE PROPOSED THEORETICAL MODEL.

AVAILABILITY - C. DESCAMPS, ET. AL., COMMISSARIAT A L'ENERGIE ATOMIQUE, PARIS, FRANCE

FAILURE, FUEL ELEMENT + HYDRAULIC ANALYSIS + IN PILE EXPERIMENT + REACTOR, LMCR + ACCIDENT, REACTIVITY + SODIUM + THERMAL ANALYSIS + ANALYTICAL MODEL + ACCIDENT MODEL + FAILURE, CLADDING + BOILING

6-040402 ALSO IN CATEGORIES 5 AND 17 DATED 1969
HORI M + HOSLER A
A STUDY OF GAS ENTRAINMENT AND ABSORPTION-DESORPTION IN THE ENRICO FERMI REACTOR AND THEIR EFFECT ON CORE PERFORMANCE
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-LA-6 +. 50 PAGES, FIGURES, TABLES, 19 REFERENCES, JULY 1969, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

THE POSSIBLE PRESENCE OF GAS IN THE PRIMARY COOLANT AND ITS EFFECT ON CORE HEAT TRANSFER ARE DISCUSSED. GAS COULD BE SLOWLY INTRODUCED INTO THE PRIMARY SYSTEM BY GAS BUBBLE ENTRAINMENT AND/OR DIFFERENTIAL TEMPERATURE SOLUBILITY BETWEEN THE HOT AND COLD PORTIONS OF THE PRIMARY SYSTEM. A FEW POTENTIAL SITES FOR THE ACCUMULATION OF THIS GAS HAVE BEEN FOUND AND THE MAXIMUM ACCUMULATED VOLUMES AND RATES OF GAS RELEASE FROM THESE SITES DETERMINED. THE EFFECT OF A SINGLE LARGE BUBBLE AND ALSO THE CONTINUOUS PRESENCE OF GAS ON THE CORE HEAT TRANSFER HAVE BEEN STUDIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CATEGORY 6
REACTOR TRANSIENTS, KINETICS, AND STABILITY

6-C40402 *CONTINUED*

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + HEAT TRANSFER + REACTOR, LMCR + REACTOR
STABILITY + COOLANT CHEMISTRY + INCIDENT + BUBBLE + REACTIVITY EFFECT, VOID + GAS + REACTOR, LMFBR

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-004216 DATED 1964
ELLIOTT DM + KUNKEL WP
OUT-OF-PILE TEMPERATURE EVALUATION OF FISSION PRODUCT RETENTION IN SODIUM (900 DEGREE - 1400 DEGREE F)
ATOMICS INTERNATIONAL
27 PAGES, 15 FIGURES, 3 REFERENCES- APRIL 9, 1964, NAA-SR-MEMO-9383- OTS

IT WAS FOUND IN THIS EXPERIMENT THAT 0.087 % OF THE IODINE IN THE SODIUM WAS RELEASED TO THE HELIUM COVER GAS AS THE SODIUM AND FUEL TEMPERATURE WAS ELEVATED FROM 900 DEGREE F TO 1400 DEGREE F. THIS ACTIVITY WAS OBSERVED IN THE HIGH TEMPERATURE TUBING. ONLY 2×10^{-4} % OF THE IODINE IN THE SODIUM WAS OBSERVED DOWNSTREAM IN THE COLDER PORTIONS OF THE SYSTEM TUBING.

FISSION PRODUCT RELEASE + FISSION RECOIL + IODINE + REACTOR, LMCR + SODIUM

7-004938 ALSO IN CATEGORY 5 DATED 1964
THOMPSON DH + MURPHY G + FISHER RW
ENGINEERING DESIGN CONSIDERATIONS FOR A LIQUID-METAL IN-PILE LOOP
AMES LABORATORY
80 PAGES, 12 TABLES, 65 REFERENCES- JANUARY 1, 1964, IS-760- OTS

THIS PAPER PRESENTS THE RESULTS OF AN ANALYTICAL INVESTIGATION IN THE AREAS OF FISSION-PRODUCT ACTIVITY, HEAT TRANSFER, AND MATERIALS FOR THE PRELIMINARY PLANNING OF A HIGH-TEMPERATURE CIRCULATING-LIQUID-METAL-FUEL IN-PILE LOOP. EQUATIONS FOR THE FISSION-PRODUCT NUMBERS AND ACTIVITIES IN A CIRCULATING FUEL WERE DERIVED FROM EQUATIONS EXISTING IN THE LITERATURE FOR THE GROWTH AND DECAY OF FISSION PRODUCTS RESULTING FROM ONE REACTOR OPERATION AND SHUTDOWN CYCLE. THESE EQUATIONS WERE DERIVED FOR CONTINUOUS LOOP IRRADIATION AND FOR THE MORE REALISTIC LOOP IRRADIATION IN A REACTOR OPERATING ON A PERIODIC CYCLE.

DESIGN CRITERIA + IN PILE EXPERIMENT + REACTOR, LMCR + HAZARDS ANALYSIS

7-005008 DATED 1964
TAUBE M
PLUTONIUM
INSTITUTE OF NUCLEAR RESEARCH, WARSZAWA
258 PAGES, 108 FIGURES, 49 TABLES, 373 REFERENCES- DISTRIBUTED BY THE MACMILLAN COMPANY, NEW YORK, 1964

1. NUCLEAR PROPERTIES OF PLUTONIUM
2. CHEMICAL PROPERTIES
3. PHYSIOLOGICAL EFFECTS OF PLUTONIUM AND HEALTH PHYSICS
4. PLUTONIUM TECHNOLOGY
5. PLUTONIUM NUCLEAR FUEL
6. ROLE OF PLUTONIUM IN DEVELOPMENT OF NUCLEAR POWER

AEROSOL + ALLOY + REACTOR, BWR + REACTOR, BREEDER + CARBIDE + CONTAINMENT, HIGH PRESSURE + DECONTAMINATION + EBR 1 AND 2 (RE) + REACTOR, FAST + FISSION PRODUCT RELEASE + REACTOR, GCR + REACTOR, LMCR + PLUTONIUM + REACTOR, TEST + TRITIUM + VBWR (BWR) + HANFORD (PR) + PRESSURE DROP + PROPERTY, CHEMICAL + FILTERS

7-006379 DATED 1965
KUNKEL WP + SCHLESINGER G
FISSION PRODUCT RETENTION AND PLATING BEHAVIOR IN LIQUID SODIUM SYSTEMS
ATOMICS INTERNATIONAL
23 PAGES, 8 FIGURES, 3 TABLES, 3 REFERENCES- CONF-650407, INTERNATIONAL SYMPOSIUM ON FISSION PRODUCT RELEASE AND TRANSPORT UNDER ACCIDENT CONDITIONS, APRIL 5-7, 1965, OAK RIDGE, TENN.- CFSTI

IN MANY POSTULATED REACTOR INCIDENTS, THE COOLANT SUPPLIES AN ADDITIONAL BARRIER TO THE RELEASE OF RADIOACTIVITY. THE FISSION PRODUCT RETENTION PROJECT AT ATOMICS INTERNATIONAL HAS BEEN CONDUCTING STUDIES ON THE EXTENT OF FISSION PRODUCT (PARTICULARLY IODINE) RELEASE FROM SODIUM. RECENT EXPERIMENTS IN THIS PROJECT HAVE INCLUDED HIGH-TEMPERATURE STUDIES WITH SODIUM TEMPERATURES OF 1000 DEGREE F FOR IN-PILE EXPERIMENTS AND 900 TO 1400 DEGREE F FOR OUT-OF-PILE EXPERIMENTS. THE RESULTS OF THESE STUDIES HAVE CORROBORATED PREVIOUS DATA OBTAINED AT LOWER TEMPERATURES AND INDICATE THAT UNDER THE CONDITIONS STUDIED, LESS THAN 0.02 PERCENT (2×10^{-4}) OF THE IODINE RELEASED TO THE SODIUM MAY BE AVAILABLE FOR FURTHER RELEASE TO THE ENVIRONMENT.

DECONTAMINATION + FISSION PRODUCT RELEASE + FISSION PRODUCT, IODINE + REACTOR, LMCR + SODIUM + URANIUM CARBIDE + FISSION PRODUCT TRANSPORT + AI

7-006380 ALSO IN CATEGORY 8 DATED 1965
LAUBEN GN + KOONTZ RL
SODIUM FIRES AND THE RELEASE CHARACTERISTICS OF PARTICULATES AND FISSION PRODUCTS
ATOMICS INTERNATIONAL
CONF-650407, INTERNATIONAL SYMPOSIUM ON FISSION PRODUCT RELEASE AND TRANSPORT UNDER ACCIDENT CONDITIONS, APRIL 5-7, 1965, OAK RIDGE, TENN.- CFSTI

IN THE SAFETY ANALYSIS OF SODIUM-COOLED REACTORS, POTENTIALLY SERIOUS DISPERSIONS OF RADIOACTIVITY (NA 24 AND CERTAIN FISSION PRODUCTS) ARE POSTULATED TO OCCUR AS A RESULT OF A PRIMARY COOLANT SODIUM FIRE. ONE PROGRAM AT ATOMICS INTERNATIONAL IS BEING CONDUCTED TO CHARACTERIZE SODIUM FIRES BY DETERMINING THE INFLUENCE OF INITIAL CONDITION VARIABLES AND SYSTEM DESIGN PARAMETERS ON COMBUSTION ENERGY RELEASE AND DISTRIBUTION, AND THE RELEASE OF SODIUM (AS OXIDE OR METAL) AND

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-006380 *CONTINUED*
IUDINE.

FIRE + FISSION PRODUCT RELEASE + REACTOR, LMCR + PARTICLE SIZE + SODIUM + FISSION PRODUCT TRANSPORT

7-007280 DATED 1965
EITINGER HJ + MUSS WD + BUSEY HM
AEROSOL CHARACTERISTICS OF BURNING SODIUM AND PLUTONIUM
LOS ALAMOS SCIENTIFIC LABORATORY
1 PAGE, 1 TABLE- CONF-650602, PAGE 143 OF AMERICAN NUCLEAR SOCIETY ANNUAL MEETING TRANSACTIONS, GATLINBURG, TENNESSEE, JUNE 21-25, 1965

THE LOS ALAMOS FAST REACTOR CORE TEST FACILITY WILL BE USED TO STUDY VARIOUS MOLTEN-PLUTONIUM SODIUM-COOLED FAST-REACTOR CORES. ALL EQUIPMENT CONTAINING RADIOACTIVE SODIUM WILL BE OPERATED IN A NITROGEN ATMOSPHERE CONTAINING LESS THAN 2% OXYGEN. IN THE EVENT OF AN ACCIDENTAL FIRE, THE RESULTING INCREASED VOLUME OF GAS WILL BE VENTED TO THE STACK THROUGH A FILTRATION SYSTEM. THIS STUDY WAS MADE TO DETERMINE THE AEROSOL CHARACTERISTICS RESULTING FROM BURNING SODIUM AND PLUTONIUM ALLOYS IN NITROGEN CONTAINING 0 TO 10% OXYGEN.

AEROSOL + ALLOY + REACTOR, FAST + FIRE + REACTOR, LMCR + OXIDATION + PARTICLE SIZE + PLUTONIUM + SODIUM + AEROSOL PROPERTIES + FFTF (TR)

7-007346 DATED 1965
QUARTERLY STATUS REPORT ON ADVANCED REACTOR TECHNOLOGY (ART) FOR PERIOD ENDING JULY 31, 1965
LOS ALAMOS SCIENTIFIC LABORATORY, LOS ALAMOS, NEW MEXICO
62 PAGES, 13 FIGURES, 15 TABLES- AUGUST 1965, LA-3370-MS- CFSTI

I. PLUTONIUM REACTORS AND FUEL ELEMENTS FAST REACTOR CORE TEST FACILITY- II. SUPPORTING RESEARCH AND DEVELOPMENT PLUTONIUM RESEARCH- III. ULTRA HIGH TEMPERATURE REACTOR EXPERIMENT (UHTREX) CONSTRUCTION AND EQUIPMENT INSTALLATION- IV. MHD APPLICATIONS- V. PLASMA THERMOCOUPLE- VI. HEAT PIPES. IN THE FRICTF THE COVER GAS OVER THE PRIMARY SODIUM WOULD BE CONTAMINATED WITH RADIOACTIVE XENON AND KRYPTON. A PRIMARY CLEANUP SYSTEM PERMITTING RECIRCULATION OF THE COVER GAS THROUGH HEATED SILICON DIOXIDE IS PROPOSED. FUEL CONTAINMENT STUDIES ARE BEING MADE TO EXPERIMENTALLY COMPARE THE EFFECT OF SEVERAL VARIABLES ON FUEL CONTAINMENT. PLUTONIUM ALLOYS ARE BEING INVESTIGATED. THE FRICTF PRESSURE VESSEL DESIGN CRITERIA ARE DISCUSSED.

ALLOY + REACTOR, BREEDER + CARBIDE + DESIGN CRITERIA + REACTOR, FAST + FAILURE, FUEL ELEMENT + GRAPHITE + IN PILE EXPERIMENT + KRYPTON + REACTOR, LMCR + PLUTONIUM + PRESSURE VESSELS + SODIUM + URANIUM + XENON + FFTF (TR) + IN PILE EXPERIMENT + NITRIDE + MPBE (MOLTEN PLUTONIUM BURNUP EXPERIMENT) + LAMPRE 1 (RE) + FILTERS

7-007556 ALSO IN CATEGORIES 5 AND 12 DATED 1965
BIERY JC + KENNEDY RM
MATHEMATICAL AND EXPERIMENTAL ANALYSIS OF TEMPERATURE DISTRIBUTIONS IN STORED IRRADIATED LAMPRE CAPSULES
LOS ALAMOS SCIENTIFIC LABORATORY
18 PAGES, 14 FIGURES, 12 REFERENCES- CHEMICAL ENGINEERING PROGRESS SYMPOSIUM SERIES 61(57)- 322-339 (1965)- HEAT TRANSFER-BOSTON, AMERICAN INSTITUTE OF CHEMICAL ENGINEERS

TEMPERATURE PROFILES IN STORED IRRADIATED LAMPRE (LOS ALAMOS MOLTEN PLUTONIUM REACTOR EXPERIMENT) FUEL CAPSULES WERE SIMULATED BOTH MATHEMATICALLY AND EXPERIMENTALLY. THE MATHEMATICAL ANALYSIS PRODUCED SOLUTION EQUATIONS FROM THE FOUR COUPLED SECOND-ORDER DIFFERENTIAL EQUATIONS BY AN ITERATIVE GENERATION PROCEDURE. THE GENERATED EQUATIONS WERE THEN USED IN AN ITERATIVE COMPUTER PROGRAM WHICH PRODUCED THE CONVERGENT TEMPERATURE PROFILES BY A CALCULATION AND CORRECTION TECHNIQUE. THE ITERATIVELY GENERATED EQUATIONS, COUPLED WITH THE COMPUTER PROGRAM, PRODUCED TEMPERATURES WHICH AGREED VERY SATISFACTORILY WITH THE EXPERIMENTAL SIMULATION RESULTS.

REACTOR, BREEDER + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + FUEL STORAGE + HEAT TRANSFER, CONDUCTION + HEAT GENERATION, INTERNAL + FUEL ELEMENTS

7-008031 DATED 1964
TASK B - VENTED FUEL DEVELOPMENT
GENERAL ELECTRIC, SAN JOSE, CALIFORNIA
3 PAGES- JULY 1964, GEAP-4640, SECTION III, PAGES 3-1 TO 3-3 OF SODIUM-COOLED REACTORS PROGRAM FAST CERAMIC REACTOR DEVELOPMENT PROGRAM, ELEVENTH QUARTERLY REPORT APRIL-JUNE- CFSTI, \$3.00

THE FAST CERAMIC REACTOR DEVELOPMENT PROGRAM IS AN INTEGRATED ANALYTICAL AND EXPERIMENTAL PROGRAM DIRECTED TOWARD THE DEVELOPMENT OF FAST REACTORS EMPLOYING CERAMIC FUELS, WITH PARTICULAR ATTENTION TO MIXED PLUTONIUM-URANIUM OXIDE. ITS MAJOR OBJECTIVES ARE- A. DEVELOPMENT OF A RELIABLE, HIGH PERFORMANCE FAST REACTOR HAVING NUCLEAR CHARACTERISTICS WHICH PROVIDE STABLE AND SAFE OPERATION AND B. DEMONSTRATION OF LOW FUEL CYCLE COST CAPABILITY FOR SUCH A REACTOR, PRIMARILY THROUGH ACHIEVING HIGH BURNUP OF CERAMIC FUELS OPERATING AT HIGH SPECIFIC POWER.

CERAMICS + REACTOR, FAST + FISSION GAS RELEASE + FISSION PRODUCT RELEASE + REACTOR, LMCR + OXIDE + PLUTONIUM + SODIUM + FISSION PRODUCT TRANSPORT + FUEL ELEMENTS

7-008142 DATED 1965
RUSEN SL
ARGON GENERATION IN NAH-COOLED REACTORS

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-008142 *CONTINUED*
ATOMICS INTERNATIONAL
17 PAGES, 3 FIGURES, 3 REFERENCES- NOVEMBER 10, 1965, NAA-SR-11303- CFSTI, \$1.00 CY, \$0.50 MN

ARGON-41, A GAMMA EMITTER, WAS EXPERIMENTALLY OBSERVED IN THE PRIMARY COOLANT SYSTEM OF THE SNAP 8 EXPERIMENTAL REACTOR. A FAST NEUTRON REACTION WITH THE POTASSIUM-41 IN THE NAK-78 EUTECTIC ALLOY COOLANT IS BELIEVED TO BE THE MECHANISMS FOR THE GENERATION OF THE ARGON-41. THIS MECHANISM LED TO THE PREDICTION OF THE PRESENCE OF ARGON-39. THIS PREDICTION WAS VERIFIED EXPERIMENTALLY.

REACTOR, LMCR + SNAP 8 (SR) + ARGON + POTASSIUM + NAK + METAL, LIQUID

7-009711 ALSO IN CATEGORY 6 DATED 1965
DICKERMAN CE + ROBINSON LE + AUGUST C + STEWART R + STEPHANY W + WILLIS L + CARLANDER R + MUELLER M
PHYSICS DIVISION ANNUAL REPORT, JULY 1, 1964 TO JUNE 30, 1965
ARGONNE NATIONAL LABORATORY
ANL-7110 +. 9 PAGES, FIGURES, TABLES, REFERENCES, DECEMBER 1965, SECTIONS III-15 THROUGH III-19, PP. 157-165, CFSTI, \$7.22 CY, \$2.00 MN.

IN TREAT TRANSPARENT MELTDOWN CAPSULES, THE FOLLOWING SAMPLES WERE SUBJECTED TO TRANSIENTS FOR THE PURPOSES INDICATED - U-5 W/O FISSION ALLOY SAMPLES PREIRRADIATED TO 1.3 AT. PERCENT BURNUP, MEASUREMENT OF EXPANSION. EBR-II FUEL PIN PREIRRADIATED TO 0.6 AT. PERCENT, STUDY OF FAILURE MECHANISM. TH-U METALLIC FUEL ELEMENTS OF THE EBR-II MARK I TYPE, STUDY OF FAILURE MECHANISM (SEE ALSO TRANS. AMERICAN NUCLEAR SOCIETY, 8(1), P. 309 (1965). EBR-II TYPE US FUELED, W-RE ALLOY CLAD ELEMENTS, STUDY OF FAILURE, IF ANY (SEE ALSO NUCLEAR APP. 1, P. 168 (1965). IN THE INTEGRAL SODIUM LOOP EBR-II PINS WERE SUBJECTED TO 3 TRANSIENTS WITH SODIUM FLOW AND THREE TRANSIENTS WITHOUT FLOW. FUEL-CLADDING EUTECTIC WAS FORMED BY MOLTEN FUEL PENETRATING FROM THE CENTER OF THE ROD TO THE SURFACE.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FUEL BURNUP + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, GRAPHITE MODERATED + IN PILE EXPERIMENT + REACTOR, LMCR + REACTOR, SAFETY RESEARCH + REACTOR, RESEARCH + REACTOR, TEST + THORIUM + TREAT (PRR) + URANIUM + URANIUM SULFIDE + HIGH TEMPERATURE + IRRADIATION TESTING + REACTOR, PULSED + FAILURE, CLADDING + PRESSURE, INTERNAL + FUEL ELEMENTS

7-010297 DATED 1965
SMIDT D + SOMMER W
APPLICATION OF THE VENTED FUEL CONCEPT TO A SODIUM COOLED POWER BREEDER WITH 1000 MWE
KERNFORSCHUNGSZENTRUM, KARLSRUHE
KFK-399 +. 15 PAGES, 2 FIGURES, 4 TABLES, OCTOBER 1965

THE PRINCIPLE OF VENTING THE FISSION GAS OF A FAST, CERAMIC-FUELED REACTOR TO ITS COOLANT HAS A NUMBER OF ADVANTAGES - (1) NO STRESS OF THE FUEL CLADDING BY FISSION-GAS PRESSURE, (2) MORE COMPACT SUBASSEMBLY, SINCE NO FISSION-GAS PLENUM IS NECESSARY, (3) NO STORAGE OF HIGH-PRESSURE RADIOACTIVE GAS. THE VENTING OF FISSION GASES TO THE SODIUM COOLANT ALLOWS FOR SAVINGS IN THE ORDER OF 2 PERCENT OF THE TOTAL PLANT COST, IF COMPARED WITH THE REFERENCE REACTOR. IN COMPARISON WITH A MORE OPTIMIZED REACTOR, THE SAVINGS ARE MUCH SMALLER.

REACTOR, BREEDER + CERAMICS + CONTAINMENT + CLADDING + REACTOR, FAST + REACTOR, LMCR + RELEASE RATE + SODIUM + REACTOR COOLANT + SAFETY PRINCIPLES AND PHILOSOPHY + OFF GAS + FISSION PRODUCT RETENTION + ECONOMIC STUDY + FUEL ELEMENTS

7-010533 DATED 1965
SMIDT D + SOMMER W
APPLICATION OF THE VENTED FUEL CONCEPT TO A SODIUM-COOLED POWER BREEDER WITH 1000 MWE
KERNFORSCHUNGSZENTRUM, KARLSRUHE, GERMANY
ANL-7120 +. 6 PAGES, 2 FIGURES, 4 TABLES, PP. 130-135 OF PROCEEDINGS OF THE CONFERENCE ON SAFETY, FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS, OCTOBER 11-14, 1965, CFSTI, \$9.25 CY

THE PRINCIPLE OF VENTING THE FISSION GAS OF A FAST CERAMIC REACTOR TO ITS COOLANT HAS BEEN WELLKNOWN FOR SOME YEARS. THERE IS AN IMPRESSIVE NUMBER OF ADVANTAGES - (1) NO STRESS OF THE FUEL CLADDING CAUSED BY FISSION GAS PRESSURE, (2) MORE COMPACT SUBASSEMBLY, SINCE NO FISSION GAS PLENUM IS NECESSARY, (3) NO STORAGE OF HIGH-PRESSURE RADIOACTIVE GAS. THIS PAPER ANSWERS THE FOLLOWING QUESTIONS- (1) WHAT ARE THE PRINCIPAL DESIGN SPECIFICATIONS FOR THE GAS-PURIFICATION PLANT WITH RESPECT TO OPERATION AND SAFETY OF THE REACTOR, (2) WHAT ARE THE NUMERICAL RESULTS FOR A REFERENCE REACTOR, (3) WHAT IS THE ECONOMICAL BALANCE OF THE CITED ADVANTAGES AND DISADVANTAGES OF THE VENTED-TO-COOLANT CONCEPT.

REACTOR, BREEDER + ECONOMICS + CLADDING + REACTOR, FAST + FISSION GAS RELEASE + REACTOR, LMCR + SODIUM + THERMAL ANALYSIS + FISSION PRODUCT TRANSPORT

7-010552 DATED 1965
ONEILL GL + KNIGHT FW + LEITZ FJ + SHERER DB
SAFETY AND ECONOMIC ASPECTS OF VENTED FUEL IN SODIUM-COOLED REACTORS
GENERAL ELECTRIC COMPANY, SAN JOSE
ANL-7120 +. 10 PAGES, 5 FIGURES, 3 TABLES, 3 REFERENCES, PP. 540-549 OF PROCEEDINGS OF THE CONFERENCE ON SAFETY, FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS, OCTOBER 11-14, 1965, CFSTI, \$9.25 CY

THE RELEASE OF CONSIDERABLE FRACTIONS OF GENERATED FISSION GAS FROM HIGH-BURNUP, HIGH-POWER-DENSITY FCR FUEL IS WELL-KNOWN. THUS, THE ACCOMMODATION OF THESE FISSION GASES IS A MAJOR DESIGN PROBLEM FOR THE FCR FUEL AND SYSTEM. VARIOUS WAYS FOR ACCOMMODATING FISSION GASES IN FUEL PINS

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-010552 *CONTINUED*

ARE SHOWN. THE BROAD OBJECTIVE OF ALL THESE DESIGNS IS TO RELIEVE THE PRESSURE BUILDUP DUE TO FISSION GAS WITHOUT PERMITTING RELEASE OF DAUGHTER ACTIVITY OR ENTRY OF SODIUM INTO THE FUEL-BLANKET REGION. IF THE GAS IS RETAINED IN THE CORE REGION, PRESSURE BUILDUP INSIDE THE FUEL ELEMENT REQUIRES RESTRAINING CLADDING THICKNESSES, WHICH ARE GENERALLY PRECLUDED BY CONSIDERATIONS OF BOTH THERMAL STRESS AND CORE COMPOSITION. ALTERNATIVELY, VERY-LOW-DENSITY FUEL, WHICH ALLOWS SPACE FOR FISSION GAS, ALSO LEADS TO A POORER NEUTRON ECONOMY.

REACTOR, BREEDER + FISSION GAS RELEASE + FUEL BURNUP + REACTOR, LMCR + SODIUM + SAFETY PRINCIPLES AND PHILOSOPHY + ECONOMIC STUDY

7-010558

DATED 1965

KUNKEL W + BERGER S

FISSION PRODUCT RETENTION IN SODIUM AND APPLICATIONS TO VENTED FUEL ELEMENT DESIGN

ATOMICS INTERNATIONAL, CANOGA PARK

ANL-7120 +. 7 PAGES, 6 FIGURES, 7 TABLES, 6 REFERENCES, PP. 855-861 OF PROCEEDINGS OF THE CONFERENCE ON SAFETY, FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS, OCTOBER 11-14, 1965, CFSTI, \$9.25 CY

THE USE OF A VENTED FUEL ELEMENT REDUCES CLADDING STRAIN FROM RELEASED FISSION-GAS PRESSURE, THEREBY INCREASING THE POTENTIAL OF THE FUEL FOR EXTENDED BURNUP. ALTHOUGH THERE ARE SOME RADIOLOGICAL PROBLEMS IN MOST VENTED-FUEL-ELEMENT DESIGNS, THE SUCCESSFUL OPERATION OF THE DUONREAY FAST REACTOR INDICATED THAT THESE PROBLEMS CAN BE SOLVED. THIS PAPER INDICATES SOME OF THE SAFETY PROBLEMS ASSOCIATED WITH THE VENTED-FUEL-ELEMENT CONCEPT AND APPLIES THE RESULTS OF RECENT EXPERIMENTAL AND ANALYTICAL STUDIES TO THOSE PROBLEMS. THE DATA INDICATE THAT THERE WAS ONLY 0.036 TIMES AS MUCH I-131 AS KR-85. THEREFORE, THE IODINE MUST HAVE BEEN DEPLETED BY PASSAGE THROUGH THE SODIUM BOND, BY PLATING, OR BY SOME OTHER MECHANISM. THESE DATA, WHILE SPARSE, INDICATE AN APPARENT FIGURE FOR IODINE RELEASE FROM THE SODIUM OF 0.2 PERCENT. THESE RESULTS COMPARE WITH 0.02 PERCENT FOR LABORATORY CONDITIONS.

CERIUM + CESIUM + FISSION PRODUCT RELEASE + FISSION PRODUCT, IODINE + KRYPTON + REACTOR, LMCR + NOBLE GAS + RUTHENIUM + SODIUM + STRONTIUM + XENON + ZIRCONIUM + FISSION PRODUCT TRANSPORT + FISSION PRODUCT RETENTION + FUEL ELEMENTS

7-010559

DATED 1965

PURGES KG

FUEL-FAILURE DETECTION IN SODIUM-COOLED REACTORS

ARGONNE NATIONAL LABORATORY, ARGONNE

ANL-7120 +. 6 PAGES, 1 FIGURE, 9 REFERENCES, PP. 862-867 OF PROCEEDINGS OF THE CONFERENCE ON SAFETY, FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS, OCTOBER 11-14, 1965, CFSTI, \$9.25 CY

THE ADVENT OF SODIUM COOLING PLTS AN ENTIRELY DIFFERENT EMPHASIS ON THE RATIONALE OF CLADDING-FAILURE MONITORING. SODIUM DOES NOT ATTACK THE FUEL, NOR IS THE INCREASE OF THE COOLANT ACTIVITY WHICH WOULD RESULT FROM FISSION PRODUCT CONTAMINATION (OVER AND ABOVE THE NA-24 ACTIVITY, POSSIBLY UP TO 10 MC/CM³, ALREADY PRESENT) CAUSE FOR CONCERN. THE EBR-II CLADDING-FAILURE MONITOR CONSISTS OF A FAST DELAYED-NEUTRON DETECTOR, FERDL (FUEL-ELEMENT-RUPTURE-DETECTION LOOP), AS WELL AS A WIRE PRECIPITATOR PRESENTLY MONITORING THE GAS BLANKET. THE FERDL IS INSTALLED ON A LOOP THAT SAMPLES THE COOLANT AT THE HEAT EXCHANGER, WITH A MAXIMUM LOOP FLOW OF 100 GPM OUT OF A TOTAL OF ABOUT 8000 GPM.

FISSION PRODUCT RELEASE + FAILURE, FUEL ELEMENT + FUEL INTEGRITY + INSTRUMENT, NUCLEAR + REACTOR, LMCR + SODIUM + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + INSTRUMENT, SURVEILLANCE + MONITOR

7-011795

ALSO IN CATEGORY 17

DATED 1966

FERMI FUEL ELEMENT FAILURE, OCTOBER 4, 1966

POWER REACTOR DEVELOPMENT COMPANY

1 PAGE, NUCLEONICS WEEK, 7(41), (OCTOBER 13, 1966)

AT 34 MW(TH) DURING A STARTUP, THE CONTROL RODS SEEMED TO BE WITHDRAWN FARTHER THAN NORMAL. TWO CORE-OUTLET TEMPERATURES WERE HIGHER THAN NORMAL. COVER-GAS ACTIVITY INCREASED, THE REACTOR-BUILDING RADIATION MONITORS GAVE AN ISOLATION SIGNAL, AND THE REACTOR WAS SCRAMMED. AS OF OCT. 10, 1966, 22 CENTS REACTIVITY HAD BEEN LOST, AND THE COVER-GAS XENON WAS ABOUT EQUAL TO THAT EXPECTED FROM A CORE SUBASSEMBLY. REACTIVITY BEGAN TO BE LOST AT ABOUT 15 MW(TH).

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + INCIDENT

7-011945

ALSO IN CATEGORY 17

DATED 1966

MCCARTHY WJ

FERMI FUEL DAMAGE PRELIMINARY REPORT, OCTOBER 10, 1966

POWER REACTOR DEVELOPMENT COMPANY, DETROIT, MICHIGAN

2 PAGES, ATOMIC ENERGY CLEARING HOUSE 12(42), PAGES 9-10 (OCTOBER 17, 1966) DOCKET NO. 50-16

BRIEF REVIEW OF AN INCIDENT ON OCTOBER 5, 1966. ON RAISING THE POWER LEVEL TO 34 MW(TH), A REGULATING ROD SEEMED FARTHER OUT THAN NORMAL, ALSO TWO FUEL-OUTLET TEMPERATURES WERE ABOVE NORMAL. COVER-GAS ACTIVITY HAD INCREASED. SOME FUEL DAMAGE AND REDISTRIBUTION RESULTED, AND INVESTIGATIONS ARE CONTINUING.

REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + REACTOR, LMCR

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-013541
PEELGREN ML + MCDONALD JS
INVESTIGATION OF SODIUM LEAK PLUGGING
ATOMIC INTERNATIONAL
NAA-SK-MEMO-5798 +. 16 PAGES, 4 FIGURES, 1 TABLE, 11 REFERENCES, OCTOBER 1960

DATED 1960

A PROGRAM IS REPORTED TO INVESTIGATE A METHOD OF STOPPING SODIUM LEAKS AT 1200 F BY REACTING THE LEAKING MATERIAL WITH A GAS SUCH AS CARBON DIOXIDE OR OXYGEN. SUCH A PLUGGING MECHANISM WOULD BE USEFUL IN PREVENTING SHUTDOWNS DUE TO SODIUM LEAKS SUCH AS THOSE IN THE SHELL OF A CALANDRIA CORE. RESULTS INDICATE LITTLE LIKELIHOOD OF SUCCESS FOR THIS MECHANISM ABOVE 600F.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.30 COPY, \$2.40 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + OXIDATION + CONTAINMENT ISOLATION + SODIUM + CARBON DIOXIDE + CONTAINMENT LEAKAGE CONTROL

7-013548
LAUBEN GN
INITIAL EXPERIENCE WITH LARGE SODIUM FIRES EXPERIMENTS (LF-1)
ATOMIC INTERNATIONAL
NAA-SR-12041 +. 41 PAGES, 14 FIGURES, 3 TABLES, 15 REFERENCES, AUGUST 1, 1966

DATED 1966

IN THE SAFEGUARDS ANALYSIS OF SODIUM-COOLED REACTORS, A SERIOUS DISPERSION OF RADIOACTIVITY IS POSTULATED TO RESULT FROM A LARGE PRIMARY-COOLANT (SODIUM) FIRE IN A GALLERY OR VAULT. TO EVALUATE AND/OR MITIGATE THIS DISPERSION, IT IS NECESSARY TO DETERMINE THE EFFECTS OF TIME, INITIAL-CONDITION VARIABLES, AND SYSTEM PARAMETERS ON THE SPATIAL DISTRIBUTION OF ENERGY, THE AMOUNT OF SODIUM RELEASED (PRESUMABLY AS SODIUM OXIDE), AND THE AMOUNT OF SELECTED FISSION PRODUCTS RELEASED. EARLY SODIUM-FIRE INVESTIGATIONS GAVE WIDELY VARYING RESULTS OF BURNING AND RELEASE RATES WHICH SEEMED TO BE VERY SENSITIVE TO GEOMETRY AND AIR FLOW. FOR EXAMPLE, IT HAS BEEN REPORTED THAT SHALLOW POOLS BURN AS NODULES ON AN IRREGULAR OXIDE SURFACE, WHEREAS DEEPER, WELL-INSULATED POOLS BURNED ON THE MOLTEN METAL SURFACE. REPORTED HEREIN ARE THE RESULTS OF THE CURRENT INVESTIGATION, INCLUDING RESULTS OF A FIRE, DESIGNATED AS LARGE FIRE NO. 1, IN WHICH THE GEOMETRIC AND CONVECTIVE CONDITIONS WHICH MIGHT BE FOUND IN A REACTOR GALLERY OR VAULT WERE SIMULATED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$2.00 COPY, \$0.50 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FIRE + FISSION PRODUCT, IODINE + IODINE + REACTOR, LMCR + SODIUM + FISSION PRODUCT TRANSPORT

7-013751
CHASE WL + HUEBOTTER PK + KOVACIC EC + MACLEOD RA + MOORE CR + SHOUDY AA
CARBON IN THE PRIMARY SODIUM SYSTEM OF THE ENRICO FERMI ATOMIC POWER PLANT
ATOMIC POWER DEVELOPMENT ASSOCIATES INC., DETROIT, MICHIGAN
TID-25101 +. 40 PAGES, AUGUST 1, 1963

DATED 1963

WHEN APDA CONDUCTED THE 1000 F TESTS ON THE PRIMARY SYSTEM OF THE ENRICO FERMI ATOMIC POWER PLANT, CARBONACEOUS GASES AND VAPORS WERE EVOLVED FROM THE GRAPHITE IN THE REACTOR PLUG. THE PURPOSE OF THIS REPORT IS TO SUMMARIZE THE EFFORTS EXPENDED TO DETERMINE THE AMOUNT OF CARBON EVOLVED, THE METHODS SUITABLE FOR REMOVING THE CARBON FROM THE SODIUM, THE POSSIBLE PROBLEMS THAT COULD BE CAUSED BY THE CARBON, AND METHODS OF OPERATIONAL CONTROL OF THE CARBON. BASED UPON THE STUDY REPORTED IN THIS SUMMARY REPORT, THERE SHOULD BE NO FURTHER OUTGASSING FROM THE PLUG DUE TO TEMPERATURE. SOME GAS MAY BE EVOLVED EACH YEAR DUE TO RADIATION. HOWEVER, THESE GASES SHOULD BE MOSTLY HYDROGEN. ANY CARBONACEOUS GASES THAT SHOULD BE EVOLVED CAN BE READILY DISSOLVED INTO THE SODIUM AND REMOVED IN THE CARBON REMOVAL SYSTEM, THEREBY FORMING NO CRUST.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$2.00 COPY, \$0.50 MICRONEGATIVE

REACTOR, BREEDER + CARBON + REACTOR, FAST + FERMI (LMFBR) + GRAPHITE + HYDROGEN + REACTOR, LMCR + SODIUM + REACTOR STARTUP EXPERIENCE + OFF GAS

7-013907
HART RS + NELSON CT
FISSION PRODUCT RETENTION CHARACTERISTICS OF SODIUM AT HIGH TEMPERATURES
ATOMIC INTERNATIONAL
NAA-SR-MEMO-0712 +. 19 PAGES, 5 FIGURES, 5 TABLES, JUNE 1, 1964

DATED 1964

DOCUMENTS THE RESULTS OF EXPERIMENTS TO DETERMINE THE RETENTIVENESS OF SODIUM FOR CS-137, I-131, AND SR-89 AT TEMPERATURES BETWEEN 900 AND 1500 F. A BRIEF DESCRIPTION OF THE EXPERIMENTAL APPARATUS, EXPERIMENTAL AND ANALYTICAL TECHNIQUES IS ALSO GIVEN. FOR THE SPECIFIC CONDITIONS OF THE EXPERIMENTS, THE RESULTS SHOW THAT THE MAXIMUM MEASURED RELEASE FRACTIONS FROM THE SODIUM TO THE COVER GAS ARE INDEPENDENT OF THE SODIUM TEMPERATURE AND LESS THAN $4 \times 10^{-(\text{MINUS } 4\text{TH})}$ FOR CS-137, $5 \times 10^{-(\text{MINUS } 6\text{TH})}$ FOR I-131, AND $4 \times 10^{-(\text{MINUS } 7\text{TH})}$ FOR SR-89.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$1.60 COPY, \$0.80 MICROFICHE

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-013907 *CONTINUED*
CESIUM + FISSION PRODUCT RELEASE + FISSION PRODUCT, IODINE + IODINE + REACTOR, LMCR + SODIUM + STRONTIUM +
FISSION PRODUCT TRANSPORT + ANALYTICAL TECHNIQUE + RADIOISOTOPE + FISSION PRODUCT RETENTION + CALIBRATION

7-014384 DATED 1966
POLLOCK BD + KUNKEL WP + MURBACH EW
NUCLEAR SAFETY, CHEMICAL REACTIONS. FISSION PRODUCT AND CONTAMINATION CONTROL
ATOMICS INTERNATIONAL
NAA-SR-12175 +. 8 PAGES, 5 FIGURES, 6 REFERENCES, QUARTERLY TECHNICAL PROGRESS REPORT AEC UNCLASSIFIED
PROGRAMS, JULY-SEPTEMBER 1966, PAGES 153-160

THE OBJECTIVE OF THIS PROGRAM IS TO ELUCIDATE THE BEHAVIOR OF FISSION PRODUCTS RELEASED TO THE
COOLANT IN A FAST SODIUM-COOLED LMFR REACTOR DURING NORMAL OPERATION, IN ORDER TO PROVIDE
INFORMATION NECESSARY FOR THE DEVELOPMENT OF FISSION PRODUCT TRAPPING TECHNIQUES IN SUCH SYSTEMS.
THE DISPOSITION OF FISSION PRODUCTS DURING NORMAL OPERATION MUST BE ALSO KNOWN IN ORDER TO
ASSESS THE CONSEQUENCES OF POTENTIAL ACCIDENTS. INFORMATION IS REQUIRED ON THE EXTENT OF FISSION
PRODUCT RETENTION IN SODIUM COOLANT, THE RATE AND EXTENT OF RELEASE TO THE COVER GAS, AND THE
RATE AND EXTENT OF PLATE-OUT ON SURFACES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICRONEGATIVE

CESIUM + IODINE + KRYPTON + REACTOR, LMCR + NOBLE GAS + SODIUM + TRACER, RADIOACTIVE + TRITIUM + XENON +
FISSION PRODUCT TRANSPORT + OUT OF PILE EXPERIMENT + FILTER, TRAP + METAL, LIQUID + GAMMA EMITTER

7-014385 ALSO IN CATEGORY 8 DATED 1966
KOONTZ RL + LAUBEN GN
NUCLEAR SAFETY, GASEOUS EFFLUENT STUDIES. CHARACTERIZATION OF SODIUM FIRES AND FISSION PRODUCT RELEASE
ATOMICS INTERNATIONAL
NAA-SR-12175 +. 14 PAGES, 3 TABLES, 7 FIGURES, QUARTERLY TECHNICAL PROGRESS REPORT AEC UNCLASSIFIED PROGRAMS,
JULY-SEPTEMBER 1966, PAGES 161-174

THE GENERAL OBJECTIVE OF THIS PROJECT IS TO DEVELOP EXPERIMENTAL INFORMATION AND ANALYTICAL
METHODS WHICH CHARACTERIZE THE RELEASE AND TRANSPORT OF EFFLUENTS AND ENERGY GENERATED DURING A
PRIMARY-COOLANT (SODIUM) ACCIDENT. THE SOURCE OF ENERGY GENERATION MAY BE FROM THE ESCAPING
COOLANT (BY RAPID THERMAL ENERGY TRANSFER) AND/OR SUBSEQUENT COMBUSTION OF THE SODIUM. THE
EFFLUENTS ARE SODIUM (AS SODIUM-24) OR ITS OXIDE IN PARTICULATE FORM AND SELECTED FISSION
PRODUCTS AVAILABLE AS POTENTIALLY SERIOUS DISPERSIONS OF RADIOACTIVITY. THE INFORMATION TO BE
DEVELOPED IS REQUIRED FOR THE DESIGN AND SAFEGUARDS ANALYSIS OF ECONOMICAL, SODIUM-COOLED FAST
REACTORS. A MAJOR EFFORT WILL BE DEVOTED TO EXPERIMENTS ON THE CHARACTERIZATION OF I-131 RELEASE
FROM SODIUM FIRES. THESE TESTS WILL BE CONDUCTED IN THE LABORATORY TEST CHAMBER AND IN THE
PARTICLE GENERATOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICRONEGATIVE

AEROSOL + FIRE + REACTOR, LMCR + PARTICLE SIZE + SODIUM + ANALYTICAL MODEL + FISSION PRODUCT TRANSPORT + OUT
OF PILE EXPERIMENT + SMOKE + METAL, LIQUID

7-014870 ALSO IN CATEGORY 3 DATED 1966
SCHLAPP FW
STORAGE OF MODERATOR ELEMENTS IN HNPf MODERATOR STORAGE CELLS
ATOMICS INTERNATIONAL
NAA-SR-MEMO-12247 +. 7 PAGES, NOVEMBER 28, 1966

GROUNDWATER WAS ASSUMED TO CONTACT AND START DISSOLVING THE ACTIVATED STAINLESS STEEL PORTIONS OF
THE THREE IRRADIATED MODERATOR ELEMENTS NOW STORED IN THE HNPf MODERATOR-STORAGE CELLS. THE
SOLUBILITY AND DECOMPOSITION RATES OF THE METAL AND THE CONCENTRATION OF RADIOACTIVITY RELEASABLE
TO THE GROUNDWATER WERE ANALYZED. THE RESULTS SHOWED THAT INSIGNIFICANT CONCENTRATIONS OF
RADIOACTIVE MATERIALS CAN BE RELEASED, EVEN IF THE CONCRETE CELL STRUCTURE AND THE STEEL LINERS
ARE NEGLECTED IN CALCULATING THE DISSOLUTION AND DISPERSAL OF THE MATERIAL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICRONEGATIVE

FISSION PRODUCT RELEASE + FUEL INTEGRITY + REACTOR, GRAPHITE MODERATED + GROUND WATER + HALLAM (LMR) +
REACTOR, LMCR + RADIOACTIVITY RELEASE

7-015182 DATED 1966
NORMAN EC
SUMMARIES OF FAST REACTOR FUELS AND MATERIALS DEVELOPMENT PROGRAMS
DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY (AEC), WASHINGTON, D. C.
TID-6506 (PT. 3) (4TH ED.) +. 259 PAGES, TABLES, MAY 1966

THIS DOCUMENT COMPRISES THE THIRD OF THREE PARTS OF THE REPORT TID-6506 (4TH EDITION). THIS PART
DESCRIBES THE FUELS AND MATERIALS RESEARCH AND DEVELOPMENT PROGRAMS (UNCLASSIFIED PORTION)
UNDERTAKEN IN SUPPORT OF THE LIQUID METAL FAST BREEDER REACTOR PROGRAM (LMFR) THAT ARE SPONSORED
BY THE DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY. THE OTHER PARTS OF TID-6506 (4TH EDITION)
ARE - (PART ONE) RESEARCH ON FUELS AND MATERIALS DIRECTED BY THE FUELS AND MATERIALS BRANCH OF

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-015182 *CONTINUED*

THE DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY. (PART TWO) CLASSIFIED RESEARCH ON FUELS AND MATERIALS UNDER DTI CATEGORY NO. C-44A, NUCLEAR TECHNOLOGY MATERIAL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

ALLOY + REACTOR, BREEDER + CARBIDE + REACTOR, FAST + REACTOR, LMCR + OXIDE + PLUTONIUM + URANIUM + CORROSION + NITRIDE + METAL, LIQUID + PLUTONIUM DIOXIDE

7-015212

DATED 1963

CESONI G + CHINAGLIA B + WALTHER H
PROBLEMS OF FISSION PRODUCT DIFFUSION IN GRAPHITE FUEL ELEMENTS
SOCIETA Ricerche Impianti Nucleari, SLUGGIA, ITALY
N-65-23015 + SORIN-15 +. 25 PAGES, 1963

STUDIES ARE BEING PERFORMED IN WHICH DIFFUSION PROBLEMS OF FISSION PRODUCTS IN GRAPHITE ARE CONSIDERED ON A FUNDAMENTAL BASIS, IN VIEW OF THE INFLUENCE WHICH A BETTER KNOWLEDGE OF SOME BASIC ASPECTS OF SUCH DIFFUSION MAY HAVE ON POSSIBLE PRACTICAL SOLUTIONS OF TECHNOLOGICAL INTEREST.

COATED PARTICLE + DIFFUSION + GRAPHITE + REACTOR, LMCR + REACTOR, ORGANIC COOLED + PYROLYTIC + DIFFUSION COEFFICIENT + FISSION PRODUCT RETENTION + THEORETICAL INVESTIGATION + FUEL ELEMENTS

7-015538

DATED 1957

CARTER RL
THE PROBLEM OF CORE GRAPHITE CONTAMINATION BY SODIUM IN A CALANDRIA TYPE SGR
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SK-MEMO-3697 +. 24 PAGES, FIGURES, REFERENCES, 1957

A GROWING BODY OF RELIABLE DATA ON SODIUM-GRAPHITE REACTIONS HAS DEFINED MORE CLEARLY THE CONSEQUENCES OF ACCIDENTAL CONTACT BETWEEN THE MEDIA. IN VIEW OF THE POSSIBILITY OF A FOUR-FOLD INCREASE IN THE SIZE OF THE PROJECTED SGR IT SEEMS IMPERATIVE THAT THE IMPLICATIONS OF THE CURRENT SCIENTIFIC DATA BE PRESENTED TO PERSONNEL SETTING POLICY FOR DESIGN OF THIS REACTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

GRAPHITE + HALLAM (LMR) + REACTOR, LMCR + SODIUM + CONTAMINATION + MODERATOR + METAL, LIQUID + MAIN COOLING SYSTEM

7-015934

ALSO IN CATEGORIES 8 AND 12

DATED 1966

SAVAGE HW
HISTORY OF THE AEC SODIUM COMPONENTS DEVELOPMENT PROGRAM. VOLUME I
OAK RIDGE NATIONAL LABORATORY
ORNL-3955 (VOL. 1) +. 401 PAGES, DECEMBER 1966

THIS HISTORY WILL BE OF GREAT INTEREST TO THOSE WHO HAVE HAD A PART IN THE DEVELOPMENT OF LIQUID-METAL COMPONENTS TECHNOLOGY DURING THE LAST EIGHT TO NINE YEARS, AND WE SINCERELY HOPE THAT IT WILL SERVE AS AN ACCURATE REFLECTION OF THE CONTRIBUTION OF THE SODIUM COMPONENTS DEVELOPMENT PROGRAM WORKING GROUP IS MAKING TO THAT DEVELOPING TECHNOLOGY. THIS VOLUME IS A SUMMARY OF DEVELOPMENT PROGRAMS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + FERMI (LMFBR) + REACTOR, LMCR + METAL WATER REACTION + SODIUM + TESTING + OPERATING EXPERIENCE SUMMARY + EQUIPMENT DESIGN + MAIN COOLING SYSTEM + PUMPS + INSTRUMENTS, MISC. + HEAT EXCHANGERS

7-015935

ALSO IN CATEGORIES 8 AND 12

DATED 1966

SAVAGE HW
HISTORY OF THE AEC SODIUM COMPONENTS DEVELOPMENT PROGRAM. VOLUME II
OAK RIDGE NATIONAL LABORATORY
ORNL-3955 (VOL. 2) +. 568 PAGES, DECEMBER 1966

THIS HISTORY WILL BE OF GREAT INTEREST TO THOSE WHO HAVE HAD A PART IN THE DEVELOPMENT OF LIQUID-METAL COMPONENTS TECHNOLOGY DURING THE LAST EIGHT TO NINE YEARS, AND WE SINCERELY HOPE THAT IT WILL SERVE AS AN ACCURATE REFLECTION OF THE CONTRIBUTION OF THE SODIUM COMPONENTS DEVELOPMENT PROGRAM WORKING GROUP IS MAKING TO THAT DEVELOPING TECHNOLOGY. THIS VOLUME CONSISTS OF REPORTS OF SUBCOMMITTEE MEETINGS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, LMCR + METAL WATER REACTION + SODIUM + TESTING + OPERATING EXPERIENCE SUMMARY + EQUIPMENT DESIGN + MAIN COOLING SYSTEM + PUMPS + INSTRUMENTS, MISC. + HEAT EXCHANGERS

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-015936 ALSO IN CATEGORIES 8 AND 12 DATED 1966
SAVAGE HW
HISTORY OF THE AEC SODIUM COMPONENTS DEVELOPMENT PROGRAM. VOLUME III
OAK RIDGE NATIONAL LABORATORY
ORNL-3955 (VOL. 3) +. 484 PAGES, TABLES, DECEMBER 1966

THIS HISTORY WILL BE OF GREAT INTEREST TO THOSE WHO HAVE HAD A PART IN DEVELOPING LIQUID-METAL COMPONENTS TECHNOLOGY DURING THE LAST EIGHT TO NINE YEARS, AND WE SINCERELY HOPE THAT IT WILL SERVE AS AN ACCURATE REFLECTION OF THE CONTRIBUTION OF THE SODIUM COMPONENTS DEVELOPMENT PROGRAM WORKING GROUP IS MAKING TO THAT DEVELOPING TECHNOLOGY. THIS VOLUME CONSISTS OF QUESTIONNAIRES SENT TO MEMBERS AND ALTERNATES, AND REPLIES. ALSO INCLUDED ARE THE MONTHLY COORDINATION REPORTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, LMCR + METAL WATER REACTION + SODIUM + TESTING + OPERATING EXPERIENCE SUMMARY + EQUIPMENT DESIGN + MAIN COOLING SYSTEM + PUMPS + INSTRUMENTS, MISC. + HEAT EXCHANGERS

7-018195 ALSO IN CATEGORIES 6 AND 8 DATED 1967
DICKERMAN CE + ROBINSON LE + SOWA ES + MONAWECK JH + AUGUST C
TREAT MELTDOWN OF IRRADIATED METALLIC FAST-REACTOR FUEL IN AN INERT ATMOSPHERE
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
15 PAGES, 10 FIGURES, 3 TABLES, 29 REFERENCES, NUCLEAR APPLICATIONS 3(8)- 461-473 (AUG. 1967)

BEHAVIOR OF TWO TYPES OF IRRADIATED METALLIC FAST-REACTOR FUEL PINS UNDER TRANSIENT HEATING IN AN INERT-GAS ATMOSPHERE IN THE TREAT REACTOR WAS STUDIED UNDER CONDITIONS RANGING UP TO COMPLETE MELTDOWN. IRRADIATION-INDUCED CHANGES IN FAILURE THRESHOLDS, MODES OF FAILURE, AND POST-FAILURE FUEL MOTION WERE SIGNIFICANT BUT NOT ENOUGH TO MAKE GROSS CHANGES IN THE COURSE OF AN ACCIDENT. THE AMOUNT OF SWELLING OBSERVED IN THE POSTEXPERIMENTAL FUEL RESIDUE WAS LARGE ENOUGH TO HAVE PLAYED A SIGNIFICANT ROLE IN DETERMINING THE COURSE OF A MELTDOWN ACCIDENT.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + SODIUM + TESTING + TREAT (PKR) + VOID COEFFICIENT + FAILURE, CLADDING

7-018291 DATED 1967
CLIFFORD JC
BEHAVIOR OF LONG-LIVED FISSION PRODUCTS IN SODIUM
LUS ALAMUS SCIENTIFIC LABORATORY
CONF-670602 +. 2 PAGES, ANS TRANS. 10(11), PAGE 336, 1967 ANNUAL MEETING, AMERICAN NUCLEAR SOCIETY, SAN DIEGO, CALIFORNIA, JUNE 11-15, 1967

THE BEHAVIOR OF FISSION PRODUCTS RELEASED TO SODIUM COOLANT FROM FAILED OR DELIBERATELY VENTED REACTOR FUEL ELEMENTS MAY LIMIT ACCESS TO THE PRIMARY COOLANT SYSTEM AND AFFECT THE CONSEQUENCES OF A LOSS-OF-COOLANT INCIDENT. DEPENDING ON THE FISSION-PRODUCT INVENTORY ANTICIPATED IN THE COOLANT, IT MAY BE DESIRABLE TO CONCENTRATE URANIUM, PLUTONIUM, LONG-LIVED ENERGETIC GAMMA-EMITTING ISOTOPES, AND SHORT-LIVED BIOLOGICALLY HAZARDOUS ISOTOPES AT SPECIFIC LOCATIONS WITHIN THE PRIMARY SYSTEM. LONG-LIVED FISSION PRODUCTS ARE ADDED TO A SMALL, FORCED-CONVECTION SODIUM LOOP BY IMMERGING IRRADIATED PLUTONIUM-BASE FUEL IN THE SODIUM. THE BEHAVIOR OF CS-137 RELEASED FROM A METALLIC FUEL TO HOT-TRAPPED SODIUM SUGGESTS THAT CESIUM IS A MOBILE VOLATILE SPECIES WHICH IS WEAKLY ADSORBED ON STAINLESS-STEEL SURFACES OPERATING IN THE 300 TO 600 C RANGE.

CESIUM + REACTOR, FAST + REACTOR, LMCR + RUTHENIUM + SODIUM + STRONTIUM + FISSION PRODUCT TRANSPORT + COOLANT CHEMISTRY + METAL, LIQUID + FISSION PRODUCT RETENTION + EUROPIUM

7-018343 DATED 1959
HINZE RB
OBSERVATIONS OF CARBON IN SRE SODIUM
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA
NAA-SR-MEMO-4479 +. 32 PAGES, TABLES, OCTOBER 12, 1959

FINE CARBON PARTICLES RESULTING FROM THE DECOMPOSITION OF HYDROCARBONS IN THE SODIUM DEPOSITED ON THE WALLS OF COLD (440 F) SODIUM PIPING IN THE SRE SECONDARY SYSTEM, WHERE THEY DECREASED THE ABILITY OF THE SODIUM TO WET THE PIPE WALL. THIS CHANGE IN WETTING EXPLAINS THE DECREASE IN FLOW INDICATION OBSERVED ON EM FLOWMETERS AFTER THE INTRODUCTION OF TETRALIN TO A SYSTEM. WHILE THE CARBON PARTICLES DEPOSIT ON 440 F PIPING, THERE IS NO EVIDENCE THAT CARBON MIGRATES TO COOLER SECTION OF THE SYSTEM, E.G., TO A COLD TRAP. THE TEMPERATURE DEPENDENCE OF SOLUBILITY APPEARS TO BE TOO SMALL FOR EFFECTIVE COLLECTION OF CARBON IN THE COLD TRAP. THE MESH, OF COURSE, WILL ACT AS A STRAINER FOR LARGE PARTICLES. ONLY VERY FINE CARBON PARTICLES HAVE BEEN OBSERVED IN THE SECONDARY SYSTEM, WHILE COARSE PARTICLES (UP TO 3/8 IN. MAXIMUM DIMENSION) HAVE BEEN FOUND IN THE PRIMARY SYSTEM. NO ADEQUATE EXPLANATION HAS BEEN PROPOSED FOR THIS DIFFERENCE. SYSTEM DIFFERENCES THAT MAY BE SIGNIFICANT ARE SODIUM TEMPERATURE AT PUMP, THE POINT OF TETRALIN INLEAKAGE, RADIATION, PIPING ARRANGEMENT, AND OTHER IMPURITY LEVELS SUCH AS OXYGEN AND HYDROGEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

CARBON + DECONTAMINATION + REACTOR, FAST + REACTOR, LMCR + SODIUM + SRE (RE) + CONTAMINATION + PIPES AND PIPE FITTINGS

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-018947 ALSO IN CATEGORY 17 DATED 1967
MUELLER KE + BEAUDRY RJ + FELDES JG
FISSION PRODUCT ACTIVITY ANALYSIS OF FERMI FUEL FAILURE
ATOMIC POWER DEVELOPMENT ASSOCIATED, INC.
CONF-670 602 +. 2 PAGES, 1 FIGURE, AND TRANSACTIONS 1011 PAGES 334-335, 1967 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, SAN DIEGO, CALIF., JUNE 11-15, 1967

DESCRIBES THE MEASUREMENTS, ANALYSIS TECHNIQUE, AND CONCLUSIONS FOR DETERMINING EXTENT OF FUEL FAILURE. MEASUREMENTS OF THE FISSION PRODUCT ACTIVITY IN THE COOLANT AND COVER GAS INDICATED 2% OF THE CORE MELTED, AND AGREED WITH THE REACTIVITY LOSS. OTHER OBSERVATIONS WERE - (1) NOBLE-GAS RELEASE WAS GREATER THAN THAT IN PREVIOUS EXPERIMENTS, 50% VS 10%, (2) PARTICULATE RELEASE WAS GREATER THAN EXPECTED, (3) CONSIDERABLE PLATEOUT OCCURRED, (4) SODIUM ANALYSIS IS LESS RELIABLE THAN COVER-GAS ANALYSIS.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FISSION PRODUCT RELEASE + FAILURE, FUEL ELEMENT + REACTOR, LMCR + ANALYTICAL TECHNIQUE + FISSION PRODUCT ACTIVITY, GROSS + INCIDENT + RADIOCHEMICAL ANALYSIS + MAIN COOLING SYSTEM

7-019555 ALSO IN CATEGORY 17 DATED 1962
HART RS
DISTRIBUTION OF FISSION PRODUCT CONTAMINATION IN THE SRE
ATOMICS INTERNATIONAL, A DIVISION OF NORTH AMERICAN AVIATION, INC., CANOGA PARK, CALIFORNIA
NAA-SR-6890 +. 27 PAGES, 4 FIGURES, 12 TABLES, 5 REFERENCES, MARCH 1, 1962

SUMMARIZES FISSION-PRODUCT RELEASE AND DISTRIBUTION DATA COLLECTED WHILE MODIFYING AND CLEANING THE REACTOR AFTER THE FUEL-MELTING INCIDENT IN JULY 1959 (SEE NAA-SR-4488). THE PRINCIPAL CONCLUSIONS DRAWN FROM THE DATA ARE - (1) VOLATILITY PREFERENCE IS INVOLVED IN THE RELEASE OF VARIOUS FISSION PRODUCTS. (2) ONLY XE AND KR WERE FOUND IN THE HELIUM COVER GAS. (3) THE IODINE RELEASE FRACTION WAS UNUSUALLY SMALL. (4) APPRECIABLE DEPOSITION OCCURRED THROUGHOUT THE PIPING SYSTEM, WITH MARKED SELECTIVITY. SR, CE, ZR-NB DEPOSITION WAS GREATER THAN CS OR I. (5) ALTHOUGH 5000 TO 10,000 Curies WERE RELEASED, NO RADIOLOGICAL EMERGENCY OCCURRED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

FISSION GAS RELEASE + FISSION PRODUCT RELEASE + FISSION PRODUCT, IODINE + FISSION PRODUCT, VOLATILE + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + SURFACE CONTAMINATION + SRE (RE) + FISSION PRODUCT ACTIVITY, GROSS + INCIDENT + FAILURE, CLADDING + FISSION PRODUCT RETENTION + MAIN COOLING SYSTEM

7-019756 ALSO IN CATEGORY 17 DATED 1959
LUNDHOLM JG + JARRETT AA
SRE FUEL ELEMENT DAMAGE
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-4488 +. 153 PAGES, 49 FIGURES, 30 TABLES, REFERENCES, NOVEMBER 30, 1959

INTERIM REPORT ON THE ORIGIN, NATURE, AND CONSEQUENCES OF THE DAMAGE. CLADDING FAILURE ON 13 FUEL ELEMENTS RESULTED FROM THE FORMATION OF LOW-MELTING FE-U ALLOY, ACCELERATED BY LOCAL HIGH TEMPERATURES RESULTING FROM COOLANT-CHANNEL BLOCKAGE. BLOCKAGE WAS ATTRIBUTED TO DECOMPOSITION PRODUCTS OF TETRALIN (PUMP COOLANT), WHICH HAD LEAKED INTO THE PRIMARY SYSTEM THROUGH A SMALL HOLE WORN THROUGH THE WALL OF A FREEZE-SEAL GLAND OF A MAIN PUMP. TENTATIVE CONCLUSIONS INDICATE THAT ALL THE IODINE RELEASED FROM THE FUEL WAS RETAINED BY THE COOLANT. (SEE SUPPLEMENT FOR FINAL ANALYSIS).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, EQUIPMENT + FISSION PRODUCT, IODINE + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + SRE (RE) + INCIDENT + REPORT, OPERATIONS ANALYSIS + FAILURE, CLADDING + FISSION PRODUCT RETENTION + FLOW BLOCKAGE + MAIN COOLING SYSTEM

7-021245 ALSO IN CATEGORIES 6 AND 12 DATED 1966
REPORT OF WORK IN THE 4TH QUARTER OF 1966
COMMISSARIAL A L-ENERGIE ATOMIQUE, CADARACHE, FRANCE
EURFNR-267 +. 89 PAGES, 1966

THIS REPORT IS ONE OF A SERIES OF REPORTS ON WORK PERFORMED UNDER THE UNITED STATES-EURATOM FAST REACTOR EXCHANGE PROGRAM. TOPICS COVERED ARE - CRITICALITY STUDIES, METALLURGY, RADIATION EFFECTS, RADIATION SHIELDING AND PROTECTION, REACTOR CONTROL, REACTOR ENGINEERING, REACTOR FUELS, AND REACTOR SAFETY.

AVAILABILITY - MICROCARD EDITIONS, INC., ACCOUNTING AND SHIPPING DEPT., WEST SALEM, WISCONSIN 54669

REACTOR, LMCR + ACCIDENT, REACTIVITY + REACTOR DYNAMICS + SODIUM + REACTOR CONTROL + CRITICALITY EXPERIMENT + FISSION PRODUCT RETENTION

7-022479 DATED 1967
KOONTZ RL + NELSON CT + BAUMASH L
CHARACTERISTICS OF AEROSOLS GENERATED DURING SODIUM FIRES

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-022474 *CONTINUED*

ATOMIC INTERNATIONAL

2 PAGES, 1 TABLE, 1 REFERENCE, ANS TRANSACTIONS 10(2), PAGE 689 AND 690, (NOV. 1965), PRESENTED AT THE 1967 WINTER MEETING OF THE AMERICAN NUCLEAR SOCIETY, CHICAGO, ILLINOIS, NOV. 5-9, 1967

STUDIES ARE BEING CONDUCTED TO DEVELOP EXPERIMENTAL INFORMATION AND ANALYTICAL METHODS THAT CHARACTERIZE THE RELEASE AND TRANSPORT OF EFFLUENTS GENERATED DURING A SODIUM FIRE IN A LIQUID-METAL FAST-BREEDER (LMFBR). EXPERIMENTS HAVE BEEN CONDUCTED TO STUDY THE RELEASE OF OXIDIZED SODIUM AND A SIMULATED FISSION PRODUCT CONSISTING OF NA-131-I INVOLVING VARIOUS QUANTITIES OF SODIUM IN DIFFERENT SIZED CONTAINERS AND IN VARIOUS OXYGEN ENVIRONMENTS.

AEROSOL + FIRE + IODINE + REACTOR, LMCR + OXYGEN + PARTICLE SIZE + SODIUM

7-022489

DATED 1967

KUNKEL WP + BERGER S

FISSION-PRODUCT DEPOSITION IN A FAST-BREEDER REACTOR SYSTEM CONTAINING FAILED FUEL ELEMENTS

ATOMIC INTERNATIONAL

15 PAGES, TABLES, FIGURES, REFERENCES, NOVEMBER 3, 1967, ABSTRACT IN ANS TRANSACTIONS 10(2), PAGE 496, PRESENTED AT AMERICAN NUCLEAR SOCIETY WINTER MEETING, NOVEMBER 5-9, 1967, CHICAGO, ILLINOIS

A DESIGN OBJECTIVE FOR THE LIQUID-METAL FAST-BREEDER REACTOR (LMFBR) IS TO ALLOW FOR POSSIBLE OPERATION WITH SOME FAILED FUEL ELEMENTS, AND IN ADDITION, TO PERMIT ACCESS TO THE HEAT-EXCHANGER CELLS AFTER REACTOR SHUTDOWN. THE CONCEPT OF A SYSTEM-DEPLETION FACTOR PROVIDES AN INTERIM METHOD OF ESTIMATING THE FISSION-PRODUCT DEPOSITION RESULTING FROM FAILED-FUEL-ELEMENT OPERATION. THE SYSTEM-DEPLETION CONCEPT IS PROPOSED AS AN INTERIM METHOD OF MORE CLOSELY APPROXIMATING THE ACTUAL EXTENT OF DEPOSITION AND THE CONSEQUENT MAINTENANCE PROBLEM.

CERIUM + CESIUM + DEPOSITION + REACTOR, FAST + IODINE + REACTOR, LMCR + RUTHENIUM + SODIUM + ZIRCONIUM + FISSION PRODUCT TRANSPORT + FISSION PRODUCT RETENTION

7-022514

DATED 1967

ETTINGER HJ + MUSS WD + BUSEY H

CHARACTERISTICS OF THE AEROSOL PRODUCED FROM BURNING SODIUM AND PLUTONIUM

LOS ALAMOS SCIENTIFIC LABORATORY, LOS ALAMOS, NEW MEXICO

14 PAGES, 6 TABLES, 15 FIGURES, NUCLEAR SCIENCE AND ENGINEERING 30, PAGES 1-13, (OCTOBER 1967)

SAFETY ANALYSIS OF SODIUM-COOLED PLUTONIUM-FUELED FAST REACTOR PLANTS MUST BE CONCERNED WITH THE POSSIBILITY OF FIRES INVOLVING THESE MATERIALS. DESIGN OF AN AIR CLEANING SYSTEM FOR SUCH A FACILITY REQUIRES BASIC DATA DEFINING THE AEROSOL CHARACTERISTICS OF SODIUM AND PLUTONIUM RELEASED DURING A FIRE. WHEN PLUTONIUM ALLOY WAS BURNED UNDER REDUCED OXYGEN CONDITIONS, THE FRACTION AIRBORNE RANGED FROM 2×10^{-7} TO 4×10^{-6} . FIRES INVOLVING PLUTONIUM ALLOY AND SODIUM TOGETHER PRODUCED AIRBORNE PLUTONIUM-SODIUM RATIOS RANGING FROM 0.34 TO 0.008%.

REACTOR, FAST + FIRE + REACTOR, LMCR + PARTICLE SIZE + PLUTONIUM + PYROPHORIC REACTION + SODIUM + AEROSOL PRODUCTION + AEROSOL PROPERTIES + PARTICLE SIZE DISTRIBUTION + METAL, LIQUID

7-02443

DATED 1967

HAFFEL W + HELLER F + SCHIKARSKI W

THE PRINCIPLE OF DOUBLE CONTAINMENT AND THE BEHAVIOUR OF AEROSOLS IN ITS RELATION TO THE SAFETY OF REACTORS WITH A HIGH PLUTONIUM INVENTORY

GESELLSCHAFT FÜR KERNFORSCHUNG M.B.H., KARLSRUHE

EUR-425 + KFK 669 + EUR-3702 + CONF-671043-5 +. 17 PAGES, FIGURES, OCTOBER 1967

THIS PAPER REPORTS ON INVESTIGATIONS OF THE INTERRELATION BETWEEN CONTAINMENT SYSTEM, INVENTORY OF RADIOACTIVE MATERIAL AND THE RADIATION BURDEN TO THE ENVIRONMENT, IF THE ACTIVITY IS ACCIDENTLY RELEASED. SPECIAL EMPHASIS WAS PAID TO THE RELEASE OF VARIOUS FISSION PRODUCTS AND FUEL ISOTOPES, PARTICULARLY IODINE AND PLUTONIUM, BECAUSE THEY REPRESENT THE LARGEST HAZARD POTENTIAL TO THE ENVIRONMENT OF A LARGE FAST-BREEDER REACTOR. THE CALCULATIONS WERE PERFORMED FOR THE CASE OF A TYPICAL LARGE, FAST, SODIUM-COOLED BREEDER OF 1000 MWE POWER, WHICH IS THE ULTIMATE GOAL OF THE PRESENT PHASE OF FAST-BREEDER DEVELOPMENT. THE OVER ALL CONCLUSION IS THAT, IN VIEW OF THE FORTHCOMING ERA OF 1000-MWE FAST BREEDER REACTORS, THERE IS A LARGE AND WELL-FOUNDED INCENTIVE TO PUT GREAT EMPHASIS ON THE INVESTIGATION OF AEROSOL BEHAVIOUR IN THE CONTEXT OF FAST-REACTOR ACCIDENTS.

AVAILABILITY - ALL DEPOSITORY LIBRARIES IN THE US AND OVERSEAS

REACTOR, BREEDER + COMPUTER PROGRAM + CONTAINMENT DESIGN + REACTOR, FAST + FISSION PRODUCT, IODINE + IODINE + REACTOR, LMCR + PLUTONIUM + SODIUM + ANALYTICAL MODEL + CONTAINMENT R AND D + GERMANY

7-024369

DATED 1968

GREENBERG S + CHENG CF + RUTHER

IN-REACTOR SODIUM CORROSION OF VANADIUM AND VANADIUM-TITANIUM ALLOYS

METALLURGY DIVISION, ARGONNE NATIONAL LAB., ARGONNE, ILLINOIS

3 PAGES, 1 FIGURE, 4 TABLES, 7 REFERENCES, NUCLEAR APPLICATIONS, 4(3), PAGE 170-172, (MARCH 1968)

VANADIUM-TITANIUM, VANADIUM-CHROMIUM, AND VANADIUM-TITANIUM-CHROMIUM ALLOYS HAVE DEMONSTRATED SUFFICIENT CORROSION RESISTANCE IN LOW-OXYGEN SODIUM AT TEMPERATURES UP TO AT LEAST 650 C TO MAKE THEM OF POTENTIAL INTEREST AS FUEL-JACKETING MATERIALS FOR SODIUM-COOLED FAST REACTORS. FINAL JUDGMENT AS TO THEIR ADEQUACY MUST BE BASED UPON PERFORMANCE OF FUEL ELEMENTS JACKETED WITH THE

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-024369 *CONTINUED*

MATERIAL OF INTEREST. VANADIUM-TITANIUM ALLOYS, EXPOSED TO FLOWING SODIUM IN THE EBR-II CORE SUFFERED SMALL WEIGHT LOSSES. IN CONTRAST TO THE BEHAVIOR OF THESE MATERIALS WHEN EXPOSED EX-REACTOR, NO HARDENED LAYERS WERE PRODUCED.

ALLOY + CLADDING + REACTOR, LMCR + TITANIUM + CORROSION + METAL, ALKALI + METAL, LIQUID + PROPERTY, PHYSICAL + VANADIUM + FUEL ELEMENTS

7-024386

DATED 1967

QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY-MARCH 1967. AEC UNCLASSIFIED PROGRAMS
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA
NAA-SR-12395 +. 183 PAGES, FIGURES, TABLES, MAY 31, 1967

THIS QUARTERLY PROGRESS REPORT INCLUDES A DESCRIPTION OF CURRENT ACTIVITIES RANGING FROM CROSS-SECTION ANALYSIS TO DYNAMIC-LOOP TESTING.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00
COPY, \$0.65 MICROFICHE

FAILURE, FUEL ELEMENT + FUEL INTEGRITY + REACTOR, LMCR + RADIATION DAMAGE + SODIUM + FISSION PRODUCT TRANSPORT + OUT OF PILE EXPERIMENT + AI + COOLANT CHEMISTRY + RADIATION SAFETY AND CONTROL + METAL, LIQUID + RADIOCHEMICAL PROCESSING + FUEL ELEMENTS

7-025234

DATED 1968

SAVOLAINEN JE + COMPERE EL
STUDY OF UNCLAD-METAL-FUELED FAST BREEDER
OAK RIDGE NATIONAL LAB., TENN.
ORNL-4229 +. 2 PAGES, 1 REFERENCE, PAGE 188-189 OF REACTOR CHEMISTRY DIVISION ANNUAL PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1967

A SODIUM-COOLED FAST BREEDER REACTOR WITH UNCLAD METAL FUEL SHOWS PROMISE AS A POWER AND HEAT SOURCE, ESPECIALLY FOR DESALINATION. ITS FEASIBILITY DEPENDS ON THE OXYGEN CONTENT IN THE SODIUM, WHICH IF UNCONTROLLED COULD CAUSE EXCESSIVE CORROSION OF THE FUEL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FUEL INTEGRITY + REACTOR, LMCR + OXYGEN + SODIUM + TESTING + CORROSION + CONTAMINATION + METAL, LIQUID + THERMAL PROPERTY + DESALTING

7-027028

DATED 1968

EICHELBERGER RL + MCKISSON RL
SOLUBILITY OF COPPER IN SODIUM
ATOMICS INTERNATIONAL
AI-AEC-12671 +. 22 PAGES, 8 FIGURES, 3 TABLES, 4 REFERENCES, MAY, 1968

THE SOLUBILITY OF COPPER IN SODIUM HAS BEEN STUDIED IN THE TEMPERATURE RANGE OF 250 TO 726 C, USING A VERY HIGH PURITY SODIUM AND OXYGEN FREE HIGH CONDUCTIVITY COPPER. A CAPSULE WAS USED, WITH THE SODIUM EXPOSED ONLY TO HIGH VACUUM. CHEMICAL ANALYSIS WAS PERFORMED USING COLORIMETRY AND ATOMIC ABSORPTION SPECTROMETRY. RESULTS OF THE 21 EXPERIMENTS REPORTED BETWEEN 350 AND 726 C WERE COMBINED WITH THOSE FROM TWO OTHER STUDIES TO GIVE AN EXPRESSION BASED ON 39 DATA POINTS AND COVERING THE TEMPERATURE RANGE OF 245 TO 726 C.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + SODIUM + COPPER + ANALYTICAL TECHNIQUE + METAL, LIQUID + PROPERTY, PHYSICAL + CALIBRATION

7-027029

DATED 1967

MOREWITZ HA + KOUNTZ RL + NELSON CT + GREENFIELD MA + HAUSKNECHT DF
SODIUM BURNING, AEROSOL GENERATION, AND CONTROL OF FISSION PRODUCT RELEASE DURING AN ACCIDENT PART II.
AEROSOL GENERATION AND CONTROL OF FISSION PRODUCT RELEASE
ATOMICS INTERNATIONAL +
29 PAGES, 16 FIGURES, TABLE, 4 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST NUCLEAR REACTORS, SEPTEMBER, 1967

RESULTS ARE PRESENTED FOR WORK IN PROGRESS AT ATOMICS INTERNATIONAL TO DEVELOP EXPERIMENTAL INFORMATION AND ANALYTICAL METHODS WHICH CHARACTERIZE THE RELEASE AND TRANSPORT OF EFFLUENTS GENERATED DURING A SODIUM FIRE IN AN LMFR. DATA ARE PRESENTED FROM EXPERIMENTS WHICH DESCRIBE THE TIME-HISTORY OF PARTICLE SIZE AND MASS DISTRIBUTION OF AEROSOLS GENERATED DURING SODIUM FIRES. EXPERIMENTAL OBSERVATIONS ARE MADE TO STUDY AEROSOL DEPLETION RESULTING FROM AGGLOMERATION, FALLOUT, AND WALL PLATING MECHANISMS. PREDICTION OF SODIUM OXIDE AEROSOL BEHAVIOR USING A THEORETICAL MODEL IS IN AGREEMENT WITH EXPERIMENTAL OBSERVATIONS IN THE LABORATORY. THE INTERACTION OF THE DOMINANT AEROSOL SPECIES, SODIUM OXIDE, WITH FUEL AND FISSION PRODUCT AEROSOLS IS ALSO DESCRIBED.

AVAILABILITY - H.A. MOREWITZ, P.O. BOX 309, CANOGA PARK, CALIFORNIA

DEPOSITION + FALLOUT + FIRE + REACTOR, LMCR + OXIDE + SODIUM + ANALYTICAL MODEL + FISSION PRODUCT TRANSPORT +

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-027029 *CONTINUED*
AEROSOL PRODUCTION + AEROSOL PROPERTIES + AGGLOMERATE + METAL, LIQUID

7-027030 ALSO IN CATEGORY 8 DATED 1968
MACPHERSON RE
SODIUM BURNING AND AEROSOL RELEASE - AN EVALUATION OF THE STATE OF THE ART
DAK RIDGE NATIONAL LABORATORY, DAK RIDGE, TENNESSEE
ORNL-TM-1937 +. 33 PAGES, 26 REFERENCES, MAY 6, 1968

IN A SODIUM-COOLED REACTOR, THE POSSIBILITY OF SODIUM FIRES AND THE RESULTANT RELEASE OF AEROSOLS STRONGLY INFLUENCE CONTAINMENT VESSEL DESIGN AND OPERATING SAFETY. QUALITATIVE INVESTIGATIONS OF SODIUM POOL FIRES AND SPRAY FIRES HAVE BEEN CARRIED OUT IN CONJUNCTION WITH EACH MAJOR SODIUM-COOLED REACTOR PROJECT. RECENTLY, MORE QUANTITATIVE STUDIES WERE UNDERTAKEN IN THESE AREAS, AND THE PROBLEM OF EXPLOSIVE INJECTION OF SODIUM SPRAYS INTO CONTAINMENT VOLUMES WAS INVESTIGATED. WORK IS ALSO UNDER WAY TO CHARACTERIZE AEROSOLS RELEASED FROM SODIUM FIRES AND TO DEFINE THEIR PERFORMANCE IN THE CONTAINMENT VOLUME. THIS DOCUMENT BRIEFLY SUMMARIZES THE WORK TO DATE ON SODIUM FIRES AND AEROSOL RELEASE, REVIEWS THE SCOPE OF CURRENT PROGRAMS, AND THEN PROPOSES ADDITIONAL EFFORT TO PROVIDE SAFETY CRITERIA FOR LIQUID-METAL-COOLED FAST-REACTOR DESIGN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

FIRES + REACTOR, LMCR + OXIDE + SAFETY REVIEW + SODIUM + AEROSOL PRODUCTION + AEROSOL PROPERTIES + METAL, LIQUID

7-027031 DATED 1967
JURLAND S + BILSKI R
CONTROL OF CONTAMINANTS IN LIQUID-METAL SYSTEMS
NASA, LEWIS RESEARCH CENTER
NASA-TM-X-1432 +. 9 PAGES, 7 FIGURES, TABLE, 10 REFERENCES, DECEMBER, 1967

THE CONTAMINANT CONTROLS AND CLEANING PROCEDURES USED IN THE SNAP-8 SIMULATION LOOP, CONSISTING OF TWO SODIUM-POTASSIUM (NAK) LOOPS AND ONE MERCURY LOOP, MINIMIZED BOTH STARTUP AND OPERATIONAL PROBLEMS. MERCURY WETTING WAS INDUCED IN THE BOILER AND CONDENSER AS DETERMINED BY PROGRESSIVE IMPROVEMENT IN THE HEAT-TRANSFER PERFORMANCE. THE DEGREE OF WETTING IS AN INDICATION OF THE SYSTEM CLEANLINESS. THE BOILER WAS CONDITIONED IN APPROXIMATELY 250 HOURS OF OPERATION WITHOUT WETTING ADDITIVES. AFTER 384 HOURS OF OPERATION, ANALYSIS OF A HG SAMPLE BY A DISTILLATION METHOD INDICATED NO CONTAMINATION. OXIDE PLUGGING IN THE NAK LOOPS WAS MINIMIZED USING THE LOOP-ENTRY PROCEDURES DEVELOPED IN THE INVESTIGATION. NO PLUGS OCCURRED BECAUSE OF CHANGED OR OF REPAIRS THAT REQUIRED SYSTEM ENTRY, AND ONLY MINOR PARTIAL PLUGS OCCURRED IN OVER 1500 HOURS OF OPERATION OF THE NAK LOOPS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

DECONTAMINATION + REACTOR, LMCR + MERCURY + OXIDE + SNAP 8 (SR) + SODIUM + REACTOR, SPACE + OUT OF PILE
EXPERIMENT + NAK + METAL, LIQUID + OPERATING EXPERIENCE + PIPES AND PIPE FITTINGS

7-027032 ALSO IN CATEGORY 12 DATED 1968
KUNKEL WP + POLLUCK BD + GUON J + ZWETZIG GB + SILBERBERG M + BERGER S
LMFBR FISSION PRODUCT CONTAMINATION AND CONTROL STUDIES - A STATUS REPORT
ATOMIC INTERNATIONAL
AI-AEC-12687 +. 78 PAGES, 24 FIGURES, 13 TABLES, 29 REFERENCES, JUNE 30, 1968

THE STATUS IS PRESENTED OF STUDIES AND EXPERIMENTS BEING CONDUCTED AT ATOMIC INTERNATIONAL TO DESCRIBE THE DISTRIBUTION OF FISSION PRODUCTS AND OTHER CONTAMINATION SOURCES RELEASED TO THE COOLANT OF THE LIQUID METAL FAST BREEDER REACTOR. EXPERIMENTAL RESULTS DERIVED FROM STUDIES OF CESIUM AND SODIUM IODIDE VAPORIZATION FROM SODIUM SOLUTIONS AND CESIUM COLD TRAPPING STUDIES ARE PRESENTED. THE FISSION PRODUCT BEHAVIOR LOOP WHICH WILL BE USED FOR A MORE COMPREHENSIVE SERIES OF INVESTIGATIONS OF FISSION PRODUCT BEHAVIOR IN SODIUM IS ALSO DESCRIBED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DEPOSITION + REACTOR, FAST + FISSION PRODUCT RELEASE + REACTOR, LMCR + SODIUM + FISSION PRODUCT TRANSPORT + AI
+ RADIOACTIVITY RELEASE + EXPERIMENT.

7-027018 DATED 1968
POUCHER FW + KOHL AL
CLEANING TECHNIQUES FOR SODIUM SYSTEM COMPONENTS
ATOMIC INTERNATIONAL, CANOGA PARK, CALIFORNIA
2 PAGES, 2 REFERENCES, ANS TRANS. 11(1), PAGES 114-115, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY
AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

SPECIALIZED TECHNIQUES ARE REQUIRED FOR CLEANING SURFACES THAT HAVE BEEN IN CONTACT WITH LIQUID SODIUM. ALTHOUGH A GREAT MANY PROCESSES HAVE BEEN INVESTIGATED, ONLY TWO HAVE PROVED GENERALLY USEFUL FOR A WIDE VARIETY OF APPLICATIONS AT ATOMIC INTERNATIONAL. THESE ARE REACTION WITH STEAM (WITH OR WITHOUT AN INERT GAS DILUENT) FOR MAJOR SYSTEMS AND COMPONENTS, AND REACTION WITH LIQUID 2-ETHOXYETHANOL (ETHYLENE GLYCOL MONOBUTYL ETHER, BUTYL CELLOSOLVE, DOWANOL E.B.) FOR SMALLER ITEMS AND HOT-CELL APPLICATIONS. IN ADDITION, DIPPING COMPONENTS IN HOT OIL HAS PROVED

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-027818 *CONTINUED*

HELPFUL AS AN INITIAL STEP TO REDUCE THE QUANTITY OF SODIUM REQUIRING REACTION. LIQUID AMMONIA WASHING HAS BEEN SHOWN TO BE APPLICABLE FOR SPECIAL CASES WHERE HYDROGEN EVOLUTION MUST BE MINIMIZED.

DECONTAMINATION + REACTOR, LMCR + SODIUM + STEAM + COOLANT CHEMISTRY + METAL, ALKALI + METAL, LIQUID + REACTOR, LMFBR

7-027819

DATED 1968

MCPHEETERS CC + BIERY JC

THE DYNAMIC CHARACTERISTICS OF A SODIUM OXIDE PLUGGING INDICATOR

LOS ALAMOS SCIENTIFIC LABORATORY, NEW MEXICO

2 PAGES, 6 REFERENCES, ANS TRANS. 11(1), PAGES 115-116, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND THE CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

WORK AT LOS ALAMOS SCIENTIFIC LABORATORY HAS INDICATED THAT THE PLUGGING INDICATOR CAN BE A USEFUL ANALYTICAL INSTRUMENT FOR INDICATING OXYGEN CONCENTRATIONS IN SODIUM. AS COMPARED TO THE DISTILLATION SAMPLER OF EMF CELLS, THE PLUGGING INDICATOR DATA HAD A MUCH LOWER STATISTICAL STANDARD DEVIATION. HOWEVER, THE METER REQUIRED CALIBRATION. THE PLUGGING TEMPERATURE WAS CORRELATED WITH FLOW RATE, COOLING RATE, AND OXYGEN CONCENTRATION.

REACTOR, LMCR + OXIDE + SODIUM + ANALYTICAL TECHNIQUE + COOLANT CHEMISTRY + METAL, ALKALI + METAL, LIQUID + ORIFICE + REACTOR, LMFBR + CALIBRATION

7-027820

DATED 1968

MCPHEETERS CC

MASS TRANSFER OF OXYGEN IN SODIUM COLD TRAPS

LOS ALAMOS SCIENTIFIC LABORATORY

2 PAGES, 5 REFERENCES, ANS TRANS. 11(1), PAGES 116-117, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND THE CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

COLD TRAPS HAVE BEEN USED EXTENSIVELY FOR CONTROLLING OXYGEN CONCENTRATION IN SODIUM SYSTEMS. COLD TRAPS UTILIZE THE POSITIVE TEMPERATURE DEPENDENCE OF THE SOLUBILITY OF SODIUM MONOXIDE IN SODIUM AND THE RESULTANT PRECIPITATION OF SODIUM MONOXIDE AT LOW SODIUM TEMPERATURES. SODIUM MONOXIDE PRECIPITATION IS HERE ASSUMED TO BE HETEROGENEOUS BECAUSE OF THE STRONG EFFECT OF SURFACE AREA ON THE PRECIPITATION RATE SHOWN IN PREVIOUS PACKLESS COLD TRAP EXPERIMENTS. A PACKLESS COLD TRAP OF SIMPLE GEOMETRY AND OPERATING WITH A TEMPERATURE GRADIENT WAS USED IN THIS STUDY. IN THE PRESENCE OF THE TEMPERATURE GRADIENT, THE EFFECTIVE PRECIPITATION SURFACE AREA WAS A FUNCTION OF THE OXYGEN CONCENTRATION IN THE SODIUM ENTERING THE COLD TRAP.

REACTOR, LMCR + OXIDE + SODIUM + ANALYTICAL TECHNIQUE + FILTER, TRAP + COOLANT CHEMISTRY + METAL, ALKALI + METAL, LIQUID + ORIFICE + REACTOR, LMFBR + CALIBRATION

7-028506

DATED 1968

PRACH TF + NEWMAN L + AUERBACH C

THE DETERMINATION OF OXYGEN IN SODIUM BY THE AMALGAMATION METHOD

BROOKHAVEN NATIONAL LABORATORY, UPTON, NEW YORK

BNL 50118 + T-498 +. 22 PAGES, 2 FIGURES, 3 TABLES, REFERENCES, JUNE 1968

THE AMALGAMATION METHOD, WHEN USED WITH THE PROPER PRECAUTIONS, IS CAPABLE OF RELIABLE AND REPRODUCIBLE ANALYSIS TO ± 1 PPM, DOWN TO THE 1 PPM LEVEL. ITS MAIN DISADVANTAGE, ESPECIALLY WHEN RADIOACTIVE SAMPLES ARE INVOLVED, IS THE LARGE NUMBER OF MANIPULATIONS REQUIRED. WE ARE USING THE METHOD ROUTINELY FOR ANALYZING LOOP SAMPLES AS WELL AS SODIUM FROM A NUMBER OF SPECIAL EXPERIMENTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + OXYGEN + SODIUM + OUT OF PILE EXPERIMENT + ANALYTICAL TECHNIQUE + METAL, LIQUID + REACTOR, LMFBR + CALIBRATION

7-028508

DATED 1968

ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT JULY 1968

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ANL-7478 +. 133 PAGES, FIGURES, TABLES, REFERENCES, AUGUST 28, 1968

THE REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, ISSUED MONTHLY, IS INTENDED TO BE A MEANS OF REPORTING THOSE ITEMS OF SIGNIFICANT TECHNICAL PROGRESS WHICH HAVE OCCURRED IN BOTH THE SPECIFIC REACTOR PROJECTS AND THE GENERAL ENGINEERING RESEARCH AND DEVELOPMENT PROGRAMS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CRITICAL ASSEMBLY + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR DYNAMICS + REACTOR KINETICS + ANL + METAL, LIQUID + ZPPR (CAF) + REACTOR PHYSICS + ZPR 3 (CAF) + OPERATING EXPERIENCE + REACTOR, LMFBR

7-028511

DATED 1968

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-028511 *CONTINUED*

81KSS IR

HELIUM PRODUCTION IN REACTOR MATERIALS

UKAEA, THE REACTOR GROUP, DOUNREAY EXPERIMENTAL REACTOR ESTABLISHMENT, DOUNREAY, GREAT BRITAIN
IRG-16081G) +. 34 PAGES, 14 FIGURES, 8 TABLES, 26 REFERENCES, 1968

ALPHA PARTICLES CAN BE PRODUCED IN REACTOR MATERIAL BY ALPHA DECAY, TRIPLE FISSION OR BY INTERACTIONS BETWEEN NUCLEI AND THE NEUTRONS AND RESULT IN AN ACCUMULATION OF HELIUM GAS. THE BEST AVAILABLE DATA AND METHODS OF CALCULATING THE PRODUCTION RATES ARE GIVEN WITH PARTICULAR REFERENCE TO FAST REACTORS SINCE ALL (N,ALPHA) REACTIONS EXCEPT THOSE WITH THE B-10 ISOTOPE REQUIRE FAST NEUTRONS. ACTUAL PRODUCTION RATES TOGETHER WITH AN INDICATION OF THE CONSEQUENCES ARE GIVEN FOR REACTOR FUEL, CLADDING MATERIAL AND COOLANT, AND THE DATA ARE PRESENTED IN SUCH A WAY THAT A QUICK ASSESSMENT OF HELIUM CONTENT IN A MATERIAL FOR A PARTICULAR CASE CAN BE OBTAINED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICROFICHE

BORDON + CLADDING + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR, LMCR + REACTOR, TEST + UNITED KINGDOM + COOLANT CHEMISTRY + HELIUM + RADIOISOTOPE + DOUNREAY (LMFBR)

7-028567 ALSO IN CATEGORY 11

DATED 1968

GUNNEK JG + POREMOKA SW

COMPENDIUM OF PROPERTIES AND CHARACTERISTICS FOR SELECTED LMFBR CLADDING MATERIALS

BATTILLE MEMORIAL INST., COLUMBUS, OHIO

DMI-1900 +. 234 PAGES, MAY 15, 1968

REVIEWS WELDING, RADIATION DAMAGE, STRENGTH AND DUCTILITY AT HIGH (600 C) TEMPERATURES, COMPATIBILITY WITH FUEL AND SODIUM FOR AUSTENITIC SS AND NICKEL-BASED ALLOYS, VANADIUM ALLOYS, NIOBIUM ALLOYS FOR OXIDE, CARBIDE, NITRIDE, AND OXIDE FUELS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + RADIATION DAMAGE + STEEL, STAINLESS + PROPERTY, PHYSICAL + INCOLOY + REACTOR, LMFBR

7-029286

DATED 1968

BISCHUFF K + HARRISON JD + POTTER PE

THE PREPARATION AND CHARACTERISATION OF SINTERED URANIUM-PLUTONIUM OXIDES

UNITED KINGDOM ATOMIC ENERGY RESEARCH ESTABLISHMENT, HARWELL

AERE-R-5774 +. 12 PAGES, 10 FIGURES, 8 TABLES, 5 REFERENCES, APRIL 1968

THE PREPARATION FROM MECHANICALLY MIXED POWDERS OF SINTERED PELLETS OF URANIUM-PLUTONIUM OXIDES WITH PLUTONIUM OXIDE CONCENTRATIONS OF 10, 20, 30 AND 40 WT. % IS DESCRIBED. THE CONDITIONS FOR THE PREPARATION OF DENSE SINGLE PHASE PELLETS ARE DESCRIBED TOGETHER WITH THE CHARACTERIZATION OF THE SINTERED PRODUCTS. HOMOGENEITY OF THE PELLETS DETERMINED BY ELECTRON MICROPROBE ANALYSIS.

AVAILABILITY - SCIENTIFIC ADMINISTRATION OFFICE, ATOMIC ENERGY RESEARCH ESTABLISHMENT, HARWELL, DIDCOT, BERKSHIRE, ENGLAND

REACTOR, FAST + OXIDE + PLUTONIUM + URANIUM + FUEL, PELLET TYPE + ANALYTICAL TECHNIQUE + RADIOCHEMICAL PROCESSING + FUEL REPROCESSING + FUEL, POWDER TYPE + REACTOR, LMFBR

7-029287

DATED 1967

JUDD AM

SODIUM BOILING AND FAST REACTOR SAFETY ANALYSIS

UNITED KINGDOM ATOMIC ENERGY ESTABLISHMENT, WINFRITH

AEEW-R561 +. 58 PAGES, 19 FIGURES, 3 TABLES, 33 REFERENCES, DECEMBER 1967

DISCUSSES VARIOUS ASPECTS OF THE BOILING BEHAVIOUR OF LIQUID METALS AS IT INFLUENCES THE SAFETY OF A FAST REACTOR. IN PARTICULAR AN ATTEMPT IS MADE TO APPLY THE AVAILABLE EXPERIMENTAL DATA TO THE QUESTIONS OF THE DEGREE OF LIQUID SUPERHEAT WHICH COULD BE REACHED IN A REACTOR UNDER ACCIDENT CONDITIONS, THE RATE AT WHICH A BLOCKED COOLANT CHANNEL COULD BE VOIDED BY BOILING, AND THE EFFECTS OF THE CONDENSATION OF VAPOUR BUBBLES IN COOL LIQUID.

AVAILABILITY - WINFRITH SECRETARIAT, ATOMIC ENERGY ESTABLISHMENT, WINFRITH, DORCHESTER, DORSET, ENGLAND

HEAT TRANSFER, BOILING + REACTOR, FAST + SODIUM + VOID COEFFICIENT + VAPOR PRESSURE + METAL, ALKALI + METAL, LIQUID + BUBBLE NUCLEATION + HEAT TRANSFER EXPERIMENT + BUBBLE + REACTOR, LMFBR

7-029411

DATED 1965

GATLEY JA + WEBB J + GALLIE RR

VENTED NUCLEAR REACTOR FUEL ELEMENT

U.S. PATENT 3,357,853 +. AUGUST 23, 1963, 3 SHEETS

GIVES A FUEL ELEMENT DESIGN OF THE VENTED TYPE HAVING A FUEL-CONTAINING PROTECTIVE SHEATH OF ELONGATED FORM. A VENT PATH UNOBSTRUCTED BY FUEL HAS A PERMANENTLY OPEN COMMUNICATION WITH THE FUEL ADJACENT AT ONE END. THE OTHER END OF THE VENT PATH HAS AN OUTLET WITH THE EXTERIOR SURROUNDING OF THE SHEATH FOR DISCHARGE INTO THE SURROUNDINGS OF GASES RELEASED BY THE FUEL

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-029411 *CONTINUED*
DURING OPERATION.

AVAILABILITY - THE U.S. PATENT OFFICE, DEPARTMENT OF COMMERCE, WASHINGTON, D.C. (\$0.25 CENTS/PAGE)

REACTOR, BREEDER + DESIGN CRITERIA + REACTOR, FAST + FISSION GAS RELEASE + FISSION PRODUCT RELEASE + FISSION PRODUCT, VOLATILE + DESIGN STUDY + FISSION PRODUCT RETENTION + REACTOR, LMFBR + FUEL ELEMENTS

7-029516 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION M-7 - FISSION PRODUCT RELEASE ASSUMPTIONS FOR DESIGN OF CONTROL ROOM SHIELDING
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
9 PAGES, 1 FIGURE, 1 TABLE, PAGE M-12 THRU -20 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION,
RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE FISSION PRODUCT RELEASE ASSUMPTIONS USED IN THE DESIGN OF THE SHIELDING OF THE CONTROL ROOM. PROVIDE DETAILS OF THE VENTILATION SYSTEM FOR THE CONTROL ROOM, INCLUDING ITS MODE OF OPERATION DURING ACCIDENT CONDITIONS. PROVIDE DOSE THAT WOULD BE RECEIVED BY OPERATORS DURING COURSE OF DBA. (ANSWER ON PAGES M-12 THRU -20.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + DOSE + FISSION PRODUCT RELEASE + REACTOR, LMCR + REPORT, SAR + VENTILATION SYSTEM + ACCIDENT MODEL + PERSONNEL EXPOSURE, RADIATION + SEFOR (RE) + SHIELDING + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + SYSTEM OPERABILITY IN ACCIDENT + CONTROL PANEL/ROOM

7-029547 DATED 1968
SIEVENARI M + RENARD A
PROPAGATION OF FUEL MELTING IN A FAST REACTOR USING A CERAMIC FUEL
13 PAGES, BULL. INFORM. SCI. TECH. (PARIS), NO. 124, PAGES 3-16 (MARCH 1968)

THE CODE PROFUSION PERMITS STUDY OF PROPAGATION OF FUSION IN THE CERAMIC FUEL PINS OF FAST REACTOR AND THE REACTIVITY VARIATION WHICH CAN BE ASSOCIATED WITH A POSSIBLE FUEL SLUMPING IN THE AXIAL HOLES RESULTING FROM THE MIGRATION OF POROSITY IN THE VIRGIN FUEL. THE TEMPERATURE DISTRIBUTIONS OF THE COOLANT, FUEL CAN, AND FUEL ARE CALCULATED. ONE CALCULATES THE CORE POWER LIMIT, WHICH INITIATES THE FUEL FUSION. AFTERWARDS THE PROPAGATION OF FUEL MELTING IS CALCULATED WHEN THE CORE POWER LIMIT IS OVERPASSED. FOR EACH CONFIGURATION, MOLTEN FUEL DISPLACEMENTS ARE PERFORMED AND THE SUBSEQUENT REACTIVITY CHANGE IS EVALUATED. THE METHOD HAS BEEN USED FOR THE ENRICO FERMI OXIDE CORE DESIGN AND KAPSODIE.

REACTOR, BREEDER + CERAMICS + COMPUTER PROGRAM + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + TEMPERATURE GRADIENT + COOLANT QUALITY + FUEL ELEMENTS

7-029548 ALSO IN CATEGORY 17 DATED 1968
MCCARTHY WJ + JENS WH
REVIEW OF THE FERMI REACTOR FUEL DAMAGE INCIDENT AND A PRELIMINARY ASSESSMENT OF ITS SIGNIFICANCE TO THE DESIGN AND OPERATION OF SODIUM COOLED FAST REACTORS
POWER REACTOR DEVELOPMENT CO., MONROE, MICH.
22 PAGES, BULL. INFORM. SCI. TECH. (PARIS), NO. 124, PAGES 39-50 (MARCH 1968) (IN FRENCH)

THE STATUS OF THE INVESTIGATION OF THE EXTENT AND CAUSE OF THE OCTOBER 5, 1966, FUEL DAMAGE AT THE ENRICO FERMI SODIUM COOLED FAST BREEDER REACTOR IS GIVEN AS OF SEPTEMBER 13, 1967. A FOREIGN OBJECT HAS BEEN FOUND IN THE CORE INLET PLENUM WHICH IS BELIEVED TO HAVE BEEN THE CAUSE OF THE DAMAGE. THE RESULTS OF EXAMINATIONS OF SIX DAMAGED FUEL SUBASSEMBLIES ARE GIVEN AND SOME PRELIMINARY OBSERVATIONS ARE MADE CONCERNING POSSIBLE IMPROVEMENTS WHICH MAY BE MADE IN THE DESIGN OF SODIUM COOLED FAST REACTORS TO AID IN THE PREVENTION, DETECTION AND RECOVERY FROM SIMILAR INCIDENTS.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + REACTOR, LMCR + DAMAGE + COMPARISON, FACILITIES + FUEL ELEMENTS

7-029834 ALSO IN CATEGORY 17 DATED 1968
SMITH RR + EBERSOLE ER + THALGOTT FW
EXPERIENCE WITH A GAS-TYPE FAILURE IN AN EBR-11 DRIVER FUEL ELEMENT
ARGONNE NATIONAL LABORATORY
2 PAGES, ANS TRANS. 11(1), PAGES 279-80, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

FISSION-GAS MONITOR SHOWED SIGNAL-TO-BACKGROUND RATIO OF 12. (FAILED ELEMENT RADIATION DETECTOR GAVE NO INDICATION). BY RAISING POWER AND STUDYING Xe-133 AND 135 BUILDUP, DETERMINED FAILURE WAS FUEL LOADED IMMEDIATELY PRIOR TO STARTUP. OBSERVED THAT ONCE A GAS BUBBLE WAS RELEASED, DRIVING FORCE FOR ADDITIONAL RELEASE IS NEGLIGIBLE, AND CONCLUDED RUNNING REACTOR WITH GAS-TYPE FAILURE IS POSSIBLE.

ACTIVITY BUILDUP + LBR 1 AND 2 (RE) + REACTOR, FAST + FISSION GAS RELEASE + FISSION PRODUCT, VOLATILE + FAILURE, FUEL ELEMENT + REACTOR, LMCR + MEASUREMENT + NOBLE GAS + REACTOR, TEST + XENON + MONITOR, GAS + COVER GAS + REACTOR, ERDA OWNED + FUEL ELEMENTS

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-029940 DATED 1967
DESCAMPS C + RYGAERT J
BEHAVIOR OF FISSION PRODUCTS IN THE ATMOSPHERE SURROUNDING A SODIUM COOLED FAST REACTOR DURING ACCIDENTAL COOLANT FIRES
SOCIETE BELGE POUR L'INDUSTRIE NUCLEAIRE, BRUSSELS
14 PAGES, 3 TABLES, CONFERENCE INTERNATIONALE SUR LA SURETE DES REACTEURS A NEUTRONS RAPIDES, VI/1, SEPT. 22, 1967

AS TO SAFETY OF FAST REACTORS, THE MOST SEVERE ACCIDENT CONSIDERED MIGHT LEAD TO THE SUDDEN RUPTURE AT HIGH TEMPERATURE OF THE PRIMARY CIRCUIT. SUCH AN ACCIDENT WOULD PUT SODIUM, CONTAMINATED BY FISSION PRODUCTS, INTO CONTACT WITH THE ATMOSPHERE. IT IS IMPORTANT TO KNOW THE VARIOUS CONSEQUENCES OF THE RESULTING EXPLOSIVE REACTION FROM THE STANDPOINT OF CONTAMINATION, TEMPERATURE, AND PRESSURE. AN EXPERIMENTAL DEVICE REPRESENTING AT REDUCED SCALE THE CONDITIONS OF ACCIDENT HAS BEEN SET UP. IN THE FIRST STAGE OF THE INVESTIGATION THE FIRST FISSION PRODUCT EXAMINED WAS IODINE.

AVAILABILITY - CL. DESCAMPS, BELGONUCLEAIRE, S. A., 35, RUE DES COLONIES, BRUXELLES 1, BELGIQUE

AEROSOL + AIRBORNE RELEASE + DEPOSITION + EXPLOSION + FIRE + FISSION PRODUCT RELEASE + FISSION PRODUCT, IODINE + IODINE + OXIDATION + SODIUM + AEROSOL PROPERTIES + SAMPLING + ACCIDENT MODEL + REACTOR, LMFBR

7-029949 DATED 1968
KEILHOLTZ GW + BATTLE GC
FISSION-PRODUCT RELEASE AND TRANSPORT IN LIQUID-METAL-COOLED FAST BREEDER REACTORS
OAK RIDGE NATIONAL LABORATORY, OAK RIDGE, TENNESSEE
14 PAGES, NUCLEAR SAFETY, VOL. 9(6), PP. 494-507 NOV-DEC 1968

LIQUID-METAL-COOLED FAST BREEDERS WILL BUILD UP LARGE FISSION-PRODUCT INVENTORIES. VOLATILE FISSION PRODUCTS MAY BE VENTED FROM THE FUEL ELEMENTS TO THE PRIMARY SODIUM COOLANT. IN OPERATION AND AFTER MINOR MELTDOWNS, THE COOLANT SYSTEM WILL RETAIN MOST OF THE IODINE AND OTHER FISSION PRODUCTS FROM VENTED OR RUPTURED FUEL ELEMENTS. SOME, PARTICULARLY NOBLE GASES, WILL ENTER THE ARGON BLANKET GAS. IF THE COOLANT SYSTEM AND THE NITROGEN-FILLED CONTAINMENT SYSTEM ENCLOSING IT WERE BREACHED, SODIUM, CONTAINING ²⁴Na, FISSION PRODUCTS, AND POSSIBLY UO₂ AND PUO₂ WOULD ENTER THE AIR WITHIN THE SECONDARY CONTAINMENT SYSTEM. THE SODIUM WOULD BURN, AND SODIUM OXIDE AEROSOLS MIGHT SORB OR OCCLUDE LARGE FRACTIONS OF MOST RADIOACTIVE MATERIALS BUT NOT OF THE NOBLE GASES. HIGH-EFFICIENCY PARTICULATE-AIR FILTER SYSTEMS FOR WATER-COOLED REACTORS MAY BE READILY ADAPTABLE FOR THE REMOVAL OF SODIUM OXIDE AEROSOLS BY THE ADDITION OF EXTENSIVE PREFILTERING SYSTEMS. METHYL IODIDE MAY NOT BE A PROBLEM IN THESE REACTORS. THE MAJOR HAZARDS WILL PROBABLY BE NA-24, PLUTONIUM, FINE AEROSOLS, AND NOBLE GASES.

AEROSOL + REACTOR, FAST + FIRE + FISSION PRODUCT RELEASE + FAILURE, FUEL ELEMENT + FUEL INTEGRITY + KRYPTON + NOBLE GAS + PLUTONIUM + SODIUM + XENON + FISSION PRODUCT TRANSPORT + ARGON + ORGANIC IODIDE + COOLANT CHEMISTRY + URANIUM DIOXIDE + METAL, ALKALI + METAL, LIQUID + FISSION PRODUCT RETENTION + PLUTONIUM DIOXIDE + REACTOR, LMFBR + FUEL COOLANT INTERACTION

7-029994 ALSO IN CATEGORIES 1 AND 9 DATED 1968
LMFBR PROGRAM PLAN. VOL. 10 - SAFETY
ARGONNE NATIONAL LABORATORY, LMFBR PROGRAM, OFFICE, ARGONNE, ILLINOIS
WASH-1110 +. 256 PAGES, FIGURES, TABLES, AUGUST 1968

THE SAFETY ELEMENT OF THE LMFBR PROGRAM IS TECHNOLOGICALLY ORIENTED TO AID DESIGNERS, OPERATORS, AND REGULATORY GROUPS IN ACHIEVING THE REQUISITE LEVEL OF SAFETY FOR LMFBRs. THE OBJECTIVE IS TO PROVIDE REALISM AND CONFIDENCE IN THE UNDERSTANDING AND ANALYSIS OF ACCIDENT SITUATIONS, TO DEVELOP AND EVALUATE SAFETY SYSTEMS FOR THE PREVENTION OF ACCIDENTS AND MITIGATION OF THEIR CONSEQUENCES, AND TO DEVELOP STANDARDS AND CODES FOR THE SAFE DESIGN, SITING, CONSTRUCTION, AND OPERATION OF BOTH USAEC TEST FACILITIES AND COMMERCIAL LMFBRs. THE SCOPE INCLUDES EVALUATIONS AND EXPERIMENTATION RELATIVE TO ABNORMAL OPERATION, POTENTIAL ACCIDENTS, AND SAFETY SYSTEMS DESIGNED TO PREVENT ACCIDENTS AND TO LIMIT THEIR CONSEQUENCES. EMPHASIS IS ON PREVENTION, EARLY DETECTION, AND CONTROL OF ACCIDENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

INSTRUMENT, PROTECTIVE + REACTOR SAFETY SYSTEM + INSTRUMENT, SURVEILLANCE + SAFETY PRINCIPLES AND PHILOSOPHY + K AND D PROGRAM + REACTOR, LMFBR + JACCBS + INSTRUMENTS, MISC.

7-030253 ALSO IN CATEGORIES 5 AND 18 DATED 1968
RESPONSE TO QUESTION 5 - SOURCE TERMS IN RELEASE CALCULATION
NORTHERN STATES POWER COMPANY
11 PAGES, PP. 23-33 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

FISSION PRODUCT SOURCE TERMS USED IN ACCIDENT CALCULATIONS AND OPERATIONAL CONSIDERATIONS IS DISCUSSED. ***THREE ACCIDENTS (CORE DISASSEMBLY IN MELTDOWN, SECONDARY LOOP RUPTURE, GAS WASTE TANK RUPTURE) WERE DISCUSSED IN SAR. ESSENTIAL FEATURES OF THESE ANALYSES ARE REPEATED HERE. (A) 2/3 OF FUEL MELTS BUT 99% OF IODINE REACTS WITH SODIUM, (B) ALTHOUGH SOME PRIMARY SODIUM MIGHT LEAK THROUGH TO THE SECONDARY SYSTEM, IT WOULD SPILL IN LOW OXYGEN AREAS. ONLY 0.0187 CU FT OF PRIMARY SODIUM WAS ESTIMATED RELEASED. (C) ASSUMED ONE TANK RELEASES 15% OF GASEOUS FP FROM CENTER FUEL ELEMENTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-030253 *CONTINUED*
FISSION GAS RELEASE + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REPORT, SAR + WASTE STORAGE + ACCIDENT,
HYPOTHETICAL + WASTE TREATMENT, GAS + SEFOR (RE) + RESPONSE TO NRC QUESTION + REACTOR, LMFBR

7-030370 DATED 1967
LEONARD BH + PLANCHAN JU
CONSTRUCTION AND DESIGN PROBLEMS OF SODIUM COOLED FAST REACTORS
BECHTEL CORP., SAN FRANCISCO
CONF-670413 (AND SUPPL.) +. 6 PAGES, PP. 6.37-43 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO,
APRIL 10-12, 1967, HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY, 1967

EFFECTS OF COOLANT PROPERTIES ON CONSTRUCTION OF NA-COOLED FAST REACTORS ARE OUTLINED. PROBLEMS
WITH PLANT DESIGN FEATURES ARE REVIEWED. THE IMPORTANCE OF QUALITY ASSURANCE IS DISCUSSED.

DESIGN CRITERIA + REACTOR, FAST + SODIUM + COOLANT CHEMISTRY + METAL, LIQUID + REACTOR, LMFBR

7-030790 DATED 1968
CASTLEMAN AW + TANG IN
FISSION PRODUCT VAPORIZATION FROM SODIUM SYSTEMS
BROOKHAVEN NATIONAL LABORATORY, UPTON, NY
BNL-13099 +. NOVEMBER 1968, 7 TABLES, 5 FIGURES, 28P.

A KNOWLEDGE OF THE FACTORS GOVERNING THE RATE AND EXTENT OF FISSION PRODUCT VAPORIZATION FROM
LIQUID SODIUM IS IMPORTANT IN BOTH OPERATIONAL AND SAFETY CONSIDERATIONS OF FAST REACTORS. THE
RESULTS OF ANALYTICAL AND EXPERIMENTAL STUDIES OF THE THERMODYNAMIC PROPERTIES AND TRANSPORT
BEHAVIOR OF SOME FISSION PRODUCTS IN SODIUM ARE PRESENTED.

BARIUM + CESIUM + REACTOR, FAST + FISSION PRODUCT RELEASE + FISSION PRODUCT, VOLATILE + IODINE + REACTOR, LMCR
+ SODIUM + STRONTIUM + TELLURIUM + THERMODYNAMICS + FISSION PRODUCT TRANSPORT + ANTIMONY + CHEMICAL KINETICS
+ THEORETICAL INVESTIGATION + RUBIDIUM + REACTOR, LMFBR

7-031003 ALSO IN CATEGORY 12 DATED 1968
CLOUGH WS
THE PARTITION OF IODINE BETWEEN LIQUID SODIUM AND THE GAS PHASE AT 500C
UNITED KINGDOM ATOMIC ENERGY AUTHORITY, HARWELL
AERE-R-5848 +. 16 PAGES, 6 FIGURES, 1 TABLE, 8 REFERENCES, AUGUST 1968

THE EFFECT OF INCREASING THE IODINE CONCENTRATION IS TO DECREASE THE VOLATILITY OF IODINE
RELATIVE TO THAT OF SODIUM. AT THE HIGHEST IODINE CONCENTRATION STUDIED THE IODINE IS ABOUT 30%
LESS VOLATILE THAN AT THE LOWEST IODINE CONCENTRATION LEVELS.

AVAILABILITY - SCIENTIFIC ADMINISTRATION OFFICE, ATOMIC ENERGY RESEARCH ESTABLISHMENT, HARWELL, DIDCOT,
BERKSHIRE, ENGLAND

FISSION GAS RELEASE + IODINE + REACTOR, LMCR + SODIUM + REACTOR + FUEL REPROCESSING + REACTOR, LMFBR

7-031068 ALSO IN CATEGORY 12 DATED 1968
LMFBR PROGRAM PLAN. VOL. 7 - FUELS AND MATERIALS
ARGONNE NATIONAL LAB., ILL.
WASH-1107 +. 457 PAGES, FIGURES, TABLES, REFERENCES, AUGUST 1968

THE FUELS AND MATERIALS ELEMENT OF THE PROGRAM PLAN COVERS RESEARCH AND DEVELOPMENT OF FUELS,
CLADDINGS, OTHER MATERIALS, AND FUEL-SYSTEM CONCEPTS FOR LMFBRs. INCLUDED IS ACQUISITION OF DATA
ON INTRINSIC PROPERTIES OF MATERIALS AND ON THE NORMAL, OFF-NORMAL, AND TRANSIENT-CONDITION
PERFORMANCE OF FUEL SYSTEMS FOR SPECIFIC REACTOR MISSIONS. THE MAJOR EMPHASIS IS ON THE URANIUM-
PLUTONIUM CYCLE. A MAJOR INTERFACE OF THIS ELEMENT IS WITH THE CORE DESIGN ELEMENT, WHICH HAS
RESPONSIBILITY FOR DEVELOPMENT OF FUEL-ELEMENT DESIGNS. OTHER IMPORTANT INTERFACES ARE WITH THE
COMPONENTS, SODIUM TECHNOLOGY, AND FUEL RECYCLE ELEMENTS OF THE PROGRAM PLAN. THE FUELS AND
MATERIALS ELEMENT WILL PROVIDE PROPERTY DATA, INFORMATION, AND TECHNICAL EVALUATIONS TO THESE
ELEMENTS IN AREAS WITHIN THEIR SCOPE OF RESPONSIBILITY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ALLOY + CARBIDE + CERAMICS + DESIGN CRITERIA + FABRICATION + CLADDING + OXIDE + PLUTONIUM + TESTING + THERMAL
ANALYSIS + URANIUM + STEEL, STAINLESS + NITRIDE + PROPERTY, PHYSICAL + REACTOR, LMFBR + JACOBS + FUEL ELEMENTS

7-031338 DATED 1968
BREFM WF
RADIOACTIVE MASS TRANSPORT IN FFTF PRIMARY PIPING
BATTELLE NORTHWEST
BNWL-608 +. 29 PAGES, TABLES, REFERENCES, AUGUST 1968

MUCH OF THIS REPORT IS CONCERNED WITH AN ESTIMATE OF THE MAGNITUDE OF CORROSION AND DEPOSITION IN
THE FFTF DRIVER CORE AND HEAT TRANSPORT SYSTEM USING PRESENTLY AVAILABLE INFORMATION, AN
EVALUATION OF THE CONSEQUENCES OF THE RESULTS, AND THE LIMITATIONS OF THE METHODS USED TO OBTAIN
THEM. AN EVALUATION IS MADE OF THE VARIOUS TYPES OF EXPERIMENTS THAT CAN BE PERFORMED TO OBTAIN
MEANINGFUL DATA THAT WILL ENABLE US TO PREDICT LOCATION AND MAGNITUDE OF RADIOACTIVE MATERIAL

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-031358 *CONTINUED*

DEPOSITION IN FFTF AND FUTURE LMFBR REACTORS. THE DISTRIBUTION OF RADIOACTIVE ISOTOPES (SPECIES) THROUGHOUT A SYSTEM IN WHICH MASS TRANSFER IS OCCURRING MAY NOT COINCIDE WITH THE GROSS DISTRIBUTION OF MASS-TRANSFERRED MATERIAL. THEREFORE, CALCULATED RADIATION FIELDS COULD BE IN ERROR BY SEVERAL ORDERS OF MAGNITUDE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

COBALT + DEPOSITION + CLADDING + REACTOR, FAST + IN PILE EXPERIMENT + SODIUM + REACTOR, TEST + OUT OF PILE EXPERIMENT + ACTIVATION PRODUCT + CHROMIUM + MANGANESE + STEEL, STAINLESS + FFTF (TR) + CORROSION + RADIOISOTOPE + MASS TRANSFER + THERMAL PROPERTY + REACTOR, LMFBR

7-031339

DATED 1967

LOCKHART RW + BLAIR KC
SODIUM MASS TRANSFER. XXIII MODEL FOR CORROSION MOCKUP OF SODIUM COOLED REACTORS
GENERAL ELECTRIC COMPANY, SUNNYVALE
GEAP-4881 +. 85 PAGES, FIGURES, TABLES, JULY 1967

A METHOD OF PROVIDING A CORROSION MOCKUP FOR A LARGE SODIUM SYSTEM IS PRESENTED. THE MOCKUP LOOP IS SO DESIGNED THAT THE CORROSION PROCESSES PROVIDE THE SAME AMOUNT OF CHANGE IN SOLUTE, DELTA-S, (CORRODED MATERIAL) IN SPECIFIED LOCATIONS IN THE MOCKUP AS WOULD BE EXPECTED IN A LARGE PLANT SYSTEM. IT IS PROPOSED THAT THE EFFECTS OF SODIUM EXPOSURE IN THE MOCKUP WILL BE IDENTICAL TO THAT IN THE LARGE SYSTEM WHICH MAY BE 10 TO 100,000 TIMES LARGER IN CAPACITY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + IN PILE EXPERIMENT + REACTOR, LMCR + MOCKUP + SODIUM + REACTOR, TEST + ANALYTICAL MODEL + OUT OF PILE EXPERIMENT + EXTRAPOLATION FACTOR + FFTF (TR) + CORROSION + MASS TRANSFER + RATE + REACTOR, LMFBR

7-031340

ALSO IN CATEGORY 11

DATED 1968

MOBERLY JW + BARLOW M + GARRISON MC + PLANTING PJ
INTERACTION OF LIQUID SODIUM WITH 304 STAINLESS STEEL
STANFORD RESEARCH INSTITUTE, MENLO PARK, CALIF.
TID-24602 +. 28 PAGES, 20 FIGURES, JANUARY 31, 1968

IN PARTICULAR IT HAS BEEN OBSERVED THAT - (1) AFTER EXPOSURE OF THE FOILS AT 1200 F, SEVERE GRAIN BOUNDARY ATTACK OCCURS. (2) SURFACE PRECIPITATES, THOUGHT NOT TO BE SODIUM OR SODIUM OXIDE, ARE OBSERVED ON THE FOILS. (3) THE PRECIPITATES, WHICH MAY CONSIST OF ONE OR MORE COMPOUNDS, ARE ALSO FORMED ALONG AND WITHIN THE GRAIN BOUNDARIES AND HAVE A DEPENDENCE ON ORIENTATION OF THE 304 STAINLESS STEEL GRAINS. (4) ELECTRON DIFFRACTION STUDIES HAVE NOT AS YET IDENTIFIED THE COMPOSITION OF THESE PRECIPITATES. (5) FINALLY, TRANSMISSION ELECTRON MICROSCOPY IS AN EXCELLENT TECHNIQUE FOR FULLY EVALUATING ANY CLEANING PROCEDURE FOR METAL FOILS AFTER LIQUID SODIUM EXPOSURE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CHEMICAL REACTION + CLADDING + REACTOR, LMCR + SODIUM + REACTOR, TEST + STEEL, STAINLESS + CORROSION + COOLANT CHEMISTRY + ELECTRON MICROSCOPY + REACTOR, LMFBR

7-031505

ALSO IN CATEGORY 12

DATED 1968

GOLDSMITH P + WELLS AC + CUX LC
AEROSOL EMISSION FROM CHROMIUM ALLOYS WHEN HEATED IN AIR AND IN CARBON DIOXIDE/1% CARBON MONOXIDE
UNITED KINGDOM ATOMIC ENERGY AUTHORITY, HARWELL, BERKSHIRE
AERE-R-5426 +. 20 PAGES, 5 FIGURES, 4 TABLES, 6 REFERENCES, SEPTEMBER 1968

WHEN 80-20 NICKEL CHROMIUM ALLOY AND 20-23 STAINLESS STEEL ARE HEATED IN AIR, A COPIOUS SUPPLY OF SUBMICRON PARTICLES IS EMITTED FROM THE HOT ALLOYS AT TEMPERATURES OVER 850 C. THESE PARTICLES HAVE BEEN FOUND TO BE COMPOSED OF A MIXTURE OF CHROMIUM TRIOXIDE AND CHROMIUM SESQUIOXIDE IN THE RATIO 2-1. AT TEMPERATURES OF 1200 C AND BELOW, WELL OVER 90% OF THE EMISSION IS DUE TO THE PRODUCTION OF THIS SUBMICRON AEROSOL. AT 1250 C IN DRY AIR, AS MUCH AS 62% OF THE CHROMIUM LOSS FROM THE HEATED ALLOYS WAS IN THE FORM OF LARGE PARTICLES OF SCALE. MOISTURE INHIBITED THIS SCALING MECHANISM BUT ENHANCED THE SUBMICRON AEROSOL PRODUCTION RATE.

AVAILABILITY - SCIENTIFIC ADMINISTRATION OFFICE, ATOMIC ENERGY RESEARCH ESTABLISHMENT, HARWELL, DIDCOT, BERKSHIRE, ENGLAND

AEROSOL + ALLOY + REACTOR, LMCR + AEROSOL PRODUCTION + CHROMIUM + NICKEL + STEEL, STAINLESS + HIGH TEMPERATURE + FUEL REPROCESSING + REACTOR, LMFBR

7-032250

DATED 1967

LINDNER L + VON BAECKMANN A
METHOD FOR CHEMICAL ANALYTICAL DETERMINATION OF PLUTONIUM AND URANIUM IN OXIDIC NUCLEAR FUELS
KERNFORSCHUNGSZENTRUM, KARLSRUHE
EURFNR-525 +. 43 PAGES, 2 FIGURES, 4 TABLES, 2 REFERENCES, NOVEMBER 1967

IN THE FRAMEWORK OF THE FAST BREEDER PROJECT, DETERMINATION OF PU IN PUO₂ (DRIED AT 400 C) AND OF

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-032250 *CONTINUED*

PU AND U IN PUO₂/UO₂ MIXTURES AS WELL AS IN MIXED-OXIDE PELLETS (SINTERED AT 1600 C) WERE REQUIRED. REPORT DESCRIBES ANALYTICAL METHODS.

AVAILABILITY - U. S. DEPOSITORY LIBRARIES IN THE U.S. AND OVER SEAS

REACTOR, FAST + ANALYTICAL TECHNIQUE + GERMANY + URANIUM DIOXIDE + PLUTONIUM DIOXIDE + REACTOR, LMFBR + CALIBRATION

7-032253

DATED 1968

TANG IN + CASTLEMAN AW + MUNKELWITZ HR
ON THE TRANSPORT OF FISSION-PRODUCT CESIUM FROM SODIUM
BROOKHAVEN NATIONAL LABORATORY
BNL-1309d +. 19 PAGES, 6 FIGURES, 10 REFERENCES, NOVEMBER 1968

INVESTIGATED AT 730 K THE VAPORIZATION OF FISSION-PRODUCT CESIUM FROM LIQUID SODIUM INTO FLOWING INERT GASES. CESIUM RELEASE RATE DEPENDED ON BOTH THE LIQUID DEPTH OF SODIUM AND MOLECULAR WEIGHT OF THE INERT CARRIER GAS. DEVELOPED A GENERAL RELEASE EQUATION BY CONSIDERING DIFFUSION IN BOTH THE LIQUID AND THE GAS PHASES. EQUATION REPRESENTS EXPERIMENTAL RESULTS VERY WELL.

AVAILABILITY - I. N. TANG, A. W. CASTLEMAN OR H. R. MUNKELWITZ, BROOKHAVEN NATIONAL LABORATORY, UPTON, NEW YORK

CESIUM + REACTOR, FAST + SODIUM + URANIUM + URANIUM CARBIDE + ANALYTICAL MODEL + FISSION PRODUCT TRANSPORT + THEORETICAL INVESTIGATION + REACTOR, LMFBR

7-032265

ALSO IN CATEGORY 18

DATED 1968

HANSON JE
EXPERIMENT DESCRIPTION AND HAZARDS EVALUATION FOR THE PACIFIC NORTHWEST LABORATORY MIXED OXIDE (UO₂-PUO₂) IRRADIATIONS IN EBR-11, TASK A, SUBTAS, 1 IRRADIATIONS
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-650 +. 190 PAGES, FIGURES, TABLES, REFERENCES, JULY 1968

RECENT DEVELOPMENTS IN MIXED OXIDE FUEL IRRADIATION EXPERIENCE PERMITTED CONSIDERATION OF SOME REDUCTION IN NUMBER OF FUEL-DESIGN PARAMETERS IN THE FTR DRIVER FUEL DEVELOPMENT PROGRAM. GENERALLY, IN ADDITION TO PROVIDING FAST-FLUX IRRADIATED FUEL SPECIMENS FOR TRANSIENT IRRADIATION, THE 19-CAPSULE SUBASSEMBLIES WILL ALSO INCLUDE SPECIMENS FOR EVALUATING EFFECTS OF CLADDING THICKNESS REDUCTION AND ALTERNATE FUEL FORMS ON FUEL PERFORMANCE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + FUEL INTEGRITY + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + HAZARDS ANALYSIS + IN PILE EXPERIMENT + URANIUM DIOXIDE + PLUTONIUM DIOXIDE + REACTOR, LMFBR + FUEL ELEMENTS

7-0325d7

ALSO IN CATEGORIES 17 AND 18

DATED 1968

STEWART DC + MACIAS ES + BASILE LJ + MILSTED J
BUILDUP OF RADIOACTIVE PRODUCTS IN THERMAL REACTORS. II. RA-226 TARGETS
ARGONNE NATIONAL LAB., ILL.
ANL-7478 +. 133 PAGES, FIGURES, TABLES, REFERENCES, JULY 1968

THIS REPORT CONTINUES A SERIES DESCRIBING THE BUILDUP OF ISOTOPIC L, PHYSICS, AND COMPONENT DEVELOPMENT), EBR-2 OPERATION, (2) GENERAL NEUTRON FLUXES AT VARIOUS LEVELS. REFERENCES ARE GIVEN IN ITS DEVELOPMENT, CHEMISTRY, (3) NUCLEAR SAFETY. WEAR TESTS WERE RUN WITH USED ARE DESCRIBED IN DETAIL. AC-227 IS ALSO OF SOME INTEREST AS AT 300 F, SODIUM LUBRICATES. BUT, AT 500 F, IT WETS SURFACE AND A ALPHA EMITTERS. THIS APPLICATION, HOWEVER, WILL PROBABLY ALWAYS LIQUID SODIUM FLASHING, PRESS.-WAVE PROPAGATION, SUPERHEAT, COOLANT ANY QUANTITY. THERE ARE ALSO SEVERAL INTENSE GAMMA EMITTERS AMONG TRANSFER, AND SODIUM-AIR REACTION. PRESENT SOME PROBLEMS IN HANDLING THE MATERIAL FOR HEAT SOURCE APPLICATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CHEMICAL REACTION + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + MOLYBDENUM + REACTOR, SAFETY RESEARCH + SODIUM + REACTOR, TEST + TESTING + TREAT (PRR) + REACTOR COOLANT + METAL, LIQUID + REACTOR, PULSED + LUBRICATION + R AND D PROGRAM + REACTOR, LMFBR

7-032699

ALSO IN CATEGORY 12

DATED 1969

BREHM WF + KUVACEVICH EA + SHANNON DW
RADIOISOTOPE TRANSPORT AND DEPOSITION IN FLOWING SODIUM
BATTELLE MEMORIAL INSTITUTE
BNWL-969 +. 24 PAGES, 7 FIGURES, 2 TABLES, 10 REFERENCES, FEBRUARY 1969

STUDIED TRANSPORT AND DEPOSITION OF RADIOISOTOPES RELEASED TO FLOWING SODIUM FROM CORROSION OF IRRADIATED STAINLESS STEEL. QUANTITATIVE MEASUREMENTS OF DEPOSITION CAN BE MADE WITH SENSITIVE COUNTING TECHNIQUES. THE RESULTS OF THE STUDY INDICATE THAT CR-51 AND CO-60 ARE TRANSPORTED PRIMARILY BY ISOTOPE EXCHANGE AND DO NOT RAPIDLY MIGRATE TO LOWER TEMP. REGIONS OF LOOP SYSTEMS. HOWEVER, MN-54 IS RELEASED IN LARGER QUANTITIES TO DEPOSIT THROUGHOUT THE TEST LOOPS. PREFERENTIAL DEPOSITION SITES ARE (1) IMMEDIATELY DOWNSTREAM FROM THE SOURCE, (2) AT MINIMUM

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-032699 *CONTINUED*

TEMPERATURE LOCATIONS, AND (3) IN OXYGEN COLD TRAPS. THESE RESULTS INDICATE MN-54 IS TRANSPORTED BOTH BY EXCHANGE AND BY SOLUTION- AND- PRECIPITATION. DEPOSITED MN-54 MIGRATED RAPIDLY TO THE COLD TRAP, DEPOSITED CR-51 AND CO-60 MIGRATED SLOWLY BY EXCHANGE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DEPOSITION + REACTOR, LMCR + SODIUM + FISSION PRODUCT TRANSPORT + STEEL, STAINLESS + CORROSION + TRANSPORT PROPERTY + REACTOR, LMFBR

7-032843 ALSO IN CATEGORIES 11 AND 18 DATED 1969

BULBA E + UNDERHILL D

APPLICATION OF A PEBBLE BED HEAT-SINK TO THE FAST FLUX TEST FACILITY CLEANUP SYSTEM

HARVARD UNIVERSITY

NYO-841-17 +. 19 PAGES, 2 FIGURES, 9 REFERENCES, MARCH 1969

CONTAINS MATHEMATICAL EQUATIONS FOR GAS TEMPERATURES FOLLOWING SODIUM RELEASE AND BURNING INTO FFTF VENTILATION SYSTEM. ALLOWS FOR VARIABLE INLET GAS TEMPERATURE BUT DOES NOT ACCOUNT FOR SODIUM DEPOSITION PRIOR TO RELEASES. EXPERIMENTAL WORK ON THE LATTER NEEDED, ALSO ON GAS-FLOW SCHEMES AND ON BEST STRATEGY FOR PLACING SYSTEM IN OPERATION IN ACCIDENTS OF VARIOUS MAGNITUDES.

DEPOSITION + REACTOR, FAST + FIRE + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + SODIUM + REACTOR, TEST + VENTILATION SYSTEM + FFTF (TR) + HEAT SINK + MATHEMATICAL TREATMENT + R AND D PROGRAM + GAS + REACTOR, LMFBR + PACKED BED + THERMAL TRANSIENT

7-032881 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION 4 - RADWASTE LINE BREAK

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-20 +. 3 PAGES, PAGE 13-17 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DISCUSS THE CONTROL OF THE WASTE GAS BATCH TANK VALVES AND THE MANNER IN WHICH THE CONTROLLER WOULD PREVENT THE CONTENTS OF THE BATCH TANKS FROM BEING DISCHARGED INTO THE DECAY TANK VAULT AND FROM THERE TO THE ENVIRONS IN THE EVENT OF A PIPE BREAK BETWEEN THE PIPE TUNNEL AND THE INLET VALVES TO THE DECAY TANKS. ***ANSWER ON PAGES 13 THRU 15.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + FISSION PRODUCT RELEASE + REACTOR, LMCR + WASTE STORAGE + WASTE DISPOSAL, GAS + WASTE MANAGEMENT + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

7-033156 DATED 1968

LMFBR PROGRAM PLAN. VOLUME 5 - SODIUM TECHNOLOGY

ARGONNE NATIONAL LABORATORY

WASH-1105 +. 362 PAGES, 17 FIGURES, 9 TABLES, AUGUST 1968

THE SODIUM TECHNOLOGY ELEMENT OF THE PROGRAM PLAN IS CONCERNED WITH THE DEVELOPMENT OF A BASE TECHNOLOGY ADEQUATE TO PROVIDE REASONABLE ASSURANCE OF LMFBR PLANT LIFE AND RELIABILITY. THE DEVELOPMENT OF SODIUM AND COVER-GAS PURIFICATION SYSTEMS EXTENDS BEYOND BASE TECHNOLOGY AND METHODS DEVELOPMENT AND INCLUDES THE COMPONENT DEVELOPMENT AND PROOF-TEST STAGES. ALL OTHER LMFBR PROGRAM ELEMENTS RELY ON THE SODIUM TECHNOLOGY ELEMENT FOR ANSWERS TO THEIR SPECIFIC PROBLEM AREAS DEALING WITH THE EFFECTS OF THE SODIUM ENVIRONMENT ON THEIR OPERATIONS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

AIR CLEANING + REACTOR, BREEDER + REACTOR, FAST + SODIUM + FISSION PRODUCT TRANSPORT + CORROSION + COOLANT CHEMISTRY + TECHNICAL SPECIFICATIONS + METAL, ALKALI + METAL, LIQUID + RESEARCH CONTRACT + IMPURITY + REACTOR, LMFBR + JACOBS

7-034347 ALSO IN CATEGORY 16 DATED 1969

DICKSON CR + START GE + MARKEE EH

AERODYNAMIC EFFECTS OF THE EBR-II REACTOR COMPLEX ON EFFLUENT CONCENTRATION

AIR RESOURCES LABORATORIES, ENVIRONMENTAL SCIENCE SERVICES ADMINISTRATION

15 PAGES, 17 FIGURES, 2 TABLES, 9 REFERENCES, NUCLEAR SAFETY 10(3), PAGES 228-42 (MAY-JUNE 1969)

DISPERSION OF A TRACER RELEASED FROM THE DOWNWIND SIDE OF THE EXPERIMENTAL BREEDER REACTOR II (EBR-II) WAS MEASURED IN STABLE AND UNSTABLE ATMOSPHERES. THE TRACER, URANINE DYE, WAS SAMPLED ON FIVE CONCENTRIC SAMPLING ARCS BETWEEN 30 AND 600 M FROM THE RELEASE POINT. THE DIFFUSION RATE IN THE EBR-II WAKE CONVERGED TO THE FREE ATMOSPHERE RATE BETWEEN 400 AND 600 M DOWNWIND OF THE SOURCE POINT.

AEROSOL + AIRBORNE RELEASE + ATMOSPHERIC DIFFUSION + EBR 1 AND 2 (RE) + EFFLUENT + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + TESTING + TRACER, RADIOACTIVE + METEOROLOGY

7-034360 DATED 1969

KLOONTZ RL + MOREWITZ HA

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-034360 *CONTINUED*
AN EVALUATION OF THE BEHAVIOR OF AEROSOLS PRODUCED BY THE LMFBR DBA
ATOMIC INTERNATIONAL, CANOGA PARK, CALIF.
AI-AEC-MEMO-12761 (REV.1) +. 17 PAGES, 3 FIGURES, 8 REFERENCES, MAY 15, 1969

THE EXISTING EXPERIMENTAL DATA ON UO₂ AND PUO₂ AEROSOLS ARE EVALUATED AND AN EMPIRICAL MODEL OF THE BEHAVIOR OF AEROSOLS PRODUCED IN THE PRIMARY CONTAINMENT BY THE LMFBR DBA IS DEVELOPED. THIS MODEL IS BASED ON THE WELL KNOWN STIRRED SETTLING LAW FOR AEROSOLS. IT PERMITS A CONSERVATIVE, BUT PHYSICALLY REALISTIC, ESTIMATE OF THE SPECIFIC REDUCTION FACTOR FOR THE AEROSOL CONCENTRATION IN THE PRIMARY CONTAINMENT FOR ANY GIVEN PLANT DUE TO AGGLOMERATION AND SUBSEQUENT AND USEFUL INTERIM GUIDE FOR BOTH LMFBR DESIGNERS AND ORL STAFF. AVAL: AVAILABILITY - R.L. KOONTZ/H.A. MOREWITZ, ATOMIC INTERNATIONAL, CANOGA PARK, CALIFORNIA

AEROSOL + AIRBORNE RELEASE + FALLOUT + ACCIDENT, MAXIMUM CREDIBLE (MCA) + PARTICLE SIZE + SODIUM + ANALYTICAL MODEL + AEROSOL PRODUCTION + AEROSOL PROPERTIES + AGGLOMERATE + ACCIDENT MODEL + URANIUM DIOXIDE + THEORETICAL INVESTIGATION + PLUTONIUM DIOXIDE + REACTOR, LMFBR

7-034361 DATED 1969
ZWETZIG GB
SURVEY OF FISSION-AND-CORROSION-PRODUCT ACTIVITY IN SODIUM-OR NAK-COOLED REACTORS
ATOMIC INTERNATIONAL, CANOGA PARK, CALIF.
AI-AEC-MEMO-12790 +. 13 PAGES, FEBRUARY 28, 1969

IN THE EVENT OF SIGNIFICANT FUEL CLADDING FAILURE, THE FOLLOWING FISSION PRODUCTS WILL PROBABLY BE PRESENT AT THE LOCATIONS INDICATED - COVER GAS (KR-85, XE-133, AND XE-135), COOLANT (I-131, BA-LA-140, ZR-NB-95, AND CS-137), PRIMARY SYSTEM WALLS (ZR-NB-95, XE-141, CE-144, SR-89, SR-90, CS-137, RU-106, AND BA-LA-140). WHERE COOLANT TEMPERATURES ARE 1000 F OR ABOVE, CORROSION OF CORE STRUCTURAL AND CLADDING MATERIALS IS LIKELY TO OCCUR. SOME OF THIS MATERIAL WILL BE RADIOACTIVE, AND WILL DEPOSIT ON THE WALLS OF THE PRIMARY SYSTEM. THE PRINCIPAL ISOTOPES WHICH WILL BE FOUND ARE MN-54 AND CO-60. THE EFFECTIVENESS OF COLD TRAPS IN THE REMOVAL OF FISSION AND CORROSION PRODUCTS OTHER THAN CA-137 REMAINS TO BE DETERMINED, FOR SYSTEMS NOT CONTAINING LARGE QUANTITIES OF CARBONACEOUS MATERIALS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

BARIUM + CERIUM + CESIUM + FISSION PRODUCT RELEASE + FISSION PRODUCT, IODINE + KRYPTON + RUTHENIUM + SODIUM + STRONTIUM + XENON + ZIRCONIUM + FISSION PRODUCT TRANSPORT + FILTER, TRAP + COOLANT CHEMISTRY + REACTOR, LMFBR

7-034421 DATED 1968
GUPTA D + HELLER F + SCHROEDER R
THE AMOUNTS OF FISSION PRODUCT NUCLIDES PRODUCED IN PU-239-FUELLED FAST REACTORS AND THE RELATED HEAT GENERATION AFTER SHUT-DOWN
KERNFORSCHUNGSZENTRUM, KARLSRUHE, WEST GERMANY
KFK-722 +. 19 PAGES, 4 FIGURES, 7 TABLES, 31 REFERENCES, NOVEMBER 1968

TO OBTAIN THE DECAY HEAT GENERATED BY THE FISSION PRODUCTS OF PU-239, FIRST THE CONCENTRATION OF INDIVIDUAL FISSION PRODUCTS WAS CALCULATED AS A FUNCTION OF IN-PILE TIME, COOLING TIME, AND SPECIFIC POWER OF THE REACTOR. THE EQUATIONS USED AND THE RESULTS OF THESE CALCULATIONS ARE PRESENTED AND THE DECAY POWER OF INDIVIDUAL FISSION PRODUCT NUCLIDES WAS CALCULATED AND, FINALLY, THESE QUANTITIES WERE SUMMED UP TO RESULT IN DATA FOR GROSS FISSION PRODUCTS.

AVAILABILITY - USAEC DEPOSITORY LIBRARIES FOR THE U.S. AND OVERSEAS

DECAY HEAT + REACTOR, FAST + FUEL BURNUP + PLUTONIUM + FISSION PRODUCT ACTIVITY, GROSS + RADIONUCLIDE, INDUCED + REACTOR, LMFBR

7-034673 ALSO IN CATEGORIES 11 AND 18 DATED 1969
PATRIARCA P + RUCKER DJ
FUELS AND MATERIALS DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1968
OAK RIDGE NATIONAL LAB., TENN.
ORNL-4390 +. 308 PAGES, FIGURES, TABLES, REFERENCES, MAY 1969

ONE OF A SERIES OF SIMILAR REPORTS. TOPICS - FUELS AND CLADDINGS, COOLANTS, RADIATION DAMAGE TO STRUCTURAL MATERIALS, TECHNIQUES OF FABRICATION AND TESTING, ENGINEERING PROPERTIES OF REACTOR MATERIALS, DEVELOPMENT OF FAST REACTOR OXIDE FUELS, AND LMFBR FUELS, FISSION-GAS RELEASE, ZIRCONIUM METALLURGY, ALKALI-METAL CORROSION, TUNGSTEN METALLURGY, AND CLAD MATERIALS FOR SPACE HEAT SOURCES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + CLADDING + REACTOR, FAST + FISSION GAS RELEASE + ZIRCONIUM + ORNL + CORROSION + TUNGSTEN + IRRADIATION TESTING + DIRECT ENERGY CONVERSION DEVICES + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

7-035866 DATED 1968
DOE CB + SMITH RK + EBERSOLE EK
EXPOSED-FUEL CALIBRATION STUDIES IN EBR-II

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-035866 *CONTINUED*
ARGONNE NATIONAL LABORATORY
1 PAGE, 1 REFERENCE, ANS TRANSACTIONS, 11(2), PAGE 179 (1968)

THREE SEPARATE FISSION-PRODUCT DETECTION SYSTEMS ARE USED AT EBR-II TO ANNUNCIATE THE RELEASE OF FISSION PRODUCTS TO THE PRIMARY SYSTEM FROM A CLADDING DEFECT. THESE ARE THE FGM AND FERD SYSTEMS AND A METHOD BASED ON THE RADIO-METRIC ASSAY OF COVER-GAS SAMPLES FOR XE-133 AND XE-135. THE FGM (FISSION-GAS MONITOR) MONITORS THE CONCENTRATIONS OF SHORT-LIVED RARE-GAS SPECIES (KR-88, KR-89, AND XE-138) IN THE COVER-GAS SYSTEM ON A CONTINUOUS BASIS. THE FERD (FUEL ELEMENT RUPTURE DETECTOR) SENSES THE LEVEL OF DELAYED-NEUTRON-EMITTING FISSION PRODUCTS IN A BYPASS COOLANT STREAM. THE THIRD TECHNIQUE INVOLVES THE ANALYSIS OF PERIODIC COVER-GAS SAMPLES FOR XE-133 AND XE-135.

AVAILABILITY - C.B. DOE/R.R. SMITH/E.R. EBERSOLE, ARGONNE NATIONAL LABORATORY, IDAHO FACILITIES, P.O. BOX 2528, IDAHO, FALLS, IDAHO 83401

EBR 1 AND 2 (KE) + REACTOR, FAST + FISSION PRODUCT RELEASE + FAILURE, FUEL ELEMENT + REACTOR, LMCR + REACTOR, TEST + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + MONITORING SYSTEM, RADIATION + REACTOR, LMFR + RADIATION MONITORS

7-036473 ALSO IN CATEGORY 11 DATED 1969
PROPOSAL ON VENTED CONTAINMENT FOR A LIQUID METAL FAST BREEDER REACTOR (LMFR)
ARGONNE NATIONAL LABORATORY
ANL-7410 +. 445 PAGES, PGS. 216-323, TABLES, FIGURES, REFERENCES, JANUARY 1969

THIS IS A PROPOSAL FOR ALLEVIATING ACCIDENT CONSEQUENCES BY VENTING THE SECONDARY CONTAINMENT ATMOSPHERE THROUGH A HEAT SINK FILTERING MEDIA TO A STANDBY CONTROL VOLUME. ENOUGH OF A STUDY HAS BEEN MADE TO ASSURE THAT REASONABLY PRACTICAL DESIGN OF HEAT SINK-FILTER AND AUXILIARY CONTAINMENT BUILDING CAN BE OBTAINED WHICH WILL RESULT IN A SIGNIFICANT REDUCTION IN DESIGN PRESSURE FOR A WHOLE CONTAINMENT SYSTEM. FURTHER DETAILED STUDY OF THE TWO HEAT SINKS WHICH WERE ASSUMED, COUPLED WITH REFINEMENTS IN THE CURRENT NUMERICAL DYNAMIC ANALYSIS, IS NECESSARY BEFORE A SPECIFIC DESIGN CAN BE REALIZED AND ITS PERFORMANCE PROVED. EXPERIMENTAL STUDIES OF BEHAVIOR OF SAND-GRAVEL BEDS WITH VARIABLE INLET GAS TEMPERATURES ARE RECOMMENDED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + CONTAINMENT, PRESSURE VENTING + EXPLOSION + PLUTONIUM + HEAT SINK + REACTOR, LMFR

7-037181 DATED 1968
LIPPS AJ
TASK 1 REPORT - EVALUATION OF VENTED-TO-COOLANT DESIGN FOR SODIUM-BONDED CARBIDE FUEL RODS
WESTINGHOUSE ELECTRIC CORPORATION, MADISON, PA.
WAKU-2000-31 +. 56 PAGES, FIGURES, TABLES, REFERENCE, FEBRUARY 1968

THIS SCOPING STUDY WAS UNDERTAKEN TO ASSESS THE FEASIBILITY OF VENTING SODIUM-BONDED MIXED CARBIDE FUEL ELEMENTS. THE VENTING-TO-COOLANT CONCEPT, BECAUSE OF ITS MORE DIRECT APPROACH, WAS THE VENTING CONCEPT SELECTED FOR A COMPARISON WITH A NONVENTED DESIGN. ALTHOUGH THERE IS SOME COST INCENTIVE FOR VENTING, THE PRELIMINARY COST DIFFERENCE APPEARS TO BE SMALL AND IS NOT THE DECIDING FACTOR. THE SAFETY CONSIDERATIONS MAY BE THE DECIDING FACTOR IN EVALUATING VENTED VERSUS NONVENTED FUEL CONCEPTS. THE MAJOR SAFETY PROBLEM IN A VENTED DESIGN IS THE ENTRAINMENT OF THE FISSION GAS IN THE COOLANT. THIS PROBLEM CAN BE ELIMINATED BY PLACING SURGE TANKS IN THE PRIMARY LOOP. THE SURGE TANK WOULD LOWER THE COOLANT VELOCITY SUFFICIENTLY TO ALLOW THE FISSION GAS TO ESCAPE TO THE COVER GAS. THE SAFETY PROBLEMS IN THE NONVENTED ROD CANNOT BE SOLVED AS EASILY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

DECONTAMINATION + DESIGN CRITERIA + FISSION GAS RELEASE + FAILURE, FUEL ELEMENT + KRYPTON + SODIUM + XENON + FILTER, TRAP + FISSION PRODUCT RETENTION + REACTOR, LMFR

7-037183 DATED 1969
SMITH RR + CISSEL DW + DOE CB + EBERSOLE ER + KIRN FS
LOCATING AND IDENTIFYING THE SOURCE OF THE MAY 24, 1967 FISSION-PRODUCT RELEASE IN EBR-II
ARGONNE NATIONAL LABORATORY, ILLINOIS
ANL-7543 +. 39 PAGES, 16 FIGURES, 6 TABLES, 5 REFERENCES, APRIL 1969

THE FIRST VERIFIED RELEASE OF FISSION PRODUCTS TO THE PRIMARY SODIUM COOLANT IN EBR-II OCCURRED ON MAY 24, 1967. PRIMARY ANNUNCIATION WAS GIVEN BY THE FGM (FISSION GAS MONITOR), WHICH INDICATED AN APPROXIMATELY 3000-FOLD INCREASE IN THE CONCENTRATION OF SHORT-LIVED KRYPTON AND XENON FISSION-PRODUCT SPECIES IN THE ARGON COVER-GAS SYSTEM. CONFIRMATORY EVIDENCE OF THE RELEASE WAS PROVIDED BY ANALYSES CONDUCTED FOR THE XE-133 AND XE-135 IN THE COVER GAS, AND FOR I-131 AND CS-137 IN THE PRIMARY COOLANT. BY REMOVING GROUPS OF SUSPECT SUBASSEMBLIES, FOLLOWED BY DEPRESSION OF THE REACTOR, THE ORIGIN OF THE FISSION-PRODUCT RELEASE WAS EVENTUALLY ATTRIBUTED TO AN EXPERIMENTAL UO2-PUO2 FUEL ELEMENT (HOV-4) IN IRRADIATION SUBASSEMBLY X011.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CESIUM + EBR 1 AND 2 (KE) + REACTOR, FAST + FISSION PRODUCT RELEASE + FISSION PRODUCT, IODINE + FAILURE, FUEL

CATEGORY 7
FISSION PRODUCT RELEASE, TRANSPORT, AND REMOVAL

7-037183 *CONTINUED*
ELEMENT + KRYPTON + REACTOR, LMCK + SODIUM + REACTOR, TEST + XENON + FAILURE, CLADDING

7-037184 DATED 1969
ZWEZIG GB + ROSE PF
INTERIM DESCRIPTION OF A COMPUTER PROGRAM (STP-1) FOR ESTIMATING THE DISTRIBUTION OF FISSION AND CORROSION
PRODUCT RADIOACTIVITY IN A LIQUID METAL-COOLED FAST BREEDER REACTOR
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
AI-AEC-12647 +. 41 PAGES, 6 FIGURES, JUNE 29, 1969

THE THEORY AND INTERIM STATUS OF THE SOURCE TERM COMPUTER PROGRAM (STP-1, VERSION 2) ARE DESCRIBED. THIS PROGRAM IS DESIGNED TO COMPUTE THE DISTRIBUTION IN A LIQUID METAL-COOLED FAST BREEDER REACTOR OF RADIOACTIVITY BOTH FROM FISSION PRODUCT LEAKAGE FROM THE FUEL AND FROM ACTIVATION AND CORROSION OF PRIMARY LOOP STRUCTURAL MATERIALS. THE PROGRAM ACCOUNTS FOR THIS DISTRIBUTION BETWEEN THE FUEL, CLADDING GAS SPACE, COOLANT, PRIMARY LOOP SURFACES, COLD TRAPS, AND COVER GAS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFILM

REACTOR, BREEDER + COMPUTER PROGRAM + FAILURE, FUEL ELEMENT + SODIUM + FISSION PRODUCT TRANSPORT + CORROSION + REACTOR, LMFB

7-037186 DATED 1968
PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON SODIUM TECHNOLOGY AND LARGE FAST REACTOR DESIGN
ARGONNE NATIONAL LABORATORY
ANL-7520(P1.2) +. 455 PAGES, FIGURES, TABLES, REFERENCES, 1968

THIS REPORT IS THE PROCEEDINGS OF THE FOURTH OF A SERIES OF TOPICAL CONFERENCES SPONSORED BY ARGONNE NATIONAL LABORATORY AND THE UNITED STATES ATOMIC ENERGY COMMISSION ON SPECIFIC TOPICS RELATED TO THE DEVELOPMENT OF FAST POWER BREEDER REACTORS. IT IS BOUND IN TWO VOLUMES - PART I CONTAINS PAPERS PRESENTED AT THE SESSIONS ON SODIUM TECHNOLOGY, AND PART II CONTAINS PAPERS PRESENTED AT THE SESSIONS ON LARGE FAST REACTOR DESIGN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFILM

CARBON + DECONTAMINATION + DESIGN CRITERIA + FIRE + OXYGEN + SODIUM + STEAM + OUT OF PILE EXPERIMENT + STEEL, STAINLESS + FILTER, TRAP + CORROSION + DESIGN STUDY + KREEP + REACTOR, LMFB

7-038775 DATED 1969
BRINARD JP + KIRKPATRICK DC
SODIUM LEAK STUDIES
LOS ALAMOS SCIENTIFIC LABORATORY, NEW MEXICO
LA-4129-MS +. 10 PAGES, 7 FIGURES, 1 TABLE, 1 REFERENCE, MAY 14, 1969

CONTAINMENT DEFECTS THAT CAN RESULT IN SODIUM LEAKAGE ARE BEING INVESTIGATED. THIS IS AN INTERIM REPORT ON SODIUM LEAKAGE THROUGH STAINLESS STEEL INTO AIR. THE PROBLEMS OF LOCATING POTENTIAL DEFECTS WITH HELIUM MASS SPECTROMETER LEAK TESTS IS INTENSIFIED BY THE FACT THAT THEY CAN EASILY BE OBSCURED BY WATER, GIL, AND SURFACE OXIDATION. UNDER THESE CONDITIONS, TREATMENT AT HIGH TEMPERATURE UNDER VACUUM OR HYDROGEN IS REQUIRED FOR DETECTION. HELIUM LEAKS AS LARGE AS 0.0001 ATM CM³/SEC DO NOT APPEAR TO BE A PROBLEM IN SODIUM SYSTEMS AT 400 C. HOWEVER, HELIUM LEAKS OF THIS SIZE AND SMALLER DO APPEAR TO BE A SERIOUS PROBLEM AT THE TEMPERATURE OF 650 C.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFILM

SODIUM + OUT OF PILE EXPERIMENT + STEEL, STAINLESS + HELIUM + LEAK RATE + LEAK + REACTOR, LMFB

7-106947 ALSO IN CATEGORIES 5 AND 18 DATED 1965
UKRENT D
PROCEEDINGS OF THE CONFERENCE ON SAFETY, FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS - OCTOBER 11-14, 1965
ARGONNE NATIONAL LABORATORY, ILL.
ANL-7120 +. 910 PPS, OCT. 11-14, 1965

PRESENTS THE TEXT OF 79 PAPERS PRESENTED AT THE CONFERENCE. TOPICS INCLUDE: PHYSICS UNCERTAINTIES AND SAFETY STUDIES; FAST REACTOR CONCEPTS; SAFETY OF LARGE FAST POWER REACTORS; FUELS AND FUEL BEHAVIOR; REACTIVITY COEFFICIENTS; MAXIMUM ACCIDENT AND CONTAINMENT; FISSION PRODUCTS, REACTIONS AND INTERACTIONS; AND SODIUM BOILING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + *REACTOR, FAST + *REACTOR, LMFR + DESIGN STUDY + REACTOR, GCFR + FUEL ELEMENTS + REACTIVITY COEFFICIENT + CONTAINMENT + PRESSURE TRANSIENT + FISSION PRODUCT RETENTION + BOILING + SODIUM + ACCIDENT ANALYSIS

CATEGORY 8
SOURCES OF ENERGY RELEASE UNDER ACCIDENT CONDITIONS

8-003904 ALSO IN CATEGORY 6 DATED 1963
WOLFE B + FRIEDMAN N + RILEY D
INFLUENCE OF THE DOPPLER EFFECT ON THE MELTDOWN ACCIDENT
GENERAL ELECTRIC COMPANY, SAN JOSE
22 PAGES, 6 FIGURES, 1 TABLE, 9 REFERENCES- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 171-192- OTS

BY MEANS OF A DIGITAL COMPUTER, THE INFLUENCE OF THE DOPPLER EFFECT IN THE CORE-DISASSEMBLY PROCESS FOLLOWING A MELTDOWN ACCIDENT IS EXAMINED WITH A BETHE-TAIT MODEL IN WHICH THE DOPPLER EFFECT, AS WELL AS CORE DISASSEMBLY, IS CONSIDERED. FOR THE WORST HYPOTHETICAL ACCIDENT DESCRIBED IN THE FERMI HAZARDS REPORT, MAXIMUM EXPLOSIVE ENERGY RELEASES ARE REDUCED FROM ABOUT 650 LB OF HIGH EXPLOSIVE ENERGY WITH NO DOPPLER EFFECT TO LESS THAN 20 LB IN THE CASE OF A DOPPLER COEFFICIENT.

BETHE TAIT + REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + COMPUTER, DIGITAL + DOPPLER COEFFICIENT + EQUATION OF STATE + REACTOR, FAST + FERMI (LM+BR) + FUEL MELTDOWN + REACTOR, LMCR + EXPLOSION + ACCIDENT, CORE DISRUPTIVE

8-006305 ALSO IN CATEGORY 18 DATED 1962
DIETZ HB
HAZARD ANALYSIS OF A SODIUM-WATER REACTION IN AN SRE PIPE GALLERY
ATOMICS INTERNATIONAL
25 PAGES- SEPTEMBER 28, 1962, NAA-SR-MEMO-7743- CFSTI

THE EFFECTS OF A POSSIBLE SODIUM-WATER REACTION IN THE SRE (SODIUM REACTOR EXPERIMENT) PIPE GALLERIES WERE ANALYZED. POSSIBILITIES FOR THIS REACTION WERE CONSIDERED VERY REMOTE.

REACTOR, LMCR + METAL WATER REACTION + SRE (RE)

8-006312 ALSO IN CATEGORIES 6 AND 17 DATED 1965
AUAMS RM + GLASSNER A
SODIUM AIR REACTION + FARET CRITICALS
ARGONNE NATIONAL LABORATORY
5 PAGES, 3 FIGURES, 1 TABLE, 2 REFERENCES- PAGES 11-15 FROM REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 19, 1965, ANL-7045- CFSTI

REGARDING EBR-II, CONTROL ROD CALIBRATION, DELAYED NEUTRON KINETICS, POWER COEFFICIENT, REACTIVITY GAIN BY FUEL REPLACEMENT. (FEW NUMERICAL VALUES GIVEN.) REGARDING FARET, FAIRLY DETAILED DISCUSSION OF TEMPERATURES AND PRESSURES CAUSED BY SODIUM FIRES. DATA ON REACTIVITY COEFFICIENTS, REACTIVITY WORTH AND PROMPT-NEUTRON LIFETIME.

REACTOR, BREEDER + CHEMICAL REACTION + CONTAINMENT, HIGH PRESSURE + CONTROL ROD WORTH + DELAYED NEUTRON + EBR 1 AND 2 (RE) + FIRE + REACTOR, LMCR + POWER COEFFICIENT + REACTIVITY COEFFICIENT + SODIUM + REACTOR, TEST + AIR + FFIF (TR) + PROMPT NEUTRON LIFETIME + CONTROL ROD CALIBRATION

8-006380 ALSO IN CATEGORY 7 DATED 1965
LAUBEN GN + KOONTZ RL
SODIUM FIRES AND THE RELEASE CHARACTERISTICS OF PARTICULATES AND FISSION PRODUCTS
ATOMICS INTERNATIONAL
CONF-650407, INTERNATIONAL SYMPOSIUM ON FISSION PRODUCT RELEASE AND TRANSPORT UNDER ACCIDENT CONDITIONS, APRIL 5-7, 1965, OAK RIDGE, TENN.- CFSTI

IN THE SAFETY ANALYSIS OF SODIUM-COOLED REACTORS, POTENTIALLY SERIOUS DISPERSIONS OF RADIOACTIVITY (NA 24 AND CERTAIN FISSION PRODUCTS) ARE POSTULATED TO OCCUR AS A RESULT OF A PRIMARY COOLANT SODIUM FIRE. ONE PROGRAM AT ATOMICS INTERNATIONAL IS BEING CONDUCTED TO CHARACTERIZE SODIUM FIRES BY DETERMINING THE INFLUENCE OF INITIAL CONDITION VARIABLES AND SYSTEM DESIGN PARAMETERS ON COMBUSTION ENERGY RELEASE AND DISTRIBUTION, AND THE RELEASE OF SODIUM (AS OXIDE OR METAL) AND IODINE.

FIRE + FISSION PRODUCT RELEASE + REACTOR, LMCR + PARTICLE SIZE + SODIUM + FISSION PRODUCT TRANSPORT

8-007444 ALSO IN CATEGORIES 6 AND 17 DATED 1965
REACTOR KINETICS
ARGONNE NATIONAL LABORATORY
9 PAGES, 3 FIGURES- SEPTEMBER 21, 1965, ANL-7090, PP 69-75 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, AUGUST 1965, CFSTI, \$3.00 CY

EBR-II- MARK I FUEL PIN FAILURE IN TREAT (FUEL VIGOROUSLY EXPELLED BY SODIUM VAPOR PRESSURE). INSTRUMENTATION FOR COOLANT EXPULSION STUDIES. DURING ACCIDENTS, CRITICAL FLOW MAY EXIST AT THE DISCHARGE END OF COOLANT CHANNELS, RESULTING IN LARGE PRESSURE GRADIENTS. A PHYSICALLY MEANINGFUL WEIGHTING FUNCTION WAS DERIVED FOR SPACE-TIME REACTOR DYNAMICS. IRRADIATION OF METAL-WATER REACTION SAMPLES FOR TREAT. HFIR FUEL ELEMENT SAMPLES MELTED IN TREAT TRANSIENTS, RESULTING IN SOME METAL-WATER REACTIONS.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, GRAPHITE MODERATED + HYDRAULIC ANALYSIS + REACTOR, LMCR + SODIUM + SPACE DEPENDENT DYNAMICS + REACTOR, TEST + TREAT (PKR) + HFIR (FTR) + PRESSURE, INTERNAL

CATEGORY 8
SOURCES OF ENERGY RELEASE UNDER ACCIDENT CONDITIONS

8-010113 DATED 1965
FORD JA
LITERATURE REVIEW OF SODIUM-WATER REACTIONS
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-167 +. 46 PAGES, REFERENCES, MARCH 15, 1965, CFSTI

A REVIEW WAS MADE OF THE LITERATURE PERTAINING TO THE INTERACTION OF SODIUM AND WATER, WITH PARTICULAR EMPHASIS ON REACTION DATA AND INFORMATION APPLICABLE TO THE DESIGN AND ANALYSIS OF SODIUM-HEATED STEAM GENERATORS. THE FIRST FOUR SECTIONS OF THE REVIEW ARE A GENERAL DISCUSSION OF THE MORE IMPORTANT ASPECTS OF THE REACTION, WHILE THE FIFTH SECTION SUMMARIZES THE INFORMATION ON TESTS AND ANALYSES PERTINENT TO REACTIONS OCCURRING IN ALKALI-METAL-HEATED STEAM GENERATORS. IN ADDITION TO A DOCUMENTATION OF THE REACTIONS WHICH HAVE OCCURRED IN OPERATING MODEL AND PLANT-SIZED STEAM GENERATORS, TABULATIONS OF THE EXPERIMENTAL DATA FROM MOST OF THE ALKALI-METAL-WATER REACTION TESTS THAT HAVE BEEN CONDUCTED ARE PRESENTED.

BIBLIOGRAPHY + REACTOR, LMCR + METAL WATER REACTION + SODIUM + DESIGN STUDY + REVIEW + FAILURE, TUBING + STEAM GENERATOR

8-010242 ALSO IN CATEGORY 17 DATED 1966
GOLDMANN K + MINUSHKIN B
SODIUM TECHNOLOGY
UNITED NUCLEAR CORPORATION
TID-8541 +. 105 PAGES, 230 REFERENCES, 37 FIGURES, 14 TABLES, PP. 321-425 OF REACTOR TECHNOLOGY, SELECTED REVIEWS--1965, L. E. LINK, EDITOR, ARGONNE NATIONAL LABORATORY, JANUARY 1966, CFSTI, \$4.50 CY, \$2.00 MN

THIS REVIEW COVERS SODIUM TECHNOLOGY THAT HAS BEEN DEVELOPED IN CONJUNCTION WITH THE DESIGN, CONSTRUCTION, AND OPERATION OF LIQUID-METAL-COOLED REACTOR PLANTS. ALSO REVIEWED IS THE STATE OF THE ART OF SODIUM TECHNOLOGY AS RELATED TO NEXT-GENERATION POWER PLANTS, WHICH ARE PROJECTED TO HAVE A CAPABILITY OF PRODUCING STEAM AT 1350 F, WITH MAXIMUM LIQUID-METAL TEMPERATURES OF 1200 TO 1300 F. WITHIN THIS FRAMEWORK THE LIQUID METALS OF INTEREST ARE SODIUM AND SODIUM-POTASSIUM (NAK). REVIEWED HERE ARE MATERIALS TECHNOLOGY, SUCH AS MATERIALS SELECTION, CORROSION, CONTROL OF SODIUM PURITY, AND THE EXPERIENCE GAINED IN THESE AREAS WITH OPERATING PLANTS.

CONTAINMENT DESIGN + REACTOR, LMCR + OPERATION + SODIUM + VAK + CORROSION + METAL, LIQUID

8-010359 DATED 1965
MUENCHOW HQ
REVIEW OF ATOMIC POWER DEVELOPMENT ASSOCIATES SODIUM-WATER REACTION PROGRAM
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
CONF-650620 +. 12 PAGES, 8 FIGURES, PP. 232-243 OF PROCEEDINGS OF SODIUM COMPONENTS DEVELOPMENT PROGRAM INFORMATION MEETING, CHICAGO, JUNE 16-17, 1965, CFSTI \$3.00 CY

THE STEAM GENERATOR IS RECOGNIZED AS THE WEAKEST LINK IN THE HEAT-TRANSPORT CHAIN OF SODIUM-COOLED REACTORS. ADPA HAS BEEN AWARDED A CONTRACT TO CONDUCT A SERIES OF SODIUM-WATER REACTION TESTS WHICH WILL ESTABLISH A MEANS OF ESTIMATING THE EXTENT OF TUBE WASTAGE DURING A SMALL LEAK OF WATER INTO THE SODIUM OF A SODIUM-HEATED STEAM GENERATOR, AND DETERMINE THE DOMINANT PARAMETERS OF TUBE WASTAGE DURING A SMALL WATER LEAK SO THAT THEY MAY BE APPLIED TO THE DESIGN OF A UNIT.

FERMI (LMFBR) + REACTOR, LMCR + METAL WATER REACTION + SODIUM + EQUIPMENT DESIGN

8-013548 ALSO IN CATEGORY 7 DATED 1966
LAUBEN GN
INITIAL EXPERIENCE WITH LARGE SODIUM FIRES EXPERIMENTS (LF-1)
ATOMICS INTERNATIONAL
NAA-SR-12041 +. 41 PAGES, 14 FIGURES, 3 TABLES, 15 REFERENCES, AUGUST 1, 1966

IN THE SAFEGUARDS ANALYSIS OF SODIUM-COOLED REACTORS, A SERIOUS DISPERSION OF RADIOACTIVITY IS POSTULATED TO RESULT FROM A LARGE PRIMARY-COOLANT (SODIUM) FIRE IN A GALLERY OR VAULT. TO EVALUATE AND/OR MITIGATE THIS DISPERSION, IT IS NECESSARY TO DETERMINE THE EFFECTS OF TIME, INITIAL-CONDITION VARIABLES, AND SYSTEM PARAMETERS ON THE SPATIAL DISTRIBUTION OF ENERGY, THE AMOUNT OF SODIUM RELEASED (PRESUMABLY AS SODIUM OXIDE), AND THE AMOUNT OF SELECTED FISSION PRODUCTS RELEASED. EARLY SODIUM-FIRE INVESTIGATIONS GAVE WIDELY VARYING RESULTS OF BURNING AND RELEASE RATES WHICH SEEMED TO BE VERY SENSITIVE TO GEOMETRY AND AIR FLOW. FOR EXAMPLE, IT HAS BEEN REPORTED THAT SHALLOW POOLS BURN AS NODULES ON AN IRREGULAR OXIDE SURFACE, WHEREAS DEEPER, WELL-INSULATED POOLS BURNED ON THE MOLTEN METAL SURFACE. REPORTED HEREIN ARE THE RESULTS OF THE CURRENT INVESTIGATION, INCLUDING RESULTS OF A FIRE, DESIGNATED AS LARGE FIRE NO. 1, IN WHICH THE GEOMETRIC AND CONVECTIVE CONDITIONS WHICH MIGHT BE FOUND IN A REACTOR GALLERY OR VAULT WERE SIMULATED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$2.00 COPY, \$0.50 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FIRE + FISSION PRODUCT, IODINE + IODINE + REACTOR, LMCR + SODIUM + FISSION PRODUCT TRANSPORT

8-014385 ALSO IN CATEGORY 7 DATED 1966
KOONTZ RL + LAUBEN GN
NUCLEAR SAFETY, GASEOUS EFFLUENT STUDIES. CHARACTERIZATION OF SODIUM FIRES AND FISSION PRODUCT RELEASE

CATEGORY 8
SOURCES OF ENERGY RELEASE UNDER ACCIDENT CONDITIONS

8-014385 *CONTINUED*

ATOMICS INTERNATIONAL
NAA-SK-12175 +. 14 PAGES, 3 TABLES, 7 FIGURES, QUARTERLY TECHNICAL PROGRESS REPORT AEC UNCLASSIFIED PROGRAMS,
JULY-SEPTEMBER 1966, PAGES 161-174

THE GENERAL OBJECTIVE OF THIS PROJECT IS TO DEVELOP EXPERIMENTAL INFORMATION AND ANALYTICAL METHODS WHICH CHARACTERIZE THE RELEASE AND TRANSPORT OF EFFLUENTS AND ENERGY GENERATED DURING A PRIMARY-COOLANT (SODIUM) ACCIDENT. THE SOURCE OF ENERGY GENERATION MAY BE FROM THE ESCAPING COOLANT (BY RAPID THERMAL ENERGY TRANSFER) AND/OR SUBSEQUENT COMBUSTION OF THE SODIUM. THE EFFLUENTS ARE SODIUM (AS SODIUM-24) OR ITS OXIDE IN PARTICULATE FORM AND SELECTED FISSION PRODUCTS AVAILABLE AS POTENTIALLY SERIOUS DISPERSIONS OF RADIOACTIVITY. THE INFORMATION TO BE DEVELOPED IS REQUIRED FOR THE DESIGN AND SAFEGUARDS ANALYSIS OF ECONOMICAL, SODIUM-COOLED FAST REACTORS. A MAJOR EFFORT WILL BE DEVOTED TO EXPERIMENTS ON THE CHARACTERIZATION OF I-131 RELEASE FROM SODIUM FIRES. THESE TESTS WILL BE CONDUCTED IN THE LABORATORY TEST CHAMBER AND IN THE PARTICLE GENERATOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICRONEGATIVE

AEROSOL + FIRE + REACTOR, LMR + PARTICLE SIZE + SODIUM + ANALYTICAL MODEL + FISSION PRODUCT TRANSPORT + OUT
OF PILE EXPERIMENT + SMOKE + METAL, LIQUID

8-014787 ALSO IN CATEGORY 17 DATED 1966
DOUGLAS RE
EFFECTS OF WATER LEAKAGE INTO TANKS CONTAINING SODIUM
ATOMICS INTERNATIONAL, CANOGA PARK
NAA-SK-MEMO-12235 +. 14 PAGES, NOVEMBER 10, 1966

ONE METHOD FOR DISPOSING OF THE HALLAM PRIMARY SODIUM IS TO BURY THE STORAGE TANKS WITHOUT PRIOR REACTION OF THE SODIUM. A TEST WAS PERFORMED TO DETERMINE THE EFFECTS OF GROUND WATER LEAKAGE INTO THE TANKS THROUGH PINHOLES OR CRACKS. A HALF QUART CAN WAS SUBMERGED AND VARIOUS SIZED HOLES DRILLED. RESULTS INDICATE THAT THE SODIUM-WATER REACTION WOULD TAKE PLACE AT A SELF-REGULATING RATE, AND NO EXCESSIVE INTERNAL PRESSURE INCREASE OR EXPLOSIVE CONDITION WOULD BE CREATED IN THE TANKS UNDER CONDITIONS SIMILAR TO THOSE IMPOSED FOR THE TEST.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U. S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

EXPLOSION + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMR + METAL WATER REACTION + SODIUM + WASTE
DISPOSAL, TERRESTRIAL + DECOMMISSIONING

8-015934 ALSO IN CATEGORIES 7 AND 12 DATED 1966
SAVAGE HW
HISTORY OF THE AEC SODIUM COMPONENTS DEVELOPMENT PROGRAM. VOLUME I
OAK RIDGE NATIONAL LABORATORY
ORNL-3955 (VOL. 1) +. 401 PAGES, DECEMBER 1966

THIS HISTORY WILL BE OF GREAT INTEREST TO THOSE WHO HAVE HAD A PART IN THE DEVELOPMENT OF LIQUID-METAL COMPONENTS TECHNOLOGY DURING THE LAST EIGHT TO NINE YEARS, AND WE SINCERELY HOPE THAT IT WILL SERVE AS AN ACCURATE REFLECTION OF THE CONTRIBUTION OF THE SODIUM COMPONENTS DEVELOPMENT PROGRAM WORKING GROUP IS MAKING TO THAT DEVELOPING TECHNOLOGY. THIS VOLUME IS A SUMMARY OF DEVELOPMENT PROGRAMS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + FERMI (LMFBR) + REACTOR, LMR + METAL WATER REACTION + SODIUM + TESTING + OPERATING EXPERIENCE
SUMMARY + EQUIPMENT DESIGN + MAIN COOLING SYSTEM + PUMPS + INSTRUMENTS, MISC. + HEAT EXCHANGERS

8-015935 ALSO IN CATEGORIES 7 AND 12 DATED 1966
SAVAGE HW
HISTORY OF THE AEC SODIUM COMPONENTS DEVELOPMENT PROGRAM. VOLUME II
OAK RIDGE NATIONAL LABORATORY
ORNL-3955 (VOL. 2) +. 568 PAGES, DECEMBER 1966

THIS HISTORY WILL BE OF GREAT INTEREST TO THOSE WHO HAVE HAD A PART IN THE DEVELOPMENT OF LIQUID-METAL COMPONENTS TECHNOLOGY DURING THE LAST EIGHT TO NINE YEARS, AND WE SINCERELY HOPE THAT IT WILL SERVE AS AN ACCURATE REFLECTION OF THE CONTRIBUTION OF THE SODIUM COMPONENTS DEVELOPMENT PROGRAM WORKING GROUP IS MAKING TO THAT DEVELOPING TECHNOLOGY. THIS VOLUME CONSISTS OF REPORTS OF SUBCOMMITTEE MEETINGS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, LMR + METAL WATER REACTION + SODIUM + TESTING + OPERATING EXPERIENCE SUMMARY +
EQUIPMENT DESIGN + MAIN COOLING SYSTEM + PUMPS + INSTRUMENTS, MISC. + HEAT EXCHANGERS

8-015936 ALSO IN CATEGORIES 7 AND 12 DATED 1966
SAVAGE HW

CATEGORY 8
SOURCES OF ENERGY RELEASE UNDER ACCIDENT CONDITIONS

8-015936 *CONTINUED*
HISTORY OF THE AEC SODIUM COMPONENTS DEVELOPMENT PROGRAM. VOLUME III
OAK RIDGE NATIONAL LABORATORY
ORNL-3953 (VOL. 3) +. 484 PAGES, TABLES, DECEMBER 1966

THIS HISTORY WILL BE OF GREAT INTEREST TO THOSE WHO HAVE HAD A PART IN DEVELOPING LIQUID-METAL COMPONENTS TECHNOLOGY DURING THE LAST EIGHT TO NINE YEARS, AND WE SINCERELY HOPE THAT IT WILL SERVE AS AN ACCURATE REFLECTION OF THE CONTRIBUTION OF THE SODIUM COMPONENTS DEVELOPMENT PROGRAM WORKING GROUP IS MAKING TO THAT DEVELOPING TECHNOLOGY. THIS VOLUME CONSISTS OF QUESTIONNAIRES SENT TO MEMBERS AND ALTERNATES, AND REPLIES. ALSO INCLUDED ARE THE MONTHLY COORDINATION REPORTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, LMCR + METAL WATER REACTION + SODIUM + TESTING + OPERATING EXPERIENCE SUMMARY + EQUIPMENT DESIGN + MAIN COOLING SYSTEM + PUMPS + INSTRUMENTS, MISC. + HEAT EXCHANGERS

8-017576 ALSO IN CATEGORIES 5 AND 6 DATED 1967
SAVINI JM + SIEGEL R
EXPERIMENTAL AND ANALYTICAL STUDY OF THE TRANSIENT SOLIDIFICATION OF A WARM LIQUID FLOWING OVER A CHILLED FLAT PLATE
LEWIS RESEARCH CENTER, CLEVELAND, OHIO. NASA
NASA-TN-D-4015 +. 63 PAGES, 12 FIGURES, 26 REFERENCES, JUNE 1967

ANALYTICAL PREDICTIONS AND EXPERIMENTAL DATA ARE PROVIDED FOR THE TRANSIENT GROWTH OF A FROZEN LAYER THAT FORMS WHEN A WARM LIQUID FLOWS OVER ONE SIDE OF A CHILLED FLAT PLATE. THE PLATE IS COOLED TO A TEMPERATURE BELOW THE FREEZING POINT OF THE LIQUID BY A COOLANT FLOWING ALONG THE OTHER SIDE OF THE PLATE. THIS COULD BE ENCOUNTERED, FOR EXAMPLE, IN A HEAT EXCHANGER WHERE WATER IS BEING COOLED BY A FLUID AT LOW TEMPERATURE. THREE ANALYTICAL PROCEDURES WERE USED AND COMPARED FOR ACCURACY AND CONVENIENCE IN APPLICATION. ONE IS AN ITERATIVE TECHNIQUE THAT YIELDED THREE CLOSED-FORM SOLUTIONS, EACH SUCCESSIVE SOLUTION BEING A MORE ACCURATE HIGHER ORDER APPROXIMATION. NUMERICAL RESULTS FROM THE THIRD APPROXIMATION ARE PRESENTED GRAPHICALLY IN SUFFICIENT DETAIL SO THAT RAPID ESTIMATES OF FROZEN LAYER GROWTH CAN BE MADE. IN THE EXPERIMENT, WATER FLOWED OVER ONE SIDE OF A THIN INCONEL PLATE WHILE THE OTHER SIDE OF THE PLATE WAS COOLED BY FLOWING CHILLED ALCOHOL. TRANSIENT ICE GROWTH WAS INITIATED BY A SUDDEN INCREASE OF THE ALCOHOL FLOW. THE ICE THICKNESS AND THE PLATE SURFACE TEMPERATURES WERE MEASURED THROUGHOUT THE TRANSIENT GROWTH PERIOD AND COMPARED FAVORABLE WITH THEORETICALLY PREDICTED VALUES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REACTOR, MOLTEN SALT + FILTER, LIQUID + HEAT TRANSFER, NATURAL CONVECTION + HEAT TRANSFER ANALYSIS + CRYOGENICS + HEAT EXCHANGERS

8-018195 ALSO IN CATEGORIES 6 AND 7 DATED 1967
DICKERMAN CE + ROBINSON LE + SOWA ES + MONAWICK JH + AUGUST C
TREAT MELTDOWN OF IRRADIATED METALLIC FAST-REACTOR FUEL IN AN INERT ATMOSPHERE
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
13 PAGES, 10 FIGURES, 3 TABLES, 29 REFERENCES, NUCLEAR APPLICATIONS 3(8)- 461-473 (AUG. 1967)

BEHAVIOR OF TWO TYPES OF IRRADIATED METALLIC FAST-REACTOR FUEL PINS UNDER TRANSIENT HEATING IN AN INERT-GAS ATMOSPHERE IN THE TREAT REACTOR WAS STUDIED UNDER CONDITIONS RANGING UP TO COMPLETE MELTDOWN. IRRADIATION-INDUCED CHANGES IN FAILURE THRESHOLDS, MODES OF FAILURE, AND POST-FAILURE FUEL MOTION WERE SIGNIFICANT BUT NOT ENOUGH TO MAKE GROSS CHANGES IN THE COURSE OF AN ACCIDENT. THE AMOUNT OF SWELLING OBSERVED IN THE POSTEXPERIMENTAL FUEL RESIDUE WAS LARGE ENOUGH TO HAVE PLAYED A SIGNIFICANT ROLE IN DETERMINING THE COURSE OF A MELTDOWN ACCIDENT.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + SODIUM + TESTING + TREAT (PRR) + VOID COEFFICIENT + FAILURE, CLADDING

8-018634 ALSO IN CATEGORY 11 DATED 1967
GAST K + SCHIKARSKI W
CONTAINMENT OF A LARGE SODIUM-COOLED FAST BREEDER REACTOR
KERNFORSCHUNGSZENTRUM, KARLSRUHE, FEDERAL REPUBLIC OF GERMANY
SM-89/7 + CONF-670-402 +. 15 PAGES, 7 FIGURES, 1 TABLE, 5 REFERENCES, APRIL 1967

FOLLOWING THE CONCEPTUAL DESIGN STUDY FOR A 1000-MW SODIUM-COOLED FAST BREEDER REACTOR, THE CONCEPTUAL DESIGN OF A 300-MW PROTOTYPE REACTOR IS CURRENTLY UNDER WAY AT KARLSRUHE RESEARCH CENTRE. FOR THE PROTOTYPE REACTOR, A RATHER CONSERVATIVE SAFETY ASSESSMENT WAS CHOSEN TO COMPENSATE FOR INCOMPLETE KNOWLEDGE OF DETAILED NUCLEAR AND THERMODYNAMIC BEHAVIOR IN A LARGE ACCIDENT. DOUBLE CONTAINMENT WILL THEREFORE BE PROVIDED, THE INNER CONTAINMENT BEING DESIGNED FOR THE POTENTIAL CONSEQUENCES OF A NUCLEAR EXCURSION WITH DESTRUCTIVE ENERGY RELEASE, AND FOR PREVENTION OF LARGE SODIUM FIRES. THE OUTER CONTAINMENT WILL SERVE AS A FINAL BARRIER FOR FISSION PRODUCTS AND OTHER RADIOACTIVE AEROSOLS. THE CONTAINMENT CONCEPT IS AIMED AT INDEPENDENCE IN SITE SELECTION.

AVAILABILITY - NATIONAL AGENCY FOR INTERNATIONAL PUBLICATIONS, INC., 317 EAST 34TH STREET, NEW YORK, NEW YORK. 10016

CONTAINMENT ANALYSIS + CONTAINMENT, MULTIPLE + ENERGY SOURCE + REACTOR, FAST + REACTOR, LMCR + ACCIDENT,

CATEGORY 8
SOURCES OF ENERGY RELEASE UNDER ACCIDENT CONDITIONS

8-018034 *CONTINUED*
MAXIMUM CREDIBLE (MCA) + DESIGN STUDY + GERMANY

8-023870 ALSO IN CATEGORIES 6 AND 18 DATED 1966
DEFELICE J + HYLAND M
FAST TEST REACTOR HYPOTHETICAL MELTDOWN ACCIDENT. PROGRESS REPORT. OCTOBER 1965 - DECEMBER 1965
BATTELLE NORTHWEST, RICHLAND, WASH.
CUU-1430-3 +. 65 PAGES, 3 FIGURES, 17 TABLES, 10 REFERENCES, JAN. 21, 1966

A STUDY OF THE ENERGY YIELD FROM A MELTDOWN IS IN PROGRESS. FUEL-ZONING EFFECTS WERE EXPLORED IN DETAIL WITH AX-1 AND WEAK-EXPLOSION COMPUTER CODES. AX-1 INDICATES A SMALLER ENERGY RELEASE DUE TO FUEL ZONING THAN EXPECTED, AND INSENSITIVENESS TO DEGREE OF ZONING. WEAK-EXPLOSION RESULTS IN A LARGE ENERGY RELEASE INCREASE WITH ZONING, HOWEVER ITS LACK OF TREATMENT OF SURFACE EFFECTS UNDOUBTEDLY OVERESTIMATES ZONING EFFECTS. WORK CONTINUES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00
COPY, \$0.05 MICROFICHE

COMPUTER PROGRAM + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REACTIVITY + REACTOR, TEST + FFTF (TR) + CORE
MELTDOWN + ACCIDENT, LOSS OF COOLANT

8-027030 ALSO IN CATEGORY 7 DATED 1968
MACPHERSON RE
SODIUM BURNING AND AEROSOL RELEASE - AN EVALUATION OF THE STATE OF THE ART
OAK RIDGE NATIONAL LABORATORY, OAK RIDGE, TENNESSEE
ORNL-TM-1937 +. 33 PAGES, 26 REFERENCES, MAY 6, 1968

IN A SODIUM-COOLED REACTOR, THE POSSIBILITY OF SODIUM FIRES AND THE RESULTANT RELEASE OF AEROSOLS STRONGLY INFLUENCE CONTAINMENT VESSEL DESIGN AND OPERATING SAFETY. QUALITATIVE INVESTIGATIONS OF SODIUM POOL FIRES AND SPRAY FIRES HAVE BEEN CARRIED OUT IN CONJUNCTION WITH EACH MAJOR SODIUM-COOLED REACTOR PROJECT. RECENTLY, MORE QUANTITATIVE STUDIES WERE UNDERTAKEN IN THESE AREAS, AND THE PROBLEM OF EXPLOSIVE INJECTION OF SODIUM SPRAYS INTO CONTAINMENT VOLUMES WAS INVESTIGATED. WORK IS ALSO UNDER WAY TO CHARACTERIZE AEROSOLS RELEASED FROM SODIUM FIRES AND TO DEFINE THEIR PERFORMANCE IN THE CONTAINMENT VOLUME. THIS DOCUMENT BRIEFLY SUMMARIZES THE WORK TO DATE ON SODIUM FIRES AND AEROSOL RELEASE, REVIEWS THE SCOPE OF CURRENT PROGRAMS, AND THEN PROPOSES ADDITIONAL EFFORT TO PROVIDE SAFETY CRITERIA FOR LIQUID-METAL-COOLED FAST-REACTOR DESIGN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.05 MICROFICHE

FIRE + REACTOR, LMCR + OXIDE + SAFETY REVIEW + SODIUM + AEROSOL PRODUCTION + AEROSOL PROPERTIES + METAL, LIQUID

8-031420 ALSO IN CATEGORY 5 DATED 1967
IVINS RO + TEVEBAUGH AD
SECTIONS FROM ANL-7425 CHEMICAL ENGINEERING DIVISION SUMMARY REPORT JULY-DEC 1967. SECTION IV REACTOR SAFETY
ARGONNE NATIONAL LABORATORY
ANL-KCV-4500 +. DEC. 1967, 111P., FIGURES, TABLES

PREPRINT OF SECTION IV, REACTOR SAFETY, OF ANL-7425, CHEMICAL ENGINEERING DIVISION SUMMARY REPORT, JULY-DECEMBER 1967. THE ANL WORK FOR THIS PERIOD WAS CONCENTRATED ON (1) THERMAL (WATER-COOLED) REACTOR STUDIES AND (2) FAST (SODIUM-COOLED) REACTOR STUDIES. ITEM (1) EFFORTS WERE DIRECTED TO ANALYTICAL STUDIES DESCRIBING THE CORE BEHAVIOR DURING A LOSS-OF-COOLANT ACCIDENT, ANALYTICAL AND EXPERIMENTAL STUDIES OF THE BEHAVIOR OF ZIRCALOY-2 CLAD UO₂ FUEL RODS DURING NUCLEAR EXCURSIONS AND OF THE POSSIBILITY OF PRESSURE PULSES DURING EITHER OF THESE TWO ACCIDENTS. ITEM (2) EFFORTS WERE DIRECTED TO EXPERIMENTALLY DETERMINING THE TRANSPORT PROPERTIES OF LIQUID URANIUM DIOXIDE, ANALYTICAL AND EXPERIMENTAL STUDIES OF THE THERMAL AND MECHANICAL INTERACTIONS BETWEEN FUEL, CLADDING, AND COOLANT, AND OF THE SODIUM-AIR REACTION, AND EXPERIMENTAL STUDIES OF FUEL MIGRATION.

HEAT TRANSFER, BUILDING + CHEMICAL REACTION + COMPUTER PROGRAM + FUEL MELTDOWN + ACCIDENT, LOSS OF COOLANT + OXIDATION + REACTOR, SAFETY RESEARCH + ACCIDENT, REACTIVITY + SODIUM + TREAT (PRR) + ANALYTICAL MODEL + MOLTEN FUEL + ACCIDENT MODEL + AIR + ANL + IN PILE EXPERIMENT + REACTOR, LWR + URANIUM DIOXIDE + REACTOR, PULSED + HEAT TRANSFER ANALYSIS + HEAT TRANSFER EXPERIMENT + ZIRCALOY + PROPERTY, PHYSICAL + PRESSURE PULSE + PRESSURE TRANSIENT + REACTOR, LMFR + GALLIUM + THERMAL TRANSIENT + FUEL COOLANT INTERACTION

8-033056 ALSO IN CATEGORIES 5 AND 12 DATED 1967
POPLAVSKY VM + AKISTARKOV NN + BAGUASAROV YE + BLAGOVOLIN SM + VYBLOV NF + PINKHASIK MS
SAFETY PROBLEMS WITH SINGLE WALL SODIUM/WATER HEAT EXCHANGERS FOR FAST REACTORS
UKAEA, DOONREAY EXPERIMENTAL REACTOR ESTABLISHMENT
CONF-670916-4 +. 26 PAGES, 4 FIGURES, 6 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPTEMBER 19-22, 1967

SUMMARY OF RUSSIAN WORK REGARDING SODIUM-WATER REACTIONS DUE TO A TUBE RUPTURE IN A SINGLE-WALL STEAM GENERATOR. DISCUSSION INCLUDED EXPERIMENTAL WORK AND OTHER FACTORS REGARDING THIS REACTION INFLUENCING THE DESIGN OF THE BN-350 AND THE BOR STEAM GENERATORS. FACTORS CONSIDERED WERE SHOCK, HYDRODYNAMIC PHENOMENA IN THE SODIUM LOOP INCLUDING STRUCTURAL LOADING, THERMAL EFFECTS, EFFICIENCY OF THE REACTION PRODUCT EVACUATION SYSTEM, AND DETERIORATION OF THE STRUCTURAL COMPONENTS.

CHEMICAL REACTION + METAL WATER REACTION + SODIUM + STEAM + WATER + PRESSURE RELIEF + USSR + METAL, ALKALI +

CATEGORY 8
SOURCES OF ENERGY RELEASE UNDER ACCIDENT CONDITIONS

8-033056 *CONTINUED*
METAL, LIQUID + EQUIPMENT DESIGN + STEAM GENERATOR + PRESSURE PULSE + REACTOR, LMFBR + HEAT EXCHANGERS

8-036468 ALSO IN CATEGORIES 5 AND 11 DATED 1966
EXPLOSION CONTAINMENT IN THE PROTOTYPE FAST REACTOR SINGLE TANK SYSTEM
SAFEX/P.69 +. 30 PAGES, 25 FIGURES, 3 TABLES, 3 REFERENCES, JANUARY 1966

DESCRIBES RESULTS OF SECOND TEST ON 1/16TH-SCALE MODEL, CHECKING EFFECTS OF DETONATION OF TNT.
RESULTS OF ROUND 11 AGREED WITH THOSE OF ROUND 10. ROUND-11 TEST SHOWED LIMITED DAMAGE.

AVAILABILITY - UNITED KINGDOM ATOMIC ENERGY AUTHORITY, 11 CHARLES 11 STREET, LONDON, S.W. 1, ENGLAND

REACTOR, BREEDER + CONTAINMENT + REACTOR, FAST + REACTOR, LMCR + UNITED KINGDOM + EXPLOSIVE, CONVENTIONAL +
PFR (LMFBR) + MODEL TESTING

8-036802 ALSO IN CATEGORIES 5 AND 18 DATED 1969
HIGH-TEMPERATURE LIQUID-METAL TECHNOLOGY REVIEW
BROOKHAVEN NATIONAL LABORATORY
80 PAGES, 11 FIGURES, 1 TABLE, APRIL 1969

SUMMARIZES PROGRESS IN VARIOUS R AND D PROGRAMS ON MATERIAL DEVELOPMENT, CORROSION, HEAT TRANSFER,
FLUID DYNAMICS, INSTRUMENTATION, PHYSICAL PROPERTIES, CHEMICAL ANALYSIS PERTINENT TO FAST-BREEDER
DEVELOPMENTAL PROGRAM. PROGRAMS REVIEWED IN CATEGORY OF SAFETY INCLUDE IN-CORE FLOWMETER
DEVELOPMENT, EFFECT OF FLOWING SODIUM ON CREEP RATE OF FUEL JACKET (VANADIUM ALLOYS), DEVELOPMENT
OF HIGH-TEMP.-BOILING DETECTOR, SODIUM-FIRE FISSION-PRODUCT RELEASE, BOILING STUDIES, SODIUM
CHEMISTRY, AND NOISE-ANALYSES INSTRUMENTATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFILME

CHEMICAL REACTION + FIRE + FISSION PRODUCT RELEASE + SODIUM + NOISE ANALYSIS + CORROSION + METAL, LIQUID +
REVIEW + R AND D PROGRAM + REACTOR, LMFBR + BOILING + INSTRUMENTS, MISC.

8-036957 ALSO IN CATEGORIES 5 AND 11 DATED 1969
UKAEA SAFETY RESEARCH BULLETIN ISSUE NUMBER 4
UKAEA, AUTHORITY HEALTH & SAFETY BRANCH, RISLEY
UKAEA SAFETY RESEARCH BULLETIN ISSUE NO. 4 +. 50 PAGES, 62 REFERENCES, 1969

PROGRESS REVIEW OF UKAEA WORK IN NUCLEAR REACTOR SAFETY. SUMMARIZED ARE STUDIES PERTAINING TO THE
SAFETY OF WATER COOLED REACTORS, LIQUID METAL FAST BREEDER REACTORS, AND THE FRACTURE OF CONCRETE
AND STEEL PRESSURE VESSELS. INCLUDED IN THE WATER REACTOR STUDIES ARE COOLANT DEPRESSURIZATION,
NATURAL CONVECTION HEAT TRANSFER, HEAT TRANSFER DURING DRYOUT CONDITIONS, BUBBLE GROWTH, ALUMINUM-
WATER INTERACTION, AND RELIABILITY OF EMERGENCY CORE-COOLING SYSTEMS. INCLUDED IN THE LIQUID
METAL FAST BREEDER REACTOR STUDIES ARE HEAT TRANSFER AND PRESSURE GENERATION FROM CORE DEBRIS
FOLLOWING A MELTDOWN, CONTAINMENT OF RADIOACTIVITY, CONSEQUENCES OF A SODIUM FIRE, AND PRESSURE
EFFECTS DUE TO A SODIUM FIRE.

AVAILABILITY - HER MAJESTY'S STATIONERY OFFICE, P.O. BOX 569, LONDON S.E.1, ENGLAND

ACCIDENT ANALYSIS + PRESSURE VESSELS + THERMAL ANALYSIS + WATER + CONCRETE, PRESTRESSED + CORE MELTDOWN +
REACTOR, LWR + METAL, LIQUID + HEAT TRANSFER ANALYSIS + BLOWDOWN + BUBBLE + EMERGENCY COOLING + PRESSURE
TRANSIENT + REACTOR, LMFBR + UKAEA + JACOBS

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-003407 ALSO IN CATEGORIES 6 AND 17 DATED 1963
KEATEN RW + PEARSON EN
REACTIVITY CALCULATIONS AND MEASUREMENTS AT THE SRE
ATOMICS INTERNATIONAL
33 PAGES, 5 TABLES, 10 FIGURES, 21 REFERENCES- ANS TOPICAL MEETING NUCLEAR PERFORMANCE OF POWER-REACTOR CORES,
SAN FRANCISCO, CALIFORNIA, SEPTEMBER 26-27, 1963- TID-7672- PP. 123-156

ORIGINAL CALCULATIONS USING STANDARD TWO-GROUP TECHNIQUES ARE COMPARED TO MEASURED VALUES OF
CRITICAL MASS, SODIUM WORTH, STATISTICAL WEIGHT, FLUX DISTRIBUTIONS, AND TEMPERATURE
COEFFICIENTS. A DISCREPANCY LARGER THAN THE EXPECTED CALCULATIONAL ERROR WAS REVEALED. REVISED
PHYSICS CALCULATIONS HAVE UTILIZED THORIUM AND U-235 RESONANCE INTERVALS CALCULATED BY A MONTE
CARLO CODE, AND THE RESULTS AGREE VERY WELL WITH THE MEASURED VALUES.

COMPUTER, ANALOG + COMPUTER, DIGITAL + REACTOR, LMCR + SODIUM COEFFICIENT + TEMPERATURE COEFFICIENT + SRE (RE)

9-003460 ALSO IN CATEGORY 6 DATED 1963
KEATEN RW + GUNBY AL + CARLSON WJ
A MATHEMATICAL MODEL DESCRIBING THE STATIC AND DYNAMIC INSTABILITY IN THE SRE CORE II
ATOMICS INTERNATIONAL
19 PAGES, 13 FIGURES, 9 REFERENCES- PROCEEDINGS OF A SYMPOSIUM ON REACTOR KINETICS AND CONTROL HELD AT THE
UNIVERSITY OF ARIZONA, MARCH 25-27, 1963- APRIL 1964, TID-7662, PP. 528-547

MATERIAL DESCRIBED IN TOPICAL REPORTS.

FUEL ELEMENT BOWING + REACTOR, LMCR + TRANSFER FUNCTION + SRE (RE)

9-003587 DATED 1964
SMITH CR + SPRINGER TH
THE CONTINUOUS NEUTRON FLUX MONITOR PROJECT, AN INTERIM REPORT
ATOMICS INTERNATIONAL
73 PAGES, 6 TABLES, 27 FIGURES, 15 REFERENCES- AUGUST 1, 1964, NAA-SR-9748

NEUTRON-SENSITIVE FLUID PASSES THROUGH A REACTOR. CONCENTRATION OF NEUTRON-FORMED PRODUCTS IN THE
EFFLUENT STREAM MEASURES NEUTRON FLUX AT A CHOSEN INCORE LOCATION. THREE FLUIDS WERE EVALUATED-
URANIUM HEXAFLUORIDE WHICH BY AN (N,F) REACTION FORMS DELAYED NEUTRON EMITTING NUCLIDES- BORON
TRIFLUORIDE GAS WHICH BY AN (N, ALPHA) REACTION PRODUCES LI 7- ARGON-40 GAS WHICH BY (N, GAMMA)
REACTION PRODUCES RADIOACTIVE AR 41. ARGON PROVED MOST PRACTICAL. IN KEWB REACTOR, IT SHOWED
LINEAR RESPONSE OVER THREE DECADES OF POWER. THERE WERE NO INHERENT LIMITATIONS IN EXTENDING
THIS RANGE. ARGON METHOD ALSO WAS USED IN FLUX MAPPING THE L-77 REACTOR. DEVELOPMENT OF
TEMPERATURE COMPENSATED SYSTEM IS BASIS OF SUBSEQUENT WORK. NOTE- SEE CRITICISM OF THIS METHOD
IN HW-81966, PAGE 5, LINE 19 FF.

INSTRUMENT, IN CORE + INSTRUMENT, NUCLEAR + REACTOR, LMCR + SRE (RE) + RADIATION MONITORS

9-003588 ALSO IN CATEGORY 6 DATED 1964
PELLET MR
CORE II PHYSICS TESTS ON THE SODIUM REACTOR EXPERIMENT
ATOMICS INTERNATIONAL
27 PAGES, 13 FIGURES- AUGUST 1, 1964, NAA-SR-9510

EXPERIMENTS WERE PERFORMED DURING THE OPERATION OF THE SRE-CORE II. TRANSFER FUNCTION WAS
MEASURED WITH A PILE OSCILLATOR. MEASUREMENTS AT-POWER WERE FIRST PERFORMED WHILE THE CORE HAD A
POSITIVE PROMPT POWER COEFFICIENT. AFTER FUEL CLUSTERS CONSTRAINED, TRANSFER FUNCTION AT 10, 15,
AND 20 MW DEMONSTRATED ABSENCE OF POSITIVE COEFFICIENT. COMBINED FUEL TEMPERATURE COEFFICIENT
WAS -0.12 CENTS/DEGREE F. INLET TEMPERATURE COEFFICIENT WAS DETERMINED TO BE -0.23CENTS/DEGREE
F. XENON PARAMETERS PREVIOUSLY DETERMINED HAVE BEEN VERIFIED. PRESENT SHIM ROD WORTH IS 94% OF
THAT INITIALLY DETERMINED. INTEGRATED ISOTHERMAL TEMPERATURE COEFFICIENT WAS 72.7 CENTS AS
COMPARED TO 81.9 CENTS PREVIOUSLY.

CONTROL ROD WORTH + FUEL ELEMENT BOWING + REACTOR, LMCR + OSCILLATOR, REACTIVITY + REACTIVITY COEFFICIENT +
XENON + SRE (RE)

9-003589 ALSO IN CATEGORIES 6 AND 17 DATED 1964
DARLEY DK + JENKINS OG + LARSON CL + MCCANN PH + MILLS RM + STIENS RR + WALTERS RJ
**HALLAM NUCLEAR POWER FACILITY - REACTOR OPERATIONS ANALYSIS PROGRAM SEMIANNUAL PROGRESS REPORT NO. 2, MARCH 1,
1963 - AUGUST 31, 1963**
ATOMICS INTERNATIONAL
162 PAGES, 23 TABLES, 51 FIGURES- MARCH 1, 1964, NAA-SR-9265

SECONDARY-TO-PRIMARY SODIUM LEAK WAS IN INTERMEDIATE HEAT EXCHANGER. IN SECONDARY SODIUM PUMP NO.
1, THERE WAS INTERFERENCE BETWEEN THE IMPELLER AND DIFFUSER. CONDENSATE LEAK IN STEAM DUMP WAS
TRACED TO A FAULTY GASKET. SODIUM FLOWMETER CALIBRATION LED TO OVERSHOOTS IN POWER. THE THREE
SECONDARY PUMPS WERE BINDING. ONE CONTROL ROD THIMBLE HAD A VERTICAL CRACK APPROXIMATELY 6-1/2
IN. LONG. FURTHER REACTIVITY CHECKS AND CONTROL ROD CALIBRATIONS WERE MADE. THREE FAILURES OF
ZIRCALOY-2 THIMBLES LED TO REPLACE ALL COMPONENTS. CODE HAS BEEN DEVELOPED FOR THE COMPONENT
RELIABILITY PROGRAM.

CONTAINMENT, PRESSURE VENTING + CONTROL ROD BURNUP + FAILURE, EQUIPMENT + HALLAM (LMR) + REACTOR, LMCR +

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-003589 *CONTINUED*
FAILURE, PIPE + RELEASE RATE

9-JJ3921
CHERNICK J

DATED 1963

CHARACTERISTICS OF FAST SETTLED BED REACTORS
ARGONNE NATIONAL LABORATORY

7 PAGES, 5 FIGURES, 3 REFERENCES- DECEMBER 1963, ANL-6792- (CONF-334), PROCEEDINGS OF THE CONFERENCE ON BREEDING, ECONOMICS AND SAFETY IN LARGE FAST POWER REACTORS, OCTOBER 7-10, 1963, PP. 587-593- OTS

DESCRIPTION OF FAST SETTLED BED REACTOR CONCEPT. DISCUSSION OF CONTROL, INCLUDING THE IDEA OF USING LI DISSOLVED IN THE COOLANT NA.

REACTOR, FAST + REACTOR, LMCR + PCISCA, SOLUBLE + TITANIUM

9-005263

ALSO IN CATEGORIES 12 AND 17

DATED 1965

LIQUID-METAL-COOLED REACTORS - EBR-II - EXAMINATION OF FAILED OSCILLATOR ROD AND THIMBLE

ARGONNE NATIONAL LABORATORY

2 PAGES- ANL-7014 (REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT), PAGES 2 AND 3 OF CHAPTER I- MARCH 18, 1965- CFSTI

THE FAILED OSCILLATOR ROD AND THIMBLE WAS EXAMINED, AND 27 BALLS FROM A BALL BEARING WERE FOUND. THE OSCILLATOR ROD WAS JAMMED BY 19 BALLS IN THE LOWER PORTION OF THE GUIDE SLEEVE. VERY HEAVY SCORING WAS OBSERVED IN TWO OF THE THREE LOWER GUIDES ON THE OSCILLATOR ROD AND IN THE CORRESPONDING GUIDE SLOTS OF THE GUIDE TUBE.

EBR 1 AND 2 (RE) + FAILURE + REACTOR, LMCR + OSCILLATOR, REACTIVITY + FAILURE, SCRAM MECHANISM

9-005333

DATED 1964

PELLET MK

AN ONLINE REACTIVITY METER FOR THE SRE

ATOMIC INTERNATIONAL

21 PAGES, 4 FIGURES- OCTOBER 15, 1964, NAA-SR-10186- OTS

AT THE SRE, THE EXCESS REACTIVITY IS USED AS A CONTINUOUS MONITOR OF CORE NORMALCY DURING OPERATION. TO PROVIDE AN IMMEDIATE INDICATION OF ABNORMALCY, AN ONLINE REACTIVITY METER WAS DEVELOPED. THE METER CONSISTS OF AN ANALOG CIRCUIT WHICH CONTINUOUSLY COMPUTES THE REACTIVITY CHANGES DUE TO TEMPERATURE, POWER, AND XENON EFFECTS. EXPERIENCE GAINED ON THE SRE CORE II METER HAS PROVEN ITS FEASIBILITY. ALSO, THE DATA AND EXPERIENCE GAINED ARE BEING USED TO DESIGN AN OPERATIONAL METER FOR CORE III.

COMPUTER, ANALOG + CONTROL ROD WORTH + REACTIVITY, EXCESS + INSTRUMENT, PROCESS + REACTOR, LMCR + REACTIVITY EFFECT + SAMARIUM + TEMPERATURE COEFFICIENT + XENON + SRE (RE) + MEASUREMENT, TEMPERATURE

9-006357

ALSO IN CATEGORIES 6 AND 17

DATED 1964

DARLEY DK + JENKINS GG + LARSON CL + LEW DE + MILLS RM + PEARSON EN + RUBINSTEIN HJ + STIENS RR + TESZLER O + WALTERS RJ + WEAVER LM

NUCLEAR ANALYSIS

ATOMIC INTERNATIONAL

45 PAGES, 25 FIGURES, 7 TABLES, 3 REFERENCES- PAGES 9-53, SECTION III, FROM HALLAM NUCLEAR POWER FACILITY REACTOR OPERATIONS ANALYSIS PROGRAM SEMI-ANNUAL PROGRESS REPORT NO. 3, SEPTEMBER 1, 1963 - FEBRUARY 29, 1964- SEPTEMBER 15, 1964, NAA-SR-9799, CFSTI

REPLACEMENT OF ZIRCALOY-II CONTROL ROD THIMBLES WITH STAINLESS STEEL THIMBLES AND ADDITION OF 10 (UL) FUEL ELEMENTS. MODERATOR ELEMENT FAILURES LED TO SODIUM FLOODING OF THE GRAPHITE MODERATOR. EFFECT OF FUEL BURNUP IS NOT AS GREAT AS CURRENTLY USED. FUEL ELEMENT VARIABLE ORIFICES BECOMING STUCK. 20 REACTOR SCRAMS HAVE BEEN ANALYZED WITH RESPECT TO CAUSE, CONSEQUENCES, AND MEANS OF REDUCING SCRAM FREQUENCY.

FAILURE, COMPONENT + CONTAINMENT, PRESSURE VENTING + CONTROL SYSTEM + HALLAM (LMR) + REACTOR, LMCR + SCRAM, REAL + URANIUM CARBIDE + REPORT, OPERATIONS ANALYSIS

9-006561

ALSO IN CATEGORIES 6 AND 17

DATED 1965

EBR-II

ARGONNE NATIONAL LABORATORY

6 PAGES, 1 FIGURE, 2 TABLES- JUNE 25, 1965, ANL-7046- PP. 1-6, SECTION A, OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 1965- CFSTI

MEASUREMENT OF POWER COEFFICIENT- FISSION PRODUCT MONITOR- LEAK RATE TESTS, CONTROL ROD MALFUNCTION, REACTIVITY MONITORING DEVICES, FEEDBACK OF AXIAL FUEL EXPANSION.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + REACTIVITY EFFECT, EXPANSION + REACTOR, FAST + FISSION PRODUCT RELEASE + REACTOR, LMCR

9-007436

ALSO IN CATEGORIES 6 AND 11

DATED 1965

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-007436 *CONTINUED*

FARET

ARGONNE NATIONAL LABORATORY

17 PAGES, 7 FIGURES, 4 TABLES- AUGUST 24, 1965, ANL-7082, PP 7-17 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT JULY 1965, CFSTI, \$3.00 CY, \$0.75 MN

FOR FARET, TESTS OF CONTROL ROD DRIVES, LEAK TEST OF VAULT HATCH SEAL, MEASUREMENTS OF CRITICAL CONTROL ROD WORTH. DANGER COEFFICIENTS OF COMPONENTS FROM OTHER ZONES OF SAME ASSEMBLIES.

REACTOR, BREEDER + CONTROL ROD WORTH + REACTOR, FAST + REACTOR, LMCR + TEST, LEAK RATE + FFTF (TR) + DANGER COEFFICIENT + FUEL HANDLING MACHINE + CONTROL ROD DRIVE MECHANISMS

9-007441

ALSO IN CATEGORY 6

DATED 1965

FARET

ARGONNE NATIONAL LABORATORY

7 PAGES, 1 FIGURE, 1 TABLE- SEPTEMBER 21, 1965, ANL-7090, PP 7-14 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, AUGUST 1965, CFSTI, \$3.00 CY

FOR FARET- TESTS OF CONTROL ROD DRIVES, DANGER COEFFICIENT OF PU, U-235, U-238 AND B-10, NA, STAINLESS STEEL, DEPLETED U, UO₂, GRAPHITE, ZR.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + FFTF (TR) + DANGER COEFFICIENT + CONTROL ROD DRIVE MECHANISMS

9-007695

ALSO IN CATEGORIES 1 AND 6 AND 17 DATED 1965

OKRENT D

A LOOK AT FAST REACTOR SAFETY

ARGONNE NATIONAL LABORATORY

25 PAGES, 1 FIGURE, 2 TABLES, 131 REFERENCES- NUCLEAR SAFETY 6(4)- 317-342 (SUMMER, 1965)

DESIGN HISTORY (DOUNREAY REACTOR, EBR-II, ENRICO FERMI, PATTERNS AND TRENDS, SAFETY-RELATED DESIGN AND OPERATING EXPERIENCE), DIFFERENCES IN NUCLEAR SAFETY FOR LARGE-SODIUM-COOLED FAST REACTORS (DELAYED-NEUTRON FRACTION, DOPPLER COEFFICIENT, SODIUM COEFFICIENT AND VOID EFFECT, STABILITY, MAXIMUM ACCIDENT CALCULATIONS), LARGE-SODIUM-COOLED-FAST REACTOR DESIGNS AND ACCIDENT STUDIES, NON-NUCLEAR SAFETY CONSIDERATIONS, COOLANTS OTHER THAN SODIUM (STEAM, GAS, OTHER METAL COOLANTS), SAFETY PHILOSOPHY AND SAFETY RESEARCH, REFERENCES.

REACTOR, BREEDER + DELAYED NEUTRON + DOPPLER COEFFICIENT + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LARGE + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR STABILITY + SODIUM + SODIUM COEFFICIENT + STEAM + VOID COEFFICIENT + OPERATING EXPERIENCE SUMMARY + HELIUM + DOUNREAY (LMFBR) + SAFETY PRINCIPLES AND PHILOSOPHY

9-007720

ALSO IN CATEGORY 17

DATED 1965

MUELLEK RE + BRANYAN CE + KLICKMAN AE

CONTROL-ROD CALIBRATIONS FOR THE ENRICO FERMI REACTOR

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

2 PAGES, 1 TABLE- PAGES 21-22 OF TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JULY 28-29, 1965, JACKSON LAKE LODGE, WYOMING

LOW-POWER TESTING CALIBRATIONS WERE MADE TO CONFIRM DESIGN AND TO INVESTIGATE DIFFERENT CALIBRATION TECHNIQUES. UNSHADOWED AND SHADOWED INDIVIDUAL ROD WORTHS SHOWED MEASURED ROD WORTHS LESS THAN PREDICTED BASED ON CRITICAL-EXPERIMENT. AN INCONSISTENCY WAS DISCOVERED BETWEEN THE MEASURED INDIVIDUAL ROD WORTHS AND THE GANGED ROD WORTH. FUTURE POWER-CONTROL RODS WILL BE MEASURED BY SUBCRITICAL MULTIPLICATION.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + SHUTDOWN MARGIN + COMPARISON, THEORY AND EXPERIENCE + OPERATING EXPERIENCE SUMMARY + REACTOR STARTUP TESTING + CONTROL ROD CALIBRATION

9-007721

ALSO IN CATEGORY 17

DATED 1965

HORNE RE + KLICKMAN AE

A TECHNIQUE FOR DETERMINATION OF THE SAFETY-ROD INSERTION TIME FOR THE ENRICO FERMI REACTOR

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

1 PAGE, 1 TABLE- PAGE 22 OF TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JULY 28-29, 1965, JACKSON LAKE LODGE, WYOMING

THE RELEASE OF THE SAFETY RODS PREVENTS USE OF CONVENTIONAL TIMING METHODS TO DETERMINE INSERTION TIME. A SIMPLE NUCLEAR METHOD OF MEASURING BOTH THE ROD WORTH AND THE INSERTION TIME HAS BEEN FOUND. A SANBURN TRACE OF THE REACTOR POWER AS A FUNCTION OF TIME, AFTER INITIATION OF A SCRAM IS ANALYZED TO OBTAIN THE DESIRED INSERTION TIME. IN ADDITION THE OUTPUT OF THE LATCH-OPEN MICROSWITCH AND THE LATCH MAGNET CURRENT WERE RECORDED ON THE SAME TRACE TO DETERMINE THE RODS TIME. THE TRACE OF THE ION-CHAMBER CURRENT WAS MANUALLY TRANSFORMED TO A DIGITAL FORM, AND THE SPACE-INDEPENDENT REACTOR KINETICS EQUATIONS WERE SOLVED BY A COMPUTER TO SHOW THE REACTIVITY AS A FUNCTION OF TIME AFTER SCRAM. THE INSERTION TIME DEFINED AS THE TIME BETWEEN RELEASE AND INSERTION OF 95% OF THE TOTAL ROD WORTH. THIS METHOD IS CONSIDERED TO BE ACCURATE TO 10 MSEC. MOST OTHER EXPERIMENTERS HAVE USED ROD-DROP EXPERIMENTS BY ANALYZING A LOW-SPEED RECORDING WITH THE PROMPT-JUMP APPROXIMATION. USING HIGH-SPEED RECORDING DOES NOT REQUIRE THIS APPROXIMATION.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + CONTROL ROD SCRAM MECHANISM + TEST, DROP

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-007728 ALSO IN CATEGORY 17 DATED 1965
ALEXANDERSON EL + HARRIGAN PM
ADDED SOURCE-RANGE DETECTORS FOR THE ENRICO FERMI ATOMIC POWER PLANT
POWER REACTOR DEVELOPMENT CO., DETROIT
2 PAGES, 1 FIGURE- PAGES 26-27 OF TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING
EXPERIENCE, JULY 28-29, 1965, JACKSON LAKE LODGE, WYOMING

OPERATION AT LOWER POWER LEVELS AND PLANT FACTORS WOULD REQUIRE A STRONGER NEUTRON SOURCE, BUT
ZIRCONIUM HYDRIDE WAS PLACED AROUND THE SOURCE RANGE DETECTORS. SOFTENING THE SPECTRUM INCREASED
THE SHUTDOWN COUNT RATE ADEQUATELY (2/3 DECADE).

REACTOR, BREEDER + ECONOMICS + REACTOR, FAST + FERMI (LMFBR) + INSTRUMENT, NUCLEAR + REACTOR, LMCR + SOURCE,
NEUTRON + OPERATING EXPERIENCE SUMMARY

9-008052 ALSO IN CATEGORY 17 DATED 1965
MCCOOL WJ
EQUIPMENT FAILURE AT THE EXPERIMENTAL BREEDER REACTOR-II (EBR-II)
DIVISION OF OPERATIONAL SAFETY, AEC
2 PAGES- JULY 12, 1965, REACTOR OPERATING EXPERIENCE 65-13- AEC-DOS

TWO OF 11 CONTROL RODS FAILED TO SCRAM BECAUSE STEEL BALLS HAD CAUSED BINDING BETWEEN THE ROD
SHAFTS AND THE REACTOR COVER. BALLS CAME FROM FAILED BALL-BUSHING, WHICH WAS FOUND TO BE A
CARBON STEEL VERSION OF THE SPECIFIED STAINLESS-STEEL BALL BUSHING.

REACTOR, BREEDER + EBR 1 AND 2 (KEI + REACTOR, FAST + REACTOR, LMCR + MATERIAL + REACTOR, ERDA OWNED +
FAILURE, SCRAM MECHANISM + CONTROL ROD DRIVE MECHANISMS

9-008211 ALSO IN CATEGORY 6 DATED 1964
SHAW PF
HNPFC CONTROL SYSTEM RESPONSE DURING TRANSIENT CONDITIONS
ATOMIC INTERNATIONAL
36 PAGES, 7 FIGURES, 7 TABLES, 1 REFERENCE- JUNE 15, 1964, NAA-SR-9584- CFSTI, \$0.75 CY

THIS REPORT DOCUMENTS THE RESULTS OF POWER RAMPS IMPOSED ON THE HNPFC DURING THE MONTH OF JANUARY
1964. ANALYSIS AND EVALUATION OF THE DATA ACQUIRED DURING THESE TESTS VERIFIES THE ADEQUACY OF
PERFORMANCE OF THE HNPFC PLANT CONTROL SYSTEM. IT WAS DEMONSTRATED, WITHIN THE CONDITIONS OF THE
TEST, THAT THE HNPFC IS CAPABLE OF AUTOMATICALLY FOLLOWING STATION LOAD CHANGE RATES IN EXCESS OF
THE 5 MWE/MIN. TEST OBJECTIVE. DATA FROM THE RAMP TESTS AND OTHER PERTINENT INFORMATION ARE
INCLUDED IN THIS REPORT TO PERMIT THE READER TO MAKE AN INDEPENDENT EVALUATION OF CONTROL SYSTEM
RESPONSE TO THE IMPOSED POWER RAMPS.

CONTROL SYSTEM + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + TEST, PLANT RESPONSE + REACTOR
POWER + TEST, SYSTEM OPERABILITY

9-008456 DATED 1960
JOHNSON LE
COUPLED DIAPHRAGM SAFETY DEVICE FOR POWER REACTORS
ATOMIC INTERNATIONAL
NAA-SR-4434 +. 26 PAGES, 17 FIGURES, 4 TABLES, 3 REFERENCES- JANUARY 15, 1960- CFSTI

THE COUPLED DIAPHRAGM TRIGGER IS A NEUTRON FLUX SENSOR WHICH, AT EXCESSIVE FLUX LEVELS, WILL TRIP
THE POWER REACTOR SAFETY DEVICE AND CAUSE REACTOR SHUT DOWN. EXPERIMENTAL RESULTS INDICATE THAT
MECHANICALLY COUPLED DIAPHRAGMS CAN BE EMPLOYED IN A SAFETY DEVICE TRIGGER TO MINIMIZE THE
EFFECTS OF AMBIENT TEMPERATURE ON THE TRIP POINT. TEMPERATURE TESTS OF TRIGGER COMPONENTS
INDICATED SATISFACTORY PERFORMANCE AT TEMPERATURES UP TO 1200 DEGREE F. TESTS OF A PROTOTYPE
TRIGGER AT ROOM TEMPERATURE INDICATE A RESPONSE TIME OF ABOUT 1 SECOND.

REACTOR FUSE + REACTOR, GRAPHITE MODERATED + INSTRUMENT, IN CORE + REACTOR, LMCR + SRE (RE) + TEST, INSTRUMENT
RESPONSE

9-008500 DATED 1964
INSTRUMENTATION, CONTROL AND SAFETY
COMBUSTION ENGINEERING, INC.
LEND-200, VOL II +. 23 PAGES, 10 FIGURES- JUNE 6, 1964- PP. V-1-V-13 OF LIQUID METAL FAST BREEDER REACTOR
DESIGN STUDY, VOLUME II- CFSTI \$3.50 CY

THE CONTROL PHILOSOPHY FOR THE LIQUID METAL FAST BREEDER PLANT, A DESCRIPTION OF THE BASIC REACTOR
INSTRUMENTATION, AND A DISCUSSION OF THE SAFETY OF THE REACTOR ARE PRESENTED IN THIS SECTION.
THE NORMAL OPERATION OF THE PLANT, INCLUDING STARTUP, SHUTDOWN, NORMAL CHANGE IN LOAD, AND THE
STOPPING AND STARTING OF COOLANT LOOPS, HAS BEEN CONSIDERED. THE MORE UNUSUAL SITUATIONS WHICH
MAY ARISE DUE TO EQUIPMENT MALFUNCTION AND ALSO THE MORE SEVERE ACCIDENT CONDITIONS HAVE BEEN
SKETCHED AND THESE ARE DESCRIBED IN SOME DETAIL. IN THE ANALYSES WHICH SERVED AS A BASIS FOR
THESE EVALUATIONS, USE WAS MADE OF AN ANALOG COMPUTER FOR SOLVING THE KINETICS EQUATIONS.

ACCIDENT ANALYSIS + CONTROL SYSTEM + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + ACCIDENT, LOSS OF FLOW +
ACCIDENT, REACTIVITY + ACCIDENT, CONTROL ROD WITHDRAWAL + REACTOR SAFETY SYSTEM + INSTRUMENT, CONTROL +
SAFETY PRINCIPLES AND PHILOSOPHY

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-009107 ALSO IN CATEGORY 17 DATED 1964
HALL RJ + AVERY RE + MOORE EA
FUEL CLADDING FAILURE MONITOR FOR SODIUM REACTORS
ATOMICS INTERNATIONAL
NAA-SR-6463 +. 26 PAGES, 9 FIGURES, 1 TABLE- DECEMBER 1, 1964- CFSTI \$0.50

FILTERED HELIUM COVER GAS FOR THE SRE FLOWS PAST A GAMMA SCINTILLATION SPECTROMETER MONITORING XENON-133 PEAK. WHILE NO FUEL FAILED DURING COKE II OPERATION, THE BACKGROUND WAS A FACTOR OF 1000 BELOW SATURATION WHILE REMAINING HIGH ENOUGH TO EASILY CHECK THE INSTRUMENT RESPONSE.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + SRE (RE) + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + FAILURE, CLADDING

9-009717 DATED 1965
PORGES KG
DESIGN IMPROVEMENTS AND SENSITIVITY TESTS OF THE FERD SYSTEM FOR EBR-II. PHYSICS DIVISION ANNUAL REPORT, JULY 1, 1964 TO JUNE 30, 1965
ARGONNE NATIONAL LABORATORY
ANL-7110 +. 2 PAGES, 1 FIGURE, 2 REFERENCES, DECEMBER 1965, SECTION IV-22, PP. 294-295, CFSTI, \$7.22 CY, \$2.00 MN.

THE FUEL ELEMENT RUPTURE DETECTOR (FERD), NOW INSTALLED IN EBR-II, HAS ALREADY BEEN DESCRIBED IN GENERAL OUTLINE. IN THIS REPORT, A BRIEF ACCOUNT IS GIVEN OF THE ACCRUED MAINTENANCE EXPERIENCE WHICH LED TO CERTAIN IMPROVEMENTS AS WELL AS OF A RECENT TEST TO WHICH THE FERD SYSTEM AND CONFIRMING CHARGED-WIRE DETECTOR WERE SUBJECTED. (K. G. PORGES, EBR-II FUEL ELEMENT FAILURE DETECTOR, REACTOR PHYSICS DIVISION ANNUAL REPORT JULY 1, 1963 TO JUNE 30, 1964, ANL-7010 (1965), P. 231).

REACTOR, BREEDER + EBR 1 AND 2 (KE) + REACTOR, FAST + REACTOR, LMCR + INSTRUMENT, DETECTION FAILED FUEL ELEMENT

9-009794 ALSO IN CATEGORY 17 DATED 1965
SUNDERMAN IO
MONTHLY OPERATING REPORT NO. 38, SEPTEMBER, 1965
HALLAM NUCLEAR POWER FACILITY
TID-22454 +. 22 PAGES, 1 FIGURE, 6 TABLES, SEPTEMBER 1965, PDR

THE HALLAM OPERATING CONTRACT WITH AEC WAS TERMINATED SEPTEMBER 9, 1965. CONSUMERS POWER WILL CARETAKEN THE FACILITY. REACTOR-BUILDING PRESSURE-DIFFERENTIAL SYSTEM MALFUNCTIONED DUE TO WATER IN THE CUPS AT THE END OF THE SENSING LINES. SIX MODERATOR-CLAD FAILURES WERE DUE TO LONG-TERM STRESS RUPTURE. THE SEVENTH OCCURRED BECAUSE THREE OTHER MODERATOR ELEMENT FAILURES CAUSED A MECHANICALLY PRODUCED GOUGE WHICH FAILED ON THE NEXT HEATUP, DUE TO THERMAL STRESS.

CONTAINMENT, PRESSURE VENTING + HALLAM (LMR) + REACTOR, LMCR + FAILURE, INSTRUMENT + MODERATOR + FAILURE, SEQUENTIAL

9-010139 ALSO IN CATEGORY 6 DATED 1964
KIRN FS + LOEWENSTEIN WB
EBR-II WET CRITICAL EXPERIMENTS
ARGONNE NATIONAL LABORATORY
ANL-6864 +. 46 PAGES, 15 TABLES, 25 FIGURES, OCTOBER 1964, CFSTI, \$2.00 CY

WET-CRITICAL EXPERIMENTS WERE PERFORMED IN THE EBR-II WITH SODIUM COOLANT IN THE REACTOR SYSTEM. MEASUREMENTS INCLUDED CONTROL ROD WORTH, NEUTRON-SOURCE STRENGTH, NUCLEAR-INSTRUMENT RESPONSE, CRITICAL SIZE, ISOTHERMAL TEMPERATURE COEFFICIENT, AND REACTIVITY WORTH OF THE SODIUM COOLANT. THE MEASUREMENTS AGREED WELL WITH CALCULATED VALUES AND WITH VALUES EXTRAPOLATED FROM THE DRY CRITICAL EXPERIMENTS AND THE ZERO-POWER MOCKUP.

REACTOR, BREEDER + CONTROL ROD WORTH + EBR 1 AND 2 (RE) + REACTOR, FAST + INSTRUMENT, NUCLEAR + REACTOR, LMCR + SOURCE, NEUTRON + TEMPERATURE COEFFICIENT + CRITICALITY EXPERIMENT + CONTROL ROD CALIBRATION + MEASUREMENT, REACTIVITY + REACTOR STARTUP EXPERIENCE

9-010150 DATED 1965
BONYHADY KA
ANALYSIS OF PROCESS CONTROL IN AN SGR 200 MWE PROTOTYPE PLANT
ATOMICS INTERNATIONAL
NAA-SR-MEMO-11339 +. 23 PAGES, JUNE 26, 1965, CFSTI, \$1.00 CY, \$0.50 MN

A STUDY WAS MADE OF METHODS FOR CONTROLLING THE TEMPERATURE OF THE STEAM ENTERING THE TURBINE IN A 200-MWE SGR PLANT USING A ONCE-THROUGH STEAM GENERATOR. THE STEAM TEMPERATURE COULD NOT BE CONTROLLED WITH CONTROLLERS BY USING THE UNBALANCE BETWEEN THE TWO SODIUM FLOWS IN THE INTERMEDIATE HEAT EXCHANGER. DESUPERHEATING BETWEEN THE EVAPORATOR AND SUPERHEATER GAVE SATISFACTORY CONTROL. HOWEVER, DESUPERHEATING AT THE SUPERHEATER OUTLET DID NOT. A CONSTANT-TEMPERATURE FEEDWATER PREHEATER IMPROVED THE CONTROL PERFORMANCE.

CONTROL SYSTEM + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + STEAM + ANALYTICAL MODEL + MATHEMATICAL STUDY

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-010435 DATED 1966
BUMP TR + SIMMONS WR
HOLD-DOWN REQUIREMENTS FOR POISON CONTROL RODS
ARGONNE NATIONAL LABORATORY, ARGONNE
3 PAGES, 3 REFERENCES, PRESENTED AT ANS ANNUAL MEETING, JUNE 20-23, 1966, DENVER, COLORADO

PROBLEM--IN REACTORS WITH LIQUID-METAL COOLANTS, THE VESSEL OR CLOSURE DESIGN MAY BE DICTATED BY THE ROD HOLD-DOWN REQUIREMENTS, RATHER THAN BY COOLANT PRESSURE. IT IS NECESSARY TO ENSURE THAT A MINOR ACCIDENT CANNOT FORCE THE RODS OUT OF THE REACTOR CORE AND CAUSE A MORE SERIOUS ACCIDENT. APPROACH--FOUR POSSIBLE MECHANISMS THAT MIGHT FORCE THE RODS FROM THE CORE WERE EXAMINED FOR THE FARET DESIGN.

DESIGN CRITERIA + REACTOR, LMCR + ACCIDENT, REACTIVITY + EXCURSION, LARGE + CONTROL ROD DRIVE MECHANISMS

9-010498 DATED 1965
KLINE KH
THE NEW HEAT TRANSFER PROTECTION SYSTEM FOR THE SODIUM REACTOR EXPERIMENT
ATOMICS INTERNATIONAL
NAA-SK-MEMO-11648 +. 11 PAGES, SEPTEMBER 14, 1965, CFSTI, \$1.00 CY, \$0.50 MN

PROBLEM - IMPROVE THE HEAT-TRANSFER PROTECTION SYSTEM FOR THE SRE AS PART OF A POWER-EXPANSION PROGRAM. SOLUTION - THIS PORTION OF SAFETY SYSTEM INCLUDES THREE CHANNELS OF DIFFERENTIAL TEMPERATURE PROTECTION, THREE CHANNELS OF OUTLET TEMPERATURE PROTECTION, ONE CHANNEL OF COLD-LEG TEMPERATURE PROTECTION, AND ONE CHANNEL OF PRIMARY-COOLANT-FLOW PROTECTION. THE FIRST TWO VARIABLES ARE OPERATED ON A TWO-CUT-OF-THREE BASIS.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + REACTOR SAFETY SYSTEM + SRE (RE) + EQUIPMENT DESIGN

9-010504 DATED 1965
QUARTERLY TECHNICAL PROGRESS REPORT - AEC UNCLASSIFIED PROGRAMS - JANUARY TO MARCH 1965
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA
NAA-SR-11200 +. 256 PAGES, FIGURES, TABLES, APRIL 15, 1965, CFSTI, \$6.00 CY

ATOMICS INTERNATIONAL PROGRESS REPORT, JANUARY TO MARCH, 1965 - THIS REPORT IS ONE OF A SERIES OF SUCH REPORTS ON THE SUBJECTS ENUMERATED BELOW. NO FURTHER REPORTS IN THE SERIES WILL BE SUMMARIZED UNLESS A SIGNIFICANT CHANGE IN THE PROGRAM OR A SIGNIFICANT ITEM OF INTEREST TO NSIC IS REPORTED. TOPICS COVERED IN THIS REPORT ARE FAST-REACTOR DEVELOPMENT, BOILING STUDIES FOR SODIUM REACTOR SAFETY, SODIUM FIRES AND FISSION PRODUCT RELEASE, FUEL ELEMENT TRANSIENT HEATING, HALLAM FUEL SURVEILLANCE PROGRAM, MEASUREMENT OF DOPPLER COEFFICIENTS, FAST-SPECTRUM DOPPLER MEASUREMENTS, FISSION-PRODUCT RETENTION BY REACTOR COOLANTS, CONTINUOUS NEUTRON FLUX MONITOR AND ATMOSPHERIC DECONTAMINATION IN REACTOR ACCIDENTS.

AIR CLEANING + DOPPLER COEFFICIENT + REACTOR, FAST + FIRE + FISSION PRODUCT RELEASE + FUEL INTEGRITY + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + INSTRUMENT, IN CORE + REACTOR, LMCR + SODIUM + FISSION PRODUCT RETENTION + FILTERS

9-010509 ALSO IN CATEGORY 6 DATED 1964
ANALOG MODELS FOR HNPF CONTROL AND PROTECTION STUDIES
ATOMICS INTERNATIONAL, CANOGA PARK
NAA-SR-65579 +. 69 PAGES, 16 FIGURES, 7 TABLES, 10 REFERENCES, JUNE 1964, CFSTI, \$1.50 CY

PROBLEM - DEVELOP A MODEL FOR THE DYNAMIC BEHAVIOR OF THE HNPF. APPROACH - AN ANALOG COMPUTER MODEL WAS DEVELOPED FOR USE IN STUDIES OF THE CONTROL AND SAFETY SYSTEMS. THE MODEL INCLUDES - NEUTRON KINETICS, AFTERHEAT, CORE HEAT-TRANSFER FOR AVERAGE CHANNEL AND HOT CHANNEL, PRIMARY-COOLANT HYDRAULICS FOR FORCED AND CONVECTION FLOWS, ROD DRIVES, CONTROL SYSTEM, AND PROTECTIVE SYSTEM. THE SIMULATIONS OF THE VARIOUS SECTIONS ARE DESCRIBED IN DETAIL.

COMPUTER, ANALOG + COMPUTER PROGRAM + CONTROL SYSTEM + HALLAM (LMR) + REACTOR, LMCR + REACTOR DYNAMICS + REACTOR SAFETY SYSTEM + ANALYTICAL MODEL + MATHEMATICAL STUDY

9-010523 DATED 1965
HURNE RE + KLICKMAN AE
DETERMINATION OF THE SAFETY ROD DROP TIME IN THE ENRICO FERMI REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-NTS-3 +. 39 PAGES, JANUARY 1965, DEP.MN

PROBLEM - MEASURE SAFETY-ROD DROP TIMES AND WORTH OF THE RODS. APPROACH - IT WAS NECESSARY TO CALCULATE THE ROD-DROP TIMES FROM THE BEHAVIOR OF THE NUCLEAR KINETICS, BECAUSE IT WAS NOT POSSIBLE TO MAKE ACCURATE MECHANICAL MEASUREMENTS OF SCRAM TIME. RESULTS - (1) ROD-DROP TIMES (630 MSEC) AGREED WITH PREDICTED VALUE, (2) DROP TIMES WERE NOT INFLUENCED BY VARIATION OF COOLANT FLOW OR COOLANT TEMPERATURE, (3) ROD WORTHS AGREED WITH PREDICTED VALUES.

REACTOR, BREEDER + CONTROL ROD WORTH + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + TESTING + CONTROL ROD SCRAM MECHANISM + MEASUREMENT, REACTIVITY

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-010978 ALSO IN CATEGORY 6 DATED 1966
REACTOR PHYSICS EFFORTS REQUIRED IN SUPPORT OF THE FAST BREEDER DEVELOPMENT PROGRAM
DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY, U.S. ATOMIC ENERGY COMMISSION, WASHINGTON, D. C.
WASH-1066 +. 41 PAGES, 3 TABLES, JANUARY 1966, CFSTI, \$3.00 CY, \$0.50 MN

SCATTERED THROUGHOUT THE REPORT ARE REFERENCES TO SODIUM COEFFICIENT, DOPPLER COEFFICIENT, CONTROL SYSTEM, AND DYNAMICS OF LARGE SYSTEMS.

REACTOR, BREEDER + CONTROL SYSTEM + CRITICAL ASSEMBLY + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LARGE + REACTOR, LMCR + REACTOR DYNAMICS + SODIUM COEFFICIENT + SPACE DEPENDENT DYNAMICS + ZPR 9 (CAF) + ZPR 6 (CAF) + SEFOR (RE) + ZPPR (CAF)

9-011424 ALSO IN CATEGORIES 6 AND 12 DATED 1966
HORST KM
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. THIRD QUARTERLY REPORT, OCTOBER 1964 - JANUARY 1965
GENERAL ELECTRIC, SAN JOSE
GEAP-4799 +. 150 PAGES, FEBRUARY 1965, CFSTI \$4.00 CY, \$1.00 MN

THIS REPORT IS ONE OF A SERIES OF SUCH REPORTS ON THE FOLLOWING SUBJECTS - PLANNING, EXPERIMENTAL PROGRAM IN ZPR-III, FUEL DEVELOPMENT, INSTRUMENTATION, CORE DYNAMICS, REACTIVITY CONTROL, FUEL HANDLING, AND CORE DESIGN

REACTOR, BREEDER + CRITICAL ASSEMBLY + DOPPLER COEFFICIENT + REACTOR, FAST + FUEL HANDLING + HEAT TRANSFER + REACTOR, LMCR + INSTRUMENT, TEMPERATURE + TEST, PHYSICS + SEFOR (RE) + RESEARCH CONTRACT + INSTRUMENTS, MISC. + FUEL ELEMENTS

9-011441 DATED 1966
REICKS GH
SODIUM LEVEL DETECTORS IN THE ENRICO FERMI ATOMIC POWER PLANT
ATOMIC POWER DEVELOPMENT ASSOCIATES
APDA-300 +. 44 PAGES, NOVEMBER 30, 1966, CFSTI, \$2.00 CY, \$0.50 MN

ALL LEVEL DETECTORS FOR HIGH-TEMPERATURE OPERATION IN LIQUID METALS WERE EVALUATED. THE MOORE SPECIAL MODEL 15 AND, LATER THE TAYLOR NAK-FILLED DETECTOR WERE INSTALLED. RESULTS OF THE FERMI-PLANT EXPERIENCE LEADS TO THE CONCLUSION THAT A DESIGN AND DEVELOPMENT PROGRAM IS NEEDED FOR LIQUID-METAL PROBES.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + INSTRUMENT, PROCESS + REACTOR, LMCR + MOLTEN FUEL + INSTRUMENT, LIQUID LEVEL + METAL, LIQUID

9-012069 DATED 1965
PRELIMINARY SAFEGUARDS SUMMARY REPORT - APPLICATION TO THE UNITED STATES ATOMIC ENERGY COMMISSION FOR CONSTRUCTION PERMIT AND OPERATING LICENSE
SOUTHWEST ATOMIC ENERGY ASSOCIATION + GENERAL ELECTRIC COMPANY
219 PAGES, 101 FIGURES, 24 TABLES, JANUARY 1965, DOCKET NO. 50-231, PDR

THE OPERATION AND CONTROL OF THE SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DURING NORMAL CONDITIONS WILL BE FROM A CENTRAL CONTROL ROOM. THE ROOM WILL CONTAIN THE NECESSARY READOUTS, ALARMS, AND MANUAL CONTROL DEVICES TO ALLOW THE OPERATORS TO START, MAINTAIN CONTROL OF THE MAIN PROCESS DURING NORMAL AND TRANSIENT CONDITIONS, AND TO SHUT THE PROCESS DOWN. THE PLANT INSTRUMENTATION IMPORTANT FOR REACTOR CONTROL AND SAFETY IS DIVIDED INTO SYSTEMS - NEUTRON MONITORING, REFLECTOR CONTROL, AREA AND PROCESS RADIATION MONITORING, AND PROCESS INSTRUMENTATION. EQUIPMENT USED FOR SAFETY-SYSTEM FUNCTIONS WILL BE INDEPENDENT OF THOSE USED FOR CONTROL PURPOSES.

CONTROL SYSTEM + REACTOR, FAST + REACTOR, LMCR + REACTOR, SAFETY RESEARCH + REACTOR SAFETY SYSTEM + INSTRUMENT, CONTROL + SEFOR (RE) + REPORT, PSAR + REACTOR, LMFBR + RADIATION MONITORS

9-012086 DATED 1964
CARLSON WJ + GUNBY AL
ANALYSIS OF POWER RAMPS WITH AN ANALOG COMPUTER
ATOMICS INTERNATIONAL, CANUGA PARK
NAA-SR-7929 +. 39 PAGES, 11 FIGURES, 2 TABLES, 7 REFERENCES, JUNE 15, 1964, CFSTI \$0.75 CY

ANALYSIS OF POWER RAMPS ON THE SODIUM REACTOR EXPERIMENT BY USE OF THE ANALOG COMPUTER IS A FAST, ACCURATE, AND SIMPLE METHOD FOR DETERMINING THE HEAT TRANSFER PARAMETERS. THIS METHOD REQUIRES A MINIMUM OF REACTOR TIME. RESULTS ON THE SRE CORE II ARE REPRODUCIBLE TO PLUS OR MINUS 5 PERCENT, SHOWING A PROMPT POWER COEFFICIENT OF REACTIVITY OF MINUS 2.05 CENTS/MWT AT 1000 GAL/MIN MAIN PRIMARY FLOW. THE SINGLE-LAG TEMPERATURE FEEDBACK APPROXIMATION IS ADEQUATE FOR MOST OPERATING CONDITIONS OF THE REACTOR.

COMPUTER, ANALOG + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + REACTOR DYNAMICS + ANALYTICAL MODEL + SRE (RE) + MEASUREMENT, REACTIVITY

9-012130 DATED 1964

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-012130 *CONTINUED*
MOORE EA + PECKINPAUGH CL
ON-LINE SHIM ROD CALIBRATION
ATOMIC INTERNATIONAL, CANOGA PARK
NAA-SK-9511 +. 21 PAGES, 6 FIGURES, AUGUST 1, 1964, CFSTI \$0.50 CY

A TECHNIQUE FOR CALIBRATING SHIM RODS WHEN ON-LINE, USING AN ANALOG REACTOR SIMULATOR, WAS PROVEN FEASIBLE. THIS NEW METHOD INCREASES PLANT AVAILABILITY, ELIMINATES OPERATIONAL HAZARDS, AND USES INEXPENSIVE ANALOG EQUIPMENT. IT IS NOT NECESSARY TO ATTACH SPECIAL DEVICES TO DRIVE THE CONTROL ROD OR TO LEAVE A PILE OSCILLATOR IN THE REACTOR CORE. AN ACCURACY OF PLUS OR MINUS 5 PERCENT, COMPARED WITH RESOLUTION OF PLUS OR MINUS 2 PERCENT OBTAINED WHEN USING ZERO POWER PERIOD MEASURING TECHNIQUES, WAS DEMONSTRATED WITH SRE CORE II. IT IS PREDICTED THAT AN ACCURACY OF PLUS OR MINUS 3 PERCENT CAN BE ACHIEVED FOR THE SRE CORE III. THIS IS SUFFICIENT TO MAINTAIN AN ACCURATE REACTIVITY HISTORY PROGRAM FOR MONITORING CORE NORMALCY WITHOUT DISRUPTING REACTOR OPERATION.

COMPUTER, ANALOG + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + SRE (RE) + TEST, PLANT RESPONSE + CONTROL ROD CALIBRATION + INSTRUMENT, TESTING

9-012256 DATED 1964
HEITEMES CJ
REACTOR CONTROL MECHANISMS
UNITED STATES ATOMIC ENERGY COMMISSION, DIVISION OF REACTOR DEVELOPMENT
WASH-1054(PIL) +. 191 PAGES, 4 TABLES, 24 FIGURES, SEPTEMBER 1964, PROCEEDINGS OF THE MEETING ON FAST REACTOR CONTROL MECHANISMS, HELD AT ATOMIC ENERGY COMMISSION HEADQUARTERS BUILDING GERMANTOWN, MARYLAND, SEPTEMBER 16 AND 17, 1964, CFSTI \$5.00

IN SEPTEMBER 1964, THE U.S. AEC HELD AN INFORMAL MEETING TO DISCUSS AND EXAMINE A NUMBER OF FAST-REACTOR CONTROL SYSTEMS. THE PURPOSE OF THE MEETING WAS TO COORDINATE THE EFFORTS OF VARIOUS CONTRACTORS NOW ENGAGED IN THE DEVELOPMENT OF SUCH SYSTEMS AND TO ACQUAINT COMMISSION PERSONNEL AND CONTRACTORS WITH THE EQUIPMENT AND TECHNIQUES. THE FOLLOWING PAPERS CAME OUT OF THE CONFERENCE - FAST-REACTOR-CONTROL FUNDAMENTALS, EBR-I CONTROL-DRIVE MECHANISMS, EBR II CONTROL DRIVES, FALLET CONTROL DRIVES, SEFOR CONTROL-DRIVE SYSTEM, LAMPRE I CONTROL-ELEMENT-DRIVE AND FERMI CONTROL RODS.

REACTOR, BREEDER + CONTROL SYSTEM + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + FFTF (TR) + SEFOR (RE) + LAMPRE 1 (RE) + CONTROL RODS + CONTROL ROD DRIVE MECHANISMS

9-012303 ALSO IN CATEGORY 17 DATED 1963
CASHETTA J
NUCLEAR INSTRUMENT ANOMALIES DURING RISE-TO-POWER TESTING AT THE HALLAM NUCLEAR POWER FACILITY
ATOMIC INTERNATIONAL, CANOGA PARK, CALIFORNIA
NAA-SR-8714 +. 47 PAGES, JULY 13, 1963, CFSTI \$2.00 CY

SEVERAL PROBLEMS ASSOCIATED WITH THE NUCLEAR INSTRUMENTATION WERE DISCOVERED DURING THE RISE-TO-POWER TESTING PHASES OF THE HALLAM NUCLEAR POWER FACILITY. THE PROBLEMS RANGED IN SEVERITY FROM THE TYPE JUSTIFYING PREVENTATIVE MAINTENANCE TO THOSE DICTATING ESSENTIAL MODIFICATION. BECAUSE OF THE LIMITED TIME AVAILABLE FOR EXPERIMENTATION AND INVESTIGATION, IT WAS NOT POSSIBLE TO PERFORM A COMPLETE ANALYTICAL STUDY OF THE PROBLEMS. THE REPORT ENCOMPASSES THE REMEDIAL ACTION INITIATED TO PROVIDE SATISFACTORY OPERATION OF THE NUCLEAR INSTRUMENTATION.

REACTOR, GRAPHITE MODERATED + HALLAM (LMK) + INSTRUMENT, NUCLEAR + REACTOR, LMCR + TEST, PLANT RESPONSE + FAILURE, INSTRUMENT + MODIFICATION

9-012383 ALSO IN CATEGORY 17 DATED 1966
HESS J
DESIGN AND OPERATING EXPERIENCE WITH THE CONTROL AND SAFETY ROD DRIVE MECHANISMS FOR THE ENRICO FERMI FAST BREEDER REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT
APDA-301 +. 221 PAGES, 42 FIGURES, 6 TABLES, JUNE 1, 1966, CFSTI \$6.00 CY, \$1.25 MN

TRACES THE EVOLUTION OF THE DESIGN AND POLICY FOR THE CONTROL-ROD DRIVES. TESTING AND OPERATING EXPERIENCE ARE SUMMARIZED. DURING THE FIRST YEAR OF REACTOR OPERATION, ONLY TWO SIGNIFICANT MALFUNCTIONS OCCURRED - FAILURE OF A LATCH BELLOWS ALLOWED SODIUM TO FREEZE ONE SAFETY ROD, AND OPENING A BLOCKING VANE TOO FAST WITH A ROTARY SOLENOID DISTURBED THE LIMIT-SWITCH SETTINGS, SOLVED BY OPENING THE VANE WITH A SMALL GEAR MOTOR. EXPERIENCE WITH THESE UNITS ARE EVALUATED AND SUGGESTIONS MADE FOR IMPROVEMENT. MANY PROBLEMS RESULTED FROM LACK OF OPERATING EXPERIENCE, AND THE MOST SERIOUS DIFFICULTIES CAME FROM NOT TESTING COMPONENTS OR PROTOTYPES IN HOT SODIUM BEFORE ATTEMPTING TO INSTALL THEM IN THE REACTOR. EXPERIENCE TO DATE INDICATES THAT CONSIDERATION SHOULD BE GIVEN TO DETAILING PROCEDURES FOR OPERATION, CALIBRATION, AND TROUBLESHOOTING. 140 PAGES OF APPENDICES PROVIDE DETAILED SPECIFICATIONS AND CALCULATIONS.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + CONTROL ROD, SHIM SAFETY + CONTROL ROD SCRAM MECHANISM + DESIGN STUDY + MODIFICATION + TEST, CONTROL ROD DRIVE + FAILURE, SCRAM MECHANISM + CONTROL ROD DRIVE MECHANISMS

9-013337 DATED 1962
ULTRASONIC INSTRUMENTATION FOR NUCLEAR APPLICATIONS - BIMONTHLY PROGRESS REPORT NO. 5
AEROPROJECTS INCORPORATED, WEST CHESTER, PENNSYLVANIA

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-013337 *CONTINUED*
NYO-10555 +. 18 PAGES, 8 FIGURES, 5 REFERENCES, AUGUST 1962, CFSTI \$1.60 CY, \$0.80 MN

A DESIGN FOR AN ULTRASONIC PROBE TO DETECT THE POSITION OF THE FOAM-WATER INTERFACE IN A BOILING WATER REACTOR IS DESCRIBED. AN EXPERIMENTAL ULTRASONIC INSTRUMENT FOR DETERMINING LIQUID LEVELS IN FUSED-SALT AND LIQUID-METAL REACTORS WAS MADE AND TESTED ON STAND-IN LIQUIDS. WITH A MODIFIED VERSION IN WHICH INTERFACIAL REFLECTION PROBLEMS WERE GREATLY REDUCED, LIQUID LEVELS WERE DETERMINED SEMICONTINUOUSLY. A DESIGN FOR AN IMPROVED PROBE IS PRESENTED.

REACTOR, BWR + INSTRUMENT, PROCESS + REACTOR, LMCR + INSTRUMENT, LIQUID LEVEL

9-013355
BURKE OW
REACTOR BEHAVIOR ANALYSIS USING ANALOG COMPUTERS
OAK RIDGE NATIONAL LABORATORY
3 PAGES, NUCLEAR SAFETY 4(3), PAGES 50-52, MARCH 1963

DATED 1963

THE REPORT OF HOLLEY CARBURETOR CO. ON THE SYSTEM ANALYSIS AND SIMULATION OF THE ENRICO FERMI POWER PLANT IS REVIEWED BRIEFLY. THE DEVICES USED TO SIMULATE VARIOUS COMPONENTS AND PHENOMENA OF THE PLANT ARE DESCRIBED BRIEFLY.

COMPUTER, ANALOG + REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR

9-014190
SRE ROD DROP-TIME MEASUREMENT
GENERAL NUCLEAR ENGINEERING CORP.
3 PAGES, 2 FIGURES, POWER REACTOR TECHNOLOGY 7(4), PAGES 377-379, (FALL 1964)

DATED 1964

A SYSTEM WAS DEVELOPED TO DETERMINE THE SCRAM TIMES OF THE SAFETY RODS IN THE SODIUM REACTOR EXPERIMENT (SRE). THE SYSTEM USES TWO TIMERS (ELECTROMAGNETIC) TO MEASURE THE INDIVIDUAL COMPONENTS OF THE TOTAL TIME. THE FREE-FALL COMPONENT AND SNUBBER COMPONENT OF TOTAL DROP TIME ARE BOTH MEASURED. THE TIMING SYSTEM APPEARS TO BE ONE THAT WOULD BE ADAPTABLE TO VARIOUS REACTORS.

REACTOR, LMCR + SRE (RE) + RESPONSE TIME + CONTROL ROD SCRAM MECHANISM + TEST, CONTROL ROD DRIVE

9-016164
MACFARLANE CR
AN ANALYTIC STUDY OF THE TRANSIENT BOILING OF SODIUM IN REACTOR COOLANT CHANNELS
ARGONNE NATIONAL LABORATORY, ILLINOIS
ANL-7222 +. 117 PAGES, 48 FIGURES, 11 TABLES, JUNE 1966

ALSO IN CATEGORIES 5 AND 6

DATED 1966

TWO TYPES OF TRANSIENTS WERE STUDIED - (1) MILD REACTIVITY ADDITION AND (2) PUMP SHUTOFF WITH FLOW COASTDOWN. NUMERICAL SOLUTIONS OBTAINED FOR THE NONLINEAR FORM OF THE COUPLED FLUID DYNAMIC AND HEAT CONDUCTION PARTIAL DIFFERENTIAL EQUATION BY MEANS OF IMPLICIT FINITE-DIFFERENCE EQUATIONS AND AN ITERATIVE ALGOL COMPUTER PROGRAM. SONIC EFFECTS AND CRITICAL FLOW PHENOMENA WERE NOT TREATED. TWO-PHASE COOLANT DENSITY IS PRESSURE DEPENDENT, AND TEMPORAL- AND SPATIAL-VARYING SLIP VELOCITY ARE INCLUDED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTIVITY, EXCESS + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + REACTOR TRANSIENT + SODIUM + VOID COEFFICIENT + METAL, LIQUID + FLOW, TWO PHASE

9-016943
PUPPER GF + KNOX AE
FARET IN-CORE INSTRUMENT DEVELOPMENT
ARGONNE NATIONAL LAB., ILL.
ANL-7161 +. 75 PAGES, 33 FIGURES, 19 TABLES, 46 REFERENCES, JULY 1966

DATED 1966

PROBLEM - MEASURE FUEL-PIN AND COOLANT TEMPERATURES, COOLANT FLOW RATES WITHIN FUEL ASSEMBLIES, FISSION-GAS PRESSURE IN FUEL PINS, AND NEUTRON AND GAMMA FLUXES IN AN EXPERIMENTAL SODIUM-COOLED FAST REACTOR. APPROACH - THE FOLLOWING INSTRUMENTATION WAS DEVELOPED FOR INSTRUMENTED FUEL SUBASSEMBLIES - FUEL-PIN THERMOCOUPLES, COOLANT THERMOCOUPLES, TURBINE FLOWMETERS, MAGNETIC FLOWMETERS, THERMAL FLOWMETERS, THERMAL FLOWMETERS (USING A HEATER AND MEASUREMENT OF TEMPERATURE USE), AND ELECTRICAL CONNECTORS AND CABLES. RESULTS - A PORTION OF THE INSTRUMENTATION WAS TESTED WITH TEMPERATURE AND RADIATION BEFORE PROJECT WAS TERMINATED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + INSTRUMENT, IN CORE + REACTOR, LMCR + OXIDE + SODIUM + COMPARISON, THEORY AND EXPERIENCE + THORIUM + STEEL, STAINLESS + FFTF (TR) + TANTALUM + MEASUREMENT, TEMPERATURE + HIGH TEMPERATURE + TUNGSTEN + DESIGN STUDY + INSTRUMENT, FLOW + INSTRUMENT, TEMPERATURE + TEST, INSTRUMENT RESPONSE + EQUIPMENT DESIGN + RHENIUM + REFRACTORY MATERIAL + THERMOCOUPLE + FUEL ELEMENTS

9-017132 ALSO IN CATEGORY 6

DATED 1967

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-017132 *CONTINUED*
KLINKMAN AE + ZAMAN SU

THE STABILITY OF THE ENRICO FERMI REACTOR UNDER AUTOMATIC CONTROL
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

CONF-670-713 +. 2 PAGES, 2 FIGURES, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 10, PAGE 6 AND 7, (JULY 1967),
CONFERENCE ON REACTOR OPERATING EXPERIENCE, REACTOR OPERATIONS DIVISION OF THE AMERICAN NUCLEAR SOCIETY,
ATLANTIC CITY, JULY 23-26, 1967

TRANSFER-FUNCTION MEASUREMENTS SHOWED A RESONANCE WHEN UNDER AUTOCONTROL (REG.-ROD VELOCITY MADE
PROPORTIONAL TO TIME DERIVATIVE OF NEUTRON FLUX). NYQUIST ANALYSIS SHOWED THAT THIS WAS NOT A
SIGN OF INSTABILITY (BECAUSE OF PHASE-ANGLE CONSIDERATIONS), THE MARGIN BEING 68 DEG AT 67 MW AND
INCREASING WITH POWER. ANOTHER NYQUIST ANALYSIS (ONE GROUP OF DELAYED NEUTRONS AND NO FEEDBACK
OTHER THAN AUTOCONTROL) SHOWED A BROAD RANGE OF STABLE VALUES, BUT A RANGE OF REALISTIC VALUES
FOR WHICH INSTABILITY COULD OCCUR.

CONTROL SYSTEM + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + REACTOR STABILITY + TRANSFER FUNCTION

9-017479 ALSO IN CATEGORY 17 DATED 1966
SMITH RK + DUE CB
CLADDING FAILURE SIMULATION TESTS IN EBR-II
ARGONNE NATIONAL LAB., ILL.
ANL-7067 +. 24 PAGES, 17 FIGURES, 1 TABLE, REFERENCES, DECEMBER 1966

TESTS, USING A SINGLE UNCLAD FUEL ROD, ON (1) CWD (CHANGED WIRE DETECTOR, BASED ON DETECTION OF
FISSION GAS IN THE CORE COVER GAS), AND (2) FERD (FUEL ELEMENT RUPTURE DETECTOR), BASED ON
DETECTION OF DELAYED-NEUTRON-EMITTING FISSION PRODUCTS IN PRIMARY COOLANT. COUNTING RATES WERE
MEASURED WITH NORMAL CONDITIONS AND WITH A SINGLE UNCLAD MARK-1A FUEL PIN IN THE CORE. SIGNAL-TO-
NOISE RATIOS ARE 12 FOR CWD AND 6.7 FOR FERD. FERD HAS A MUCH SHORTER TIME RESPONSE (20 SEC) BUT
WILL NOT USUALLY RESPOND TO FAILURES IN THE FUEL-ROD GAS PLENUM. SINCE CWD IS SENSITIVE TO BOTH
BOND-TYPE AND GAS-TYPE FAILURES, THE SYSTEMS COMPLEMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + REACTOR, LMCR + TEST, PROOF + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + REPORT,
OPERATIONS ANALYSIS + REACTOR, ERDA OWNED

9-018506 DATED 1963
RADCLIFF JG
DESIGN AND DEVELOPMENT OF THE SAFETY-ROD DROP-TIME TEST SYSTEM
ATOMICS INTERNATIONAL
NAA-SR-7880 +. 16 PAGES, 5 FIGURES, OCTOBER 31, 1963

THE SRE SAFETY-ROD DROP-TIME SYSTEM CONSISTS OF ELECTRONIC TIME-INTERVAL METERS AND MAGNETIC
PICKUPS TO MEASURE FREE-FALL DROP TIME AND SNUBBER TIME OF THE SAFETY-ROD POISON ELEMENTS PRIOR
TO ALL NORMAL REACTOR STARTUPS. THE ELECTRONIC TIMING SYSTEM IS PANEL MOUNTED IN THE REACTOR
CONTROL ROOM. THE SNUBBER TIME AND FREE-FALL TIME ARE DISPLAYED ON FOUR-DECADE LIGHTED DIALS ON
THE TWO TIMERS AND ARE READ DIRECTLY IN UNITS OF TIME. THE COMPLETE DROP-TIME MEASUREMENTS CAN
BE MADE AT THE TEST PANEL BY ONE INDIVIDUAL IN LESS THAN TEN MINUTES. THE TIMING SYSTEM WAS
DEVELOPED TO PROVIDE A RAPID TIME-MEASUREMENT TECHNIQUE TO CHECK FOR OBSTRUCTIONS TO FREE-FALL
AND TO MEASURE SNUBBER WEAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + SRE (RE) + CONTROL ROD, SHIM SAFETY + CONTROL ROD SCRAM MECHANISM +
INSTRUMENT, TESTING

9-018814 DATED 1967
SCOTT CC + HUEBOTTER PR + CALLEN RC
POTENTIAL APPLICATIONS FOR AN UNDER-SODIUM ULTRASONIC SCANNING DEVICE
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-205 +. 38 PAGES, FIGURES, REFERENCES, JULY 1967

PREVIOUS AEC-SPONSORED DEVELOPMENT WORK ON VARIOUS CONCEPTS OF UNDER-SODIUM SCANNING DEVICES
RESULTED IN THE SELECTION OF A SINGLE APPROACH, DESCRIBED HERE, THAT REPRESENTS THE BEST
COMPROMISE OF SIMPLICITY, FLEXIBILITY, RANGE, AND RESOLUTION. IN THE CONTEXT OF THE CAPABILITIES
AND LIMITATIONS OF THIS APPROACH, WHICH EMPLOYS A NARROW-BEAM ULTRASONIC TRANSDUCER WITH A
METALLIC FOCUSING LENS OPERABLE IN SODIUM AT 600 F, NUMEROUS APPLICATIONS IN EXISTING AND FUTURE
SODIUM-COOLED REACTORS AND ASSOCIATED RESEARCH AND DEVELOPMENT PROJECTS ARE POSSIBLE. A SURVEY
OF THESE APPLICATIONS HAS BEEN MADE, EVALUATED, AND REDUCED TO A SINGLE DEMONSTRATION OBJECTIVE
WHICH REPRESENTS A COMMON DENOMINATOR OF MOST OF THE APPLICATIONS OF GREATEST POTENTIAL INTEREST
IN THE LIQUID METAL FAST BREEDER REACTOR (LMFBR) DEVELOPMENT PROGRAM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR, LMCR + SODIUM + INSTRUMENT, COMPONENT +
INSTRUMENT, SURVEILLANCE + SURVEY

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-019115 DATED 1966
STEELE UP + KELLOGG LG + DEWART WG
DEVELOPMENT OF THE CONTROL SYSTEM FOR SNAP REACTORS
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
18 PAGES, 26 FIGURES, 6 TABLES, IEEE TRANS. NUCL. SCIENCE 13(1), PAGES 206-223 (FEB. 1966)

THE SNAP 10A, 2, AND 8 NUCLEAR REACTOR SYSTEMS ARE DESIGNED TO PROVIDE ELECTRICAL POWER ON SPACE VEHICLES OR ENVIRONMENTS OF ULTRAHIGH VACUUM. ALTHOUGH EACH SYSTEM HAS DIFFERENT POWER CAPABILITIES, THE CONTROL COMPONENTS ARE BASICALLY THE SAME. DEVELOPMENT OF THE VARIOUS CONTROL COMPONENTS AND SUBCOMPONENTS IS DISCUSSED. THE ELECTROMECHANICAL ACTUATOR, BEARINGS, AND GEARS FOR EACH SNAP SYSTEM ARE REVIEWED IN DETAIL. A CONTROL-DRUM-DRIVE MECHANISM THAT PROVIDES FAIL SAFE SCRAM CAPABILITIES FOR GROUND TESTING OF THE REACTOR SYSTEM HAS BEEN DESIGNED AND PROOF TESTED. THE DETAILS OF THIS DESIGN AS IT APPLIES TO SNAP 10A SYSTEMS ARE PRESENTED.

CONTROL SYSTEM + REACTOR, LMCR + SNAP 10A (SR) + SNAP 2 (SR) + SNAP 8 (SR) + REACTOR, SPACE + SERVOMECHANISM + TEST, COMPONENT

9-019442 DATED 1966
GLASGOW LE
ASSESSMENT OF THE SODIUM REACTOR EXPERIMENT RELIABILITY
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA
NAA-SK-MEMU-12224 +. 10 PAGES, FIGURES, TABLES, REFERENCES, OCTOBER 31, 1966

THE AIM OF THE PROGRAM WAS TO COMPLETE THE CONVERSION OF THE SODIUM REACTOR EXPERIMENT FROM AN EXPERIMENTAL FACILITY INTO A ROUTINELY OPERABLE, HIGH-TEMPERATURE, SODIUM-COOLED, TEST FACILITY. CONSEQUENTLY, GREAT EMPHASIS WAS PLACED ON IMPROVING PLANT RELIABILITY. IN ADDITION TO CONSERVATIVE DESIGN, CONSIDERABLE ATTENTION WAS GIVEN TO QUALITY ASSURANCE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, EQUIPMENT + REACTOR, LMCR + RELIABILITY ANALYSIS + OPERATING EXPERIENCE SUMMARY + QUALITY ASSURANCE + INSTRUMENTS, MISC.

9-019500 DATED 1963
PATEKS NT
SPECIFICATION FOR CHECK VALVE FOR LIQUID SODIUM SYSTEM. SPECIFICATION NO. 60-2. REVISION NO. 3
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
NP-16127 +. 10 PAGES, 1 FIGURE, JULY 1, 1963

THIS SPECIFICATION COVERS THE DESIGN REQUIREMENTS, MATERIALS, FABRICATION, INSPECTION, TESTING, AND DELIVERY OF THREE CHECK VALVES THAT WILL REPLACE THOSE NOW IN THE PRIMARY SYSTEM OF THE ENRICO FERMI ATOMIC POWER PLANT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FASI + FERMI (LMFBR) + REACTOR, LMCR + SODIUM + OPERATING EXPERIENCE SUMMARY + INSTRUMENT, FLOW + ORIFICE + VALVES

9-019758 ALSO IN CATEGORY 17 DATED 1959
LUNDHCLM JG + JARRETT AA
SRE POWER EXCURSION NOT HALTED BY FAST PERIOD
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SK-4488 +. 3 PAGES, 2 TABLES, PAGES 111-11 TO 13 FROM SRE FUEL ELEMENT DAMAGE, NOVEMBER 30, 1959

AFTER STARTING A POWER INCREASE, THE LEVEL PERSISTED IN RISING FASTER THAN EXPECTED EVEN THOUGH CONTROL RODS WERE BEING SLOWLY INSERTED TO HOLD IT BACK. THREE 50-SEC POSITIVE-PERIOD TRANSIENTS OCCURRED, FOLLOWED BY A 7-1/2 SEC PERIOD. THE REACTOR WAS SCRAMMED. PEAK POWER WAS 24 MW. A SETBACK WAS NOT INITIATED AUTOMATICALLY AT A 10-SEC PERIOD BY A SWITCH ACTUATED BY A NOTCH IN A CAM, BECAUSE IT WOULD OPERATE WELL ONLY IF THE RATE OF CHANGE OF THE PERIOD WAS SLOW. THE CAM WAS MODIFIED TO OPERATE AT ALL RATES OF CHANGE. A POSSIBLE EXPLANATION FOR THE TRANSIENT IS THE PARTING AND DROPPING OF A LOWER SECTION OF A SINGLE ELEMENT OR THE DOPPLER EFFECT (SEE SUPPL. FOR FINAL ANALYSIS).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

DOPPLER EFFECT + FAILURE, FUEL ELEMENT + REACTOR, LMCR + REACTOR TRANSIENT + SRE (RE) + FAILURE, INSTRUMENT + REACTIVITY EFFECT, ANOMALOUS + INSTRUMENT, PERIOD + RECORDERS

9-019862 ALSO IN CATEGORY 17 DATED 1961
MODIFICATIONS TO REACTOR
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SK-4488 (SUPPL) +. 14 PAGES, 9 FIGURES, FROM SRE FUEL ELEMENT DAMAGE, FINAL REPORT, JUNE 30, 1961

MODIFICATIONS TO AVOID A SECOND MAJOR FUEL-ELEMENT FAILURE INCLUDE ELIMINATING THE USE OF TETRALIN AS A SERVICE COOLANT, INCREASING CLEARANCE BETWEEN ELEMENT AND CHANNEL, RELOCATION OF THE ELEMENT

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-019862 *CONTINUED*

ORIFICE PLATE, ADDING A STRUCTURAL MEMBER TO THE ROD CLUSTER, ATTACHING A STRAINER AT THE BOTTOM OF THE ELEMENT, INSTALLING A CONTINUOUS COVER-GAS MONITOR, INSTALLING A TRANSIENT MONITOR (ACTIVATED BY AN ABNORMAL SIGNAL SUCH AS A SCRAM) TO MONITOR FLUX, PERIOD, TEMPERATURES, FLOW, AND ROD POSITION. TWO RECORDERS WERE INSTALLED TO MEASURE FUEL-SLUG TEMPERATURES. THREE 20-PEN RECORDERS WERE INSTALLED TO MONITOR PROTECTIVE SYSTEM ACTUATIONS, AND AN ADDITIONAL PROTECTIVE SYSTEM WAS ADDED TO PREVENT A LARGE MISMATCH BETWEEN FLUX AND POWER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, FUEL ELEMENT + INSTRUMENT, PROTECTIVE + REACTOR, LMCR + SRE (RE) + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + MODIFICATION + FAILURE, CLADDING + INSTRUMENTS, MISC. + FUEL ELEMENTS + RECORDERS

9-020638 ALSO IN CATEGORY 17 DATED 1966
MCCARTHY WJ + JENS WH
ENRICO FERMI FAST BREEDER REACTOR
POWER REACTOR DEVELOPMENT CO. + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
4 PAGES, NUCLEAR NEWS 10(11), PAGE 54-57, (NOVEMBER 1966)

SHORT DISCUSSION OF CURRENT STATUS, MOSTLY WHAT HAS BEEN LEARNED. AN INLET STRAINER, CORE-PLATE STANDOFF PINS, OR PROVISION FOR CROSS FLOW WOULD HELP ELIMINATE FUTURE MELTDOWNS. REACTIVITY RATE-OF-CHANGE INFORMATION AND A FASTER FISSION-PRODUCT MONITOR (DELAYED NEUTRONS AT COKE EXIT TO SUPPLEMENT PRESENT SENSITIVE COVER-GAS AS DETECTOR WITH 13-MIN TRANSPORT LAG) IS NEEDED. FEW PROVISIONS WERE MADE FOR INCIDENT RECOVERY, E.G., DIFFICULT ACCESS TO WHOLE CORE, NO PROVISION FOR DRAINING VESSEL OR FOR ON-SITE INSPECTION OF IRRADIATED FUEL.

FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + INCIDENT + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + INCIDENT, RECOVERY FROM + FUEL HANDLING MACHINE + REMOTE MANIPULATING AND VIEWING + FLOW BLOCKAGE + INSTRUMENT, OPERATING REACTIVITY

9-020709 DATED 1967
CALLEN RL + PROUTEAU LP + FRIEDLAND AJ
RESPONSE OF FERMI FUEL SUBASSEMBLY OUTLET THERMOCOUPLES UNDER NORMAL AND FUEL-FAILURE CONDITIONS
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + ELECTRICITE DE FRANCE
2 PAGES, ANS TRANSACTIONS 10(2), PAGE 637 AND 638, (NOVEMBER 9, 1967), PRESENTED AT THE 1967 WINTER MEETING OF THE AMERICAN NUCLEAR SOCIETY, CHICAGO, ILLINOIS, NOVEMBER 5-9, 1967

HIGH-POWER TESTS INCLUDED MEASUREMENT OF CORE AND INNER RADIAL BLANKET LOCAL COOLANT OUTLET TEMPERATURES TO DETERMINE STATIC THERMAL RESPONSE. RESPONSE OF THE TEMPERATURE INSTRUMENTATION WAS EXAMINED IN ISOTHERMAL TESTS. REACTOR POWER AND ELECTROMAGNETIC FLOWMETERS WERE CALIBRATED DURING HEAT-BALANCE MEASUREMENTS ON THE STEAM-FEEDWATER SIDE OF THE STEAM GENERATORS.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + COMPARISON, THEORY AND EXPERIENCE + INSTRUMENT CALIBRATION + OPERATING EXPERIENCE SUMMARY + INSTRUMENT, FLOW + INSTRUMENT, TEMPERATURE + TEST, INSTRUMENT RESPONSE + INSTRUMENT, POWER RANGE

9-022457 ALSO IN CATEGORY 17 DATED 1967
EBR 2 STEAM GENERATOR LEVEL CONTROLLER MALFUNCTION
ARGONNE NATIONAL LAB., ARGONNE, ILL.
ANL-7399 +. 1 PAGE, PAGE 94 OF ARGONNE NATIONAL LABORATORY. REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, NOVEMBER 1967, DECEMBER 1967

ON THE NOV. 12, 1967, STARTUP, AT 20 MW THE PRIMARY BULK-SODIUM TEMPERATURE BEGAN INCREASING AND COULD NOT BE CONTROLLED BY ADJUSTING SECONDARY FLOW. A MALFUNCTION OF THE STEAM-DRUM WATER-LEVEL CONTROLLER/ALARM ALLOWED THE DRUM TO BOIL DRY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, COMPONENT + EBR 1 AND 2 (RE) + CONTROLLER + REACTOR, LMCR + INCIDENT, EQUIPMENT + INSTRUMENT, LIQUID LEVEL + STEAM GENERATOR + INDEPENDENCE

9-023146 DATED 1959
ANALYSIS AND SIMULATION STUDY OF THE OPERATING CONTROL SYSTEM FOR THE ENRICO FERMI ATOMIC POWER PLANT. REPORT 59
HOLLEY CARBUETOR CO., ELECTRO MECHANICAL DIV., WARREN, MICHIGAN
TID-11,025 +. 260 PAGES, FIGURES, TABLES, NOV. 1, 1959

DEVELOPED A PLANT SIMULATOR AND THE CONCEPTUAL PLANT CONTROL SYSTEM FOR ONE OF THE THREE CIRCUITS OF THE ENRICO FERMI ATOMIC POWER PLANT. AN ANALOG SIMULATOR WITH SPECIALLY DESIGNED ELECTRO-MECHANICAL EQUIPMENT IS DEVELOPED. INCLUDED IN THE SIMULATION ARE ALL ELEMENTS BUT THE STEAM TURBINE, I.E., THE REACTOR, INTERMEDIATE HEAT EXCHANGER, STEAM GENERATOR, MIXING AND TRANSPORT DELAY PHENOMENA, AND STEAM-SYSTEM HYDRAULICS. WITH THE SUPPORT OF A STEADY-STATE DIGITAL ANALYSIS OF THE PLANT, A POLICY OF PLANT CONTROL IS EVOLVED. CONCEPTUAL DESIGNS WERE TESTED ON THE SIMULATOR AND AN OPTIMUM DESIGN WAS SELECTED AND TUNED.

COMPUTER, ANALOG + REACTOR, BREEDER + CONTROL SYSTEM + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + INSTRUMENT, CONTROL + REACTOR CONTROL + TEST, PLANT RESPONSE + ANALOG SIMULATION + TEST, SYSTEM OPERABILITY +

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-023146 *CONTINUED*
SYSTEM DESCRIPTION + THEORETICAL INVESTIGATION + SIMULATION + MATHEMATICAL TREATMENT

9-023327 DATED 1967
SWANSON CD + WOOD MR + CUNE KR
FAST FLUX TEST FACILITY INSTRUMENTATION AND CONTROL PROGRAM
BATTTELLE MEMORIAL INSTITUTE, RICHLAND, WASHINGTON
ANL-7380 +. 5 PAGES, 1 FIGURE, REFERENCES, PAGES 4-8 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL
INSTRUMENTATION AND CONTROL, MARCH 2, 1967

DESCRIBES THE RELATION OF THE FFTF INSTRUMENTATION AND CONTROL PROGRAM TO THE LMFBR PROGRAM.
INSTRUMENTATION AND CONTROL TECHNOLOGY NEEDED TO FULFILL FFTF FUNCTIONAL TEST REQUIREMENTS, AND
OPERATIONAL AND SAFETY REQUIREMENTS ARE SUMMARIZED. A BRIEF DESCRIPTION OF THE MAJOR FFTF
REACTOR CONCEPTS PRESENTLY UNDER STUDY AND THE TYPES OF INSTRUMENTATION PROBLEMS ASSOCIATED WITH
THESE CONCEPTS IS PRESENTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY,
\$0.65 MICROFICHE

DESIGN CRITERIA + REACTOR, FAST + INSTRUMENT, IN CORE + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + TEST,
COMPONENT + INSTRUMENT, TEMPERATURE + INSTRUMENT, TESTING + IN CORE MEASUREMENT + SYSTEM DESCRIPTION +
EQUIPMENT DESIGN + INSTRUMENTS, MISC. + FUEL ELEMENTS

9-023328 DATED 1967
KNOX AE + PCPPER GF
ANL PARTICIPATION IN THE FFTF INSTRUMENT DEVELOPMENT EFFORT
ARGONNE NATIONAL LABORATORY
ANL-7380 +. 6 PAGES, 4 FIGURES, 6 REFERENCES, PAGES 9-14 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL
INSTRUMENTATION AND CONTROL, MARCH 2, 1967

IN-CORE INSTRUMENT DEVELOPMENT IS PROCEEDING IN THE FOLLOWING CATEGORIES - (1) FAILED-FUEL-
LOCATION DETECTION, (2) SIGNAL-LEAD CONNECTORS, (3) FISSION-GAS-PRESSURE TRANSDUCERS, (4) IN-CORE
FLOWMETERS, AND (5) FUEL-PIN THERMOCOUPLES. ONE OF THE PRIME OBJECTIVES OF THE PROGRAM IS TO
PROVIDE BOTH THE FFTF AND THE LMFBR PROGRAM WITH THE SPECIFICATIONS NECESSARY FOR MATERIALS AND
FABRICATION PROCEDURES AS WELL AS THE QUALITY CONTROL AND INSPECTION METHODS NEEDED SO THAT
RELIABLE COMMERCIAL INSTRUMENTS CAN BE MADE AVAILABLE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY,
\$0.65 MICROFICHE

INSTRUMENT, IN CORE + REACTOR, LMCR + INSTRUMENT, COMPONENT + INSTRUMENT, DETECTION FAILED FUEL ELEMENT +
INSTRUMENT, FLOW + INSTRUMENT, TEMPERATURE + EQUIPMENT DESIGN + INSTRUMENT, PRESSURE

9-023329 DATED 1967
HINES DP + HORST KM
SEFOR DEVELOPMENT ACTIVITIES
GENERAL ELECTRIC CO., SUNNYVALE, CALIF.
ANL-7380 +. 7 PAGES, 6 FIGURES, 3 REFERENCES, PAGES 15-21 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL
INSTRUMENTATION AND CONTROL, MARCH 2, 1967

DEVELOPED INSTRUMENTATIONS TO MEASURE FUEL-PELLET CENTRAL TEMPERATURE, FUEL-ASSEMBLY COOLANT
TEMPERATURE AND FLOW RATE, AND TRANSIENT CORE FLUX. THE DATA FROM THESE SENSORS, IN CONJUNCTION
WITH OTHER SEFOR PLANT PROCESS INSTRUMENTATIONS, WILL PROVIDE THE BASIC INFORMATION FROM WHICH
THE REACTIVITY EFFECTS AND CORRELATIONS WITH CORE POWER AND TEMPERATURE WILL BE DETERMINED DURING
THE SEFOR EXPERIMENTAL PROGRAM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY,
\$0.65 MICROFICHE

CHAMBER, FISSION + INSTRUMENT, IN CORE + INSTRUMENT, NUCLEAR + REACTOR, LMCR + HIGH TEMPERATURE + INSTRUMENT,
FLOW + INSTRUMENT, TEMPERATURE + EQUIPMENT DESIGN

9-023330 DATED 1967
MORIARTY KJ
EBR-II INSTRUMENTATION EXPERIENCES
ARGONNE NATIONAL LABORATORY
ANL-7380 +. 7 PAGES, 3 FIGURES, PAGES 22-28 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL INSTRUMENTATION
AND CONTROL, MARCH 2, 1967

EXPERIENCES WITH THE EBR-II PRESSURE INSTRUMENTATION, BOTH STATIC AND DIFFERENTIAL, THE
TEMPERATURE-MEASURING DEVICES, AND WITH THE LEVEL INDICATIONS ARE PRESENTED. THE PROBLEMS
ENCOUNTERED ARE OUTLINED, AND THE CORRECTIVE ACTION TAKEN TO RECTIFY A PROBLEM IS GIVEN WHERE
POSSIBLE. THE IMPORTANCE OF PRE-INSTALLATION DATA IS STRESSED. CALIBRATION AND REPAIR
PROCEDURES USED FOR FILLED SYSTEMS ARE INCLUDED FOR GENERAL INFORMATION AND TO POINT OUT THE TYPE
OF INFORMATION THAT MUST BE INCLUDED IN THE PRE-INSTALLATION CALIBRATION CHECK ON A COMPONENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY,
\$0.65 MICROFICHE

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-023330 *CONTINUED*
REACTOR, BREEDER + EBR 1 AND 2 (KE) + INSTRUMENT, PROCESS + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + INSTRUMENT, LIQUID LEVEL + INSTRUMENT, TEMPERATURE + INSTRUMENT, PRESSURE

9-023331 DATED 1967
SCOTT CC
FERMI PROCESS INSTRUMENTATION
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN
ANL-7380 +. 8 PAGES, 12 FIGURES, PAGES 29-36 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL INSTRUMENTATION AND CONTROL, MARCH 2, 1967

OPERATING EXPERIENCE AND EQUIPMENT DESIGN ARE REVIEWED FOR THREE KINDS OF PROCESS INSTRUMENTS ON THE FERMI REACTOR. PROBLEMS ASSOCIATED WITH PLUGGED PROCESS LINES, IN-PLACE CALIBRATION, AND THE DIFFICULTY OF REMOVING UNITS FOR MAINTENANCE ARE DISCUSSED COMPONENTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + FERMI (LMFBR) + INSTRUMENT, PROCESS + REACTOR, LMCR + INSTRUMENT CALIBRATION + OPERATING EXPERIENCE SUMMARY + INSTRUMENT, FLOW + INSTRUMENT, TEMPERATURE + EQUIPMENT DESIGN + INSTRUMENT, PRESSURE

9-023332 DATED 1967
STRAHL H
SNAP SYSTEM INSTRUMENTATION
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
ANL-7380 +. 3 PAGES, 2 TABLES, 1 REFERENCE, PAGES 37-39 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL INSTRUMENTATION AND CONTROL, MARCH 2, 1967

THE EXPERIENCE GAINED DURING OPERATION OF SNAP REACTORS WITH LIQUID-METAL PROCESS INSTRUMENTATION IS PRESENTED. THIS EXPERIENCE IS BASED ON THE OPERATION OF SEVERAL SNAP REACTOR SYSTEMS AT TEMPERATURES UP TO 1300 F FOR PERIODS OF UP TO 10,000 HR. SEVERAL TYPES OF TEMPERATURE AND PRESSURE DEVICES WERE INSTALLED IN A LOOP, AND THEIR PERFORMANCES WERE COMPARED AND EVALUATED PRIOR TO SELECTION OF INSTRUMENTATION FOR THE S80K TESTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

INSTRUMENT, PROCESS + REACTOR, LMCR + REACTOR, SPACE + OPERATING EXPERIENCE SUMMARY + HIGH TEMPERATURE + INSTRUMENT, FLOW + INSTRUMENT, LIQUID LEVEL + INSTRUMENT, TEMPERATURE + INSTRUMENT, TESTING + SNAP (SR) + INSTRUMENT, PRESSURE

9-023334 DATED 1967
DUNSMORE CL
TEMPERATURE PROTECTIVE CIRCUITS FOR THE HMPF FUEL-CHANNEL EXIT
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
ANL-7380 +. 4 PAGES, 4 FIGURES, 2 REFERENCES, PAGES 43-46 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL INSTRUMENTATION AND CONTROL, MARCH 2, 1967

THE HMPF PLANT PROTECTIVE SYSTEM INCLUDES CIRCUITS WITH THE FUNCTION OF PRECLUDING UNSAFE HIGH TEMPERATURE AND SUSTAINED FAST CHANGES IN FUEL-CHANNEL COOLANT-EXIT TEMPERATURES, EITHER IN THE INCREASING OR DECREASING DIRECTION. THE DEVELOPMENT OF THE FUEL-CHANNEL-EXIT HIGH-TEMPERATURE AND RATE-CHANGE-OF-TEMPERATURE CIRCUITS RESULTED IN A TRANSISTORIZED SYSTEM OF PLUG-IN COMPONENTS OF HIGH RELIABILITY. THE CIRCUITS HAVE BEEN DESIGNED, WHERE POSSIBLE, FOR FAIL-SAFE OPERATION IN CASE OF COMPONENT MALFUNCTION, AND FOR EASE OF MAINTENANCE AND CALIBRATION. IN ADDITION TO PROVIDING SHUTDOWN OR SCRAM SIGNALS, THE CIRCUITS ALSO PROVIDE SETBACK AND ALARM SIGNALS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

INSTRUMENT, PROTECTIVE + REACTOR, LMCR + INSTRUMENT, CONTROL + INSTRUMENT, TEMPERATURE

9-023335 DATED 1967
STEELE OP
STABLE HIGH-TEMPERATURE DETECTION
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
ANL-7380 +. 5 PAGES, 10 FIGURES, PAGES 47-51 OF PROCEEDINGS OF SYMPOSIUM ON LIQUID METAL INSTRUMENTATION AND CONTROL, MARCH 2, 1967

THE PLATINUM RTD AFFORDS A MORE STABLE REFERENCE TEMPERATURE-MEASURING SYSTEM THAN A THERMOCOUPLE IN THE 1200-1300 F RANGE. COMPARATIVE DRIFT RATES ARE SHOWN FOR THE PLATINUM RTD AND FOR A VARIETY OF THERMOCOUPLES FOR 10,000 HR AT THESE TEMPERATURES. THE ACCURATE DETERMINATION OF THIS DRIFT RATE BY THE USE OF ZINC AND ALUMINUM MELTING-POINT PHYSICAL STANDARDS IS DESCRIBED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + MEASUREMENT, TEMPERATURE + HIGH TEMPERATURE + INSTRUMENT, TEMPERATURE + EQUIPMENT DESIGN

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-023347

DATED 1968

DRIVER GE

THE LMFBR COMPUTER CONTROL CENTER

BATTELLE MEMORIAL INSTITUTE, RICHLAND, WASHINGTON

ANL-7380 +. 4 PAGES, PAGES 128-131 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL INSTRUMENTATION AND CONTROL, MARCH 2, 1967

IF THE PRESENT TREND IN INSTRUMENTATION AND CONTROL SYSTEMS CONTINUES, IT IS HIGHLY LIKELY THAT LMFBRs WILL USE DIGITAL COMPUTERS TO HANDLE DATA AND CONTROL REACTORS, HEAT EXCHANGERS, AND POWER-GENERATION EQUIPMENT. WHAT IS NEEDED IS A SINGLE ORGANIZATION WHICH COMPRISES THE BEST OF KNOWLEDGE AND EXPERIENCE IN ALL PHASES OF REACTOR CONTROL, IN SHORT, A COMPUTER CONTROL CENTER. THE CENTERS MAIN JOB WOULD BE TO PROVIDE THE LMFBR PLANT DESIGNER WITH THE SPECIALIZED HELP NEEDED TO ACQUIRE, INSTALL, AND OPERATE A COMPUTER-BASED DATA-HANDLING AND CONTROL SYSTEM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

CONTROL SYSTEM + REACTOR, LMCR + DATA PROCESSING + INSTRUMENT, CONTROL + COMPUTER CONTROL

9-023350

DATED 1967

LEWIS RH

ALL-SOLID IN-CORE POWER MONITORS FOR LMFBR SERVICE

THE BABCOCK AND WILCOX COMPANY, LYNCHBURG, VA.

ANL-7380 +. 6 PAGES, 3 FIGURES, 1 TABLE, 10 REFERENCES, PAGES 140-45 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL INSTRUMENTATION AND CONTROL, MARCH 2, 1967

AN ALL-SOLID, SELF-POWERED, IN-CORE NEUTRON DETECTOR (RHODIUM TYPE) FOR MEASURING THE POWER DISTRIBUTION IN LARGE THERMAL POWER REACTORS WAS DEVELOPED BY BABCOCK AND WILCOX. DETECTORS OF THIS TYPE ARE ALSO NECESSARY TO ENSURE OPTIMUM PERFORMANCE OF LMFBRs. THUS, AN IN-CORE FLUX-MONITOR DEVELOPMENT PROGRAM WAS INITIATED BASED ON AN EXTENSION OF THE SAME DETECTION PRINCIPLE USED IN THE RHODIUM-TYPE DETECTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

INSTRUMENT, IN CORE + REACTOR, LMCR + SOLID STATE DEVICE + REACTOR, THERMAL + HIGH TEMPERATURE + RHODIUM + SELF POWERED FLUX DETECTOR

9-023351

DATED 1967

KINZER JE + PARK TA

A MINIATURIZED NEUTRON FLUX SENSOR FOR LMFBR ENVIRONMENT

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

ANL-7380 +. 2 PAGES, 2 FIGURES, 1 REFERENCE, PAGE 146-47 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL INSTRUMENTATION AND CONTROL, MARCH 2, 1967

THE ADIT (ADVANCED DEVELOPMENT IRRADIATION TECHNIQUES) FLUX SENSOR WAS DEVELOPED AT ATOMICS INTERNATIONAL TO MEET THE NEED FOR IN-CORE NEUTRON-SENSING DEVICE. THE DEVICE CONSISTS OF A THERMOCOUPLE IN WHICH FISSIONABLE MATERIAL HAS REPLACED THE OXIDE INSULATION AROUND THE MEASURING JUNCTION. THE TYPE OF FISSIONABLE MATERIAL USED MAY BE CHOSEN TO MATCH THE NEUTRON-FLUX ENVIRONMENT OF INTEREST.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

INSTRUMENT, IN CORE + INSTRUMENT, NUCLEAR + REACTOR, LMCR + FLUX, INTEGRATED + NEUTRON

9-023358

DATED 1967

ROUX DP

SUBCRITICALITY MEASUREMENTS BY NEUTRON NOISE ANALYSIS

OAK RIDGE NATIONAL LABORATORY

ANL-7380 +. 3 PAGES, 8 REFERENCES, PAGES 180-182 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL INSTRUMENTATION AND CONTROL, MARCH 2, 1967

EVALUATES THE APPLICABILITY OF USING REACTOR FLUCTUATIONS (NOISE) TO MEASURE REACTOR PARAMETERS ON LIQUID METAL FAST BREEDER REACTORS (LMFBR) AND AS AN ON-LINE SAFETY DEVICE. REACTOR SUBCRITICALITY MEASUREMENT BY NOISE ANALYSIS HAS A FAIR CHANCE TO BE SUCCESSFULLY APPLIED IF THE MEASUREMENTS ARE LIMITED TO THOSE OF ONLY SMALL SHUTDOWN MARGIN. IN THE RANGE OF 1 TO 2 DOLLARS SUBCRITICAL, AN ACCURACY OF 10% CAN BE EXPECTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

CRITICALITY SAFETY + REACTOR, LMCR + SHUTDOWN MARGIN + NOISE ANALYSIS + NEUTRON

9-023359

DATED 1967

RANDALL RL + LOGAN D

APPLICATION OF NOISE-ANALYSIS TECHNIQUES TO HYDRAULIC MEASUREMENTS IN LIQUID-METAL SYSTEMS

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-023359 *CONTINUED*
ANL-7380 +. 6 PAGES, 7 FIGURES, 12 REFERENCES, PAGES 183-88 OF PROCEEDINGS OF THE SYMPOSIUM ON LIQUID METAL INSTRUMENTATION AND CONTROL, MARCH 2, 1967

INSTRUMENTATION AND ANALYSIS TECHNIQUES ARE DESCRIBED FOR DETECTING AND STUDYING THE FOLLOWING - ONSET OF BOILING, VOID FORMATIONS AND MOVEMENT, MECHANICAL VIBRATION OF SYSTEM COMPONENTS, VELOCITY PROFILES, FLOW PATTERNS, AND MIXING AND CROSS FLOW IN HYDRAULIC SYSTEMS. FREQUENCY AND TIME-CORRELATION ANALYSIS TECHNIQUES ARE APPLIED TO PRACTICAL PROBLEMS IN THE FIELD OF REACTOR-SYSTEM HYDRAULICS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

HYDRAULIC ANALYSIS + HYDRAULIC EXPERIMENT + REACTOR, LMCR + NOISE ANALYSIS + INSTRUMENT, FLOW + INSTRUMENT, TEMPERATURE + FLOW, TWO PHASE + VIBRATION + FUEL ELEMENTS

9-023946 ALSO IN CATEGORY 17 DATED 1967
BALL GL
RESULTS OF THE LOW-POWER NUCLEAR TEST PROGRAM FOR THE ENRICO FERMI FAST BREEDER REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-302 +. 126 PAGES, 29 FIGURES, 16 TABLES, REFERENCES, FEB. 1967

(PG 41) 1/M FUEL-LOADING PLCTS EXTRAPOLATED VERY CLOSE TO CRITICAL VALUE, HOWEVER WAS NOT USED FOR OFFICIAL PREDICTIONS BECAUSE LINEAR EXTRAPOLATION GAVE UNACCEPTABLE OVERESTIMATES. (PG 67) NEUTRON-DETECTOR GUIDE TUBES WERE FLUX-MAPPED (GAMMA AND NEUTRON) TO SET DETECTOR LOCATIONS. GAMMA PEAKS WERE FOUND NEAR SODIUM PIPES, AND NEUTRON PEAKS DUE TO BORON WINDOW AND TO TRANSITION DECK. (PG 87) INDIVIDUAL SAFETY RODS WERE WORTH 15% LESS THAN PREDICTED, AND THE GROUP WAS SHADOWED BY 11%, GIVING AN \$8 WORTH. (PG 118) RELATIVELY LARGE INCREASES IN REACTIVITY FOLLOWING SHORT PERIODS WITH THE COOLANT SYSTEM ISOTHERMAL AT 700 F, ARE CALCULATED TO BE FROM PHASE-TRANSFORMATION SHRINKAGE (27 IN-HOURS) AND HYDRIDING OF THE CLAD (22 IN-HOURS).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + CONTROL ROD INTERACTION + CONTROL ROD WORTH + REACTIVITY EFFECT, EXPANSION + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + INSTRUMENT CALIBRATION + REACTOR STARTUP TESTING + HYDRIDE + ZIRCALOY + REACTIVITY EFFECT, ANOMALOUS + INSTRUMENT, ABNORMAL INDICATION

9-023947 ALSO IN CATEGORY 18 DATED 1966
QUARTERLY TECHNICAL PROGRESS REPORT ON AEC-SPONSORED ACTIVITIES. MARCH 15-JUNE 15, 1966
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-193 +. 107 PAGES, 26 FIGURES, 9 TABLES, REFERENCES, MARCH 15, 1966

INDIVIDUAL PROJECTS ARE - (PROJECT 8) PASTE CONCEPT FOR BLANKETS (29 PG) - TEST LOOP OPERATED, IN-PILE LOOP DESIGNED, 1000-MWE REF.-DESIGN MADE. (10) SMALL-LEAK SODIUM-WATER REACTION (10 PG) - SOME TESTS ON TUBE WASTAGE MADE. (11) SODIUM LOOP OPERATION FOR IMPURITY DETECTION AND REMOVAL (43 PG). (14) UNDER-SODIUM SCANNER (8 PG) - DISPLAY OF ULTRASONIC SIGNAL ON TV TUBE BEING ASSEMBLED. (15) COMPILATION OF FERMI EXPERIENCE (3 PG) - LISTS TOPICAL REPORTS IN PROGRESS. (16) FAST-FLUX TEST FACILITY (3 PG) - WORK REPORTED IN APDA 186, 187, 188.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + SODIUM + REACTOR, TEST + OUT OF PILE EXPERIMENT + FFTF (TR) + CORROSION + REMOTE MANIPULATING AND VIEWING + COOLANT PURIFICATION SYSTEM + FAILURE, TUBING + INSTRUMENT, COOLANT QUALITY + R AND D PROGRAM

9-024332 DATED 1967
BATCH ML + KLINKMAN AE
TRANSFER FUNCTIONS FROM REACTIVITY PULSES GENERATED BY THE REGULATING ROD. ENRICO FERMI REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-NTS-14 +. 31 PAGES, DECEMBER 1967

DESCRIBES THE MEASUREMENT OF REACTOR POWER AND TEMPERATURE RESPONSES TO REACTIVITY PULSES GENERATED BY A SINGLE, TRAPEZOIDAL MOVEMENT OF THE REGULATING ROD. DATA OBTAINED AT THREE POWER LEVELS ARE REDUCED BY FOURIER ANALYSIS OF THEIR TIME DERIVATIVES. RESULTS ARE PRESENTED IN TERMS OF TRANSFER FUNCTIONS BETWEEN REACTIVITY AND POWER AND BETWEEN POWER AND OUTLET TEMPERATURES. THE ADVANTAGES AND DISADVANTAGES OF THIS TECHNIQUE WITH RESPECT TO AN ALTERNATE OSCILLATOR-ROD TECHNIQUE ARE EXAMINED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + TRANSFER FUNCTION + TEST, PLANT RESPONSE + PULSED NEUTRON TECHNIQUE + REACTOR POWER + INSTRUMENT, PULSE + COMPARISON, REACTOR CHARACTERISTICS

9-025112 DATED 1967
VAUTREY L
MECHANISM FOR EFFECTING THE POSITIONAL ADJUSTMENT OF NUCLEAR REACTOR CONTROL RODS

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-025112 *CONTINUED*
COMMISSARIAT A L'ENERGIE ATOMIQUE
BRITISH PATENT 1,086,099 +. 5 PAGES, 6 FIGURES, OCT. 4, 1967, PATENT (BRITISH)

THE DESIGN IS GIVEN OF A MECHANISM FOR POSITIONING CONTROL RODS IN A LIQUID METAL COOLED FAST REACTOR. THE THREE OPERATORS OF THE MECHANISM ARE A VERTICAL OPERATING ROD ATTACHED TO THE CONTROL ROD GRAB, THE HORIZONTAL LEVER ARM CONNECTING THE FIRST MEMBER AND THE VERTICAL ARM PROJECTING OUTSIDE THE REACTOR VESSEL.

AVAILABILITY - THE PATENT OFFICE, 25 SOUTHAMPTON BUILDING, LONDON, W.C. 2, ENGLAND (\$0.49 CENTS/COPY)

REACTOR, FAST + REACTOR, LMCR + UNITED KINGDOM + PATENT + CONTROL ROD DRIVE MECHANISMS

9-025177 DATED 1967
DEMITT TF
FFTF IN-LINE INSTRUMENTATION FOR SODIUM PURITY MEASUREMENT
BATTELLE-NORTHWEST, RICHLAND
BNWL-562 +. 24 PAGES, REFERENCES, OCTOBER 1967

THE STATUS OF DEVELOPMENT WORK ON IN-LINE METERS FOR MEASURING IMPURITIES IN SODIUM IS DISCUSSED. THE METERS ARE DISCUSSED IN TERMS OF THEIR POTENTIAL FOR USEFUL SERVICE IN THE FAST TEST REACTOR (FTR). THE PLUGGING METER, THE ELECTROCHEMICAL OXYGEN METER, THE CARBON DIFFUSION METER, AND THE HYDROGEN DIFFUSION METER ARE DEVICES THAT SHOULD BE DEVELOPED IN TIME FOR USE IN FTR DURING STARTUP OPERATIONS. THE ELECTRICAL RESISTIVITY METER MAY ALSO FIND USEFUL APPLICATION AT FTR IF EFFORTS TO IMPROVE PERFORMANCE CHARACTERISTICS ARE SUCCESSFUL. NO OTHER METERS APPEAR TO HAVE SUFFICIENT MERIT AT PRESENT TO WARRANT USE IN THE FTR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + INSTRUMENT, IN CORE + INSTRUMENT, PROCESS + REACTOR, LMCR + SODIUM + REACTOR, TEST + FFTF (TR) + METAL, LIQUID + INSTRUMENT, COOLANT QUALITY

9-025615 DATED 1960
PROCEEDINGS OF THE POWER REACTOR IN CORE INSTRUMENTATION MEETING, WASHINGTON, D.C.
AEC, DIVISION OF REACTOR DEVELOPMENT
TID-7598 +. 115 PAGES, FIGURES, MARCH, 1961

THE OBJECTIVES OF THIS MEETING WERE - (1) TO REVIEW THE STATUS OF THE CURRENT DEVELOPMENT PROGRAMS FOR IN-CORE INSTRUMENTATION, (2) TO DISCUSS THE MAJOR PROBLEMS IN THIS AREA, (3) TO REVIEW THE OVERALL PRESENT AND FUTURE REQUIREMENTS FOR IN-CORE INSTRUMENTATION, PARTICULARLY IN POWER REACTORS, (4) TO PROVIDE A FREE EXCHANGE OF INFORMATION AND IDEAS THAT MIGHT LEAD TO IMPROVEMENTS IN REACTOR-CORE PERFORMANCE AND REACTOR SAFETY THROUGH IMPROVEMENTS IN IN-CORE INSTRUMENTATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BWR + CHAMBER, ION + REACTOR, LCR + INSTRUMENT, IN CORE + REACTOR, LMCR + REACTOR, ORGANIC COOLED + REACTOR, PWR + REACTOR, POWER + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + INSTRUMENT, FLOW + INSTRUMENT, TEMPERATURE + REACTOR, LWR + TEST, INSTRUMENT RESPONSE + IN CORE MEASUREMENT + OPERATING EXPERIENCE

9-025835 DATED 1967
PUPPER GF + HAKKER JM
THE PERFORMANCE OF A COUNTING-MEAN-SQUARE VOLTAGE CHANNEL IN THE EBR-II
ARGONNE NATIONAL LABORATORY
6 PAGES, 4 FIGURES, 3 TABLES, 13 REFERENCES, IEEE TRANS. ON NUCLEAR SCIENCE NS-15(1), PAGES 22-27 (FEBRUARY 1968)

A WIDE-RANGE NEUTRON-FLUX-MONITORING TEST CHANNEL BASED ON COUNTING AND MEAN-SQUARE VOLTAGE TECHNIQUES HAS BEEN INSTALLED AT THE EBR-II. THE MEASURED RESPONSES TO NEUTRON FLUX ARE PRESENTED AND COMPARED TO THOSE PREDICTED. DATA TAKEN WHILE THE REACTOR IS ON AN ESSENTIALLY CONSTANT PERIOD DURING TYPICAL RE-STARTS ARE SHOWN. THE GAMMA DISCRIMINATION RATIO FOR THIS SYSTEM HAS BEEN DETERMINED AND IS DISCUSSED.

REACTOR, BREEDER + COUNTER + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + FLUX, INTEGRATED + NEUTRON + INSTRUMENT, WIDE RANGE

9-026513 ALSO IN CATEGORIES 10 AND 11 DATED 1967
VERBER F
FURTHER DEVELOPMENTS IN CONTAINMENT-BUILDING ELECTRICAL PENETRATIONS
8 PAGES, 13 FIGURES, 1 TABLE, 6 REFERENCES, REACTOR AND FUEL-PROCESSING TECHNOLOGY 11(1), PAGES 5-12 (WINTER 1967-1968)

REVIEWS SOME ELECTRICAL PENETRATION PROBLEMS AND SOLUTIONS EXPERIENCED DURING PLANT OPERATIONS AT FERMI AND AT PEACH BOTTOM. DESCRIBES FIRE-PROTECTION MODIFICATIONS TO THE PENETRATIONS AND TO THE ELECTRICAL DESIGN AT PEACH BOTTOM AFTER A FIRE. ALSO PRESENTS RECENT DEVELOPMENTS IN THE DESIGN OF THE DRESDEN-2 ELECTRICAL PENETRATION SYSTEM AND ASSEMBLIES.

REACTOR, BWR + DESIGN CRITERIA + FERMI (LMFBR) + FIRE + PEACH BOTTOM 1 (HTGR) + DRESDEN 2 (BWR) + INSTRUMENT,

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-026513 *CONTINUED*

COMPONENT + EQUIPMENT DESIGN + REACTOR, HTGR + ELECTRICAL CONDUCTION + CONTAINMENT PENETRATION, ELECTRICAL + OPERATING EXPERIENCE + REACTOR, LMFBR

9-027519

DATED 1967

HOPKINSON A + LEVELL RW

ANALOGUE AND DIGITAL COMPUTING TECHNIQUES IN PLANT MODELLING, REDUCTION AND CONTROL STUDIES

UNITED KINGDOM ATOMIC ENERGY AUTHORITY, WINFRITH

8 PAGES, 2 FIGURES, 7 REFERENCES, FROM SECOND U.K.A.C. CONTROL CONVENTION ON ADVANCES IN COMPUTER CONTROL, THE UNIVERSITY OF BRISTOL, APRIL 11-14, 1967

DESCRIBES THE TECHNIQUES USED AND EXPERIENCE GAINED DURING DYNAMIC STUDIES OF A PROPOSED PROTOTYPE FAST REACTOR POWER PLANT. THE STUDIES INCLUDED DEVELOPMENT OF DYNAMIC MODELS FOR INVESTIGATING THE UNCONTROLLED PLANT UNDER NORMAL CONDITIONS AND FOLLOWING VARIOUS PLANT FAULTS AND FOR STUDYING PLANT CONTROL OVER A FAIRLY WIDE RANGE OF POWER LEVELS.

AVAILABILITY - C. BALDWIN LTD., TUNBRIDGE WELLS, KENT, ENGLAND

COMPUTER, ANALOG + COMPUTER PROGRAM + CONTROL SYSTEM + COMPUTER, DIGITAL + REACTOR, FAST + REACTOR, LMCR + SODIUM + TRANSFER FUNCTION + ANALYTICAL MODEL + REACTOR CONTROL + THEORETICAL INVESTIGATION

9-027541

DATED 1968

MATTE J

A PROPOSED PROGRAM FOR THE DESIGN AND DEMONSTRATION OF A PROTOTYPE UNDER-SODIUM ULTRASONIC SCANNER

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT

APDA-220 +. 41 PAGES, 8 FIGURES, REFERENCES, FEBRUARY 1968

THE APPLICABILITY TO LARGE SODIUM-COOLED REACTORS OF THE UNDER-SODIUM ULTRASONIC TRANSDUCER IS SHOWN TO DEPEND ON THE DEVELOPMENT OF MECHANISMS SUITABLE FOR ACCURATE AND VERSATILE MANIPULATION OF THE TRANSDUCER WITHIN THE REACTOR. A PROGRAM OF PROTOTYPE DESIGN AND TESTING IS PROPOSED TO DEMONSTRATE THE FEASIBILITY AND DESIRABILITY OF SUCH A DEVICE FOR EIGHT TYPICAL APPLICATIONS TO SUCH REACTORS. THE FUNCTIONAL REQUIREMENTS OF A PROTOTYPE SCANNER AND OF A SUITABLE DEMONSTRATION TEST RIG ARE PRESENTED, TOGETHER WITH BRIEF DESCRIPTIONS OF INITIAL CONCEPTS OF THE PROTOTYPE AND THE TEST RIG. FINALLY, THE PROPOSED PROGRAM OF DEMONSTRATION TESTS IS OUTLINED, TOGETHER WITH THE BASIS FOR EVALUATING OF THE PROTOTYPE PERFORMANCE IN EACH APPLICATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + SODIUM + SOLID STATE DEVICE + DESIGN STUDY + INSTRUMENT, COMPONENT + TEST, INSTRUMENT RESPONSE + METAL, LIQUID

9-027582

DATED 1968

RILEY DR + FLECK J

REACTOR AVAILABILITY DRIVER FUEL INSTRUMENTATION

GENERAL ELECTRIC, SCHENECTADY + GENERAL ELECTRIC, SUNNYVALE, CALIF.

1 PAGE, 1 FIGURE, 1 TABLE, AEE TRANSACTIONS 11(1), PAGE 330, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO CANADA, JUNE 10-13, 1968

DURING THE EVOLUTION OF THE CONCEPTUAL DESIGN OF THE FFTF BACKUP DESIGN, SPECIAL EMPHASIS WAS PLACED ON MAXIMIZING REACTOR PLANT AVAILABILITY. SEVEN SCHEMES FOR CORE TEMPERATURE SENSING WERE CONCEIVED, RANGING FROM A SCHEME WITH NO REDUNDANCY WHATEVER TO AN ELABORATE SCHEME USING FOUR THERMOCOUPLES PER CHANNEL AND EXTENSIVE FAILURE-DETECTING CIRCUITRY. THE UNAVAILABILITY OF EACH SCHEME HAS BEEN PREDICTED, AND THE ARRANGEMENT RECOMMENDED USES FOUR THERMOCOUPLES PER CHANNEL, WITH THREE OF THEM ACTIVELY CONNECTED AND ONE AS A SPARE. ONE THERMOCOUPLE FROM EACH CHANNEL IS CONNECTED TO EACH OF THREE OR CIRCUITS, AND THEN THE THREE ORS ARE CONNECTED TO TWO-OUT-OF-THREE MAJORITY LOGIC TO PROVIDE THE SCRAM SIGNAL.

REACTOR, FAST + INSTRUMENT, IN CORE + REACTOR, LMCR + FFTF (TR) + INSTRUMENT, TEMPERATURE

9-027595

ALSO IN CATEGORIES 5 AND 17

DATED 1968

HARRIGAN PM + MATTEL J + JOTH WL

OPTICAL, LIGHTING, AND PHOTOGRAPHIC EQUIPMENT FOR THE ENRICO FERMI ATOMIC POWER PLANT REPAIR PROGRAM

POWER REACTOR DEVELOPMENT COMPANY + ATOMIC POWER DEVELOPMENT ASSOCIATES

1 PAGE, 1 FIGURE, AEE TRANSACTIONS 11(1), PAGE 358, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

A MELTDOWN OF TWO CORE SUBASSEMBLIES REQUIRED THE USE OF VERY LONG OPTICAL INSTRUMENTS AND LIGHTING EQUIPMENT FOR THE PURPOSE OF INSPECTION OF THE CORE AND REACTOR COMPONENTS. BECAUSE LITTLE INFORMATION WAS AVAILABLE ON THE DESIGN OF SUITABLE VIEWING AND LIGHTING EQUIPMENT FOR USE IN SODIUM REACTOR ENVIRONMENTS, MUCH DEVELOPMENTAL WORK AND TESTING HAD TO BE DONE TO PROVIDE SUITABLE EQUIPMENT. MUCH DEVELOPMENT WAS REQUIRED TO PROVIDE LIGHTING EQUIPMENT WITH ADEQUATE LAMP LIFE AND MANEUVERABILITY. THE USE OF PHOTOGRAPHY THROUGH THE VIEWING INSTRUMENTS WAS NOT SUCCESSFUL INITIALLY, BUT IMPROVEMENTS BOTH IN TECHNIQUE AND EQUIPMENT LED TO REMARKABLY GOOD RESULTS IN THE LATTER PART OF THE PROGRAM.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + INSTRUMENT, IN CORE + REACTOR, LMCR + MAINTENANCE AND REPAIR + INSTRUMENT, SURVEILLANCE + EQUIPMENT DESIGN + REMOTE MANIPULATING AND VIEWING + MAINTENANCE, REMOTE

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-027596 ALSO IN CATEGORIES 5 AND 17 DATED 1968
DUFFY JG + MATTE J + RYAN JD + WARBERG FS
OBJECT RETRIEVAL TOOLS FOR THE ENRICO FERMI ATOMIC POWER PLANT REPAIR PROGRAM
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN
3 PAGES, 2 FIGURES, AND TRANSACTIONS 11(1), PAGES 358-360, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY
AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

THE METHOD OF RETRIEVING FROM THE BOTTOM OF THE FERMI REACTOR AN OBJECT THAT CAUSED A MELTDOWN
INVOLVED PROCEDURES AND TOOLS WHICH ARE UNIQUE AND COULD HAVE APPLICATION IN FUTURE PROBLEMS.
TOOLS WERE DEVELOPED TO CUT THE OBJECT INTO PIECES SMALL ENOUGH TO BE REMOVED UPWARD THROUGH THE
SUPPORT PLATE. SEVERAL GRIPPERS FOR THIS PURPOSE WERE ALSO DEVELOPED. SIMULTANEOUSLY, SPECIAL
TOOLS WERE DEVELOPED TO REMOVE THE OBJECT INTACT VIA THE SODIUM INLET PIPE.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + INSTRUMENT, IN CORE + REACTOR, LMCR + MAINTENANCE AND
REPAIR + EQUIPMENT DESIGN + REMOTE MANIPULATING AND VIEWING + MAINTENANCE, REMOTE

9-027774 ALSO IN CATEGORIES 6 AND 17 DATED 1968
ZINIUK MA + WILBER HA
SCRAM TESTS TO DETERMINE SHUTDOWN TRANSIENT CHARACTERISTICS OF THE ENRICO FERMI REACTOR AND ASSOCIATED HEAT
TRANSPORT SYSTEM
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-NTS-15 +. 45 PAGES, 8 FIGURES, 7 TABLES, 4 REFERENCES, APRIL 1968

REACTOR JULET TEMPERATURE FLUCTUATION AFTER 67 MWTH OPERATION BEGAN 15 SEC AFTER SCRAM AND LASTED
TIL 220 SEC. THE RATE WAS 44 F/SEC, OVER A CHANGE OF 76 F. TIME CONSTANTS AND TRANSPORT LAGS
WERE DETERMINED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + SCRAM, REAL + RESPONSE TIME + INSTRUMENT, TEMPERATURE + REPORT, OPERATIONS ANALYSIS + TEST,
INSTRUMENT RESPONSE + MAIN COOLING SYSTEM + REACTOR, LMFBR + THERMAL TRANSIENT

9-028016 DATED 1968
MUELLER RE + BRANNAN CE
CALIBRATION MEASUREMENTS OF CONTROL AND SAFETY RODS IN THE ENRICO FERMI REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT
APDA-NTS-18 +. 59 PAGES, 12 FIGURES, 4 TABLES, 13 REFERENCES, APRIL 1968

DURING THE LOW-POWER TESTING OF THE ENRICO FERMI REACTOR, DETAILED CALIBRATION MEASUREMENTS OF THE
CONTROL AND SAFETY RODS WERE MADE TO CONFIRM THE ROD DESIGN AND ALSO TO INVESTIGATE THE
ADAPTABILITY OF DIFFERENT CALIBRATION TECHNIQUES. WORTHS OF INDIVIDUAL UNSHADOWED AND SHADOWED
RODS WERE OBTAINED AS WELL AS GAGED ROD WORTH. THE MEASURED ROD WORTH WERE ABOUT 6 TO 15 PER
CENT LESS THAN HAD BEEN PREDICTED FROM CRITICAL EXPERIMENT DATA. HOWEVER, THIS AGREEMENT IS
CONSIDERED SATISFACTORY IN VIEW OF DESIGN CHANGES MADE SINCE THE EXPERIMENT. MEANS BY WHICH THE
CALIBRATION PROCEDURE COULD BE IMPROVED ARE DISCUSSED.

AVAILABILITY - USAEC DEPOSITORY LIBRARIES IN THE U.S. AND OVERSEAS

REACTOR, BREEDER + CONTROL ROD INTERACTION + CONTROL ROD WORTH + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR
+ COMPARISON, THEORY AND EXPERIENCE + TESTING + CONTROL ROD CALIBRATION

9-028163 ALSO IN CATEGORIES 1 AND 6 DATED 1968
GREGGIER P + HENLEY EJ
ADVANCES IN NUCLEAR SCIENCE AND TECHNOLOGY
GENERAL ELECTRIC COMPANY + UNIVERSITY OF HOUSTON
391 PAGES, VOLUME 4, 1968, ACADEMIC PRESS, NEW YORK AND LONDON

THIS VOLUME IS CONCERNED WITH ADVANCED REACTOR CONCEPTS. TWO OF THE CHAPTERS DEAL WITH THE
TECHNOLOGY OF, RESPECTIVELY, SODIUM-COOLED FAST BREEDER POWER REACTORS AND GAS-COOLED POWER
REACTORS, THE LATTER INCLUDING BOTH THERMAL CONVERTERS AND FAST BREEDERS. COGNIZANCE IS TAKEN OF
THE CENTRAL ROLE OF REACTOR SAFETY IN THE DEVELOPMENT OF FAST BREEDER REACTORS IN TWO CHAPTERS
DEALING WITH, RESPECTIVELY, THE DOPPLER EFFECT IN FAST REACTORS AND THE ANALYSIS OF MAXIMUM
ACCIDENTS. THREE CHAPTERS COVER RECENT DEVELOPMENTS IN AREAS OF MORE GENERAL APPLICABILITY,
NAMESLY, OPTIMAL CONTROL THEORY, MODERN PERTURBATION THEORY, AND INDUSTRIAL APPLICATION OF
IONIZING RADIATIONS. MAIN HEADINGS FOR THE 7 CHAPTERS ARE - GAS-COOLED REACTOR TECHNOLOGY,
SAFETY AND ECONOMIC CHARACTERISTICS OF A 1000-MWE FAST SODIUM-COOLED REACTOR DESIGN, AND THE
DOPPLER EFFECT IN FAST REACTORS.

BETHE TAIT + DOPPLER EFFECT + ECONOMICS + REACTOR, FAST + REACTOR, GCR + REACTOR, LMCR + REACTOR DYNAMICS +
REACTOR SAFETY SYSTEM + REACTOR CONTROL + THEORETICAL INVESTIGATION + MATHEMATICAL TREATMENT + REACTOR, LMFBR

9-028929 DATED 1969
LIBBY HL + JENSEN JD
FEASIBILITY STUDY OF PROBE-TYPE LIQUID-METAL FLOWMETERS FOR FTR SUBASSEMBLIES
BATTILL-NORTHWEST, WASHINGTON
BNWL-911 +. 52 PAGES, 26 FIGURES, 8 REFERENCES, FEBRUARY 1969

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-028929 *CONTINUED*

GIVES RESULTS OF A FEASIBILITY STUDY OF PROBE-TYPE LIQUID-METAL FLOWMETERS. A COMPARISON OF INSIDE-CHANNEL EDDY CURRENT FLOWMETERS INDICATES THAT THE EDDY CURRENT FLOWMETER IS MOST SUITABLE FOR THE PROPOSED APPLICATION IN THE FTR SUBASSEMBLIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + COMPARISON, THEORY AND EXPERIENCE + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + INSTRUMENT, COMPONENT + INSTRUMENT, FLOW + METAL, LIQUID + EQUIPMENT DESIGN + R AND D PROGRAM

9-029002 ALSO IN CATEGORIES 5 AND 12 DATED 1968

DAVIDSON DF + ROACH PF
AN EXPERIMENTAL CONTINUOUS-INDICATION PLUGGING METER FOR IMPURITY MONITORING IN LIQUID ALKALI METALS
UNITED KINGDOM ATOMIC ENERGY AUTHORITY
IRG REPORT 1640 (R) +. 7 PAGES, FIGURES, TABLES, REFERENCES, 1968

A CONTINUOUS READING PLUGGING METER WAS DEMONSTRATED THAT MONITORED SODIUM SATURATION TEMPERATURE 125 TO 280 C. THE LARGEST CYCLE VARIATION IN THE TEST TRACE WERE ± 5 C AT 125 C. UNDER STABLE CONDITIONS SATURATION TEMPERATURE COULD BE READ TO ± 2 C AT 125 AND ± 1 C AT 280.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + SODIUM + UNITED KINGDOM + COOLANT CHEMISTRY + INSTRUMENT, FLOW + INSTRUMENT, SURVEILLANCE + IMPURITY

9-029040 ALSO IN CATEGORY 17 DATED 1968

BATCH ML + KLICKMAN AE
EVALUATION OF NOISE ANALYSIS FOR THE ENRICO FERMI REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-NTS-13 +. 90 PAGES, FIGURES, TABLES, REFERENCES, JAN. 1968

THE ANALYSIS OF RESULTS COVERED THE FOLLOWING ASPECTS - (1) SEARCH FOR INDICATIONS OF INSTABILITY, (2) EXAMINATION OF POWER SPECTRAL DISTRIBUTION OF VARIOUS SIGNALS, (3) MEASUREMENT OF THE AMPLITUDE OF THE TRANSFER FUNCTION BETWEEN VARIOUS PARAMETERS, (4) MEASUREMENT OF THE TIME LAG BETWEEN THE VARIOUS PARAMETERS, AND (5) MEASUREMENT OF THE RELATIVE AMOUNT OF COHERENT SIGNAL BETWEEN INPUT AND OUTPUT PARAMETERS. THE STUDY INDICATES THAT THROUGH NOISE ANALYSIS MEANINGFUL RESULTS CAN BE OBTAINED IN EACH OF THESE ASPECTS AND POINTS UP CHANGES IN RECORDING TECHNIQUES AND SELECTION OF PARAMETERS WHICH COULD LEAD TO IMPROVED ACCURACY, RELIABILITY, AND UNDERSTANDING OF FUTURE WORK.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + TRANSFER FUNCTION + NOISE ANALYSIS + INSTABILITY + CORRELATION

9-029120 DATED 1968

LMFBR PROGRAM PLAN. VOL. 4 - INST. + CONTROL
ARGONNE NATIONAL LABORATORY
WASH-1104 +. 351 PAGES, TABLES, 1 FIGURE, AUGUST 1968

PRESENTS THE INSTRUMENTATION AND CONTROL SECTION OF THE LMFBR PLAN PROGRAM. IT IS ONE OF NINE SECTIONS THAT COMPRISE THE OVERALL PROGRAM. THE SECTION ASSESSES THE STATE OF THE ART FOR ITS AREA OF TECHNOLOGY, DEFINES THE OBJECTIVES, ESTABLISHES THE CRITERIA AND REQUIREMENTS, AND DESCRIBES THE R AND D PROGRAM INCLUDING MILESTONES, SCHEDULES, AND ESTIMATED COSTS NECESSARY TO SATISFY THE REQUIREMENTS AND MEET THE OBJECTIVES ON TIME. COVERED IN THE TECHNICAL PROGRAM ARE TEST FACILITIES, CIRCUITS, MEASUREMENT SYSTEMS, CONTROL SYSTEMS, SIGNAL TRANSMISSION, INSTRUMENT COMPONENTS, AND SENSORS. LARGE BIBLIOGRAPHY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

CONTROL SYSTEM + INSTRUMENT, IN CORE + INSTRUMENT, NUCLEAR + INSTRUMENT, CONTROL + INSTRUMENT, COMPONENT + INSTRUMENT, TESTING + TEST, INSTRUMENT RESPONSE + R AND D PROGRAM + REACTOR, LMFBR + JACOBS + INSTRUMENTS, MISL.

9-029122 DATED 1968

MURIARTY KJ + JUST FH + ROVIG MD
CONTROL SYSTEM FOR TRACE HEATING ON EBR-II SECONDARY SODIUM PIPING
ARGONNE NATIONAL LABORATORY
ANL-7433 +. 48 PAGES, 33 FIGURES, 1 TABLE, MARCH 1968

THE CONTROL SYSTEM FOR TRACE HEATING ON EBR-II SECONDARY SODIUM PIPING IS A STATIC SOLID-STATE SCAN AND CONTROL SYSTEM FOR COORDINATED CONTROL OF 76 INDUCTIVE AND RESISTIVE HEATER CIRCUITS USED FOR HEATING THE SECONDARY SODIUM SYSTEM WHEN REACTOR HEAT IS NOT AVAILABLE. THIS REPLACEMENT SYSTEM, FOR THE ORIGINAL CONTROLS, IS DESCRIBED AND SCHEMATICALLY DEPICTED. DESIGN CRITERIA, PHILOSOPHY, AND JUSTIFICATIONS ARE GIVEN.

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-029122 *CONTINUED*
AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTROL SYSTEM + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + SOLID STATE DEVICE + REACTOR, TEST +
INSTRUMENT, CONTROL + DESIGN STUDY + REACTOR, ERDA OWNED + HEAT TRANSFER, CONDUCTION + EQUIPMENT DESIGN +
HEAT PIPE

9-029128 DATED 1968
COLEMAN JH + HILL EF
FEASIBILITY STUDY OF IN-LINE DETECTOR FOR HYDROGEN IN SODIUM
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-222 +. 45 PAGES, MARCH 1968

A NEW METHOD FOR CONTINUOUSLY DETECTING HYDROGEN IN SODIUM HAS BEEN DEVELOPED. A CATALYTIC
SURFACE IS USED TO OXIDIZE HYDROGEN TO WATER AS IT EMERGES FROM A DIFFUSION MEMBRANE WHICH IS IN
CONTACT WITH SODIUM CONTAINING HYDROGEN. THE AMOUNT OF WATER FORMED IS PROPORTIONAL TO THE
AMOUNT OF HYDROGEN CONTAINED IN THE SODIUM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + METAL WATER REACTION + INSTRUMENT, COMPONENT + COOLANT PURIFICATION SYSTEM + INSTRUMENT,
COOLANT QUALITY

9-029195 DATED 1966
CONTROL MECHANISM FOR CONTROL RODS
COMMISSARIAT L'ENERGIE ATOMIQUE
FRENCH PATENT 1,458,620 +. OCT. 3, 1966

THE MECHANISM IS USED IN LIQUID METAL-COOLED REACTORS AND COMPRISES GRIPPING ORGANS FOR THE HEAD
OF THE CONTROL ROD, AND A CONTROL STEM FOR THESE ORGANS IN A COAXIAL TRANSFER TUBE. MEANS ARE
PROVIDED OUTSIDE THE REACTOR VESSEL FOR THE TRANSFER OF THE TUBE THROUGH THE VESSEL. A SEALING
LABYRINTH JOINT IS DISPOSED BETWEEN THE TUBE AND THE VESSEL AND A METALLIC BELLOWS BETWEEN AND
THE CONTROL SYSTEM. THE TUBE INCLUDES OPENINGS, ALLOWING FREE CIRCULATION OF THE LIQUID METAL
THROUGH THE TUBE WHILE GUIDING ORGANS FOR THE CONTROL STEM HAVING A POINT CONTACT WITH THE
LATTER. THE TUBE MAY BE GUIDED IN ITS TRANSFER BY A FIXED EXTERNAL TUBE HAVING OPENINGS FOR THE
FREE CIRCULATION OF THE LIQUID FROM THE VESSEL.

AVAILABILITY - FOREIGN PATENTS MAY BE OBTAINED FROM THE U.S. PATENT OFFICE, DEPARTMENT OF COMMERCE,
WASHINGTON, D.C. (\$0.30 CENTS/PAGE)

FRANCE + REACTOR, LMCR + PATENT + CONTROL ROD DRIVE MECHANISMS

9-029321 ALSO IN CATEGORY 6 DATED 1968
PHLASTERER GR
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 16, FEBRUARY-APRIL 1968
GENERAL ELECTRIC COMPANY
GEAP-5629 +. 32 PAGES, REFERENCES, MAY 1968

THIS REPORT IS ONE OF A SERIES OF SUCH REPORTS CONCERNED WITH SUMMARIZING THE WORK PERFORMED ON
THE RESEARCH AND DEVELOPMENT PROGRAM FOR THE SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR (SEFOR).
THE SEFOR PROJECT CONSISTS OF TWO MAJOR PARTS - THE DESIGN AND CONSTRUCTION OF THE FACILITY AND
THE RELATED RESEARCH DEVELOPMENT PROGRAM. SEFOR IS PARTICULARLY DESIGNED FOR THE SYSTEMATIC
DETERMINATION OF THE DOPPLER COEFFICIENT OF REACTIVITY AT TEMPERATURES UP TO THE VICINITY OF FUEL
MELTING. THIS REPORT SUMMARIZES THE TASKS SCHEDULED FOR ACTIVITY AT THIS TIME. REMAINING TASKS
WILL BE INCLUDED IN FUTURE REPORTS AS WORK PROGRESSES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + DESIGN STUDY + REACTOR, PULSED + SEFOR (RE) + REVIEW + R
AND D PROGRAM

9-029440 ALSO IN CATEGORIES 2 AND 18 DATED 1968
RESPONSE TO QUESTION A-9 - SEISMIC EFFECTS ON CRITICAL INSTRUMENTATION
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, PAGE A-30 THRU A-32 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CERTAIN CONTROLS AND INSTRUMENTATION IN THE CONTROL ROOM AND OTHER PARTS OF THE FACILITY ARE
CRITICAL TO SAFE SHUTDOWN OF THE FACILITY. PROVIDE A DISCUSSION OF THE CRITERIA AND METHODS OF
ANALYSIS EMPLOYED IN THE DESIGN OF THESE CRITICAL CONTROLS AND INSTRUMENTATION FOR SEISMIC
EFFECTS INCLUDING SEISMIC LOADINGS, POSSIBLE TILT (I.E., MERCURY SWITCHES AND RELAYS), ETC.
(ANSWER ON PGS A-30 THRU -32.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-029440 *CONTINUED*
DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + INSTRUMENT, CONTROL + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + CONTROL PANEL/ROOM

9-029461 ALSO IN CATEGORIES 11 AND 18 DATED 1968
RESPONSE TO QUESTION D-11 - INSTRUMENT LEAD PENETRATION
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE D-14 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DUCKET 50-231, MAY 24, 1968

SLIDING FRICTION IS THE SOLE MEANS OF HOLDING SHIELDING PLUGS IN PLACE WHEN INSTRUMENT LEADS PASS THROUGH THE PENETRATION. DISCUSS RELATIONSHIP OF THE REFUELING CELL PRESSURE AND FORCE REQUIRED TO SLIDE PLUGS OUT. (ANSWER) WHEN IN USE, UTILITY WALL SLEEVE PLUG WOULD BE REPLACED, WITH CONFIGURATION HELD IN PLACE BY BATTING. NO DESIGN AT PRESENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION + ELECTRICAL CONDUCTION + CONTAINMENT PENETRATION, ELECTRICAL + INSTRUMENTS, MISC.

9-029471 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION G-1 - PROTECTION INSTRUMENTS DURING TRANSIENT TEST

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, 2 REFERENCES, PAGE G-1 AND -2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DUCKET 50-231, MAY 24, 1968

IDENTIFY AND EVALUATE ADEQUACY OF PROTECTION INSTRUMENT SYSTEM TO BE USED DURING PLANNED TRANSIENT TEST PROGRAM. (ANSWER ON PGS G-1 AND -2.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

INSTRUMENT, PROTECTIVE + REACTOR, LMCR + REACTOR TRANSIENT + REPORT, SAR + TESTING + SEFOR (RE) + RESPONSE TO NRC QUESTION

9-029472 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION G-2 - INSTRUMENTATION DURING FUEL HANDLING

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, PAGE G-3 THRU -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DUCKET 50-231, MAY 24, 1968

DESCRIBE INSTRUMENTATION USED DURING FUEL HANDLING OPERATIONS. JUSTIFY ADEQUACY. (ANSWER ON PGS G-3 THRU -5.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FUEL HANDLING + REACTOR, LMCR + REPORT, SAR + MONITOR + SEFOR (RE) + RESPONSE TO NRC QUESTION + RADIATION MONITORS

9-029473 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION G-3 - RELIABILITY OF ION CHAMBER NITROGEN COOLING SYSTEM

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

7 PAGES, 2 FIGURES, PAGE G-5 THRU -11 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DUCKET 50-231, MAY 24, 1968

DISCUSS RELIABILITY OF ION CHAMBER NITROGEN COOLING SYSTEM AND ASSOCIATED TEMPERATURE SENSORS. CAN TEMPERATURE MONITORING SYSTEM BE DISABLED BY SINGLE FAILURE. (ANSWER ON PGS G-5 THRU -11.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CHAMBER, ION + REACTOR, LMCR + REPORT, SAR + NITROGEN + INSTRUMENT, TEMPERATURE + SEFOR (RE) + RESPONSE TO NRC QUESTION + INSTRUMENT, POWER RANGE + AUXILIARY COOLING SYSTEM

9-029474 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION G-4 (PT. 1) - BYPASS MODE OF PROTECTIVE INSTRUMENTS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE G-12 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DUCKET 50-231, MAY 24, 1968

PROVIDE SCHEMATIC DIAGRAM OF THOSE PORTIONS OF THE PROTECTION CIRCUITRY INSTRUMENT WHICH ACTUATE AND REMOVE THE OPERATIONAL BYPASSES. (ANSWER) SEE SUPPLEMENT 2 OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-029474 *CONTINUED*
INSTRUMENT, PROTECTIVE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION

9-029475 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION G-4 (PT. 2) - PROTECTION SYSTEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
8 PAGES, 4 FIGURES, PAGE G-12 THRU -15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE ABILITY TO MEET PROPOSED IEEE STANDARD, NUCLEAR POWER PROTECTION SYSTEMS. (ANSWER ON PGS G-12 THRU -15.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CODES AND STANDARDS + INSTRUMENT, PROTECTIVE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION

9-029481 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION G-10 - IN-CORE INSTRUMENT TO SENSE PRESSURE PULSE DURING TRANSIENT TEST
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE G-23 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ARE THERE IN-CORE INSTRUMENTS CAPABLE OF SENSING PRESSURE PULSES WHICH MAY OCCUR DURING TRANSIENT TEST PROGRAM. (ANSWER) ANALYSIS OF EXCURSIONS DID NOT INDICATE A SIGNIFICANT PRESSURE PULSE. NO INSTRUMENTS PROVIDED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

INSTRUMENT, IN CORE + REACTOR, LMCR + REACTOR TRANSIENT + REPORT, SAR + TEST, PHYSICS + SEFOR (RE) + RESPONSE TO NRC QUESTION + PRESSURE PULSE + INSTRUMENT, PRESSURE

9-029482 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION H-1 - SINGLE CONTROL SCHEME
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, PAGE H-1 THRU -3 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS BASIS FOR YOUR CONCLUSION THAT ONE CONTROL SCHEME IS SUFFICIENT FOR SEFOR. INCLUDE CONSIDERATIONS GIVEN TO FAILURES THAT COULD DISABLE THE REACTIVITY CONTROL SYSTEM (I.E., SHIELD PLUG MOVEMENT). (ANSWER ON PGS H-1 THRU -3.) NOT SURE OF INTENT OF QUESTION. DRL PREVIOUSLY CONCLUDED THAT CONCEPT IS SOUND.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + REACTOR SAFETY SYSTEM + INSTRUMENT, CONTROL + SEFOR (RE) + RESPONSE TO NRC QUESTION

9-029484 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION H-3 - MEASURING FRICTIONAL LOADS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE H-4 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THE FRICTIONAL FORCES RESISTING MOTION OF THE REFLECTORS CAN BE ESTABLISHED ONLY INDIRECTLY BY MEANS OF A CHANGE IN THE SCRAM TIME. DISCUSS SENSITIVITY OF THIS METHOD AND FEASIBILITY OF PROVIDING A MORE DIRECT MEANS OF MEASURING FRICTIONAL LOADS THROUGHOUT THE STROKE OF UNITS. (ANSWER) TEST PROGRAM INDICATED NO CHANGE FOR LIFETIME. MEASUREMENT CAN BE MADE BY UNBOLTING DRIVE UNIT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

GUIDE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + REFLECTOR + RESPONSE TO NRC QUESTION + CORE COMPONENTS + ABLATION + DISPLACEMENT DESIGN + CONTROL ROD DRIVE MECHANISMS

9-029485 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION H-4 - ACCUMULATOR DRIVING FORCE DURING SCRAM
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE H-5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ACCUMULATORS GIVE A RELATIVELY SMALL DOWNWARD FORCE TO OVERCOME POSSIBLE LARGE FRICTION FORCES OR TO ACCELERATE THE REFLECTOR. WHAT ARE THE CONSIDERATIONS THAT LED TO THIS LIMITED DRIVING FORCE

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-029485 *CONTINUED*

ASSISTANCE AVAILABLE DURING A SCRAM. (ANSWER) THE 200 PSI ACCUMULATOR PRESSURE RESULTS IN 600 LB OF ADDED FORCE FOR INITIALLY DRIVING THE REFLECTOR CONTROL ELEMENT DOWN ON A SCRAM. ACCUMULATOR ACTS AS BACKUP TO LOW PRESSURE SYSTEM WHICH PROVIDES ADDITIONAL FORCE TO START ROD MOTION. FRICTION OF RODS OF 300 LB MEASURED, SO ACCUMULATOR COURSES INCREASE IN DOWNWARD FORCE FROM 1100 LB TO 1700 LB.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + REACTOR SAFETY SYSTEM + SCRAM, REAL + CONTROL ROD SCRAM MECHANISM + SEFOR (RE) + REFLECTOR + RESPONSE TO NRC QUESTION + ACCELERATION + ABLATION + CONTROL ROD DRIVE MECHANISMS + ACCUMULATORS

9-029486 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION H-5 - OPERATING PRESSURES AND RANGES FOR ACCUMULATORS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, 1 FIGURE, PAGE H-6 AND -7 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT WAS THE BASIS FOR ESTABLISHING THE OPERATING PRESSURES AND THEIR RANGES FOR THE ACCUMULATORS. IT APPEARS THAT THE FUNCTION OF THE ACCUMULATORS COULD BE NULLIFIED IF THE OPERATING PRESSURES APPROACHED THE EXTREMES OF THE PRESENTLY ALLOWABLE RANGES. (ANSWER) PRESSURE NOW SET OF 200 PSI. CHANGE WILL BE MADE DURING FINAL CHECKOUT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + OPERATION + REPORT, SAR + CONTROL ROD SCRAM MECHANISM + SEFOR (RE) + REFLECTOR + RESPONSE TO NRC QUESTION + ACCUMULATORS

9-029671

DATED 1968

PADILLA A + PETERSON RE + REGIMBAL JJ

FFTF CORE INSTRUMENTATION BASES AND REQUIREMENTS

BATTELLE NORTHWEST LABORATORY, PACIFIC NORTHWEST LAB., RICHLAND, WASH.

2 PAGES, 1 FIGURE, 2 REFERENCES, ANS TRANS. 11(2), PAGES 693-94, 1968 WINTER MEETING OF THE AMERICAN NUCLEAR SOCIETY, WASHINGTON, D.C., 20345 NOVEMBER 10-15, 1968

REQUIREMENTS FOR REACTOR CORE INSTRUMENTATION IN THE FAST FLUX TEST FACILITY WERE ESTABLISHED FROM A COMPREHENSIVE EVALUATION OF POTENTIAL LIMITING ACCIDENT CONDITIONS AND FROM CONSIDERATION OF EXPERIMENTAL OBJECTIVES. PROTECTION OF THE CORE AGAINST THE CONSEQUENCES OF LOSS OF COOLANT FLOW, OVERPOWER, AND FAILED FUEL IS THE FUNDAMENTAL SAFEGUARDS OBJECTIVE OF IN-CORE INSTRUMENTATION. THEREFORE, THE STUDY WAS FOCUSED UPON TIME-TEMPERATURE CHARACTERISTICS DURING THESE EVENTS.

DESIGN CRITERIA + REACTOR, FAST + INSTRUMENT, IN CORE + INSTRUMENT, PROTECTIVE + REACTOR, LMCR + REACTOR, TEST + FFTF (TR)

9-029877

DATED 1968

FORD JA + MESHII T

CALIBRATION OF ELECTROMAGNETIC FLOW-METERS IN THE ENRICO FERMI ATOMIC POWER PLANT

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

2 PAGES, 1 REFERENCE, ANS TRANS. 11(2), PAGES 632-33, 1968 WINTER MEETING OF THE AMERICAN NUCLEAR SOCIETY, WASHINGTON, D. C., NOV. 10-15, 1968

INITIAL CALIBRATION OF THE SODIUM FLOW METERS WAS DETERMINED FROM THE EXPERIMENTALLY MEASURED PUMP CHARACTERISTICS OBTAINED AT PUMP SPEEDS LESS THAN 50% OF FULL SPEED. A NEW CALIBRATION WAS OBTAINED USING THE EXPERIMENTAL HYDRAULIC PUMP CHARACTERISTICS AT HIGH PUMP SPEED. A COMPARISON OF THE MEASURED SODIUM FLOW USING THE NEW CALIBRATION WITH THE CALCULATED VALUE FROM HEAT BALANCE TESTS SHOWED EXCELLENT AGREEMENT WITH AN AVERAGE DEVIATION OF 1.7%. PREVIOUS CALIBRATIONS WERE TOO HIGH BY AN AVERAGE OF 10%.

FERMI (LMFR) + REACTOR, LMCR + SODIUM + INSTRUMENT, FLOW + METAL, LIQUID

9-029961

DATED 1968

REGIMBAL JJ

TECHNICAL BASES FOR FFTF OPEN TEST POSITION AND CLOSED-LOOP INSTRUMENTATION

BATTELLE-NORTHWEST, RICHLAND, WASH.

BNWL-779 +. 75 PAGES, APRIL 1968

THE OBJECTIVE OF THIS REPORT IS TO ADDRESS IN DETAIL THE PROBLEM OF RESOLVING CORE INSTRUMENTATION REQUIREMENTS FOR THE TEST CHANNELS AND TO IDENTIFY AND EVALUATE THOSE TECHNICAL FACTORS WHICH INFLUENCE SUCH REQUIREMENTS. OPEN TEST POSITIONS UTILIZE PRIMARY REACTOR SYSTEM SODIUM, WHEREAS THE CLOSED LOOPS EMPLOY INDIVIDUAL COOLANT SYSTEMS THAT ARE ISOLATED FROM THE REACTOR COOLANT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CONTROL SYSTEM + DESIGN CRITERIA + REACTOR, FAST + INSTRUMENT, IN CORE + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + INSTRUMENT, TESTING

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-029994 ALSO IN CATEGORIES 1 AND 7 DATED 1968
LMFBR PROGRAM PLAN. VOL. 10 - SAFETY
ARGONNE NATIONAL LABORATORY, LMFBR PROGRAM, OFFICE, ARGONNE, ILLINOIS
WASH-1110 +. 256 PAGES, FIGURES, TABLES, AUGUST 1968

THE SAFETY ELEMENT OF THE LMFBR PROGRAM IS TECHNOLOGICALLY ORIENTED TO AID DESIGNERS, OPERATORS, AND REGULATORY GROUPS IN ACHIEVING THE REQUISITE LEVEL OF SAFETY FOR LMFBRs. THE OBJECTIVE IS TO PROVIDE REALISM AND CONFIDENCE IN THE UNDERSTANDING AND ANALYSIS OF ACCIDENT SITUATIONS, TO DEVELOP AND EVALUATE SAFETY SYSTEMS FOR THE PREVENTION OF ACCIDENTS AND MITIGATION OF THEIR CONSEQUENCES, AND TO DEVELOP STANDARDS AND CODES FOR THE SAFE DESIGN, SITING, CONSTRUCTION, AND OPERATION OF BOTH USAEC TEST FACILITIES AND COMMERCIAL LMFBRs. THE SCOPE INCLUDES EVALUATIONS AND EXPERIMENTATION RELATIVE TO ABNORMAL OPERATION, POTENTIAL ACCIDENTS, AND SAFETY SYSTEMS DESIGNED TO PREVENT ACCIDENTS AND TO LIMIT THEIR CONSEQUENCES. EMPHASIS IS ON PREVENTION, EARLY DETECTION, AND CONTROL OF ACCIDENTS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
INSTRUMENT, PROTECTIVE + REACTOR SAFETY SYSTEM + INSTRUMENT, SURVEILLANCE + SAFETY PRINCIPLES AND PHILOSOPHY + R AND D PROGRAM + REACTOR, LMFBR + JACOBS + INSTRUMENTS, MISC.

9-030254 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 2 - FUEL FAILURE DETECTION
NORTHERN STATES POWER COMPANY
7 PAGES, PP. 11-17 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10 1968, DOCKET 50-231

FUEL FAILURE DETECTION DISCUSSED. ***FOLLOWING EACH TRANSIENT TWO FUEL RODS WILL BE EXAMINED AND COVER GAS SAMPLES ANALYZED BEFORE/AFTER. THREE TYPES OF CONTINUOUS MONITORS DISCUSSED (DELAYED NEUTRON, CHARGED WIRE PRECIPITATOR, GROSS GAMMA OF COVER GAS). THE THIRD WAS CHOSEN AND WILL BE INSTALLED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REPORT, SAR + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + SEFOR (RE) + RESPONSE TO NRC QUESTION + REACTOR, LMFBR

9-030351 ALSO IN CATEGORY 6 DATED 1968
HYNDMAN RW + NICHOLSON RB
THE EBR-II FEEDBACK FUNCTION
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
ANL-7476 +. JULY 1968, 37 FIGURES, 4 REFERENCES, 56P.

CONSIDERABLE EFFORT HAS BEEN MADE TO ASSESS THE SYSTEMS STABILITY OF EBR-II BECAUSE OF BOTH THE GENERAL NEED TO GAIN INFORMATION ON FAST-REACTOR SYSTEMS AND THE PARTICULAR NEED TO MAINTAIN SAFE OPERATION OF EBR-II, WHICH PLAYS A CRUCIAL ROLE AS AN IRRADIATION FACILITY IN THE LMFBR PROGRAM. THIS REPORT IS THE RESULT OF AN ATTEMPT TO DEDUCE AS MUCH INFORMATION AS POSSIBLE ON THE EBR-II FEEDBACK FUNCTION IN RUN 26 FROM A SERIES OF OSCILLATOR AND ROD-DROP EXPERIMENTS PERFORMED IN OCT. AND NOV. 1967.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (KE) + REACTOR, FAST + REACTOR, LMCR + REACTOR STABILITY + REACTOR, TEST + TRANSFER FUNCTION + REACTOR, EKDA OWNED + TEST, DROP + CONTROL RODS

9-030743 DATED 1968
NEUENHAIN P
SIMULATION AND CONTROL OF SODIUM-COOLED THERMAL REACTORS
TECHNISCHE HOCHSCHULE, MUNICH
4 PAGES, ATOMPRAXIS 14, PAGES 157-60 (APRIL-MAY 1968)

USE OF AN ANALOG COMPUTER FOR THE SIMULATION OF A SODIUM-COOLED THERMAL REACTOR IS DESCRIBED. DYNAMIC RESPONSE OF THE REACTOR IS DISCUSSED. EFFECTS OF MODERATOR TEMPERATURE COEFFICIENTS ON REACTIVITY ARE SHOWN. USE OF AN ON-OFF CONTROLLER WITH THE REACTOR IS DISCUSSED. A BLOCK DIAGRAM OF THE REACTOR AND CONTROL SYSTEM IS SHOWN. CONTROL OPTIMIZATION IS DISCUSSED.

COMPUTER, ANALOG + REACTOR, LMCR + REACTOR DYNAMICS + INSTRUMENT, CONTROL + REACTOR CONTROL + SIMULATION

9-030748 ALSO IN CATEGORY 11 DATED 1967
ELIAS P + GOLDFUSS G
A PROTOTYPE PORTABLE PUMP AND OXIDE-CLEANUP SYSTEM FOR EBR-II ROTATING SHIELD PLUG SEALS
ARGONNE NATIONAL LABORATORY
ANL-7379 +. 23 PAGES, 9 FIGURES, AUGUST 1967

INVESTIGATED THE OCCASIONAL STICKING OF TWO, MOTOR-DRIVEN, ROTATING SHIELD PLUGS PREPARATORY TO EXCHANGE OF FUEL AND OR SPECIAL SUBASSEMBLIES CONTAINING IRRADIATION-TEST SPECIMENS. POTENTIAL SOURCES OF STICKING - ACCUMULATIONS OF OXIDE MATERIAL IN THE BISMUTH-TIN ALLOY ROTARY FREEZE SEAL ON EACH PLUG. THIS REPORT DESCRIBES THE MEASURES USED TO REDUCE OXIDATION, MANUAL TECHNIQUES

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-030748 *CONTINUED*

USED FOR BATCH-TYPE REMOVAL OF OXIDE MATERIAL, AND A PORTABLE PUMP AND CLEANUP SYSTEM DESIGNED TO EXPEDITE REMOVAL OF OXIDE MATERIAL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DECONTAMINATION + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + OXIDATION + REACTOR, TEST + INSTRUMENT, TEMPERATURE + REACTOR, ERDA OWNED + SHIELDING + SEAL + PUMPS

9-030945
HUCK CE

DATED 1968

FFTF COOLANT THERMOCOUPLED DEVELOPMENT

BATTELLE-NORTHWEST, RICHLAND, WASHINGTON

BNWL-SA-1857 + CONF-680526-1 +. 15 PAGES, REFERENCES, FROM ERA/ISA SYMPOSIUM, SEATTLE, WASHINGTON, MAY 14, 1968

A GROUP OF THERMOCOUPLES WAS EXPOSED TO A FAST NEUTRON FLUENCE AS PART OF A PROGRAM TO DEVELOP SODIUM COOLANT THERMOCOUPLES FOR THE FAST FLUX TEST FACILITY. CHROMEL/ALUMEL IS THE MOST PROMISING CANDIDATE THERMOCOUPLE BASED ON A NEGLIGIBLE AMOUNT OF DECALIBRATION AND OTHER THERMAL NEUTRON REACTOR EFFECTS ON THIS TYPE OF THERMOCOUPLE. THE TESTS ARE BEING REPEATED AFTER EBR-II FAST NEUTRON REACTOR IRRADIATION TO DETERMINE THE EXTENT OF TRANSMUTATION OR OTHER PROPERTY CHANGES WHICH WOULD AFFECT THEIR PERFORMANCE IN THE FFTF.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + INSTRUMENT CALIBRATION + INSTRUMENT, TEMPERATURE + TEST, INSTRUMENT RESPONSE + IRRADIATION TESTING + INSTRUMENT, ABNORMAL INDICATION + R AND D PROGRAM

9-031241
SWANSON CD

DATED 1968

FAST FLUX TEST FACILITY INSTRUMENTATION REQUIREMENTS

BATTELLE-NORTHWEST, RICHLAND, WASH., PACIFIC NORTHWEST LAB.

BNWL-SA-1826 +. SEPTEMBER 1968, 54P.

THE FAST FLUX TEST FACILITY IS THE MAJOR IRRADIATION TEST FACILITY PLANNED FOR THE LIQUID METAL FAST BREEDER REACTOR PROGRAM. THIS PAPER DISCUSSES THE FACILITY INSTRUMENTATION REQUIREMENTS, PLACING PARTICULAR EMPHASIS ON OUT-OF-CORE INSTRUMENTATION. IN-CORE INSTRUMENTATION REQUIREMENTS ARE DESCRIBED IN MORE GENERAL TERMS DUE TO THEIR GREATER DEPENDENCE ON REACTOR CONCEPT DECISIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT INSTRUMENTATION + INSTRUMENT, PROCESS + FFTF (TR) + INSTRUMENT, FUEL SCANNING + INSTRUMENT, TEMPERATURE + MONITOR + INSTRUMENT, COOLANT QUALITY + IRRADIATION FACILITY + REACTOR, LMFB + INSTRUMENTS, MISC.

9-032526 ALSO IN CATEGORY 5
VOLK ER + JENS WH + KISHIDA K + NIMS JB
MALFUNCTION DETECTION IN FAST REACTORS
ATOMIC PWK. DEVELOP. ASSOC. INC.

DATED 1967

CONF-670916 +. 13 PAGES, PAGES VA2.1-13 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPT. 19-22, 1967

REVIEWS THE METHODS OF MALFUNCTION DETECTION IN FAST REACTORS. EQUIPMENT AND OPERATIONAL PROCEDURES REQUIRED FOR SUCH SYSTEMS ARE OUTLINED. THE PROPOSED MALFUNCTION DETECTION SYSTEM FOR THE FERMI FAST BREEDER REACTOR IS DESCRIBED. CALCULATIONAL PROCEDURES USED IN ON-LINE ANALYSES ARE DETAILED. COMPUTER AND DATA ACQUISITION SYSTEM SPECIFICATIONS ARE TABULATED.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + DATA PROCESSING + SAFETY ANALYSIS + TEST, ON LINE + REACTOR, LMFB + FAILURE, INCIPENT

9-032527 ALSO IN CATEGORIES 5 AND 6
STACHUKA SJ + BLAINE RA + BERLAND RF + HAUSKNECHT DF
AIROS-II - REACTOR DYNAMICS SIMULATOR FOR LMFB DESIGN
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

DATED 1967

CONF-670916 +. 9 PAGES, PAGES IVA5.1-9 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

MODIFICATIONS TO THE DIGITAL KINETICS CODE, AIROS, HAVE BEEN INCORPORATED TO PROVIDE CAPABILITIES FOR TREATING PHASE CHANGES DURING TRANSIENTS SUCH AS COOLANT VOIDING AND FUEL MELTING BASED UPON SIMPLIFIED MODELS. THE GENERALIZED NATURE OF THE FEEDBACK EQUATIONS PERMITS DETAILED TREATMENT OF EITHER NEUTRONIC, THERMAL, OR FLOW BEHAVIOR.

COMPUTER PROGRAM + FUEL MELTDOWN + REACTOR DYNAMICS + VOID COEFFICIENT + ANALYTICAL MODEL + PHASE CHANGE + SIMULATION + REACTOR, LMFB

PAGE 140

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-032590 DATED 1900
AN EVALUATION OF THE REQUIREMENT FOR INSTRUMENTING ALL FUEL CHANNELS IN FFTF
BATTTELLE-NORTHWEST, RICHLAND, WASH.
BNWL-CC-1012 +. 10 PAGES

PRESENTS CONSIDERATIONS OF THE VARIOUS PURPOSES FOR INSTRUMENTING TEST AND DRIVER FUEL. PROPOSES
MINIMUM REQUIREMENTS FOR THIS INSTRUMENTATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + INSTRUMENT, IN CORE + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + MEASUREMENT, TEMPERATURE +
DESIGN STUDY + INSTRUMENT, DETECTION FAILED FUEL ELEMENT

9-032627 ALSO IN CATEGORIES 5 AND 17 DATED 1968
REPORT ON THE FUEL MELTING INCIDENT IN THE ENRICO FERMI ATOMIC POWER PLANT
POWER REACTOR DEVELOPMENT COMPANY
50 PAGES, FIGURES, TABLES, REFERENCES, DECEMBER 15, 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--
COMMONWEALTH ASSOC.

DESCRIBES THE OCT. 5 HAPPENINGS (FLOW BLOCKAGE DUE TO ZIRCONIUM PLATE CAUSED PARTIAL MELTDOWN,
RELEASING FISSION GAS AND LOSING EXCESS REACTIVITY), RESULTS OF INVESTIGATIONS (ZIRCONIUM FROM
CONICAL FLOW GUIDE, FUEL FAILURE DID NOT PROPAGATE, FLOW REDUCTIONS CAME ONLY FROM ZIRCONIUM
SEGMENT), PLANT MODIFICATION (ON-LINE COMPUTER WILL ALARM ON DEVIATION FROM CALCULATED VALUE OF
REACTIVITY BALANCE, FISSION PRODUCTS, OR SUBASSEMBLY DELTA T AND WILL LATER CAUSE SCRAM).
STARTUP TEST PROGRAM OUTLINED. APPENDICES GIVE DETAILS ON FUEL EXAMINATION. *** OCT. 5
OPERATION WAS TO CHECK EARLIER HIGH DELTA T IN ELEMENTS WHICH MELTED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM
CHARGE \$2.00)

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + INCIDENT + INSTRUMENT,
TEMPERATURE + REPORT, OPERATIONS ANALYSIS + INCIDENT, RECOVERY FROM + EXAMINATION + REMOTE MANIPULATING AND
VIEWING + FLOW BLOCKAGE + INSTRUMENT, ABNORMAL INDICATION + INSTRUMENT, OPERATING REACTIVITY + COMPUTER
CONTROL + REACTOR, LMFBR + FUEL ELEMENTS

9-032628 ALSO IN CATEGORY 17 DATED 1968
FERMI MELTDOWN INDICATIONS
POWER REACTOR DEVELOPMENT COMPANY
32 PAGES, FIGURES, TABLES, PP. 5-37 OF REPORT ON FUEL MELTING INCIDENT, DECEMBER 15, 1968, DOCKET 50-16, TYPE--
LMFBR, MFG--APDA, AE--COMMONWEALTH

AUGUST OPERATION SHOWED HIGH OUTLET TEMPERATURES FROM SEVERAL FUEL ELEMENTS. ELEMENTS WERE MOVED
TO SEE IF THE CAUSE WAS POOR FLOW DISTRIBUTION (THE OCT. 5 OPERATION WAS TO TEST THIS). HOT
REACTIVITY AT 1 MW WAS OK. DELAYS DURING THE DAY LEAD TO RECORDING THERMOCOUPLE DATA DURING
POWER USE, RATHER THAN INCREASING POWER IN STEPS AND MONITORING TEMPERATURES. AT 20 MWTH IN
AUTOMATIC CONTROL THE DN/UT SIGNAL BECAME ERRATIC. AS THIS HAD BEEN EXPERIENCED BEFORE AT THIS
POWER LEVEL AND AS THE NOISE DISAPPEARED IN MANUAL CONTROL, REACTOR POWER AGAIN RAISED IN
AUTOMATIC TIL ALL RODS BEING 9 IN. WITHDRAWN INSTEAD OF 6 IN. CAUSED THE SRO TO HALT POWER
INCREASE, THEN HIGH ELEMENT OUTLET TEMPERATURES (3X NORMAL) AND RADIATION ALARMS RESULTED IN
MANUAL SHUTDOWN. ANALYSIS SHOWED A LOSS OF \$0.005/MW FROM 6 TO 18 MW, THEN LOSS OF \$0.013/MW
ABOVE THIS, WITH A NET LOSS OF \$0.26 (11.5 FUEL LOSS AND 6.9 FUEL BOWING). THE FISSION PRODUCT
DETECTOR (CHARGED WIRE IN REACTOR VESSEL OFFGAS) WAS BEING CALIBRATED DURING POWER INCREASES, AND
REQUIRED 3 OR 7 MIN SUAK TIMES. THE LAST MEASUREMENT WAS OFF SCALE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM
CHARGE \$2.00)

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + NOISE + INSTRUMENT,
DETECTION FAILED FUEL ELEMENT + REPORT, OPERATIONS ANALYSIS + PERIOD METER + FLOW BLOCKAGE + REACTIVITY
EFFECT, ANOMALOUS + INSTRUMENT, ABNORMAL INDICATION + OPERATING EXPERIENCE + REACTOR, LMFBR

9-032807 ALSO IN CATEGORY 18 DATED 1968
REFLECTOR CONTROL SYSTEM
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-11 +. 160 PAGES, 24 FIGURES, TABLES, SEFOR, PSAR, SUPPLEMENT 11, FEBRUARY 29, 1968, DOCKET 50-231

DESCRIBES CONTROL SYSTEM AND DISCUSSES THE VARIOUS COMPONENTS FROM DESIGN TO INSTALLATION AND
TESTING. SYSTEM HAS BEEN SUBJECTED TO MORE THAN 800,000 INCHES OF TRAVEL AT NORMAL DRIVE SPEEDS
AND 130,000 INCHES AT SCRAM SPEED, WHICH IS APPROXIMATELY 12 TIMES EXPECTED USAGE, AND HAS NEVER
FAILED TO FUNCTION. GUIDE-PAD DEVELOPMENT AND TEST PROGRAM DEMONSTRATED PADS AND RAILS HAVE LIFE
EXPECTANCY CONSIDERABLY LONGER, AND AT LOADS HIGHER THAN EXPECTED OPERATING CONDITIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CONTROL SYSTEM + REACTOR, FAST + REACTOR, LMCR + TESTING + REACTOR, PULSED + SEFOR (RE) + REFLECTOR + SYSTEM
DESCRIPTION + REPORT, PSAR + REACTOR, LMFBR + CONTROL RODS + CONTROL ROD DRIVE MECHANISMS

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-032820 ALSO IN CATEGORY 18 DATED 1967
SPECIFICATION AND ELEMENTARY DIAGRAMS OF REACTOR SAFETY SYSTEM
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-2 +. 75 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 2, DEC. 5, 1967, DOCKET 50-231

GIVES DETAILED DRAWINGS OF ELECTRICAL MAKEUP OF SAFETY SYSTEM. SYSTEM IS DESCRIBED,
SPECIFICATIONS AND QUALITY ASSURANCE ARE GIVEN.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (108 CENTS/PAGE -- MINIMUM
CHARGE \$2.00)

DESIGN CRITERIA + REACTOR, FAST + REACTOR, LMCR + REACTOR SAFETY SYSTEM + REACTOR, PULSED + SEFOR (RE) +
SYSTEM DESCRIPTION + REPORT, PSAR + REACTOR, LMFBR

9-032836 ALSO IN CATEGORIES 5 AND 11 DATED 1969
RYDEN J
NONDESTRUCTIVE TESTING OF SMALL DIAMETER, STAINLESS STEEL FUEL CLAD TUBING
BATTELLE-NORTHWEST, RICHLAND
BNWL-SA-2275 + CONF-690311-1 +. 20 PAGES, FROM NATIONAL SPRING CONFERENCE OF THE AMERICAN SOCIETY FOR
NONDESTRUCTIVE TESTING, LOS ANGELES, 1969

DISCUSSES THE TECHNIQUES USED BY THE PACIFIC NORTHWEST LAB. TO INSPECT FFTF FUEL CLADDING FOR
DEFECTS. DESCRIBES THE THREE DIFFERENT TESTERS USED TO INSPECT THIS TUBING. THEY ARE A
SINUSOIDAL EDDY CURRENT TESTER, AN ULTRASONIC PULSE-ECHO TESTER, AND AN ULTRASONIC TESTER THAT
USES A SEND-RECEIVE SYSTEM OF TRANSDUCERS. MICROGRAPHS, STRIP CHARTS, AND STATISTICAL DATA
OBTAINED FROM THE INSPECTION OF 20,000 FEET OF TUBING ARE PRESENTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + TESTING + STEEL, STAINLESS + FFTF (TR) + INSTRUMENT
CALIBRATION + TEST, NONDESTRUCTIVE + R AND D PROGRAM + TUBING

9-032884 ALSO IN CATEGORIES 6 AND 18 DATED 1968
RESPONSE TO QUESTION 12 - FRED OPERATING VARIABLES
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 3 PAGES, 1 FIGURE, PAGE 41 THRU 43 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

PROVIDE INFORMATION ON OPERATIONAL VARIABLES TO BE USED DURING THE FAST REACTIVITY EXCURSION
DEVICE EXPERIMENTS AND INDICATE WHICH VALUES HAVE BEEN OBTAINED BY MEANS OTHER THAN DIRECT
MEASUREMENT. *** ANSWER ON PAGES 41 THRU 43.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + MEASUREMENT, REACTIVITY +
REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + INSTRUMENTS, MISC.

9-032890 ALSO IN CATEGORIES 6 AND 18 DATED 1968
RESPONSE TO QUESTION 13 - INSTALLATION AND REMOVAL OF TEST DEVICES
GENERAL ELECTRIC
50231-20 +. 4 PAGES, 1 FIGURE, PG. 44 THRU 48 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

FROM INFORMATION SUBMITTED, IT APPEARS THAT DUE TO HEIGHT AND OTHER LIMITATIONS, THE EXPERIMENTAL
EQUIPMENT FOR EXCURSION MEASUREMENT MUST BE PARTIALLY ASSEMBLED AND DISASSEMBLED IN PLACE.
DESCRIBE SEQUENTIALLY THE STEPS INVOLVED AND THE CHECKS THAT WILL BE MADE IN REMOVING AND
INSTALLING THE DEVICES. *** ANSWER ON PAGES 44 THRU 48

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + EXCURSION, LARGE + MEASUREMENT, REACTIVITY + REACTOR,
PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + INSTRUMENTS, MISC.

9-032894 ALSO IN CATEGORIES 6 AND 11 AND 18 DATED 1968
RESPONSE TO QUESTION 17 - DESIGN OF PRESSURE VESSELS IN FRED
GENERAL ELECTRIC
50231-10 +. 2 PAGES, 1 FIGURE, PG. 53 AND 54 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

DESCRIBE BLOWOUT PATCH AND ACCUMULATOR ON THE EXCURSION DEVICE. *** ANSWER ON PAGES 53 AND 54.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) +
SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + INSTRUMENTS, MISC.

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-032916 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 1 - CONTROL SYSTEM MODIFICATIONS
 GENERAL ELECTRIC, SAN JOSE
 50231-21 +. 5 PAGES, 2 TABLES, PAGES 1 THRU 5 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

WHAT MODIFICATIONS ARE TO BE MADE TO CONTROL SYSTEM TO ENSURE GREATER FORCE AVAILABLE TO WITHDRAW
 SEGMENT THAN AVAILABLE TO INSERT. *** ANSWER ON PAGES 1 THRU 5.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

CONTROL SYSTEM + REACTOR, FAST + REACTOR, LMCR + REACTOR CONTROL + MODIFICATION + REACTOR, PULSED + SEFOR (RE)
 + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFB + CONTROL ROD DRIVE MECHANISMS

9-032919 ALSO IN CATEGORIES 6 AND 18 DATED 1968
RESPONSE TO QUESTION 4 - DESCRIPTION OF EXCURSION TESTING DEVICE
 GENERAL ELECTRIC, SAN JOSE
 50231-21 +. 35 PAGES, 13 FIGURES, 1 TABLE, PAGES 16 THRU 51 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET
 50-231

PROVIDE ADDITIONAL INFORMATION ON THE EXCURSION TESTING DEVICE (FRED) INCLUDING DETAILS OF ITS
 DESIGN AND HOW IT WILL BE UTILIZED IN THE SEFOR EXPERIMENTAL PROGRAM. *** ANSWER ON PAGES 18
 THRU 51.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) +
 SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + EQUIPMENT + REPORT, PSAR + REACTOR, LMFB

9-032920 ALSO IN CATEGORIES 5 AND 18 DATED 1968
RESPONSE TO QUESTION 5 - COOLANT LEAK EFFECT ON CONTROL
 GENERAL ELECTRIC, SAN JOSE
 50231-21 +. 4 PAGES, 1 FIGURE, PAGES 52 THRU 56 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

DISCUSS POSSIBILITY OF SODIUM LEAK IN PRIMARY COOLANT SYSTEM THAT MIGHT INTERFERE WITH OPERATION
 OF REFLECTOR CONTROL SYSTEM. *** ANSWER ON PAGES 52 THRU 57.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + CONTROL SYSTEM + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + REACTOR
 CONTROL + REACTOR, PULSED + SEFOR (RE) + REFLECTOR + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFB
 + CONTROL RODS

9-033051 ALSO IN CATEGORIES 1 AND 12 DATED 1968
LIQUID METAL ENGINEERING CENTER - SEMIANNUAL TECHNICAL PROGRESS REPORT, JULY-DECEMBER 1968
 LIQUID METAL ENGINEERING CENTER
 LMEC-69-1 +. 281 PAGES, 87 FIGURES, 17 TABLES, REFERENCES, DECEMBER 1968

SUBJECTS COVERED - VALVES AND SODIUM TECHNOLOGY, PUMP STUDIES, PIPING STUDIES, LMFB TEST AND
 FACILITY PLANNING, CODES AND STANDARDS FOR LMFB AND RELATED ACTIVITIES (REACTOR VESSEL,
 CIRCULATING COLD TRAP, MECHANICAL FREE-SURFACE SODIUM PUMPS, ELECTRO-MAGNETIC PUMPS, SODIUM
 PIPING, VALVES, AND QUALITY), MAINTENANCE AND MALFUNCTION ANALYSIS, COMBINED UNIT REACTOR VESSEL
 STUDY, QUALITY ASSURANCE, OPERATION OF SODIUM COMPONENTS TEST INSTALLATION, GENERAL OPERATING
 ACTIVITIES, LIQUID METAL INFORMATION CENTER, SODIUM PUMP TEST FACILITY, SODIUM PUMP SEAL TEST
 PROGRAM, BEAMING TEST PROGRAM, STEAM GENERATOR AND INTERMEDIATE HEAT EXCHANGER PERFORMANCE
 EVALUATION, INSTRUMENTATION TESTING, EVALUATION, CALIBRATION, AND STANDARDS, OPERATION AND
 MAINTENANCE OF THE LCTL, LETS, AND HNPFP PUMP TEST LOOP, MODIFICATIONS TO THE LCTL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
 COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + CODES AND STANDARDS + PRESSURE VESSELS + SODIUM + TESTING + STRESS ANALYSIS + EQUIPMENT + STEAM
 GENERATOR + PUMP, MAGNETIC + REACTOR, LMFB + PUMPS + VALVES + PIPES AND PIPE FITTINGS + INSTRUMENTS, MISC.

9-033128 DATED 1969
 ZINIUK MA + WILBER HA
REACTOR PERFORMANCE ON APPROACH TO POWER BY AUTOMATIC CONTROL - ENRICO FERMI REACTOR
 ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., MICH.
 APDA-NTS-22 +. 31 PAGES, FEBRUARY 1969

PRESENTS RESULTS OF TESTS AT THE ENRICO FERMI REACTOR AT POWER LEVELS OF 13.3, 20, AND 110 MWT FOR
 APPROACH TO POWER UNDER CONTROL OF THE AUTOMATIC LOADING PROGRAMMER. A PERCEPTIBLE POWER
 OSCILLATION, THOUGH WELL DAMPED, WAS OBSERVED ON LEVEL-OFF TO 13.3 AND 20 MWT, BUT 110 MWT WAS
 OBTAINED SMOOTHLY, WITHOUT OSCILLATION. THE TESTS PROVIDED DATA THAT CONFIRM THAT TEMPERATURE
 TRANSIENTS WERE WITHIN ESTABLISHED LIMITS AND THAT PROGRAMMER PERFORMANCE WAS SATISFACTORY.

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-033128 *CONTINUED*
AVAILABILITY - M.A. ZINIUK, ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN

CONTROL SYSTEM + FERMI (LMFBR) + INSTRUMENT, NUCLEAR + INSTRUMENT, CONTROL + REACTOR CONTROL + INSTRUMENT, COMPONENT + INSTRUMENT, TEMPERATURE + INSTRUMENT, STARTUP + INSTRUMENT, STARTUP RANGE + REACTOR, LMFBR

9-033364 DATED 1968
MACLEOD CJ
REACTOR INSTRUMENTATION
UNIVERSITY OF STRATHCLYDE, GLASGOW
4 PAGES, MEAS. INSTRUM. REV., 15, PAGES 741-4 (OCTOBER 1968)

SURVEYS INSTRUMENTATION LIKELY TO BE REQUIRED BY FOUR TYPES OF REACTORS WHICH MAY, IN THE FUTURE, SUPERSEDE THE GRAPHITE-MODERATED, GAS-COOLED REACTORS USED AT PRESENT BY THE CENTRAL ELECTRICITY GENERATING BOARD OF UK. THE INFORMATION IS PRESENTED IN TABULAR FORM, THE REACTORS CONSIDERED BEING AGR, FAST REACTORS, WATER REACTORS, AND FLUID-FUEL REACTORS.

REACTOR, FAST + REACTOR, GCR + LIQUID FUEL + REACTOR, LMCR + REACTOR, LWR + SURVEY + INSTRUMENTS, MISC.

9-033375 DATED 1968
RAVEN AD
CONTROLLING NUCLEAR REACTORS
ENGLISH ELECTRIC CO., LTD.
BRITISH PATENT 1,131,103 +. OCTOBER 23, 1968

DESCRIBES A LIQUID-METAL-COOLED REACTOR WHICH HAS LI DISPERSED IN THE LIQUID METAL COOLANT TO ADD IN REACTIVITY CONTROL. THE PROPORTION OF LI IN THE COOLANT IS 1% BY VOLUME.

AVAILABILITY - THE PATENT OFFICE, 25 SOUTHAMPTON BUILDING, LONDON W.C. 2, ENGLAND

REACTOR, LMCR + REACTIVITY EFFECT + UNITED KINGDOM + REACTOR CONTROL + LITHIUM + PATENT

9-033729 DATED 1900
ANDERSON DR
FUEL HANDLING MECHANISMS FOR LMFBR SYSTEMS
ATOMICS INTERNATIONAL, CALIF.
CONF-680419 +. 19 PAGES, PP. 403-421.

GENERAL REQUIREMENTS FOR FUEL-HANDLING MECHANISMS ARE IDENTIFIED BY EXAMINING INTERFACES WITH THE OTHER REACTOR PLANT SYSTEMS. THE IMPACT OF FUEL HANDLING ON CONTAINMENT, CORE, VESSEL, FUEL MANAGEMENT, AND DECAY STORAGE ARE DISCUSSED. THE PRESERVATION OF FUEL INTEGRITY DURING HANDLING IS EMPHASIZED. THE TECHNOLOGY OF REFUELING SYSTEMS IN EXISTENCE TODAY IS EXPLORED IN TERMS OF ADVANTAGES, DISADVANTAGES, AND THE TECHNICAL BASIS FOR FUTURE SYSTEMS. IN-VESSEL VS EX-VESSEL STORAGE AND ON-LINE VS OFF-LINE REFUELING SCHEMES ARE COMPARED, AND THE ECONOMICS OF IN-VESSEL STORAGE FOR VARIOUS REFUELING INTERVALS IS PRESENTED.

ECONOMICS + FUEL HANDLING + REFUELING + REVIEW + REACTOR, LMFBR

9-033950 ALSO IN CATEGORY 18 DATED 1968
SMAARDYK A + DIVONA CJ + DLP RH + HUITER E
DESIGN AND ANALYSIS OF INSTRUMENTED SUBASSEMBLY SYSTEM FOR EBR-II
ARGONNE NATIONAL LAB., ILLINOIS
ANL-7423 +. 48 PAGES, 21 FIGURES, 3 TABLES, 7 REFERENCES, JUNE 1

DESCRIBES THE RESEARCH, DESIGN, AND DEVELOPMENT WORK PERFORMED AFTER A FEASIBILITY STUDY OF INSTALLING AN INSTRUMENTED SUBASSEMBLY SYSTEM CAPABLE OF MEASURING PERTINENT OPERATING PARAMETERS OF SPECIMEN FUELS AND RELATED COMPONENTS WHILE UNDER IRRADIATION IN THE EBR-II CORE. IMMEDIATE OBJECTIVE IS CONSTRUCTION OF A PROTOTYPE SYSTEM FOR INSTALLATION AND PERFORMANCE EVALUATION IN THE EBR-II DURING THE FIRST QUARTER OF CALENDAR YEAR 1969 FOR EVALUATING ADVANCED FUELS OF INTEREST TO THE AEC LMFBR PROGRAM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + REACTOR, FAST + FUEL BACKUP + INSTRUMENT, IN CORE + REACTOR, LMCR + REACTOR, TEST + ANL + DESIGN STUDY + IRRADIATION TESTING + R AND D PROGRAM + REACTOR, LMFBR + FUEL ELEMENTS

9-034390 ALSO IN CATEGORY 6 DATED 1966
KANDALL RL + PEKUL PJ
APPLICATIONS OF ANALOG TIME AND FREQUENCY CORRELATION COMPUTERS TO REACTOR-SYSTEM
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA
24 PAGES, 14 FIGURES, 14 REFERENCES, PAGES 357-80 OF NEUTRON NOISE, WAVES, AND PULSE PROPAGATION, MAY 1967, PROCEEDINGS OF A SYMPOSIUM AT THE UNIVERSITY OF FLORIDA, GAINESVILLE, FEBRUARY 14-16, 1966

ANALYSIS TECHNIQUES OF ANALOG-CROSS SPECTRAL DENSITY AND ANALOG CROSS CORRELATION ARE APPLIED TO PRACTICAL PROBLEMS IN REACTOR DESIGN AND PERFORMANCE EVALUATION. CROSS-SPECTRAL-DENSITY TECHNIQUES AND EQUIPMENT ARE DESCRIBED FOR EXPERIMENTAL DETERMINATION OF A NUMBER OF FEEDBACK

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-034390 *CONTINUED*

COEFFICIENTS VITAL TO REACTOR STABILITY AND CONTROL. CROSS-CORRELATION TECHNIQUES, TRANSDUCER DEVELOPMENTS, AND CORRELATION EQUIPMENT ARE DESCRIBED FOR MEASUREMENT OF LOCALIZED IN-CORE COOLANT VELOCITY AND FUEL-ELEMENT STRUCTURAL VIBRATIONS.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + INCIDENT, CONSEQUENCE + REACTOR STABILITY + SNAP 8 (SR) + REACTOR, SPACE + TESTING + ANALYTICAL MODEL + DATA PROCESSING + REACTOR CONTROL + SRE (RE) + NOISE ANALYSIS + ANALOG SIMULATION + FLOW

9-034663

DATED 1969

PURGES KGA

FUEL-FAILURE DETECTION IN LMFBR POWER PLANTS

ARGONNE NATIONAL LABORATORY, ARGONNE, ILL.

ANL-7533 +. 45 PAGES, 4 FIGURES, 6 TABLES, 21 REFERENCES, FEBRUARY 1969

DISCUSSES APPLICATION OF DELAYED-NEUTRON MONITORING TO THE LATTER TASK. THE VARIOUS EFFECTS ON THE SIGNAL OBTAINED FROM SUCH A CHANNEL OF THE TYPE OF FUEL, CONJECTURAL FAILURE MECHANISM, FISSION-PRODUCT TRANSPORT TO THE DETECTOR SITE, AND, FINALLY, ELECTRONIC PROCESSING ARE DISCUSSED AND EVALUATED IN THE LIGHT OF TRANSIENT-DETECTION STATISTICS. SUCH STATISTICAL CONSIDERATIONS STRONGLY EMPHASIZE THE NEED FOR ADEQUATE ELECTRONIC PROCESSING AND PROVIDE GUIDANCE IN THE CHOICE OF DETECTOR SITE AND OTHER HARDWARE DECISIONS. WITH WELL-DESIGNED ELECTRONICS AND A FAVORABLE SITE, A DELAYED NEUTRON-MONITORING SYSTEM CAN GIVE A FAILURE INDICATION WITHIN ABOUT 5 SEC FROM THE ACTUAL MOMENT OF FAILURE, WHEREAS A FLOWMETER GIVES AN INDICATION WITHIN 1 SEC FROM THE MOMENT THE CHANNEL BECOMES BLOCKED BY DEBRIS. THE DELAYED-NEUTRON SIGNAL SHOULD ALLOW CERTAIN INFERENCES REGARDING THE TYPE AND EXTENT OF FAILURE FROM THE SIGNAL SHAPE. THE FLOWMETER, ON THE OTHER HAND, LOCATES THE EVENT. THE TWO SYSTEMS THUS COMPLEMENT EACH OTHER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

DELAYED NEUTRON + REACTOR, FAST + INSTRUMENT, NUCLEAR + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + REACTOR, LMFBR

9-034784

ALSO IN CATEGORY 12

DATED 1969

ASTLEY ER

FAST FLUX TEST FACILITY - PERIODIC TECHNICAL REPORT - OCTOBER 1968 THROUGH FEBRUARY 1969

BATTELLE NORTHWEST, RICHLAND, WASHINGTON

BNWL-941 +. 200 PAGES, FIGURES, TABLES, MAY, 1969

COVERS PROGRAM MANAGEMENT, AND FFTF OPERATIONS, PLANT DESIGN, COMPONENTS, INSTRUMENTATION AND CONTROLS, SODIUM TECHNOLOGY, CORE DEVELOPMENT AND DESIGNS, SAFEGUARDS, AND CLADDING DEVELOPMENT

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151 \$3.00 COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + REACTOR, LMCR + SODIUM + REACTOR, TEST + FFTF (TR) + INSTRUMENT, CONTROL + CORROSION + COOLANT CHEMISTRY + DESIGN STUDY + CORE COMPONENTS + SAFETY ANALYSIS + FUEL ELEMENTS

9-035244

DATED 1969

BARTON GB

INSTRUMENTATION FOR ANALYSIS OF THE INERT GASES USED IN FFTF

BATTELLE-NORTHWEST LABORATORY

BNWL-957 +. 56 PAGES, 4 TABLES, 11 REFERENCES, MARCH 1969

REVIEWS INERT-GAS MONITORING NEEDS OF THE FFTF. DEFINES THE SELECTION CRITERIA FOR THE REQUIRED INSTRUMENTATION. CURRENT TECHNOLOGY AND COMMERCIAL ON-LINE GAS-ANALYTICAL INSTRUMENTS ARE REVIEWED AND COMPARED TO FFTF REQUIREMENTS. RESULTS OF THIS STUDY CONCLUDE THAT FFTF CELL-BLANKET GAS-ANALYTICAL NEEDS CAN BE MET IN ALMOST ALL INSTANCES BY COMMERCIAL INSTRUMENTS. CURRENT TECHNOLOGY IS INADEQUATE TO MEET SODIUM COVER-GAS-MONITORING NEEDS, AND A DEVELOPMENT EFFORT IS REQUIRED (GAS CHROMATOGRAPHY AND POSSIBLY MASS SPECTROMETRY APPEAR TO HOLD THE GREATEST PROMISE).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + REACTOR, FAST + CHROMATOGRAPHY + REACTOR, LMCR + REACTOR, TEST + ANALYTICAL TECHNIQUE + COVER GAS + FFTF (TR) + DESIGN STUDY + INSTRUMENT, SURVEILLANCE + REVIEW + GAS + INSTRUMENTS, MISC.

9-036078

ALSO IN CATEGORIES 6 AND 17

DATED 1969

SMITH RR + BUMP TR + CUSHMAN RA + HYNDMAN RW + KIRN FS + LUEWENSTEIN WB + LONG KJ + MADELL JT + PERSIANI PJ + WALLIN WR

THE EFFECTS OF AN OVER-COOLED STAINLESS STEEL REFLECTOR ON THE EBR II POWER COEFFICIENT

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ANL-7544 +. 115 PAGES, 60 FIGURES, 20 TABLES, 42 REFERENCES, MAY, 1969

DURING A ROUTINE STARTUP FOR EBR-II RUN 25 ON APRIL 21, 1967, SIGNIFICANT CHANGES WERE NOTED IN THE BASIC BEHAVIOR OF THE POWER COEFFICIENT. THE MAGNITUDE OF THE INTEGRAL POWER COEFFICIENT DECREASED BY A FACTOR OF APPROXIMATELY TWO WHILE THE MAGNITUDE OF THE DIFFERENTIAL POWER COEFFICIENT IN THE MIDPOWER REGION DECREASED BY A FACTOR OF APPROXIMATELY FOUR. CONSIDERABLE

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-036078 *CONTINUED*

ANALYTICAL EFFORT SHOWED THE CHANGE IN THE POWER COEFFICIENT TO BE IDENTIFIED WITH THERMOMECHANICAL DEFORMATION EFFECTS IN A STAINLESS STEEL REFLECTOR THAT WAS INSTALLED BEFORE THE STARTUP.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

POWER COEFFICIENT + REACTOR STABILITY + STEEL, STAINLESS + DEFORMATION + MEASUREMENT, REACTIVITY + REFLECTOR + SYSTEM DESCRIPTION + REACTIVITY EFFECT, ANOMALOUS + REACTOR PHYSICS + OPERATING EXPERIENCE + REACTOR, LMFBR

9-036200

DATED 1969

HIGH-TEMPERATURE PERFORMANCE TESTS ON AN EDDY-CURRENT FLOWMETER

ARGONNE NATIONAL LABORATORY

4 PAGES, 7 FIGURES, IEEE TRANSACTIONS ON NUCLEAR SCIENCE, NS-16(1), PAGES 192-5 (FEBRUARY 1969)

THE EDDY-CURRENT FLOWMETER SEEMS WELL ADAPTED TO MEASUREMENTS IN HIGH-TEMPERATURE LIQUID-METAL SYSTEMS, WHERE DETERIORATION OF THE PROPERTIES OF PERMANENT MAGNETS AND OTHER FERROMAGNETIC PARTS OR BEARING PROBLEMS MAY SERIOUSLY DEGRADE THE PERFORMANCE OF OTHER TYPES. TESTS ON AN EDDY-CURRENT FLOWMETER CONFIRM THE HIGH SIGNAL LEVELS AND THE OPTIMUM EXCITATION FREQUENCY PREDICTED BY AN ANALYSIS OF THE DEVIATION.

REACTOR, LMFBR + HIGH TEMPERATURE + INSTRUMENT, FLOW + TEST, INSTRUMENT RESPONSE + METAL, LIQUID

9-036206

DATED 1969

SWANSON CD

FAST FLUX TEST FACILITY INSTRUMENTATION REQUIREMENTS

BATTELLE MEMORIAL INSTITUTE, RICHLAND, WASHINGTON

17 PAGES, 9 FIGURES, 5 REFERENCES, IEEE TRANSACTIONS ON NUCLEAR SCIENCE, NS-16(1), PAGES 232-48 (FEBRUARY 1969)

DISCUSSES THE FACILITY INSTRUMENTATION REQUIREMENTS, PLACING PARTICULAR EMPHASIS ON OUT-OF-CORE INSTRUMENTATION. IN-CORE INSTRUMENTATION REQUIREMENTS ARE DESCRIBED IN MORE GENERAL TERMS DUE TO THEIR GREATER DEPENDENCE ON REACTOR CONCEPT DECISIONS. OUT-OF-REACTOR INSTRUMENTATION REQUIREMENTS INCLUDED IN THESE DISCUSSIONS ARE - SODIUM AND INERT GAS PURITY, SODIUM HEAT TRANSPORT SYSTEM MEASUREMENTS, FUEL HANDLING AND EXAMINATION, RADIATION MONITORING, CONTAINMENT ACTUATION AND PROCESS CONTROL PROBLEM AREAS. CONSIDERATIONS LEADING TO PROVISIONS FOR A MULTIPLE DIGITAL COMPUTER-BASED DATA HANDLING AND DISPLAY SYSTEM ARE DESCRIBED.

AVAILABILITY - C.D. SWANSON, BATTELLE MEMORIAL INSTITUTE, PACIFIC NORTHWEST, LABORATORY, RICHLAND, WASHINGTON

CONTAINMENT INSTRUMENTATION + REACTOR, FAST + INSTRUMENT, IN CORE + INSTRUMENT, NUCLEAR + INSTRUMENT, PROCESS + INSTRUMENT, PROTECTIVE + REACTOR, LMFBR + MEASUREMENT + REACTOR, TEST + FFTF (TR) + INSTRUMENT, CONTROL + PLANT PROTECTIVE SYSTEM + DESIGN STUDY + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + INSTRUMENT, TEMPERATURE + REACTOR, LMFBR + DATA COLLECTION + RADIATION MONITORS + INSTRUMENTS, MISC.

9-036209

ALSO IN CATEGORY 5

DATED 1969

ANDERSON TT + GRATE TA

ACOUSTIC BOILING DETECTION IN REACTOR VESSELS

ARGONNE NATIONAL LABORATORY

4 PAGES, 10 REFERENCES, IEEE TRANSACTIONS ON NUCLEAR SCIENCE, NS-16(1), PAGES 260-3 (FEBRUARY 1969)

REACTOR COOLANT WHICH BOILS IN A FAST REACTOR OR HIGH-FLUX RESEARCH REACTOR, ENDANGERING REACTOR SAFETY, CAN BE DETECTED ACOUSTICALLY. AS A CONTRIBUTION TO DESIGN OF ACOUSTIC BOILING DETECTORS, THE TANK WALL OF A REACTOR IS SHOWN THEORETICALLY TO AFFECT STANDING WAVES IN THAT TANK. EXPERIMENTAL RESONANCE FREQUENCIES OF A 3-FT-DIAMETER TANK DEMONSTRATE THAT FOR LOW-PRESSURE REACTOR TANKS, THE PRESSURE AT STANDING-WAVE FREQUENCIES IS NEARLY ZERO AT THE TANK WALL, IN CONTRAST TO THE RESULTS FOR A PERFECTLY RIGID WALL.

AVAILABILITY - T.A. GRATE, ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

REACTOR, FAST + REACTOR, LMFBR + NOISE + COMPARISON, THEORY AND EXPERIENCE + REACTOR, TEST + INSTRUMENT, COMPONENT + BOILING + ACOUSTICS

9-036238

ALSO IN CATEGORY 5

DATED 1968

EVALUATION ON INCONEL 718 FOR APPLICATION IN THE FAST TEST REACTOR

BATTELLE-NORTHWEST, RICHLAND, WASHINGTON

BNWL-930 +. 31 PAGES, 7 FIGURES, 6 TABLES, 28 REFERENCES, DECEMBER 1968

INCONEL 718, A PRECIPITATION HARDENABLE NICKEL-BASE ALLOY, HAS RECENTLY BEEN CONSIDERED FOR USE IN THE FAST TEST REACTOR (FTR). PROPERTIES OF INTEREST WHICH QUALIFY THIS ALLOY FOR REACTOR USE ARE EXCELLENT WEAR RESISTANCE, FATIGUE RESISTANCE, EASE OF FABRICATION, SUPERIOR HIGH TEMPERATURE MECHANICAL PROPERTIES, AND STRUCTURAL STABILITY. THREE DISTINCT AREAS OF POSSIBLE APPLICATION HAVE THUS FAR BEEN IDENTIFIED - FLOW DUCT CONTACT PADS, SAFETY ROD ACCELERATOR SPRINGS, AND FUEL-PIN GRID-SPACER ASSEMBLIES.

REACTOR, FAST + REACTOR, LMFBR + REACTOR, TEST + STRESS ANALYSIS + FFTF (TR) + CORROSION + CONTROL ROD SCRAM MECHANISM + MATERIAL + PROPERTY, PHYSICAL + INCONEL + CORE COMPONENTS + ACCELERATION + STRUCTURAL ANALYSIS, DYNAMIC + PROPERTY, MECHANICAL

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-036239 DATED 1968
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM - 19TH QUARTERLY REPORT
GENERAL ELECTRIC, SUNNYVALE, CALIFORNIA
GEAP-5754 +. 65 PAGES, FIGURES, TABLES, REFERENCES, NOVEMBER 1968

SEFOR (SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR) IS A 20 MWt, SODIUM-COOLED FAST REACTOR WITH CHARACTERISTICS SIMILAR TO THE LARGE, SLOW SPECTRUM FAST BREEDER REACTORS FUELED WITH MIXED PUO₂/U(238)O₂, FOR WHICH STUDIES INDICATE THE POTENTIAL FOR PRODUCING LOW COST POWER. SEFOR WILL BE USED TO OBTAIN PHYSICS AND ENGINEERING DATA AT FUEL COMPOSITIONS, TEMPERATURES, AND CRYSTALLINE STATES CHARACTERISTIC OF POWER REACTOR OPERATING CONDITIONS. SEFOR IS PARTICULARLY DESIGNED FOR THE SYSTEMATIC DETERMINATION OF THE DOPPLER COEFFICIENT OF REACTIVITY AT TEMPERATURES UP TO THE VICINITY OF FUEL MELTING.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

FABRICATION + INSTRUMENT, IN CORE + INSTRUMENT, NUCLEAR + REACTOR, LMCR + OSCILLATOR, REACTIVITY + SODIUM + REACTOR CONTROL + INSTRUMENT, COMPONENT + TEST, INSTRUMENT RESPONSE + TEST, SYSTEM OPERABILITY + REACTOR, PULSED + SEFOR (KE) + EQUIPMENT DESIGN + CORE COMPONENTS + R AND D PROGRAM + FUEL ELEMENTS + FILTERS

9-036482 DATED 1968
BAXTER WG + KUHLMAN WC
FAST BREEDER REACTOR THERMOCOUPLE DEVELOPMENT (TASK 1414)
GENERAL ELECTRIC COMPANY, CINCINNATI, OHIO
GEMP-1010 +. 30 PAGES, 17 FIGURES, 5 TABLES, 2 REFERENCES, PP. 260-290 OF AEC FUELS AND MATERIALS DEVELOPMENT PROGRAM PROGRESS REPORT 78, DECEMBER 31, 1968

ONE OF A SERIES OF REPORTS ON FABRICATION OF EBR FUEL ELEMENT THERMOCOUPLE MATERIALS - W AND W-RE THERMOCOUPLE SHEATHS, ALUMINA-NICKEL METAL-TO-CERAMIC SEALS, SS-TO-W25RE WELDMENT THERMAL CYCLING, FACILITIES FOR THERMAL CYCLING COOLANT THERMOCOUPLES, AND FACILITIES FOR HIGH-TEMPERATURE-RESISTIVITY MEASUREMENTS ON THERMOCOUPLE INSULATORS. TESTS OF W-25RE-SHEATHED, W/W25RE CPLS. WITH BeO OR HfO₂ INSULATORS IN HELIUM AT 2300 C SHOWED LOSS OF W WIRE DIAM. AND GAIN OF W-25RE WIRE DIAMETER DUE TO OXYGEN TRANSPORT. ONE CPL. WITH H₂-FIRED HfO₂ INSULATOR TESTED TO 303 HR AT 2300 C SHOWED NO CHANGE IN CALIBRATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + TESTING + WELDING + HIGH TEMPERATURE + TUNGSTEN + INSTRUMENT, TEMPERATURE + EQUIPMENT DESIGN + RHENIUM + R AND D PROGRAM + THERMOCOUPLE

9-036650 DATED 1965
ASAMOTO RR + NOVAK PE
A SURVEY FOR A HIGH-TEMPERATURE SENSOR FOR SEFOR
GENERAL ELECTRIC COMPANY, SAN JOSE, CALIFORNIA
GEAP-4903 +. 28 PAGES, 2 FIGURES, 3 TABLES, 63 REFERENCES, JULY 1965

A SEARCH OF THE LITERATURE WAS UNDERTAKEN TO INVESTIGATE TEMPERATURE SENSING DEVICES USEABLE UP TO 4700 F IN A PU₂O₂ ENVIRONMENT. BASED ON THIS REVIEW, IT IS CONCLUDED THAT THERMOCOUPLES WOULD BEST MEET THE REQUIREMENTS FOR USE IN SEFOR FUEL TEMPERATURE MEASUREMENTS. THE REFERENCE STEADY-STATE FUEL THERMOCOUPLE DESIGN UTILIZES AN 1/8-INCH OD W26RE SHEATH, BeO, ThO₂, AND/OR GAS INSULATION, AND W-RE ELEMENTS. THE REFERENCE TRANSIENT FUEL THERMOCOUPLE WILL INCORPORATE EITHER A BARE-WIRE REFRACTORY-METAL HOT JUNCTION OR A REDUCED-DIAMETER SHEATHED DESIGN TO ACHIEVE THE DESIRED TIME RESPONSE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

BERYLLIUM + REACTOR, FAST + REACTOR, LMCR + OXIDE + THERMAL MECHANICAL EFFECT + THORIUM + RADIATION UNIT + HIGH TEMPERATURE + VAPOR PRESSURE + TUNGSTEN + INSTRUMENT, TEMPERATURE + ENVIRONMENT + REACTOR, PULSED + SEFOR (KE) + REVIEW + RHENIUM + INSTRUMENT, CURRENT + ELECTRICAL CONDUCTION + REFRACTORY MATERIAL + INSTRUMENT, VOLTAGE + THERMOCOUPLE

9-036766 DATED 1968
NOVAK PE + ASAMOTO RR
AN IN-PILE EVALUATION OF W-RE THERMOCOUPLE SYSTEMS FOR USE TO 4700 F IN SEFOR FUEL
GENERAL ELECTRIC COMPANY
GEAP-5468 +. 87 PAGES, 36 FIGURES, 7 TABLES, 15 REFERENCES, OCTOBER 1968

IN-PILE TESTS WERE CONDUCTED TO DEVELOP RELIABLE W-RE THERMOCOUPLE SYSTEMS CAPABLE OF MEASURING FUEL TEMPERATURES UP TO 4700 F IN SEFOR. RESULTS SHOWED THAT W5RE/W26RE THERMOCOUPLES CAN BE EXPECTED TO DEGRADATE LESS THAN 45 F AT 800 F AND LESS THAN 10 F AT 3000 AFTER THEIR FULL DESIGN LIFE OF 1800 HOURS IN THE SEFOR CORE. BOTH BERYLLIA AND THORIA INSULATED THERMOCOUPLES OPERATED SATISFACTORILY FOR OVER 1200 HOURS IN-PILE AT TEMPERATURES IN EXCESS OF 4200 F. BARE-WIRE, 1/16-IN(SHEATHED DIAM) W26RE-SHEATHED THERMOCOUPLES, AND 1/8-IN W26RE-SHEATHED THERMOCOUPLES OPERATED SATISFACTORILY IN-PILE UNDER BOTH STEADY STATE AND TRANSIENT CONDITIONS WHICH ARE REPRESENTATIVE OF THOSE EXPECTED IN SEFOR.

BERYLLIUM + REACTOR, FAST + REACTOR, LMCR + THORIUM + HIGH TEMPERATURE + TUNGSTEN + IN PILE EXPERIMENT + INSTRUMENT, TEMPERATURE + TEST, INSTRUMENT RESPONSE + REACTOR, PULSED + SEFOR (KE) + IN CORE MEASUREMENT +

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-036766 *CONTINUED*
RHENIUM + INSTRUMENT, ABNORMAL INDICATION + R AND D PROGRAM + THERMOCOUPLE

9-036773 DATED 1966
NOVAK PE + ASAMOTU RR
AN OUT-OF-PILE EVALUATION OF W-RE THERMOCOUPLE SYSTEMS FOR USE TO 4700 F IN PUO2-UO2
GENERAL ELECTRIC COMPANY, SAN JOSE, CALIFORNIA
GEAP-5166 +. 39 PAGES, 26 FIGURES, 15 TABLES, 29 REFERENCES, JUNE

CHARACTERISTICS OF W3RE/W25RE AND W5RE/W26RE THERMOCOUPLE SYSTEMS WERE MADE FOR USE IN MEASURING FUEL TEMPERATURES UNDER SEFOR CONDITIONS. BARE-WIRE TUNGSTEN-RHENIUM THERMOCOUPLES WERE CALIBRATED TO TEMPERATURES ABOVE 5400 F IN VACUUM. DRIFT TESTS FOR 15 HR AT 4700 F INDICATED GOOD STABILITY. THE PERFORMANCE OF THORIA-INSULATED THERMOCOUPLES WAS ADEQUATE AND REPRODUCIBLE FOR TEMPERATURE MEASUREMENTS TO 5100 F. BERYLLIA-INSULATED THERMOCOUPLES PERFORMED SATISFACTORILY UP TO 4600 F FOR SHORT TIMES. TUNGSTEN/26 PERCENT RHENIUM THERMOCOUPLE SHEARING AND MIXED OXIDE FUEL (17 PERCENT PUO2/83 PERCENT UO2) WERE COMPATIBLE AT AVERAGE SEFOR FUEL TEMPERATURES BUT SHOWED SIGNS OF REACTION AT 4700 F. CONCLUSIONS - THORIA-INSULATED, 1/8-IN.-OD, W26RE-SHEATHED, W3RE/W25RE .010-IN.-DIAM THERMOCOUPLES CAN MEASURE TEMPERATURE REPRODUCIBLY TO 5100 F, ARE COMPATIBLE WITH SEFOR FUEL TO 2700 F FOR TIME LONGER THAN 500 HR, ARE LIMITED TO A FEW HR AT 4700 F, AND HAVE TIME CONSTANTS OF ABOUT 3 SEC. W26RE IS PREFERRED TO TA FOR SHEATH IN CONTACT WITH FUEL. BARE-WIRE W3RE/W25RE AND W5RE/W25RE COUPLES CAN MEASURE 5400 F WITHIN + 2 PERCENT. W3RE IS PREFERRED FOR ROOM-TEMP. FLEXING TOLERANCE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

BERYLLIUM + REACTOR, FAST + REACTOR, LMC + OXIDE + THORIUM + INSTRUMENT CALIBRATION + HIGH TEMPERATURE + TUNGSTEN + INSTRUMENT, TEMPERATURE + TEST, INSTRUMENT RESPONSE + REACTOR, PULSED + SEFOR (RE) + EXAMINATION + RHENIUM + THERMOCOUPLE

9-036775 DATED 1969
SHEPARD RL
THERMOMETRY FOR FAST BREEDER REACTORS
OAK RIDGE NATIONAL LABORATORY
12 PAGES, 6 FIGURES, 3 TABLES, 42 REFERENCES, REACTOR AND FUEL-PROCESSING TECHNOLOGY, 12(2), PAGES 205-16 (MARCH 1969)

MIXED-OXIDE FUEL-PIN CENTERLINE TEMPERATURES OF 2300 TO 2750 C AND FUEL-PIN CLADDING TEMPERATURES TO 1100 C MUST BE MEASURED IN FUEL DEVELOPMENT STUDIES FOR THE LMFBR PROGRAM. COMMERCIAL SHEATHED, TYPE-K THERMOCOUPLES ARE BEING EVALUATED TO MEET RELIABILITY REQUIREMENTS FOR ROUTINE COOLANT-TEMPERATURE MONITORING IN LMFBR PLANTS. HIGH-TEMPERATURE W-RF THERMOCOUPLES ARE BEING DEVELOPED FOR FUEL-PIN TEMPERATURE MEASUREMENT. NEW TECHNIQUES AND INSTRUMENTS ARE BEING STUDIED AS ALTERNATIVES FOR OVERCOMING SOME OF THE BASIC LIMITATIONS OF THERMOCOUPLES AND THUS ENSURING ACCURATE, SAFE, RELIABLE, AND ECONOMICAL INSTRUMENTATION FOR THE FAST BREEDER REACTOR.

ALLOY + BERYLLIUM + CLADDING + HAFNIUM + OXIDE + THORIUM + CHROMIUM + MEASUREMENT, TEMPERATURE + HIGH TEMPERATURE + TUNGSTEN + FREQUENCY SPECTRA + INSTRUMENT, TEMPERATURE + TEST, INSTRUMENT RESPONSE + IN CORE MEASUREMENT + MAGNESIUM + REVIEW + RHENIUM + ELECTRICAL CONDUCTION + INSTRUMENT, FLUIDICS + REACTOR, LMFBR + ULTRASONICS + THERMOCOUPLE

9-037406 ALSO IN CATEGORY 17 DATED 1969
VULKER + GREEN DM + BATH ML
DESIGN AND CAPABILITIES OF THE MALFUNCTION DETECTION ANALYZER INSTALLED ON THE ENRICO FERMI FAST BREEDER REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-L3 +. 76 PAGES, FIGURES, TABLES, 19 REFERENCES, JULY 1969

AS A RESULT OF THE FUEL MELTING INCIDENT (FERMI, OCTOBER 5, 1966), A STUDY WAS INITIATED TO DETERMINE A PROMPT, RELIABLE, AND ECONOMIC MEANS TO DETECT MALFUNCTIONS THAT COULD LEAD TO A SIMILAR INCIDENT. CONDITIONS THAT COULD RESULT IN A SECOND INCIDENT WERE REVIEWED. CALCULATIONS CONSIDERED CRITICAL FOR DETECTION WERE SELECTED FOR DETAILED ANALYSIS. THE SELECTION AND DESCRIPTION OF THE REAL-TIME COMPUTER SYSTEM INSTALLED ARE INCLUDED.

REACTOR, BREEDER + COMPUTER PROGRAM + COMPUTER, DIGITAL + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + INSTRUMENT, PROTECTIVE + REACTOR, LMC + REACTOR SAFETY SYSTEM + INCIDENT + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + MONITOR + SYSTEM DESCRIPTION + REACTIVITY EFFECT, ANOMALOUS + COMPUTER CONTROL + REACTOR, LMFBR + FAILURE, INCIDENT

9-037487 ALSO IN CATEGORY 17 DATED 1969
ENRICO FERMI ATOMIC POWER PLANT REPORT FOR MARCH 1969
POWER REACTOR DEVELOPMENT COMPANY
EP-67 +. 11 PAGES, MARCH 1969, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

RESTORATION OF PRIMARY PIPING WAS COMPLETED AND THE VESSEL FILLED WITH SODIUM AND HEATED TO 450 F FOR SODIUM CLEANUP. 53 HR OF STARTUP TRAINING WERE LOGGED BY THE OPERATORS ON THE SIMULATOR. A DEFECTIVE RELAY COIL IN THE BUS THROWOVER CIRCUIT CAUSED THE CONTROL POWER FUSE TO BLOW. ELECTRICAL CONTAINMENT PENETRATIONS WERE LEAK-RATE TESTED. AN IBM SERIES-1800 COMPUTER WAS RECEIVED. THE COMPUTER WILL PERFORM COMPUTATIONS EVERY 5 SEC UNDER NORMAL CONDITIONS FOR ANOMALOUS REACTIVITY, SUBASSEMBLY OUTLET TEMP., AND FISSION-PRODUCT BEHAVIOR VS POWER.

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-037487 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + COMPUTER, DIGITAL + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR
+ REPORT, OPERATIONS + TEST, LEAK RATE + TRAINING + FAILURE, INSTRUMENT + INSTRUMENT, SURVEILLANCE +
SIMULATION + CONTAINMENT PENETRATION, ELECTRICAL + REACTOR, LMFBR + RELAYS

9-037965

DATED 1966

JOHNSON DL + STEARNS JD
EXAMINATION OF TYPE 304 STAINLESS STEEL PIPE EXPOSED TO NITROGEN IN THE SRE
ATOMICS INTERNATIONAL, INC.
NAA-SR-MEMO-12259 +. 3 PAGES, NOVEMBER 28, 1966

METALLOGRAPHIC EXAMINATION AND CHEMICAL ANALYSES WERE CONDUCTED ON SECTIONS OF 304 SS PIPE FROM
THE SRE PRIMARY SYSTEM TO DETERMINE IF IRRADIATION PROMOTED NITRIDING OF 304 SS AT 1200 F. NO
EVIDENCE OF NITRIDING WAS FOUND FOR UP TO 1600 HR EXPOSURE.

AVAILABILITY - D.L. JOHNSON, ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA

CHEMICAL REACTION + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + STEEL, STAINLESS + SRE (RE) + NITRIDE +
TEST, PHYSICS + EXAMINATION + RADIATION EFFECT + PIPES AND PIPE FITTINGS

9-038058

DATED 1968

CRAIG ER
SEFOR INSTRUMENTED FUEL ASSEMBLY DESIGN AND DEVELOPMENT
GENERAL ELECTRIC
GEAP- 5615 +. 61 PAGES, APRIL 1968

THIS REPORT DESCRIBES PREOPERATIONAL R AND D UNDER TASK 4.5, OTHER CORE INSTRUMENTATION. IT
COVERS DESIGN OBJECTIVE, EQUIPMENT DESIGN (THERMOCOUPLES, FLOWMETERS, INSTRUMENT LEAD CONNECTOR,
AND FUEL HANDLING FIXTURES), AND TEST AND DEVELOPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + INSTRUMENT, IN CORE + REACTOR, LMCR + INSTRUMENT, COMPONENT + TEST, INSTRUMENT RESPONSE +
REACTOR, PULSED + FUEL HANDLING MACHINE + SEFOR (RE) + R AND D PROGRAM + FUEL ELEMENTS

9-038092

DATED 1968

DUNCION PJ + CUNDILL B
COMPONENTS FOR NUCLEAR POWER STATIONS. PART A. REACTOR COMPONENTS. PART B. COMPONENTS FOR SODIUM COOLED
REACTORS
FAIREY ENGINEERING LTD., HESTON (ENGLAND) + ENGLISH ELECTRIC COMPANY, LTD., WHETSTONE (ENGLAND)
CONF-681064-2 +. 9 PAGES, OCTOBER 15, 1968, FROM BRITISH NUCLEAR FORUM AND DEUTSCHES ATOMFORUM MEETING,
FRANKFURT AM MAIN, GERMANY

PART A CONFINES ITS ATTENTION TO COMPONENTS WHICH HAVE BEEN DEVELOPED AND SUPPLIED FOR USE IN A
WIDE RANGE OF BRITISH GAS-COOLED REACTORS AND IN THE SCHWR. THE DESIGN AND MANUFACTURE OF
REACTOR COMPONENTS IS A COMMERCIAL DANGEROUS OPERATION UNLESS IT IS BACKED BY ADEQUATE
EXPERIENCE AND TECHNOLOGICAL KNOW-HOW. PART B IS INTENDED TO REVIEW FOUR MAIN GROUPS OF SODIUM
COMPONENTS, AND NAMELY, PUMPS, VALVES, SODIUM TO STEAM GENERATORS, HEAT EXCHANGERS AND SODIUM
INSTRUMENTATION. WITHIN THIS FRAMEWORK SPECIAL REFERENCES WILL BE MADE TO DEVELOPMENT ASPECTS,
THE FLOW OF INFORMATION TO INDUSTRY AND COMPONENT TESTING.

AVAILABILITY - USAEC DEPOSITORY LIBRARIES IN THE U.S. AND OVERSEAS

DESIGN CRITERIA + REACTOR, GCR + REACTOR, LMCR + UNITED KINGDOM + RELIABILITY, COMPONENT + COMPONENTS + STEAM
GENERATOR + PUMPS + VALVES + INSTRUMENTS, MISC. + HEAT EXCHANGERS

9-038093

DATED 1969

FAM SS + LYNNWORTHLC + CARNEVALE EH
ULTRASONIC THERMOMETRY IN LMFBR SYSTEMS
PANAMETRICS, WALTHAM, MASSACHUSETTS
NYO-3906-7 +. 25 PAGES, JUNE 1969

THE PROGRESS ACHIEVED DURING THE SEVENTH QUARTER, APRIL-JUNE 1969, MAY BE SUMMARIZED AS FOLLOWS -
(1) ATTEMPTS WERE MADE TO USE A COMPOSITE WIRE THERMOCOUPLE-ULTRASONIC PROBE EVEN THOUGH THE TIE-
ON LINES WERE NOT AT THE OPTIMUM POINTS FOR ULTRASONIC MEASUREMENTS. SIGNAL OVERLAP AND
INTERFERENCE WERE AGAIN OBSERVED AT ABOUT 3500 F. (2) ALUMINUM CELLS, ANODIZED AT PANAMETRICS,
WERE TESTED AT 500 F FOR 1030 HOURS, 300 F FOR 780 HOURS, 900 F FOR 100 HOURS, AND AT 1000 F FOR
230 HOURS. COIL FAILURES WERE OBSERVED AT 800 F AND 1000 F, POSSIBLY DUE TO GRADUAL DIFFUSION OF
THE PARENT METAL THROUGH AN ANODIC LAYER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

ORR (RR) + TUNGSTEN + INSTRUMENT, TEMPERATURE + RHENIUM + R AND D PROGRAM + REACTOR, LMFBR + ULTRASONICS +
THERMOCOUPLE + FUEL ELEMENTS

CATEGORY 9
NUCLEAR INSTRUMENTATION, CONTROL, AND SAFETY SYSTEMS

9-038580

DATED 1967

ASAMOTO RR + NOVAK PE

TUNGSTEN-RHENIUM THERMOCOUPLES FOR USE AT HIGH TEMPERATURE

GE-ADVANCED PRODUCTS OPERATION

6 PAGES, 5 FIGURES, 24 REFERENCES, REVIEW OF SCIENTIFIC INSTRUMENTS, VOL. 38, PAGES 1047-1052, AUGUST 1967

TUNGSTEN-RHENIUM THERMOCOUPLE SYSTEMS WERE EVALUATED FOR USE IN MEASURING TEMPERATURES BETWEEN 1600 AND 3000 C. TEMPERATURE-MILLIVOLT RELATIONSHIPS WERE EXTENDED FROM 2300 TO 3000 C FOR BARE-WIRE W3RE/W25RE AND W5RE/W26RE THERMOCOUPLES IN VACUUM. THE PERFORMANCE OF W26RE SHEATHED, HIGH-FIRED BERYLLIA-INSULATED THERMOCOUPLES WAS LIMITED ONLY BY THE MELTING POINT OF THE BERYLLIA INSULATION (2550 C). THE THERMOELECTRIC OUTPUT OF HIGH-FIRED THORIA-INSULATED THERMOCOUPLES SHEATHED IN W26RE WAS DETERMINED AND FOUND TO BE RELIABLE UP TO AT LEAST 2850 C, WHEN CORRECTED FOR ELECTRICAL SHUNTING THROUGH THE INSULATION. STABILITY OF HIGH TEMPERATURE THERMOCOUPLES USING THORIA INSULATION WAS WITHIN ± 0.3 MV (± 40 C) WHEN HELD AT 2425 C FOR 148 HR.

REACTOR, FAST + REACTOR, LMCR + HIGH TEMPERATURE + TUNGSTEN + INSTRUMENT, TEMPERATURE + REACTOR, PULSED + SEFOR (RE) + RHENIUM + THERMOCOUPLE + FUEL ELEMENTS

CATEGORY 10
ELECTRICAL POWER SYSTEMS

10-004693 DATED 1960
VINCENS E
ELECTRIC POWER SYSTEMS OF NUCLEAR POWER PLANTS
OAK RIDGE NATIONAL LABORATORY
11 PAGES- NUCLEAR SAFETY, 2(1)- 32-42 (SEPTEMBER 1960)- GPO

THE IMPORTANCE OF POWER PRODUCTION AND CONTINUED FUNCTIONING OF ESSENTIAL ELECTRICAL EQUIPMENT ESTABLISH A REQUIREMENT FOR RELIABILITY SELDOM MATCHED IN INDUSTRY. DEPENDABILITY OF EMERGENCY POWER IS A COMMON REQUIREMENT OF ALL TYPES OF NUCLEAR POWER PLANTS. DESCRIPTIONS ARE GIVEN OF THE ELECTRIC POWER SYSTEMS OF SEVENTEEN U. S. POWER REACTORS.

BIG ROCK POINT (BWR) + REACTOR, BWR + BONUS (ISR) + REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + CONTAINMENT, MULTIPLE + CVTR (PWR) + DRESDEN 1 (BWR) + EGCR (AGR) + ELK RIVER (BWR) + EMERGENCY POWER, ELECTRIC + REACTOR, FAST + FERMI (LMFBR) + REACTOR, GCR + REACTOR, HWR + HUMBOLDT BAY (BWR) + INDIAN POINT 1 (PWR) + REACTOR, LMCR + NS SAVANNAH (PWR) + PATHFINDER (ISR) + PEACH BOTTOM 1 (HTGR) + PIQUA (OCR) + REACTOR, PWR + SHIPPINGPORT (PWR) + YANKEE ROWE (PWR) + ELECTRIC POWER

10-004694 DATED 1959
VINCENS E
AUXILIARY AND EMERGENCY ELECTRIC POWER SYSTEMS
OAK RIDGE NATIONAL LABORATORY
3 PAGES- NUCLEAR SAFETY, 1(2)- 33-35 (DECEMBER 1959)- GPO

THE REVIEW ARTICLE GIVES CRITERIA FOR EVALUATING AUXILIARY AND EMERGENCY ELECTRIC POWER SYSTEMS FOR NUCLEAR POWER PLANTS. BRIEF DISCUSSIONS OF THESE SYSTEMS ARE GIVEN FOR SEVERAL U. S. REACTORS.

REACTOR, BWR + REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, MULTIPLE + DRESDEN 1 (BWR) + EMERGENCY POWER, ELECTRIC + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + REACTOR, MARITIME + NS SAVANNAH (PWR) + REACTOR, PWR + SHIPPINGPORT (PWR) + YANKEE ROWE (PWR) + ELECTRIC POWER

10-005262 ALSO IN CATEGORIES 12 AND 17 DATED 1965
LIQUID-METAL-COOLED REACTORS - EBR-II - OPERATIONS
ARGONNE NATIONAL LABORATORY
2 PAGES- ANL-7017 (REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT), PAGES 1 AND 2 OF CHAPTER I- MARCH 18, 1965- CFSTI

DURING A SITE-WIDE POWER FAILURE, THE 400-KW DIESEL GENERATOR STARTED, LOADED NORMALLY, AND THEN STOPPED. THIS FAILURE OF THE DIESEL WAS ACCOMPANIED BY A LOSS OF ANNUNCIATION IN THE CONTROL ROOM. INVESTIGATION ALSO REVEALED THAT THE DIESEL CANNOT BE RESTARTED AFTER A FAILURE OF THIS KIND UNTIL THE DIESEL-CONTROL SWITCH IN THE CONTROL ROOM HAS BEEN PLACED IN OFF MOMENTARILY AND BACK TO STANDBY. LACK OF THIS INFORMATION PREVENTED MANUAL RESTARTING OF THE DIESEL DURING THE POWER OUTAGE, CAUSES AND REMEDIAL ACTION.

CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + EMERGENCY POWER, ELECTRIC + FAILURE + REACTOR, LMCR

10-018293 DATED 1960
ELECTRIC POWER SYSTEMS OF NUCLEAR POWER PLANTS
12 PAGES, REFERENCES, NUCLEAR SAFETY 2(1), PAGES 32-43 (SEPTEMBER 1960)

THIS ARTICLE DESCRIBES THE ELECTRIC POWER SYSTEMS OF PRESENTLY OPERATING AND CURRENTLY PLANNED NUCLEAR POWER PLANTS, AND IT ATTEMPTS TO EVALUATE THEIR RELIABILITY. SOME DESIGNS REPRESENT AN ADAPTATION FROM CONVENTIONAL PLANT METHODS, AND OTHER DESIGNS INDICATE AN EARNEST ENDEAVOR TO INTEGRATE THE PECULIARITIES OF A REACTOR PLANT INTO A SYSTEM THAT WILL PROVIDE ACCEPTABLE RELIABILITY. NUCLEAR POWER PLANTS COVERED ARE - SHIPPINGPORT, INDIAN POINT, N.S. SAVANNAH, YANKEE, DRESDEN, ELK RIVER, EGCR, FERMI, PATHFINDER, HUMBOLDT BAY, BIG ROCK POINT, PEACH BOTTOM, CVTR, PIQUA, BONUS, AND FUCNG.

BIG ROCK POINT (BWR) + BONUS (ISR) + CVTR (PWR) + DRESDEN 1 (BWR) + EGCR (AGR) + ELK RIVER (BWR) + FERMI (LMFBR) + REACTOR, GCR + HUMBOLDT BAY (BWR) + INDIAN POINT 1 (PWR) + REACTOR, LMCR + REACTOR, MARITIME + NS SAVANNAH (PWR) + REACTOR, ORGANIC COOLED + PATHFINDER (ISR) + PEACH BOTTOM 1 (HTGR) + PIQUA (OCR) + REACTOR, PRESSURE TUBE + SHIPPINGPORT (PWR) + YANKEE ROWE (PWR) + ELECTRIC POWER + FUCNG

10-020579 ALSO IN CATEGORY 17 DATED 1966
MCCARTHY JF
SHORT ACROSS BOTH 480V FEEDERS AT FERMI
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + POWER REACTOR DEVELOPMENT COMPANY + DETROIT EDISON COMPANY
APDA-CFE-1 +. 1 PAGE, PAGE 19 OF COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT, AUGUST 1966

(PG 19) THE JUNE 29 POWER OUTAGE FAILED BOTH 480-VOLT FEEDS. ONE PHASE OF ONE PRIMARY SIDE OF THE UNGROUNDED DELTA-WOUND 4800/480-V TRANSFORMER HAD SHOWN PREVIOUS SPORADIC GROUNDING. THIS GROUNDS THE AFFECTED PHASE WITHOUT CURRENT FLOW AND RAISE THE OTHER PHASES FROM 2770 TO 4800 V. ON JUNE 29, THE RAISED VOLTAGE PROBABLY TRIGGERED A WEAK SPOT IN THE OTHER FEED, RESULTING IN A PHASE-TO-PHASE SHORT THROUGH GROUND, OPENING THE 4800-V CIRCUIT BREAKERS AND DESTROYING THE CABLES. DIESEL CARRIED THE DIESEL BUS LOADS.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

CATEGORY 10
ELECTRICAL POWER SYSTEMS

10-020579 *CONTINUED*
EMERGENCY POWER, ELECTRIC + FERMI (LMFBR) + ACCIDENT, LOSS OF POWER + FAILURE, SEQUENTIAL + FAILURE, SIMULTANEOUS + REPORT, OPERATIONS SUMMARY + REACTOR, LMFBR

10-026513 ALSO IN CATEGORIES 9 AND 11 DATED 1967
VERBER F
FURTHER DEVELOPMENTS IN CONTAINMENT-BUILDING ELECTRICAL PENETRATIONS
8 PAGES, 13 FIGURES, 1 TABLE, 6 REFERENCES, REACTOR AND FUEL-PROCESSING TECHNOLOGY 11(1), PAGES 5-12 (WINTER 1967-1968)

REVIEWS SOME ELECTRICAL PENETRATION PROBLEMS AND SOLUTIONS EXPERIENCED DURING PLANT OPERATIONS AT FERMI AND AT PEACH BOTTOM. DESCRIBES FIRE-PROTECTION MODIFICATIONS TO THE PENETRATIONS AND TO THE ELECTRICAL DESIGN AT PEACH BOTTOM AFTER A FIRE. ALSO PRESENTS RECENT DEVELOPMENTS IN THE DESIGN OF THE DRESDEN-2 ELECTRICAL PENETRATION SYSTEM AND ASSEMBLIES.

REACTOR, BWR + DESIGN CRITERIA + FERMI (LMFBR) + FIRE + PEACH BOTTOM 1 (HTGR) + DRESDEN 2 (BWR) + INSTRUMENT, COMPONENT + EQUIPMENT DESIGN + REACTOR, HTGR + ELECTRICAL CONDUCTION + CONTAINMENT PENETRATION, ELECTRICAL + OPERATING EXPERIENCE + REACTOR, LMFBR

10-028796 ALSO IN CATEGORY 17 DATED 1967
COSTELLO RH
COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT, NOVEMBER 1967
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH. POWER REACTOR DEVELOPMENT CO., DETROIT, MICH.
DETROIT EDISON CO., MICH.
APDA-CFE-16 +. 45 PAGES, 25 FIGURES, 2 TABLES, JANUARY 1968

MORE DESCRIPTION OF TOOLS FOR CUTTING UP AND REMOVING UNIDENTIFIED FOREIGN OBJECTS FROM THE REACTOR VESSEL. OVERCROWDED CABLE TRAYS IN THE HIGH AMBIENT TEMPERATURE NEAR SODIUM PIPING CAUSED HEATER FEEDER WIRES INSULATION TO CRACK. CABLES WERE REROUTED AND REPLACED WITH ASBESTOS INSULATED CABLES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, 22151, \$3.00 COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + REPORT, OPERATIONS + REMOTE MANIPULATING AND VIEWING + DAMAGE + ELECTRICAL CONDUCTION + REACTOR, LMFBR

10-028896 DATED 1968
DUNSMORE CL
INTEGRATED EMERGENCY POWER SUPPLY FOR NUCLEAR PLANTS
4 PAGES, 1 FIGURE, 1 TABLE, 7 REFERENCES, POWER ENGINEERING, 72(8), PAGE 39-42, (AUG. 1968)

LOCAL GENERATORS FOR EMERGENCY ELECTRIC POWER ARE DISCUSSED FOR A SODIUM-COOLED REACTOR. THESE GENERATORS, OPERATING ON-LINE, WOULD PROVIDE A HIGHLY RELIABLE SYSTEM. UNIT SIZE AND SOME ECONOMIC ASPECTS ARE DISCUSSED.

ECONOMICS + EMERGENCY POWER, ELECTRIC + REACTOR, LMCOR + ELECTRIC POWER + DESIGN STUDY

10-028901 DATED 1968
MONTEITH OB
PRELIMINARY RELIABILITY PREDICTION FOR THE FFTF CONCEPTUAL BUILDING ELECTRICAL POWER SYSTEM, SYSTEM 12
BATTELLE-NORTHWEST, RICHLAND, WASH. PACIFIC NORTHWEST LAB.
BNWL-813 +. 55 PAGES, 14 FIGURES, 1 TABLE, 8 REFERENCES, AUGUST 1968

GIVES A PRELIMINARY RELIABILITY ANALYSIS OF THE FFTF BUILDING ELECTRICAL POWER SYSTEM. RESULTS OF THE ANALYSIS ARE PRELIMINARY BECAUSE THEY ARE BASED ON A CONCEPTUAL DESIGN OF THE SYSTEM AND ON OTHER SYSTEMS OF THE FFTF WHICH INFLUENCE THE DESIGN OF THE POWER SYSTEM BUT ARE NOT YET FULLY DEFINED. THIS RELIABILITY ANALYSIS PREDICTS THE AVAILABILITY, MEAN TIME BETWEEN FAILURES, AND THE MEAN TIME TO RESTORE ELECTRICAL POWER TO THE VARIOUS BUSES THAT MAKE UP THE FFTF BUILDING ELECTRICAL POWER SYSTEM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCOR + RELIABILITY ANALYSIS + REACTOR, TEST + FFTF (TR) + ELECTRIC POWER, NORMAL + ELECTRIC POWER, ALTERNATE

10-029439 ALSO IN CATEGORIES 2 AND 18 DATED 1968
RESPONSE TO QUESTION A-8 - BATTERIES AND RACKS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE A-30 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ARE THE BATTERIES AND BATTERY RACKS LOCATED UNDER THE CONTROL ROOM CLASS 1 ITEMS. DISCUSS THE CRITERIA AND METHODS OF ANALYSIS USED IN THE DESIGN OF THE BATTERIES, AND THE RACKS HOLDING THE BATTERIES, TO ENSURE THAT THEY WILL FUNCTION PROPERLY UNDER THE MAXIMUM EARTHQUAKE LOADING.
(ANSWER) BATTERIES AND RACKS ARE CLASS 1 ITEMS, CERTIFIED EARTHQUAKE RESISTANT BY MANUFACTURER.

CATEGORY 10
ELECTRICAL POWER SYSTEMS

10-029439 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + EMERGENCY POWER, ELECTRIC + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + BATTERIES & CHARGERS

10-029478 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION G-7 - EFFECT OF SODIUM VAPOR ON ELECTRICAL SYSTEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE G-19 AND -20 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231 MAY 24, 1968

DISCUSS ADVERSE EFFECTS, IF ANY, OF SODIUM VAPOR ON ELECTRICAL EQUIPMENT SUCH AS CABLES, INSULATORS, CONNECTORS, AND TERMINAL BOARDS. (ANSWER) SEE SECT. 3.4 OF SUPPLEMENT 12 TO FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SODIUM + VAPOR PRESSURE + ENVIRONMENT + SEFOR (RE) + RESPONSE TO NRC QUESTION + DAMAGE + ELECTRICAL CONDUCTION

10-030092 DATED 1968
MONTEITH DB
PRELIMINARY RELIABILITY ANALYSIS OF THE PRIMARY ELECTRICAL POWER SYSTEM, SYSTEM 11
PACIFIC NORTHWEST LABORATORY, RICHLAND, WASHINGTON
BNWL-691 +. 2 PAGES, FEBRUARY 1968

REPORTS A PRELIMINARY RELIABILITY PREDICTION FOR THE FFTF PRIMARY ELECTRICAL POWER SYSTEM. THE EFFECT OF LIGHTNING (OR UNKNOWN) SURGES ON SYSTEM RELIABILITY IS INCLUDED, AND THE SYSTEM AVAILABILITY IS DETERMINED. A CALCULATION OF EXPECTED UNAVAILABILITY OF THE FFTF CAUSED BY SYSTEM OUTAGES IS PRESENTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + RELIABILITY ANALYSIS + REACTOR, TEST + FFTF (TR) + ELECTRIC POWER, AUXILIARY + ELECTRIC POWER, NORMAL

10-030255 ALSO IN CATEGORIES 5 AND 12 AND 18 DATED 1968
RESPONSE TO QUESTION 4 - EMERGENCY COOLING
NORTHERN STATES POWER COMPANY
2 PAGES, PP. 21-22 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

EMERGENCY COOLING IS THAT FOLLOWING LOSS OF SECONDARY HEAT SINKS AND TOTAL LOSS OF SITE POWER. ***REFERS TO SUPPLEMENT 19, QUESTION 2, WHERE SAME QUESTION ANSWERED. A 50-KW AUXILIARY POWER UNIT HAS BEEN PROVIDED FOR AUX. PC PUMP, AND ALSO POWERS A GAS BLOWER TO COOL THIS PUMPS WINDINGS. LATTER GAS BLOWER SYSTEM IS DESCRIBED HERE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (108 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

EMERGENCY POWER, ELECTRIC + ACCIDENT, LOSS OF POWER + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + AUXILIARY COOLING SYSTEM + REACTOR, LMCR + PUMPS

10-030767 DATED 1968
DUNSMOKE CL
INTEGRATED EMERGENCY POWER
2 PAGES, POWER ENGINEERING 72(5), PAGES 48-50 (MAY 1968)

GENERAL CRITERIA ARE SPECIFIED AND PROPOSED AS STANDARD FOR ALL NUCLEAR PLANTS. SOME OF THESE RULES ARE DISCUSSED, ALONG WITH DESIGN AND OPERATIONAL DETAILS. THE USE OF SMALL TURBOGENERATORS (SEPARATE FROM THE MAIN MACHINE) IS SHOWN TO BE AN EXCELLENT WAY OF MEETING BOTH NORMAL AND EMERGENCY AUXILIARY POWER NEEDS. TYPICAL DESIGNS OF AUXILIARY AND EMERGENCY SYSTEMS ARE DISCUSSED, AND A DETAILED DESIGN OF STATION AUXILIARIES FOR A SODIUM-COOLED FAST BREEDER WITH THREE HEAT TRANSFER CIRCUITS IS CONSIDERED. THOUGH THE DETAILED DESIGN WAS MADE FOR A SODIUM PLANT, IT MAY BE ADAPTED TO OTHER EMERGENCY POWER PROBLEMS.

EMERGENCY POWER, ELECTRIC + REACTOR, LMCR + ELECTRIC POWER + DESIGN STUDY + ELECTRIC POWER, AUXILIARY + ELECTRIC POWER, NORMAL + REACTOR + REVIEW + ENGINEERED SAFETY FEATURE + TURBINE + REACTOR, LMCR + GENERATOR, DIESEL

10-032808 ALSO IN CATEGORY 18 DATED 1968
ELECTRIC POWER SYSTEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

CATEGORY 10
ELECTRICAL POWER SYSTEMS

10-032808 *CONTINUED*

50231-12 +. 60 PAGES, 4 TABLES, SEFOR, PSAR, SUPPLEMENT 12, MARCH 11, 1968, DOCKET 50-231

TOPICS DESCRIBED AND DISCUSSED INCLUDE REQUIREMENTS OF NORMAL AND EMERGENCY ELECTRIC SYSTEMS, GROUNDING AND CATHODIC PROTECTION, COMMUNICATION SYSTEM, DISTRIBUTION, DC SYSTEM, AND PUMP-POWER SUPPLIES.

CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCK + ELECTRICAL POWER + REACTOR, PULSED + SEFOR (RE) + SYSTEM DESCRIPTION + REPORT, PSAR + REACTOR, LMFBR

10-032917 ALSO IN CATEGORIES 2 AND 18 DATED 1968

RESPONSE TO QUESTION 2 - LOSS OF SITE POWER - TORNADO

GENERAL ELECTRIC, SAN JOSE

50231-21 +. 5 PAGES, 7 FIGURES, PAGES 6 THRU 11 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

WHAT PROVISIONS HAVE BEEN MADE FOR LOSS OF SITE POWER DUE TO A TORNADO PASSING THROUGH OR NEAR THE REACTOR SITE. *** ANSWER ON PAGES 6 THRU 11.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCK + ACCIDENT, LOSS OF POWER + ELECTRIC POWER, NORMAL + REACTOR, PULSED + SEFOR (RE) + SAFE FAILURE CRITERION + RESPONSE TO NRC QUESTION + DESTRUCTIVE WIND + REPORT, PSAR + REACTOR, LMFBR

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-000749
MCGUIRE W + FISHER GP
CONTAINMENT STUDIES OF THE ENRICO FERMI ATOMIC POWER PLANT
POWER REACTOR DEVELOPMENT COMPANY, INC., ITHACA, N. Y.
126 PAGES, 14 FIGURES, 1 TABLE, MAY 1959, SOURCE

DATED 1959

EFFECTS ON CONTAINMENT OF CERTAIN HYPOTHETICAL ACCIDENTS WERE MADE. THE STRUCTURAL LOADS CAUSED BY THESE ACCIDENTS AND THE RESPONSE OF THE STRUCTURE WERE DETERMINED BY THE AUTHORS. THE BEST AVAILABLE PUBLISHED INFORMATION WAS USED TO ESTIMATE THE PROPERTIES OF THE STATED ACCIDENTS. IN SEVERAL INSTANCES, RESULTS OF OTHER INVESTIGATORS WERE ACCEPTED ON FAITH, E.G., U. S. NAVAL GRUNANCE LABORATORY ESTIMATE OF THE SIZE AND VELOCITY OF THE MOST CRITICAL INTERNAL MISSILE AND ARMOUK RESEARCH FOUNDATION ESTIMATE OF THE ENERGY AVAILABLE TO HEAT THE AIR FOLLOWING AN EXPLOSION. THE POSTULATED ACCIDENTS TREATED ARE - 300- AND 1000-LB EQUIVALENT EXPLOSION, SODIUM FIRE, EXTERNAL MISSILES, AND REQUIREMENTS FOR A MODEL TEST PROGRAM. THEY FIND NO MEANS BY WHICH THE STIPULATED ACCIDENTS COULD BREACH CONTAINMENT.

REACTOR, BREKEDER + ACCIDENT, CONSEQUENCES + EXPLOSION + REACTOR, FAST + FERMI (LMFBR) + FIRE + REACTOR, LMCR + MISSILE GENERATION AND PROTECTION + CONTAINMENT INTEGRITY

11-002801 ALSO IN CATEGORY 2 DATED 1965
COTTKELL WB + SAVOLAINEN AW
U. S. REACTOR CONTAINMENT EXPERIENCE. A HANDBOOK OF CURRENT PRACTICE AND ANALYSIS, DESIGN, CONSTRUCTION, TEST, AND OPERATION. CHAPTER 1. - INTRODUCTION
OAK RIDGE NATIONAL LABORATORY
97 PAGES, 12 TABLES, 16 FIGURES, 41 REFERENCES, AUGUST 1965, ORNL-NSIC-5, VOL. 1, CFSTI

THIS CHAPTER IS THE INTRODUCTION TO A COMPREHENSIVE REVIEW OF U.S. REACTOR CONTAINMENT EXPERIENCE. 1-1 PHILOSOPHY - 1-1.1 - DEFINITIONS - 1-1.2 - NEED FOR CONTAINMENT - 1-1.3 - SAFETY FACTORS, 1-2 RAMIFICATIONS OF REACTOR CONTAINMENT - 1-2.1 - REACTOR SITE CRITERIA - 1-2.2 - NUCLEAR MERCHANT SHIP SITE CRITERIA - 1-2.3 - RELIABILITY OF REACTOR SYSTEMS - 1-2.4 - NUCLEAR SAFETY ECONOMICS - 1-2.5 - LIABILITY AND INDEMNITY, 1-3 GENERAL DESCRIPTION OF REACTOR TYPES, 1-4 - GENERAL DESCRIPTION OF CONTAINMENT TYPES - 1-4.1 - PRESSURE CONTAINMENT - 1-4.2 - PRESSURE-SUPPRESSION CONTAINMENT - 1-4.3 - PRESSURE-RELEASE CONTAINMENT - 1-4.4 - MULTIPLE CONTAINMENT - 1-4.5 - STATE OF THE ART, 1-5 GUIDE TO THE USE OF HANDBOOK - 1-5.1 PURPOSE - 1-5.2 LIMITATIONS AND USE.

REACTOR, BWR + CODES AND STANDARDS + CONTAINMENT DESIGN + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + CONTAINMENT, MULTIPLE + CONTAINMENT, PRESSURE RELIEF + CONTAINMENT, PRESSURE SUPPRESSION + CONTAINMENT, PRESSURE VENTING + DOSE + ECONOMICS + REACTOR, FAST + REACTOR, GCR + REACTOR, HWR + INSURANCE + REACTOR, LMCR + REACTOR, MARITIME + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR, MOLTEN SALT + CONTAINMENT SYSTEM, OPERATION + REACTOR, ORGANIC COOLED + REACTOR, PWR + RELIABILITY ANALYSIS + REACTOR, RESEARCH + REACTOR SAFETY SYSTEM + SITING + SITING, REACTOR + REACTOR, TRAINING + ACCIDENT, PROBABILITY OF + CONTAINMENT STRUCTURE + CONTAINMENT LEAKAGE CONTROL + ENGINEERED SAFETY FEATURE

11-002803 DATED 1965
PIPER HB
U. S. REACTOR CONTAINMENT EXPERIENCE. A HANDBOOK OF CURRENT PRACTICE AND ANALYSIS, DESIGN, CONSTRUCTION, TEST, AND OPERATION. CHAPTER 3. - CREDIBLE ACCIDENTS
OAK RIDGE NATIONAL LABORATORY
43 PAGES, 8 TABLES, 10 FIGURES, 47 REFERENCES, AUGUST 1965, ORNL-NSIC-5, VOL. 1, CFSTI

A DESCRIPTION OF VARIOUS CREDIBLE ACCIDENTS WITH PARTICULAR EMPHASIS ON THE MAXIMUM CREDIBLE ACCIDENT (MCA) IS PRESENTED. A COMPILATION OF REACTOR ACCIDENT EXPERIENCE IS INCLUDED. 3-1 - INTRODUCTION, 3-2 TYPICAL ACCIDENTS, - 3-2.1 - THE STARTUP ACCIDENT - 3-2.2 - THE CONTROL ROD WITHDRAWAL ACCIDENT, 3-2.3 - THE POISON REMOVAL ACCIDENT, 3-2.4 - THE COLD COOLANT ACCIDENT, 3-2.5 - CONTROL ROD DROP ACCIDENT, 3-2.6 - LOSS OF COOLANT FLOW, 3-2.7 - LOSS OF LOAD ACCIDENT, 3-2.8 - STEP INCREASE IN LOAD ACCIDENT, 3-2.9 - RELEASE OF PRIMARY FLUID AND INTERCHANGE OF PRIMARY AND SECONDARY FLUIDS, 3-2.10 - LOSS OF COOLANT ACCIDENT, 3-2.11 - FUEL HANDLING ACCIDENTS, 3-2.12 - FUEL ELEMENT FAILURES, 3-2.13 - GENERAL, 3-3 MAXIMUM CREDIBLE ACCIDENT, 3-4 COMPONENT, 3-5 PROBABILITIES, 3-6 REACTOR ACCIDENT EXPERIENCE.

ACCIDENT ANALYSIS + REACTOR, BWR + ACCIDENT, COLD COOLANT + FAILURE, COMPONENT + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + CONTAINMENT, PRESSURE VENTING + FAILURE, FUEL ELEMENT + FUEL HANDLING + FUEL MELTDOWN + REACTOR, GCR + HALLAM (LMR) + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + ACCIDENT, LOSS OF POWER + ACCIDENT, LOSS OF PRESSURE + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR, ORGANIC COOLED + FAILURE, PIPE + PIQUA (OCR) + FAILURE, PRESSURE VESSEL + REACTOR, PWR + ACCIDENT, REACTIVITY + REACTOR, RESEARCH + ACCIDENT, CONTROL ROD WITHDRAWAL + REACTOR, INTERNAL SUPERHEAT + REACTOR, TEST + YANKEE ROWE (PWR) + SL 1 (BWR) + HANFORD (PR) + ACCIDENT, PROBABILITY OF + CORE MELTDOWN + REACTOR, MILITARY + REACTOR, LMFBR

11-002808 DATED 1965
APPLEFORD AM + DYER GH + GRIFFIN RF + HAWTHORNE JR + PIPER HB + STEELE LE
U. S. REACTOR CONTAINMENT EXPERIENCE- A HANDBOOK OF CURRENT PRACTICE AND ANALYSIS, DESIGN, CONSTRUCTION, TEST, AND OPERATION. CHAPTER 8.- STRUCTURAL DESIGN CONSIDERATIONS
OAK RIDGE NATIONAL LABORATORY + BECHTEL CORPORATION + U. S. NAVAL RESEARCH LABORATORY
92 PAGES, 15 TABLES, 48 FIGURES, 88 REFERENCES- AUGUST 1965, ORNL-NSIC-5, VOL. 11- CFSTI

THIS CHAPTER DISCUSSES SOME OF THE FACTORS CONSIDERED IN DETERMINING THE OPTIMUM CONTAINMENT TYPE, SIZE, SHAPE, AND MATERIAL TO MEET THE BASIC CONTAINMENT REQUIREMENTS AND ATTEMPTS TO INDICATE WHY SOME CONTAINMENT DESIGNS ARE SUITABLE FOR SOME SITUATIONS AND NOT FOR OTHERS. ALTERNATE

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-002808 *CONTINUED*

TECHNIQUES OF CONTAINMENT CONSTRUCTION ARE ALSO DISCUSSED AND THE MOST IMPORTANT STEPS OF EACH ARE OUTLINED. THROUGHOUT THIS CHAPTER, THE DETAILS AND THE PROBLEM AREAS UNIQUE OR MOST SIGNIFICANT TO THOSE AREAS WHICH, ALTHOUGH THEY MAY BE OF MAJOR IMPORTANCE, ARE COMMON TO THE ENGINEERING AND CONSTRUCTION CONVENTIONAL STRUCTURES.

- 9-1 INTRODUCTION
- 9-2 GENERAL DESIGN CONSIDERATIONS
- 9-3 CHOICE OF MATERIALS
- 9-4 SPECIAL DESIGN FEATURES
- 9-5 CONSTRUCTION
- 9-6 RADIATION EMBRITTLEMENT OF STEEL
- 9-7 CONTAINMENT PROTECTION
- 9-8 INSPECTION AND CONTROL
- 9-9 REFERENCES

REACTOR, BWR + BONUS (ISR) + REACTOR, BREEDER + CODES AND STANDARDS + CONCRETE + CONTAINMENT DESIGN + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + CONTAINMENT, MULTIPLE + CONTAINMENT, PRESSURE SUPPRESSION + CONTAINMENT, PRESSURE VENTING + CVTR (PWR) + DRESDEN 1 (BWR) + EBR 1 AND 2 (RE) + EBWR (BWR) + EGCR (AGR) + ELK RIVER (BWR) + REACTOR, FAST + FERMI (LMFBR) + REACTOR, GCR + HALLAM (LMR) + REACTOR, HWR + HUMBOLDT BAY (BWR) + HWCTR (TR) + INDIAN POINT 1 (PWR) + REACTOR, LMCR + MISSILE GENERATION AND PROTECTION + PATHFINDER (ISR) + PEACH BOTTOM 1 (HTGR) + REACTOR, PRESSURE TUBE + REACTOR, PWR + SAXTON (PWR) + SEISMOLOGY + SHIPPINGPORT (PWR) + CONTAINMENT, SHOCK GENERATION AND PROTECTION + STEEL + REACTOR, INTERNAL SUPERHEAT + REACTOR, TEST + VBWR (BWR) + YANKEE ROWE (PWR) + CONTAINMENT CONSTRUCTION + CONCRETE, PRESTRESSED + CONTAINMENT INSPECTION AND MAINTENANCE + VESR (ISR) + SRE (RE) + CONTAINMENT STRUCTURE + PENETRATIONS, PRIMARY CONTAINMENT

11-002809

DATED 1965

GRIFFIN RF + DYER GH + WAHL HW + WALKER CS

U. S. REACTOR CONTAINMENT EXPERIENCE- A HANDBOOK OF CURRENT PRACTICE AND ANALYSIS, DESIGN, CONSTRUCTION, TEST, AND OPERATION. CHAPTER 9.-CONTAINMENT ACCESSORIES

BECHTEL CORPORATION + OAK RIDGE NATIONAL LABORATORY

87 PAGES, 4 TABLES, 42 FIGURES, 56 REFERENCES- AUGUST 1965, ORNL-NSIC-5, VOL. II- CFSTI

A COMPLETE CONTAINMENT SYSTEM MUST INCLUDE NUMEROUS ACCESSORIES AS WELL AS THE BASIC CONTAINMENT STRUCTURE IN ORDER TO PROPERLY PERFORM ITS REQUIRED FUNCTIONS. THIS CHAPTER DISCUSSES THE REQUIREMENTS FOR ACCESS, PIPING, FUEL TRANSFER AND ELECTRICAL PENETRATIONS, AND PRESENTS INFORMATION ON CERTAIN SPECIFIC DESIGNS OF EACH OF THESE GENERAL TYPES OF PENETRATIONS. VENTILATION AND PRESSURE CONTROL SYSTEMS AND FISSION PRODUCT REMOVAL SYSTEMS ARE DISCUSSED IN MORE GENERAL TERMS. INSTRUMENTATION AND POWER REQUIREMENTS FOR OPERATION OF THE CONTAINMENT SYSTEM ARE DISCUSSED BRIEFLY.

- 9-1 INTRODUCTION
- 9-2 ACCESS PENETRATIONS
- 9-3 PIPING PENETRATIONS
- 9-4 FUEL TRANSFER PENETRATIONS
- 9-5 ELECTRICAL PENETRATIONS
- 9-6 VENTILATION AND PRESSURE CONTROL SYSTEMS
- 9-7 FISSION PRODUCT REMOVAL AND TRAPPING SYSTEMS

AIR CLEANING + BIG ROCK POINT (BWR) + REACTOR, BREEDER + CANADA + CODES AND STANDARDS + CONTAINMENT DESIGN + CONTAINMENT INSTRUMENTATION + CONTAINMENT SPRAY + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, MULTIPLE + CONTAINMENT, PRESSURE RELIEF + CONTAINMENT, PRESSURE SUPPRESSION + CONTAINMENT, PRESSURE VENTING + DRESDEN 1 (BWR) + EBR 1 AND 2 (RE) + EBWR (BWR) + EGCR (AGR) + EMERGENCY POWER, ELECTRIC + REACTOR, FAST + FERMI (LMFBR) + FOAM + FOG + REACTOR, GCR + HUMBOLDT BAY (BWR) + IODINE + REACTOR, LMCR + REACTOR, MARITIME + ORR (RR) + PEACH BOTTOM 1 (HTGR) + CONTAINMENT ISOLATION + REACTOR, PWR + REACTOR, RESEARCH + SCRUBBER + STEEL + REACTOR, TEST + VENTILATION SYSTEM + YANKEE ROWE (PWR) + ELECTRIC POWER + CONTAINMENT CONSTRUCTION + CHARCOAL + HANFORD (PR) + CONTAINMENT FILTERING SYSTEM + CONTAINMENT STRUCTURE + CONTAINMENT PENETRATION, ELECTRICAL + CONTAINMENT PENETRATION, PIPE + FILTERS + PENETRATIONS, PRIMARY CONTAINMENT

11-002810

DATED 1965

GRIFFIN RF + DYER GH

U. S. REACTOR CONTAINMENT EXPERIENCE- A HANDBOOK OF CURRENT PRACTICE AND ANALYSIS, DESIGN, CONSTRUCTION, TEST, AND OPERATION. CHAPTER 10.-PERFORMANCE TESTS

BECHTEL CORPORATION

116 PAGES, 9 TABLES, 15 FIGURES, 58 REFERENCES- AUGUST 1965, ORNL-NSIC-5, VOL. II- CFSTI

A COMPREHENSIVE REVIEW OF CONTAINMENT PROOF AND LEAK TESTING PRACTICE IN THE U. S.

- 10-1 INTRODUCTION
- 10-2 STRENGTH TESTING
- 10-3 LOCAL LEAK TESTING TECHNIQUES
- 10-4 LEAKAGE RATE TESTING
- 10-5 TYPICAL PROOF TESTING PROCEDURE
- 10-6 RETESTING
- 10-7 MAINTENANCE OF CONTAINMENT
- 10-8 REFERENCES

BIG ROCK POINT (BWR) + REACTOR, BWR + BONUS (ISR) + REACTOR, BREEDER + CODES AND STANDARDS + CONCRETE +

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-002810 *CONTINUED*

CONTAINMENT, UNDERGROUND + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + CONTAINMENT, MULTIPLE + CONTAINMENT, PRESSURE SUPPRESSION + CVTR (PWR) + DRESDEN 1 (BWR) + EBWR (BWR) + ELK RIVER (BWR) + REACTOR, FAST + FERMI (LMFBR) + REACTOR, HWR + HUMBOLDT BAY (BWR) + INDIAN POINT 1 (PWR) + REACTOR, LMCR + REACTOR, MAKITIME + NS SAVANNAH (PWR) + CONTAINMENT SYSTEM, OPERATION + PATHFINDER (ISR) + REACTOR, PRESSURE TUBE + REACTOR, PWR + SAXTON (PWR) + SHIPPINGPORT (PWR) + STEEL + REACTOR, INTERNAL SUPERHEAT + REACTOR, TEST + TEST, LEAK RATE + TEST, PROOF + VBWR (BWR) + YANKEE ROWE (PWR) + CONTAINMENT AIR LOCK + PLUM BROOK (TR) + CONTAINMENT REFERENCE MEASURING SYSTEM + CONTAINMENT INSPECTION AND MAINTENANCE + VESR (ISR) + CONTAINMENT STRUCTURE + TEST, LEAK LOCATION + WTR (TR) + RAVENSWOOD (PWR) + REACTOR, MILITARY + PENETRATIONS, PRIMARY CONTAINMENT

11-002811 ALSO IN CATEGORY 2

DATED 1965

BERNSEN SA + GRIFFIN RF

U. S. REACTOR CONTAINMENT EXPERIENCE- A HANDBOOK OF CURRENT PRACTICE AND ANALYSIS, DESIGN, CONSTRUCTION, TEST, AND OPERATION. CHAPTER 11--ECONOMICS
BECHTEL CORPORATION

57 PAGES, 20 TABLES, 18 FIGURES, 30 REFERENCES- AUGUST 1965, ORNL-NSIC-5, VOL. 11- CFSTI

THIS CHAPTER DISCUSSES THE ECONOMIC FACTORS IN DESIGNING AND EVALUATING CONTAINMENT SYSTEMS AND VARIOUS CONTAINMENT FEATURES. ITS PURPOSE IS TO PROMOTE AN UNDERSTANDING OF THE ASPECTS OF CONTAINMENT WHICH ARE OF GREATER ECONOMIC SIGNIFICANCE AND TO PROVIDE GUIDANCE IN MAKING ECONOMIC COMPARISONS OF ALTERNATE DESIGNS. CONTAINS THE FOLLOWING - (1) INTRODUCTION, (2) ECONOMIC IMPACT OF CONTAINMENT, (3) UNIT COSTS OF CONTAINMENT COMPONENTS, (4) COST EXPERIENCE ON CONSTRUCTED FACILITIES, (5) TYPICAL COST EVALUATIONS, (6) COST ESTIMATING, (7) SUMMARY, AND (8) REFERENCES.

BIG ROCK POINT (BWR) + BUDEGA BAY (BWR) + REACTOR, BWR + BONUS (ISR) + REACTOR, BREEDER + CODES AND STANDARDS + CONCRETE + CONTAINMENT, UNDERGROUND + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + CONTAINMENT, MULTIPLE + CONTAINMENT, PRESSURE RELIEF + CONTAINMENT, PRESSURE SUPPRESSION + CONTAINMENT, PRESSURE VENTING + DRESDEN 1 (BWR) + EBWR (BWR) + ECONOMICS + EGCR (AGR) + ELK RIVER (BWR) + REACTOR, FAST + REACTOR, GCR + REACTOR, HWR + HUMBOLDT BAY (BWR) + HWCTR (TR) + INDIAN POINT 1 (PWR) + REACTOR, LMCR + REACTOR, ORGANIC COOLED + PATHFINDER (ISR) + PEACH BLITZ 1 (HTGR) + PIQUA (OCR) + REACTOR, PWR + SITING + SITING, REACTOR + REACTOR, INTERNAL SUPERHEAT + VBWR (BWR) + VENTILATION SYSTEM + YANKEE ROWE (PWR) + CONTAINMENT AIR LOCK + CONNECTICUT YANKEE (PWR) + CONTAINMENT CONSTRUCTION + SAN ONOFRE 1 (PWR) + STACK + CONTAINMENT FILTERING SYSTEM + CONTAINMENT STRUCTURE + RAVENSWOOD (PWR) + ENGINEERED SAFETY FEATURE + FILTERS

11-006817 ALSO IN CATEGORIES 5 AND 12

DATED 1965

BOULANGER JR

SNAP-10A--PIPING STRESS ANALYSIS

ATOMICS INTERNATIONAL

30 PAGES- JANUARY 22, 1965, NAA-SK-MEMO-9817- CFSTI

THIS REPORT CONTAINS THE STRESS ANALYSIS OF THE SNAP 10A PIPING SYSTEM. THIS SYSTEM IS MADE UP OF VARIOUS SIZES AND SHAPES OF STAINLESS STEEL TUBING WELDED TOGETHER INTO A NAK CONTAINMENT NETWORK. STAINLESS STEEL IS USED BECAUSE OF ITS STABILITY AT THE OPERATING TEMPERATURES AND ITS COMPATIBILITY WITH THE REMAINDER OF THE SYSTEM. BACKGROUND INFORMATION REGARDING THE CONFIGURATION AND MATERIALS USED, THE LOADS AND THEIR SOURCES AND THE ANALYTICAL VERIFICATIONS ARE PROVIDED. THIS ANALYSIS VERIFIES THE STRUCTURAL INTEGRITY OF THE SYSTEM DURING BOTH THE SNAP 10A THERMAL AND VIBRATION ENVIRONMENTS.

INTEGRITY + REACTOR, LMCR + SNAP 10A (SR) + REACTOR, SPACE + STEEL, STAINLESS + STRESS ANALYSIS + MAIN COOLING SYSTEM + PIPES AND PIPE FITTINGS

11-006850

DATED 1965

CONTAINMENT

GENERAL ELECTRIC, ATOMIC POWER EQUIPMENT DEPT.

14 PAGES, 1 FIGURE, 2 TABLES- 1965, DOCKET NO. 50-231 (NP-15077), SECTION III, PP 3-1--3-14 OF PRELIMINARY SAFEGUARDS SUMMARY REPORT, SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR, CFSTI, \$10.10 CY

THE SEFOR REACTOR AND ITS PRIMARY COOLANT SYSTEM COMPONENTS WILL BE PLACED IN CELLS HAVING REINFORCED CONCRETE WALLS. THESE CELLS WILL BE THE FIRST LINE OF DEFENSE IF RADIOACTIVE MATERIAL SHOULD ESCAPE FROM THE PRIMARY COOLANT SYSTEM. THE CELLS WILL BE INTERCONNECTED TO TAKE ADVANTAGE OF MAXIMUM AVAILABLE CONTAINMENT VOLUME. IT HAS BEEN ASSUMED THAT LEAKAGE THROUGH THE FIRST CONTAINMENT BARRIER WOULD BE 20 PERCENT OF ITS CONTAINED VOLUME OVER A 24-HOUR PERIOD AT THE DESIGN PRESSURE OF 10 PSIG, AND LEAKAGE THROUGH THE SECOND CONTAINMENT BARRIER WOULD BE 2-1/2 PERCENT OF ITS CONTAINED VOLUME AT 30 PSIG, ALSO FOR A 24-HOUR PERIOD. THE SECOND CONTAINMENT BARRIER WILL TAKE THE FORM OF A STEEL PRESSURE VESSEL APPROXIMATELY 105.5 FEET IN OVER-ALL HEIGHT WITH A DIAMETER OF 50 FEET.

REACTOR, BREEDER + CONTAINMENT DESIGN + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, MULTIPLE + REACTOR, FAST + REACTOR, LMCR + CONTAINMENT INSULATION + CONTAINMENT AIR LOCK + ARGON + CONTAINMENT CONSTRUCTION + CONTAINMENT EQUIPMENT HATCH + NITROGEN + CONTAINMENT STRUCTURE + SEFOR (RE) + THERMAL INSULATION + CONTAINMENT PENETRATION, ELECTRICAL + CONTAINMENT PENETRATION, PIPE + PENETRATIONS, PRIMARY CONTAINMENT

11-007435 ALSO IN CATEGORIES 6 AND 17

DATED 1965

EBR-II

ARGONNE NATIONAL LABORATORY

7 PAGES, 5 TABLES- AUGUST 24, 1965, ANL-7082, PP 1-7 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT JULY 1965, CFSTI, \$3.00 CY, \$0.75 MN

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-007435 *CONTINUED*

REGARDING EBR-II 10- SUBASSEMBLIES CONTAINING SPECIAL SURVEILLANCE FUEL PINS SWELLED FURTHER THAN NORMAL PINS AND WERE REPLACED. NORMAL PRODUCTION PINS ALSO SWELLED SOMEWHAT. OXIDES OF BISMUTH, TIN AND SODIUM FORMED IN THE SEAL TROUGH AND INTERFERED WITH ROTATION OF THE PLUGS AND THE TEMPERATURE DISTRIBUTION OF THE TROUGHS. PUMP M-1 FROZE TWICE, APPARENTLY BECAUSE OF SODIUM COMBUSTION IN SHAFT. CONCRETE OF PUMP SHIELD PLUGS GIVES OFF WATER. MINOR CONTROL ROD DRIVE TROUBLE. CONTAINMENT TESTING. CONTROL ROD WORTH AND POWER COEFFICIENT.

REACTOR, BREEDER + FAILURE, COMPONENT + CONTROL ROD WORTH + EBR 1 AND 2 (RE) + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR, LMCR + POWER COEFFICIENT + TEST, LEAK RATE + OPERATING EXPERIENCE SUMMARY + PUMPS + CONTROL ROD DRIVE MECHANISMS

11-007436 ALSO IN CATEGORIES 6 AND 9 DATED 1965

FARET

ARGONNE NATIONAL LABORATORY

17 PAGES, 7 FIGURES, 4 TABLES- AUGUST 24, 1965, ANL-7082, PP 7-17 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT JULY 1965, CFST1, \$3.00 CY, \$0.75 MN

FOR FARET, TESTS OF CONTROL ROD DRIVES, LEAK TEST OF VAULT HATCH SEAL, MEASUREMENTS OF CRITICAL CONTROL ROD WORTH. DANGER COEFFICIENTS OF COMPONENTS FROM OTHER ZONES OF SAME ASSEMBLIES.

REACTOR, BREEDER + CONTROL ROD WORTH + REACTOR, FAST + REACTOR, LMCR + TEST, LEAK RATE + FFTF (TR) + DANGER COEFFICIENT + FUEL HANDLING MACHINE + CONTROL ROD DRIVE MECHANISMS

11-007746 ALSO IN CATEGORY 17 DATED 1965

WHITHAM GK + TEBQ FJ + MCGINNIS FD

EBR-II REACTOR BUILDING LEAK TESTING

ARGONNE NATIONAL LABORATORY, IDAHO

2 PAGES, AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JACKSON LAKE LODGE, WYOMING, JULY 28-29, 1965, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 8, PAGES 40-41

A 3-DAY LEAK TEST SHOWED LESS THAN 100 CU FT/DAY AT 36.2 PSIA, AFTER ALL OPERATIONAL DOORS INSTALLED. SINCE THEN, YEARLY TESTS HAVE BEEN ON MAJOR PENETRATIONS AND RANDOMLY CHOSEN ELECTRICAL PENETRATIONS. THEY PROPOSE TO ADOPT 8 OF 24 ELECTRICAL PENETRATIONS FOR TESTING, INSTEAD OF RANDOM TESTING, WHICH DISRUPTS OPERATIONS.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + TEST, LEAK RATE + CONTAINMENT LEAKAGE CONTROL

11-007912 DATED 1965

WHITHAM GK + TEBQ FJ + MCGINNIS FD

EBR-II REACTOR BUILDING LEAK TESTING

ARGONNE NATIONAL LABORATORY, IDAHO FALLS, IDAHO

2 PAGES, TRANS. AM. NUCL. SOC., 8- SUPPL., 40-1 (1965)- AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, WYOMING, JULY 28-29, 1965

FOLLOWING ERECTION OF THE STEEL SHELL OF THE REACTOR CONTAINMENT BUILDING, THE STRUCTURE, INCLUDING FOUR AIR LOCKS, WAS PRESSURIZED FOR THE FIRST TIME AND GIVEN A SOAP-BUBBLE LEAK TEST FOLLOWED BY A 42.2 PSIA PRESSURE TEST AND THEN A THREE-DAY 36.2 PSIA LEAK-RATE TEST. A LARGE CONSTRUCTION OPENING WAS THEN FLAME CUT IN THE BUILDING TO FACILITATE CONSTRUCTION WORK SUCH AS POURING CONCRETE, INTERNAL STEEL ERECTION AND ACCESS FOR THE PRIMARY-TANK COMPONENTS. SEVERAL YEARS LATER THE LARGE OPENING WAS WELDED SHUT AND THE BUILDING WAS PRESSURIZED FOR A SECOND SERIES OF LEAK TESTS. THE THREE-DAY LEAK-RATE TEST DATA SHOWED LESS THAN 100 CU FT PER DAY (36.2 PSIA AND 32 DEGREE F) LEAK RATE.

REACTOR, BREEDER + CONTAINMENT INSTRUMENTATION + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + TEST, LEAK RATE + CONTAINMENT AIR LOCK + CONTAINMENT CONSTRUCTION + CONTAINMENT STRUCTURE + CONTAINMENT ABSOLUTE MEASURING SYSTEM + SEAL + PENETRATIONS, PRIMARY CONTAINMENT

11-009746 ALSO IN CATEGORY 18 DATED 1965

SIMMONS WR + PERSIANI PJ + BUMP TR

FAST REACTOR TEST FACILITY (FARET) AND INITIAL EXPERIMENTAL PROGRAM

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ANS-100(SUPPL.) + CONF. 650417(SUPPL.) + 19 PAGES, 3 FIGURES, PAGES 37-55 OF FAST REACTOR OF TECHNOLOGY, ANS NATIONAL TOPICAL MEETING, DETROIT, MICHIGAN, APRIL 26-28, 1965, SOURCE

THE FAST REACTOR TEST FACILITY SITE, BUILDING, CONTAINMENT STRUCTURE, FUEL-HANDLING SYSTEMS, SHIELDING, COOLANT SYSTEMS, REACTOR PROPER, CONTROL SYSTEMS, EXPERIMENTAL CAPABILITIES, DESIGN SPECIFICATIONS, LOADING, AND FUEL-TESTING PROGRAM AND OBJECTIVES ARE DESCRIBED.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + REACTOR, FAST + REACTOR, LMCR + PRESSURE VESSELS + REACTOR DESCRIPTION + FFTF (TR) + CONTAINMENT STRUCTURE

11-009750 DATED 1965

ROBINSON GC + HORTON TR

LEAK TESTING OF REACTOR CONTAINMENT SYSTEMS

OAK RIDGE NATIONAL LABORATORY + UNIVERSITY OF NORTH CAROLINA

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-009750 *CONTINUED*
10 PAGES, 3 TABLES, REFERENCES, NUCLEAR SAFETY, 7(2), PP. 194-203 (WINTER 1965-1966)

THE ARTICLE COMPILES AND DISCUSSES DATA ON DESIGN, LEAK TESTING, AND LEAK-RATE TESTING OF PRESSURE-TIGHT CONTAINMENT SYSTEMS. EMPHASIS IS PLACED ON COMPARISON OF THE VARIOUS CONTAINMENT DESIGNS AND HISTORY OF LEAK-RATE TESTING. METHODS OF ANALYSIS OF LEAK-RATE TESTING BY THE VARIOUS CONTAINMENT OPERATORS ARE PRESENTED. CURRENT RETESTING SCHEDULES ARE ALSO PRESENTED AND DISCUSSED.

REACTOR, BWR + CONTAINMENT DESIGN + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + CONTAINMENT, PRESSURE SUPPRESSION + CONTAINMENT, PRESSURE VENTING + REACTOR, GCR + REACTOR, GRAPHITE MODERATED + REACTOR, HWR + REACTOR, LMCR + REACTOR, ORGANIC COOLED + REACTOR, PRESSURE TUBE + REACTOR, PWR + REACTOR, INTERNAL SUPERHEAT + REACTOR, TEST + TEST, LEAK RATE + CONTAINMENT REFERENCE MEASURING SYSTEM + CONTAINMENT STRUCTURE + TEST, LEAK LOCATION + CONTAINMENT ABSOLUTE MEASURING SYSTEM + REVIEW + CONTAINMENT LEAKAGE CONTROL + TEST INTERVAL + REACTOR, MILITARY

11-010356 ALSO IN CATEGORIES 12 AND 18 DATED 1965
SAUVAGE M + GUMUCHIAN M
1000-MWE REACTOR DESIGN STUDY OF A HANDLING CELL
COMMISSARIAT A L ENERGIE ATOMIQUE, CADARACHE, FRANCE, CENTRE D ETUDES NUCLEAIRES
ORNL-TR-988 + PNR/SEPR/65/008 +. 41 PAGES, FEBRUARY 18, 1965, JCL, \$4.60 FS, \$1.43 MF

THIS REPORT REPRESENTS A SUPPLEMENTARY CHAPTER TO THE 1000 MW REACTOR DESIGN AS PRESENTED IN GENEVA (CEA REPORT 2554). IT DESCRIBES A DESIGN MAKING USE OF A CELL INSTEAD OF ROTATING PLUGS TO CARRY OUT HANDLING. THE PURPOSE OF A STUDY OF THIS VARIANT IS TO PROPOSE A REACTOR DESIGN TO SERVE AS A BASIS FOR A MORE-DETAILED STUDY WITHIN THE SCOPE OF A DESIGN PROJECT. THUS, THE PROBLEMS POSED BY THE REACTOR BLOCK, STORAGE, HANDLING AND THE HANDLING CELL, HAVE BEEN EXAMINED, WITH THE STRESS PLACED ON THE LAST TWO POINTS, WHICH PRESENT THE GREATEST DIFFICULTIES. MUCH REMAINS UNKNOWN ABOUT THE CONTAMINATION OF THIS CELL BY SODIUM AT THE TIME OF HANDLING. METHODS ARE PROPOSED TO LIMIT THIS CONTAMINATION. THE PRINCIPAL ADVANTAGES ARE THE FOLLOWING - (A) HANDLING TAKES PLACE WITHIN SIGHT. THIS VISIBILITY IN CERTAIN CASES IS DIRECT THROUGH PORT HOLES PROVIDED IN THE CELL, OR INDIRECT BY MEANS OF PERISCOPES OR TELEVISION CAMERAS. (B) SINCE THIS CELL IS LEAK-PROOF AND FILLED WITH ARGON, THE PROBLEMS OF TIGHTNESS BETWEEN REACTOR AND CELL ARE CONSIDERABLY SIMPLIFIED. (C) THIS CELL ALSO PERMITS HANDLING OF CONTAMINATED EQUIPMENT WITHOUT REQUIRING THE USE OF AUXILIARY, HIGHLY SHIELDED PROTECTIVE EQUIPMENT. (D) FINALLY, THE CELL PLAYS THE ROLE OF A TIGHT CHAMBER WHICH CAN RETAIN ACTIVE PRODUCTS, ITS STRUCTURE PERMITS IT TO OFFER CONTAINMENT OF A REACTOR EXPLOSION.

CONTAINMENT + DESIGN CRITERIA + REACTOR, FAST + HOT CELL + REACTOR, LMCR + MAINTENANCE AND REPAIR

11-011394 DATED 1965
LANDLER G
HNPF-STRESS CORROSION STUDY
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA
NAA-SR-MEMO-11607 +. 47 PAGES, AUGUST 30, 1965, CFSTI \$2.00 CY, \$0.50 MN

THE FAILURE OF EXPANSION TANK 3 CALLED ATTENTION TO THE POSSIBILITY OF THERE BEING OTHER WELDMENTS IN SODIUM-SYSTEM COMPONENTS WITH THE SAME CORROSIVE DEPOSITS AND THEREFORE POSSIBLY SUBJECT TO STRESS-CORROSION CRACKING. THE REQUIREMENTS FOR SUCH STRESS-CORROSION CRACKING ARE ONLY IMPERFECTLY KNOWN AT THIS TIME BUT APPEAR TO BE A DEPOSIT OF CORROSIVE MEDIA, SUCH AS CHLORIDE, THE PRESENCE OF MOISTURE AT SOME TIME, AND AT LEAST A MODERATE STRESS LEVEL. EXAMINATION OF DRAWINGS AND SPECIFICATIONS INDICATE THAT A LARGE NUMBER OF WELDMENTS CAN BE IDENTIFIED IN THE HNPF SODIUM SYSTEMS WHICH MAY MEET SOME OR ALL THESE REQUIREMENTS. THE ONLY SURE METHOD OF LOCATING POINTS OF INCIPENT TROUBLE IS A DESTRUCTIVE EXAMINATION, ALTHOUGH ULTRASONIC TESTING MAY BE SATISFACTORY IN CERTAIN IDEALIZED CIRCUMSTANCES. MANY OF THE SUSPECT WELDMENTS CANNOT BE INSPECTED BECAUSE OF THE RADIATION LEVELS OR PHYSICAL INACCESSIBILITY. IN THE MOST FAVORABLE CIRCUMSTANCES, IT IS NECESSARY TO REMOVE INSULATION, ELECTRICAL HEATERS, AND, IN SOME INSTANCE, HEATER JACKETS.

FAILURE, COMPONENT + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + SODIUM + WELDS + TEST, NONDESTRUCTIVE + CHLORINE + STRESS CORROSION

11-011396 ALSO IN CATEGORY 5 DATED 1965
LEMCOE M
STUDY OF INSTANTANEOUS COLLAPSE AND CREEP BUCKLING CHARACTERISTICS OF LARGE SGR PROCESS TUBES
ATOMICS INTERNATIONAL, CANOGA PARK
NAA-SR-11323 +. 47 PAGES, 4 TABLES, REFERENCES, JULY 25, 1965, CFSTI \$3.00 CY, \$0.50 MN

THE INSTANTANEOUS COLLAPSE (LESS THAN 0.2 HR) AND CREEP-BUCKLING CHARACTERISTICS (UP TO 5000 HR) OF TYPE 304 STAINLESS STEEL TUBES (4.00 IN. IN OD, WITH 0.025-IN. WALL), UNDER EXTERNAL RADIAL PRESSURE AND VARYING AXIAL LOAD, WERE DETERMINED AT TEMPERATURES UP TO 1200 F IN AN ARGON ENVIRONMENT. THE LENGTH OF EACH TUBE WAS 6 FT, WHICH IS ESSENTIALLY INFINITE IN TERMS OF BUCKLING. IN ADDITION, TUBES IN COMBINATION WITH FUEL-ELEMENT BUNDLES CORRESPONDING TO THOSE PROPOSED FOR A LARGE SODIUM GRAPHITE REACTOR (LSGR) WERE ALSO INVESTIGATED TO DETERMINE THE EXTENT TO WHICH THE FUEL-ELEMENT BUNDLE INCREASED THE RESISTANCE TO COLLAPSE OF THE PROCESS TUBE. THE RESULTS OF THE STUDY INDICATED THAT STIFFENER RINGS OR THICKER-WALLED TUBES SHOULD NOT BE NECESSARY UNDER NORMAL OPERATING CONDITIONS.

BUCKLING + REACTOR, LMCR + STEEL, STAINLESS + PIPES AND PIPE FITTINGS

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-012319 ALSO IN CATEGORY 17
SELECTED OPERATING EXPERIENCES

DATED 1965

POWER REACTOR DEVELOPMENT COMPANY

EF-28 +. 3 PAGES, POWER REACTOR DEVELOPMENT COMPANY, ENRICO FERMI ATOMIC POWER PLANT REPORT FOR DECEMBER 1965
PAGES 1-2 AND 10, DECEMBER 1965, DOCKET NO. 50-16

THE NUCLEAR TESTING AT LESS THAN 1 MW(TH) EXTENDED FROM AUGUST 1963 TO DECEMBER 1965. THE FIRST POWER INCREASE ABOVE 1 MW(TH) TOOK PLACE DECEMBER 29, 1965. THE NEW SOURCE-RANGE DETECTORS (INCLUDING TWO SURROUNDED BY ZIRCONIUM HYDRIDE) WORKED WELL. WHILE DAILY CHECKS OF ELECTRICAL PENETRATION LEAKAGE AT 5 PSIG WERE BELOW 50 SCF/DAY, TESTS AT 32 PSIG INDICATED 110.

USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

CONTAINMENT, HIGH PRESSURE + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + TEST, LEAK RATE + REPORT, OPERATIONS SUMMARY + INSTRUMENT, STARTUP RANGE

11-016576 ALSO IN CATEGORY 18

DATED 1965

SUPPLEMENT TO SRE SAFETY ANALYSIS REPORT. A PLAN FOR USE OF A FAST-REACTOR FUEL TEST SECTION IN THE SRE ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-11234 +. 78 PAGES, FIGURES, MAY 3, 1965

SRE MODIFICATIONS INCLUDE REPLACEMENT OF THE CENTRAL SEVEN PEP MODERATOR ELEMENTS BY A FAST-FUEL FLUX-TRAP TEST SECTION, ADDITION OF A 20-MW(TH) HEAT SINK TO THE HEAT-TRANSFER SYSTEM TO ALLOW 50-MW(TH) OPERATION, IMPROVEMENT OF ENVIRONMENTAL CONTROL AROUND FAST FUEL ELEMENTS DURING FUEL HANDLING, AND CHANGES IN SERVICE FACILITIES. CONCLUSIONS - (1) TECHNICALLY, NO ADDITIONAL BUILDING CONTAINMENT NEEDED BECAUSE OF ADDITION OF FAST TEST SECTION. (2) FOR UNMODIFIED REACTOR, ENERGY RELEASE OF UP TO 50 LB OF TNT WILL NOT LIFT THE LOADING FACE SHIELD OR DAMAGE CORE TANKS. (3) ADDITIONAL CONSTRAINT OF LOADING-FACE SHIELD NEEDED FOR LARGER RELEASES, BUT THOSE UP TO 200 LB OF TNT CAN BE CONTAINED. (4) PRESENCE OF PU DOES NOT MATERIALLY AFFECT SITE-BOUNDARY DOSE RATE WITH NONCREDIBLE DETONATION ACCIDENT OF UP TO 50 LB OF TNT. (5) ADDITION OF PUOZ FUEL DOES NOT SUBSTANTIALLY ALTER RADIOLOGICAL CONSIDERATIONS AT SITE BOUNDARY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + REACTOR, LMCR + REPORT, SAR + SRE (RE) + IN PILE EXPERIMENT + IRRADIATION TESTING + REACTOR, FLUX TRAP

11-018209 ALSO IN CATEGORIES 5 AND 12

DATED 1967

HARTY H + BATCH JM + PEDERSEN LT + WHEELER RG + FOX JC
ENGINEERING DEVELOPMENT DEPARTMENT QUARTERLY REPORT, APRIL-JUNE 1966

BATTELLE-NORTHWEST, RICHLAND, WASHINGTON

BNWL-367 +. 76 PAGES, 18 FIGURES, 2 TABLES, APRIL 1967

PROGRESS REPORT COVERING SUCH TOPICS AS THE FAST FLUX TEST FACILITY, HEAT TRANSFER STUDIES, CROSS-FLOW MIXING BETWEEN PARALLEL FLOW, BOILING-STABILITY STUDIES, IRRADIATION DAMAGE TO REACTOR METALS, UPSTREAM BOILING BURNOUT, THE CSE, IODINE FORMS IN STEAM CONDENSATE, AND DECONTAMINATION OF COATINGS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

HEAT TRANSFER, BOILING + BURNOUT HEAT FLUX + IODINE + REACTOR, LMCR + PLUTONIUM + STEAM + AEROSOL PROPERTIES + CONTAINMENT R AND D + CSE + SAMPLING + HEAT TRANSFER EXPERIMENT

11-018634 ALSO IN CATEGORY 8

DATED 1967

GAST K + SCHIKARSKI W

CONTAINMENT OF A LARGE SODIUM-COOLED FAST BREEDER REACTOR

KERNFORSCHUNGSZENTRUM, KARLSRUHE, FEDERAL REPUBLIC OF GERMANY

SM-89/7 + CONF-670-402 +. 15 PAGES, 7 FIGURES, 1 TABLE, 5 REFERENCES, APRIL 1967

FOLLOWING THE CONCEPTUAL DESIGN STUDY FOR A 1000-MW SODIUM-COOLED FAST BREEDER REACTOR, THE CONCEPTUAL DESIGN OF A 300-MW PROTOTYPE REACTOR IS CURRENTLY UNDER WAY AT KARLSRUHE RESEARCH CENTRE. FOR THE PROTOTYPE REACTOR, A RATHER CONSERVATIVE SAFETY ASSESSMENT WAS CHOSEN TO COMPENSATE FOR INCOMPLETE KNOWLEDGE OF DETAILED NUCLEAR AND THERMODYNAMIC BEHAVIOR IN A LARGE ACCIDENT. DOUBLE CONTAINMENT WILL THEREFORE BE PROVIDED, THE INNER CONTAINMENT BEING DESIGNED FOR THE POTENTIAL CONSEQUENCES OF A NUCLEAR EXCURSION WITH DESTRUCTIVE ENERGY RELEASE, AND FOR PREVENTION OF LARGE SODIUM FIRES. THE OUTER CONTAINMENT WILL SERVE AS A FINAL BARRIER FOR FISSION PRODUCTS AND OTHER RADIOACTIVE AEROSOLS. THE CONTAINMENT CONCEPT IS AIMED AT INDEPENDENCE IN SITE SELECTION.

AVAILABILITY - NATIONAL AGENCY FOR INTERNATIONAL PUBLICATIONS, INC., 317 EAST 34TH STREET, NEW YORK, NEW YORK. 10016

CONTAINMENT ANALYSIS + CONTAINMENT, MULTIPLE + ENERGY SOURCE + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + DESIGN STUDY + GERMANY

11-018638

DATED 1967

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-018638 *CONTINUED*

WOLFF PH

TO CONTAIN, OR NOT TO CONTAIN

NUCLEAR DESIGN AND CONSTRUCTION LIMITED, WHETSTONE, LEICESTER, UK

SM-89733 + CONF-670-402 +. 14 PAGES, 1 TABLE, 1 REFERENCE, APRIL 1967

THE REACTOR SYSTEM OFFERING LEAST POTENTIAL HAZARD IS CLEARLY ONE WHICH BY VIRTUE OF ITS INHERENT DESIGN IS INCAPABLE OF ALLOWING THE RELEASE OF RADIOACTIVE MATERIAL FROM WITHIN THE SYSTEM DURING NORMAL OPERATION OR UNDER ANY CREDIBLE FAULT CONDITIONS. HENCE NO SECONDARY SYSTEM OF CONTAINMENT IS REQUIRED. THIS CONCLUSION HAS BEEN REACHED FOR THE GAS-COOLED CIVIL NUCLEAR POWER STATIONS SO FAR CONSTRUCTED IN THE UNITED KINGDOM (7200 MWIE) SO FAR COMPLETED OR UNDER CONSTRUCTION). THE APPROACH REQUIRES THE BEHAVIOUR OF THE REACTOR SYSTEM TO BE KNOWN UNDER ALL CONDITIONS AND THE EXTENT TO WHICH THIS IS POSSIBLE IS ILLUSTRATED BY DISCUSSION OF THE DESIGN SOLUTIONS ADOPTED TO PREVENT THE ESCAPE OF RADIOACTIVE MATERIAL FROM THE ADVANCED GAS-COOLED REACTORS AT PRESENT UNDER CONSTRUCTION FOR THE SECOND PHASE OF THE BRITISH NUCLEAR POWER PROGRAMME AND FOR THE STEAM-GENERATING HEAVY-WATER MODERATED REACTOR, SODIUM-COOLED FAST REACTOR AND HIGH-TEMPERATURE GAS-COOLED REACTOR PROTOTYPES AND POWER DESIGNS UNDER DEVELOPMENT IN THE UNITED KINGDOM. SAME REFERENCE IS ALSO MADE TO THE DESIGN PROBLEMS IN OTHER SYSTEMS.

AVAILABILITY - NATIONAL AGENCY FOR INTERNATIONAL PUBLICATIONS, INC., 317 EAST 34TH STREET, NEW YORK, N.Y. 10016

ACCIDENT ANALYSIS + CONTAINMENT + CONTAINMENT ANALYSIS + REACTOR, GCR + REACTOR, HWR + REACTOR, LMCN + UNITED KINGDOM

11-020821

DATED 1967

GAST K + SCHIKARSKI W

CONTAINMENT OF A LARGE SODIUM-COOLED FAST BREEDER REACTOR

KERNFORSCHUNGSZENTRUM, KARLSRUHE, GERMANY

KFK-347 +. 10 PAGES, 1 FIGURE, 1 TABLE, 5 REFERENCES, FEBRUARY 1967

FOR THE CONCEPTUAL DESIGN OF A 300-MWE SODIUM-COOLED FAST-BREEDER PROTOTYPE REACTOR, A SEVERE NUCLEAR EXCURSION IS ASSUMED AS THE DESIGN-BASIS ACCIDENT. BASED ON A VERY CONSERVATIVE ANALYSIS OF THIS ACCIDENT, A DOUBLE-CONTAINMENT CONCEPT IS CHOSEN FOR REACTOR DESIGN. IT IS SHOWN THAT THIS CONCEPT IS CAPABLE OF CONTAINING THE DESIGN-BASIS ACCIDENT WITHOUT ANY HAZARD TO THE ENVIRONMENT. IMPROVED KNOWLEDGE IN ACCIDENT ANALYSIS, ACTIVITY RELEASE, AND ENGINEERED SAFEGUARDS ARE EXPECTED TO FACILITATE SINGLE-CONTAINMENT DESIGNS FOR LARGE FAST-BREEDER REACTORS, SODIUM COOLED.

CONTAINMENT ANALYSIS + CONTAINMENT, MULTIPLE + REACTOR, FAST + REACTOR, LMCN + ACCIDENT, MAXIMUM CREDIBLE (MCA)

11-022328

ALSO IN CATEGORY 2

DATED 1967

MCCULLOUGH CR + BROWN GM + HANCOCK RP + LYERLY RL + MITCHELL W + TURNER SE

SITING STUDIES OF TWO LARGE DUAL-PURPOSE DESALINATION REACTORS

SOUTHERN NUCLEAR ENGINEERING, INC., DUNEDIN, FLORIDA

SNE-33 +. 87 PAGES, 18 FIGURES, 16 REFERENCES, SEPTEMBER 1, 1967

PRESENTS A PRELIMINARY STUDY ON THE SITING OF A 3-GW(T) AND A 10-GW(T) REACTOR FOR DESALINATION AND POWER PRODUCTION. TWO REACTOR CONCEPTS WERE TREATED, THE HTGR AND THE LMFBR. CONTAINMENT AND ENGINEERED SAFETY FEATURES ARE DISCUSSED IN CONJUNCTION WITH THE DESIGN-BASIS ACCIDENT (DBA) AND MAXIMUM HYPOTHETICAL ACCIDENT (MHA). METHODS OF DETERMINING DOWNWIND DOSES IN THE EVENT OF ACCIDENTS ARE PRESENTED.

ATMOSPHERIC DIFFUSION + REACTOR, GCR + REACTOR, LMCN + SITING, REACTOR + REACTOR, DESALINATION + DISTANCE + DOSE CALCULATION, INTERNAL + DOSE CALCULATION, EXTERNAL + JACOBS

11-023813

DATED 1967

CRAIG ER + FULTON RH + DAMON JJ

DESIGN AND TESTING OF SOLENOID AND INSTRUMENT LEAD CONNECTOR FOR USE IN SEFOR REACTOR VESSEL

GENERAL ELECTRIC CO., SUNNYVALE, CALIF. ADVANCED PRODUCTS OPERATION

GEAP-5307 +. 26 PAGES, 9 FIGURES, MARCH 1967

DESCRIBES THE DESIGN AND TESTING OF A MULTI-PIN LEAD CONNECTOR FOR THE SEFOR INSTRUMENTED FUEL ASSEMBLY AND THE TESTING OF A SOLENOID FOR THE INSTRUMENTED FUEL ASSEMBLY LIFTING FIXTURE. TESTS IN A SODIUM-VAPOR/ARGON ENVIRONMENT AT TEMPERATURES UP TO 400 F DEMONSTRATED THAT THE OPERATIONAL PERFORMANCE OF THE COMPONENTS WOULD MEET OR EXCEED THE ANTICIPATED REQUIREMENTS FOR THE SEFOR OPERATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL HANDLING + INSTRUMENT, IN CORE + REACTOR, LMCN + TESTING + TEST, PROOF + TEST, PREOPERATIONAL + REACTOR, PULSED + SEFOR (RE) + FUEL ELEMENTS

11-023857

DATED 1964

GUEGAUD A + ROQUES C

THE USE OF AUSTENITIC STEEL AS CONSTRUCTION MATERIAL FOR THE PRESSURE VESSEL OF THE FAST REACTOR RAPSODIE

SOCIETE DES FORGES ET ATELIERS DU CREUSOT, PARIS (FRANCE)

EURFNR-92F + CONF-777-1 + EUR-1642-F +. 37 PAGES, FIGURES, REFERENCES, 1964, TRANSLATION OF REPORT CEA-IP-

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-023857 *CONTINUED*

1593, PRESENTED AT SYMPOSIUM ON MATERIALS FOR COOLING REACTORS WITH SODIUM, NEW YORK, NOV. 1963

DISCUSSES CHOICE OF AN 18/8-TYPE STEEL. ADVANTAGES ARE ITS CREEP RESISTANCE AND CORROSION RESISTANCE IN SODIUM. PRINCIPAL DISADVANTAGE IS THE HIGH EXPANSION COEFFICIENT, WHICH COULD CAUSE LARGE THERMAL STRESSES. HOWEVER, THE STRENGTH OF THE STEEL AT HIGH TEMPERATURE ALLOWS FOR A THIN WALL, REDUCING TEMPERATURE GRADIENTS. 18/8-MO STEEL WAS CHOSEN OVER 18/8-NB AS A RESULT OF TESTS, WHICH ARE DESCRIBED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + PRESSURE VESSELS + STEEL + TESTING + CREEP BEHAVIOR + TEMPERATURE GRADIENT + EXPANSION

11-024156 ALSO IN CATEGORY 18
BERGSTROM RN + CHITTENDEN WA + MCCLINTOCK DC
SODIUM-COOLED FAST REACTORS
SARGENT + LUNDY
3 PAGES, 1 FIGURE, POWER ENGINEERING, 71(10), PAGE 56-58, (OCT. 1967)

DATED 1967

QUITE GENERAL DISCUSSION OF STRUCTURAL PROBLEMS OF THE BUILDING, WHICH, WHILE NOT BASICALLY DIFFERENT FROM WATER REACTORS, ARE COMPLEX AND HARDER TO ANALYZE.

REACTOR, LMCR + STRESS ANALYSIS + BUILDING

11-025992 ALSO IN CATEGORY 18 DATED 1966
JAMROG AK
APPARATUS FOR CONTROLLING THE ATMOSPHERE OVER A NUCLEAR REACTOR
U.S. PATENT 3,282,793 +. 4 PAGES, 2 FIGURES, NOVEMBER 1, 1966

APPARATUS FOR PROVIDING AIR ATMOSPHERE DIRECTLY OVER A REACTOR INSIDE LARGER ARGON-FILLED CHAMBER COMPRISED OF AXIALLY COLLAPSIBLE BELLOWS WITH ONE END SEALED TO REACTOR AND OTHER END TO OPENING IN TOP OF CHAMBER. BELLOWS PROTECTS CONTROL-ROD ELECTRICAL CONNECTIONS AND PERMITS DIRECT-CONTACT-MAINTENANCE CONTROL-ROD DRIVES. REACTOR COVER CAN BE RAISED, COLLAPSING BELLOWS, FOR REPLACEMENT OF FUEL ELEMENTS.

AVAILABILITY - THE U.S. PATENT OFFICE, DEPARTMENT OF COMMERCE, WASHINGTON, D. C., 25 CENTS PER COPY

CONTAINMENT + CONTAINMENT DESIGN + REACTOR, LMCR + MAINTENANCE AND REPAIR + CONTAINMENT STRUCTURE + CONTAINMENT ATMOSPHERE, INERT + PATENT + CONTAINMENT PENETRATION, ELECTRICAL + CONTROL ROD DRIVE MECHANISMS

11-026513 ALSO IN CATEGORIES 9 AND 10 DATED 1967
VERBER F
FURTHER DEVELOPMENTS IN CONTAINMENT-BUILDING ELECTRICAL PENETRATIONS
8 PAGES, 13 FIGURES, 1 TABLE, 6 REFERENCES, REACTOR AND FUEL-PROCESSING TECHNOLOGY 11(1), PAGES 5-12 (WINTER 1967-1968)

REVIEWS SOME ELECTRICAL PENETRATION PROBLEMS AND SOLUTIONS EXPERIENCED DURING PLANT OPERATIONS AT FERMI AND AT PEACH BOTTOM. DESCRIBES FIRE-PROTECTION MODIFICATIONS TO THE PENETRATIONS AND TO THE ELECTRICAL DESIGN AT PEACH BOTTOM AFTER A FIRE. ALSO PRESENTS RECENT DEVELOPMENTS IN THE DESIGN OF THE DRESDEN-2 ELECTRICAL PENETRATION SYSTEM AND ASSEMBLIES.

REACTOR, BWR + DESIGN CRITERIA + FERMI (LMFBR) + FIRE + PEACH BOTTOM 1 (HTGR) + DRESDEN 2 (BWR) + INSTRUMENT, COMPONENT + EQUIPMENT DESIGN + REACTOR, HTGR + ELECTRICAL CONDUCTION + CONTAINMENT PENETRATION, ELECTRICAL + OPERATING EXPERIENCE + REACTOR, LMFBR

11-027827 ALSO IN CATEGORY 1 DATED 1968
MANUAL OF LECTURE NOTES - REACTOR SAFETY COURSE NO. 7
UNITED KINGDOM ATOMIC ENERGY AUTHORITY, RISLEY, WARRINGTON, LANCASHIRE
200 PAGES, FIGURES, TABLES, REFERENCES, SEPTEMBER 23, 1968

NINE SECTIONS. (I) INTRODUCTION, 5 LECTURES. (II) FISSION PRODUCT RELEASE, 5 LECTURES. (III) CONTAINMENT ENGINEERING, 7 LECTURES. (IV) CONTROL AND INSTRUMENTATION, 7 LECTURES. (V) GAS COOLED REACTORS, 11 LECTURES. (VI) WATER REACTORS, 12 LECTURES. (VII) GENERAL, 5 LECTURES. (VIII) FAST REACTORS, 7 LECTURES. (IX) SITING AND EMERGENCY PROCEDURES, 4 LECTURES. NEW LECTURES APPEAR TO BE (7) ASSESSMENT OF ENVIRONMENTAL HAZARDS FROM FISSION PRODUCT RELEASES. (8A) FISSION PRODUCT BEHAVIOR IN LIQUID COOLED REACTORS. (10A) CONTAINMENT OF FISSION PRODUCTS WITHIN FUEL ELEMENTS. (24A) METHODS AND PROBLEMS IN REACTOR ASSESSMENTS. (40) PROBABILITY TECHNIQUES APPLIED TO WATER REACTORS.

AVAILABILITY - AUTHORITY HEALTH AND SAFETY BRANCH, U.K.A.E.A., 11 CHARLES II STREET, LONDON, S. W.1.

REACTOR, GCR + SAFETY REVIEW + UNITED KINGDOM + REACTOR, LWR + SAFETY PRINCIPLES AND PHILOSOPHY + FISSION PRODUCT RETENTION + SAFETY ANALYSIS + REACTOR, HTGR + PROBABILITY + REACTOR, LMFBR + PARTITION COEFFICIENT + FUEL ELEMENTS

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-028987 ALSO IN CATEGORY 7 DATED 1968
CONNER JG + POREMBKA SW
COMPENDIUM OF PROPERTIES AND CHARACTERISTICS FOR SELECTED LMFBR CLADDING MATERIALS
BATTELLE MEMORIAL INST., COLUMBUS, OHIO
BML-1900 +. 234 PAGES, MAY 15, 1968

REVIEWS WELDING, RADIATION DAMAGE, STRENGTH AND DUCTILITY AT HIGH (600 C) TEMPERATURES,
COMPATIBILITY WITH FUEL AND SODIUM FOR AUSTENITIC SS AND NICKEL-BASED ALLOYS, VANADIUM ALLOYS,
NIUBIUM ALLOYS FOR OXIDE, CARBIDE, NITRIDE, AND OXIDE FUELS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + RADIATION DAMAGE + STEEL, STAINLESS + PROPERTY, PHYSICAL + INCOLOY + REACTOR, LMFBR

11-029433 ALSO IN CATEGORIES 2 AND 18 DATED 1968
RESPONSE TO QUESTION A-1. DEFINITIONS AND FUNCTION OF CLASS I AND II ITEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIFORNIA
11 PAGES, 3 FIGURES, 2 TABLES, PAGE A-1 THRU A-11 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE
APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS THE EXTENT TO WHICH THE DEFINITIONS AND FUNCTIONS OF CLASS I AND CLASS II ITEMS HAVE BEEN
MODIFIED IN THE COURSE OF FINAL DESIGN AND CONSTRUCTION. (ANSWER ON PGS A-1 THRU -11.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION +
EARTHQUAKE ENGINEERING

11-029434 ALSO IN CATEGORIES 2 AND 18 DATED 1968
RESPONSE TO QUESTION A-2 - EARTHQUAKE EXPOSURE
GENERAL ELECTRIC CO., SAN JOSE, CALIFORNIA
4 PAGES, 1 TABLE, PAGE A-12 THRU A-15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE
TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

INDICATE THE EARTHQUAKE EXPOSURE, DESCRIBED BY RESPONSE SPECTRA, USED IN THE DESIGN FOR BOTH THE
DESIGN AND MAXIMUM EARTHQUAKES. (ANSWER ON PGS A-12 THRU -15.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE
ENGINEERING + RESPONSE SPECTRUM

11-029436 ALSO IN CATEGORIES 2 AND 18 DATED 1968
RESPONSE TO QUESTION A-4 - SEISMIC ANALYSIS OF CLASS I ITEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
66 PAGES, 21 FIGURES, 2 TABLES, PAGE A-16 THRU A-21 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE
APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE IN GENERAL THE METHODS USED FOR SEISMIC ANALYSIS OF CLASS I ITEMS. PROVIDE MINIMUM
INFORMATION FOR THE CONTAINMENT AND A FEW OF THE MOST ESSENTIAL CLASS I SYSTEMS. (A) ANALYTICAL
MODEL USED, E.G., LOCATION OF LUMPED MASSES, VALUES OF PARAMETERS USED, AND SUPPORT CONDITIONS.
(ANSWER ON PAGES A-15 THRU -21 PLUS 60 PAGES OF TABLES AND FIGURES.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + REACTOR, LMCR + REPORT, SAR + ANALYTICAL TECHNIQUE + SEFOR (RE) + RESPONSE TO NRC
QUESTION + EARTHQUAKE ENGINEERING

11-029443 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION B-1 - STEADY STATE THERMAL GRADIENTS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
12 PAGES, 7 FIGURES, PAGE B-1 THRU -12 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE
TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE A RATIONAL NUMERICAL ESTIMATE OF THE ERRORS INTRODUCED INTO THE CALCULATIONS FOR THE
STRUCTURE BY THE USE OF STEADY STATE THERMAL GRADIENTS INSTEAD OF TRANSIENT GRADIENTS. COVER ALL
THREE CASES - (1) OPERATING CONDITION, (2) ACCIDENT CONDITION, (3) SPILLING OF SODIUM. CONSIDER
ONE OR TWO EXAMPLES ONLY, BUT PROVIDE NUMERICAL INFORMATION AND SKETCHES AS REQUIRED FOR
EVALUATION. (ANSWER ON PG B-1 THRU -12.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONCRETE + REACTOR, LMCR + REPORT, SAR + THERMAL MECHANICAL EFFECT + STRESS ANALYSIS + TEMPERATURE GRADIENT +

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-029443 *CONTINUED*
CONTAINMENT STRUCTURE + SEFOR (RE) + RESPONSE TO NRC QUESTION

11-029445 ALSO IN CATEGORIES 2 AND 18 DATED 1968
RESPONSE TO QUESTION B-2 - MATHEMATICAL MODEL FOR SEISMIC DESIGN
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, PAGE B-13 THRU -15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THE MATHEMATICAL MODEL USED FOR THE SEISMIC DESIGN OF THE CONTAINMENT STRUCTURE ASSUMES A VERTICAL
FLEXIBLE CANTILEVER. ROCKING AND DIFFERENTIAL HORIZONTAL TRANSLATION OF THE BASE AND THE
INTERACTION BETWEEN THE PRIMARY AND SECONDARY CONTAINMENTS APPEAR TO HAVE BEEN NEGLECTED.
DISCUSS THE ERROR INTRODUCED INTO THE DESIGN BY THESE ASSUMPTIONS. (ANSWER ON PGS B-13 THRU B-
15.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + REACTOR, LMCR + REPORT, SAR + CONTAINMENT STRUCTURE + SEFOR (RE) + RESPONSE TO NRC
QUESTION + EARTHQUAKE ENGINEERING + MATHEMATICAL TREATMENT + INTERACTION, SOIL AND STRUCTURE

11-029446 ALSO IN CATEGORIES 5 AND 18 DATED 1968
RESPONSE TO QUESTION B-3 - PRESSURE LOADING
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE B-16 AND -17 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

IT IS STATED IN THE APPLICATION THAT PRESSURE LOADING CAN BE APPLIED AS UNIFORM LOAD
SIMULTANEOUSLY IN ALL CELLS AND THAT THE COMMUNICATIONS PROVIDED BETWEEN CELLS ARE SUCH THAT THIS
ASSUMPTION IS VALID. DISCUSS A TRANSIENT PRESSURE DISTRIBUTION AND ITS INFLUENCE, OR LACK OF
INFLUENCE, ON THE DESIGN. INCLUDE A DISCUSSION ON COMMUNICATIONS BETWEEN CELLS. (ANSWER) SEE
SECTION XVI OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + PRESSURE,
INTERNAL + RESPONSE TO NRC QUESTION

11-029447 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION B-4 - CONCRETE REINFORCEMENT
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE B-18 AND -19 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DOES CONCRETE HAVE ENOUGH REINFORCING TO COVER ALL TENSION BLOCK OF HOMOGENEOUS
UNCRACKED CONCRETE SECTION. HAVE METHODS OF ACI CODE FOR CRACKED SECTION BEEN USED. ON WHAT
BASIS HAS CONCRETE COMPRESSION BEEN ESTABLISHED. HAVE PROVISIONS BEEN MADE TO PREVENT BUCKLING
OF REINFORCING BARS IN COMPRESSION ZONE, ESPECIALLY WHERE PLASTIC DEFORMATIONS ARE PERMITTED TO
OCCUR. HAVE LOCAL PLASTIC STRAINS BEEN EVALUATED. PROVIDE ASSUMED ELASTICITY AND SHEAR MODULUS
OF CONCRETE, CONCRETE POISSONS RATIO, ASSUMED SHRINKAGE STRAIN FOR CONCRETE UNDER LONG-TERM
TEMPERATURE LOADING. (ANSWER ON PAGES B-18 AND -19.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONCRETE + CONTAINMENT DESIGN + CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + CONTAINMENT
STRUCTURE + SEFOR (RE) + RESPONSE TO NRC QUESTION + STRUCTURAL INTEGRITY

11-029448 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION B-5 - LINER DESIGN
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
34 PAGES, 3 FIGURES, 2 TABLES, PAGE B-20 THRU -31 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE
APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CONCERNING LINER, DISCUSS VARIATIONS IN THICKNESS AND YIELD POINTS AND ERECTION INACCURACIES
FACTORED INTO DESIGN, UNBALANCED LOADS AT ANCHORS AND BUCKLING, DEFORMATION FROM BUCKLING,
STRAINS WHERE BUCKLING IS NOT PERMITTED, TESTING, WELDING ELECTRODE AND QUALITY CONTROL OF
PROCESS, DESIGN CRITERIA, LEAD OF MAXIMUM LOCAL STRESSES AT STRESS CONCENTRATION POINTS, USE OF
2% OF COLUMN LOAD FOR LATERAL RESTRAINT OF ANCHOR BARS, SODIUM TEMPERATURE FOR ACCIDENTAL SPILL,
DESIGN CRITERIA AND MARGINS FOR BELLOW, PENETRATION SLEEVES, AND OTHER FLEXIBLE JOINTS. (ANSWER
ON PG B-20 THRU -31 PLUS ATTACHMENT.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + STRESS + WELDING + SEFOR (RE) + RESPONSE
TO NRC QUESTION + QUALITY ASSURANCE + CONTAINMENT LINER + PENETRATIONS, PRIMARY CONTAINMENT

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-029449 ALSO IN CATEGORIES 2 AND 18 DATED 1968
RESPONSE TO QUESTION C-1 - INTERACTION OF INNER AND OUTER CONTAINMENTS DURING EARTHQUAKE
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 1 PAGE, PAGE C-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
 REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE THE LOCAL LOADING GENERATED DURING AN EARTHQUAKE BY THE INTERACTION OF THE INNER AND
 OUTER CONTAINMENTS. INCLUDE THE ROCKING AND THE DIFFERENTIAL (WITH RESPECT TO THE GROUND)
 HORIZONTAL TRANSLATION OF THE BUILDING. PROVIDE THE CORRESPONDING MAXIMUM STRESSES AND STRAINS.
 (ANSWER) SEE ANSWER TO QUESTION B-2.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE
 ENGINEERING + INTERACTION, SOIL AND STRUCTURE

11-029450 ALSO IN CATEGORIES 2 AND 18 DATED 1968
RESPONSE TO QUESTION C-2 - LOCAL STRESSES DUE TO EARTHQUAKE
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 1 PAGE, PAGE C-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
 REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE THE LOCAL STRESSES, DUE TO AN EARTHQUAKE, AT THE JUNCTION OF THE STEEL SHELL AND HEAVY
 CONCENTRATED MASSES, SUCH AS LOCKS, HATCHES, AND LARGE PENETRATIONS. FOR EXAMPLE, AN EMERGENCY
 ESCAPE LOCK PROTRUDES FROM THE CONTAINMENT SHELL AT GROUND LEVEL AND APPEARS TO BE SUPPORTED BY
 THE SHELL. THIS PROTRUSION IS MASSIVE AND LONG AND MAY REQUIRE ADDITIONAL SUPPORT TO PREVENT IT
 FROM OVERLOADING THE CONTAINMENT SHELL AT THE POINT OF FASTENING. ADDITIONAL INFORMATION IS
 REQUESTED AS TO THE DESIGN OF THIS ESCAPE LOCK AND THE PROVISIONS THAT HAVE BEEN OR WILL BE TAKEN
 TO INSURE THAT IT CANNOT IN ANY WAY JEOPARDIZE THE CONTAINMENT OF THE STRUCTURE UNDER ACCIDENT
 AND EARTHQUAKE LOADING CONDITIONS. (ANSWER) SEE SECT. 7.3 AND SUPPL. 7 TO FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + STRESS + CONTAINMENT AIR LOCK + SEFOR (RE) +
 RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING

11-029451 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION D-1 - VACUUM BREAKER SYSTEM
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 3 PAGES, 2 FIGURES, PAGE D-1 THRU D-3 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE
 TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS THE FOLLOWING REGARDING THE VACUUM BREAKER SYSTEM - TYPE, SIZE MOTIVE FORCE, AND SETTING
 OF VACUUM BREAKER VALVES, FREQUENCY AT WHICH THEIR OPERABILITY WILL BE CHECKED, AND METHOD OF
 TESTING. (ANSWER) 4 VALVES INSTALLED IN PARALLEL (2 INSIDE AND 2 OUTSIDE CONTAINMENT). VALVES
 ARE 6 INCH, 150 LB SWINGING DISC TYPE. OPERABILITY TESTS ANNUALLY. LEAKAGE TESTS BY PLACING
 BLIND FLANGES ON OUTLET PIPING AND PRESSURIZING TO 30 PSIG.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + TESTING + CONTAINMENT VACUUM BREAKER + SEFOR (RE) + SYSTEM DESCRIPTION +
 RESPONSE TO NRC QUESTION + VALVES

11-029452 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION D-2 - PROTECTION OF CLOSED LOOPS
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 2 PAGES, PAGE D-4 AND -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
 IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THERE ARE, IN ADDITION TO THE SODIUM LINES, EIGHT REFRIGERANT LINES, ONE RADIOACTIVE GASEOUS WASTE
 LINE, ONE ARGON SUPPLY LINE, ONE BREATHING AIR LINE, AND ONE NITROGEN SUPPLY LINE THAT PENETRATES
 THE OUTER CONTAINMENT BARRIER, ALL OF WHICH ARE CLASSIFIED AS TYPE 3 PENETRATIONS, I. E., THEY
 ARE CLOSED LOOPS AND, THEREFORE, DO NOT REQUIRE ISOLATION VALVES. DISCUSS EXTENT TO WHICH THESE
 LOOPS HAVE BEEN PROTECTED FROM (A) FALLING OBJECTS, (B) MISSILES, (C) SEISMIC DISTURBANCES, (D)
 WIND LOADS, (E) FIRE, (F) OUTER CREDIBLE MISHAPS. (ANSWER) SEE SECTION VII OF FDSAR AND ANSWER
 TO QUESTION D-15.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + FIRE + REACTOR, LMCR + MISSILE GENERATION AND PROTECTION + REPORT, SAR + SEFOR
 (RE) + EARTHQUAKE + RESPONSE TO NRC QUESTION + DESTRUCTIVE WIND + CONTAINMENT PENETRATION, PIPE

11-029453 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION D-3 - CLOSURE DEVICES OF SODIUM LINES FOR CONTAINMENT ISOLATION

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-029453 *CONTINUED*

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE D-5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THE TWO 10-INCH MAIN SECONDARY SODIUM LINES AND THE TWO 3-INCH AUXILIARY SECONDARY SODIUM LINES HAVE NO CONTAINMENT ISOLATION CLOSURE DEVICES AT THE OUTER CONTAINMENT BARRIER NOR AT ANY OTHER POINT THROUGHOUT THEIR CIRCUITS. DISCUSS THE OPERATIONAL CONSIDERATIONS, THE CONTAINMENT ISOLATION CAPABILITIES UNDER ALL ACCIDENT CONDITIONS AND THE ASSOCIATED RADIOLOGICAL HAZARDS THAT LED TO THE ABOVE DESIGN DECISION. (ANSWER) THEY QUALIFY AS TYPE 3 PENETRATIONS. COVER GAS PRESSURE WILL BE KEPT HIGHER THAN THAT IN PRIMARY LOOPS, SO LEAKAGE WILL BE INTO PRIMARY SYSTEM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + CONTAINMENT PENETRATION, PIPE + VALVES + PIPES AND PIPE FITTINGS

11-029454 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION D-4 - CONTAINMENT PENETRATION DESIGN

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE D-6 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FROM FIGURE VII-3 IN FUSAR, DEPICTING CONTAINMENT PENETRATION DESIGN, IT IS NOTED THAT 13 TABULATED LINES EACH HAVE ONE COMMON PENETRATION SLEEVE WHICH PASSES THROUGH BOTH INNER AND OUTER CONTAINMENT BARRIERS. IT IS NOTED ON PAGE 13 OF SUPPLEMENT 7 THAT THE SILICONE RUBBER DIAPHRAGM SEALS BETWEEN THE PENETRATION SLEEVES AND THE OUTER CONTAINMENT HAVE BEEN REPLACED WITH METALLIC SEALS. SUPPLY A DISCUSSION OF THE NEW PENETRATION DETAILS FOR THE 13 ABOVE MENTIONED LINES, INCLUDING A TYPICAL DRAWING, AND THE METHOD THAT WILL BE EMPLOYED TO VERIFY THE LEAK-TIGHT INTEGRITY OF THE JUNCTION BETWEEN THE PENETRATION SLEEVE AND (A) THE INNER CONTAINMENT AND (B) THE OUTER CONTAINMENT. (ANSWER) FIGURE GIVEN OF TYPICAL SEAL. INTEGRITY DETERMINED AS PART OF ENTIRE INNER CONTAINMENT TEST.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + INTEGRITY + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION + INDEPENDENCE + PENETRATIONS, PRIMARY CONTAINMENT

11-029455 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION D-5 - BELLOWS TEMPERATURES FROM SODIUM LINE CONTAINMENT PENETRATIONS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, 1 FIGURE, PAGE D-6 AND -7 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS EFFECTS OF LOCALIZED PEAK TEMPERATURES EXPERIENCED BY METAL BELLOWS AS A RESULT OF CALROD HEATERS IN THE SODIUM LINE CONTAINMENT PENETRATIONS. TAKE ANTICIPATED PIPE DISPLACEMENT AND COLD STARTUP INTO ACCOUNT WHEN ESTABLISHING PEAK TEMPERATURES. (ANSWER) SEE SUPPL. 7, SECT. 2,3, TO FUSAR. PEAK TEMP. OF 850 F FOR BELLOWS. DESIGN TEMP. IS 1000 F.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + THERMAL CONSIDERATION + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION + CONTAINMENT PENETRATION, PIPE

11-029456 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION D-6 - ROLLER GUIDE SUPPORT

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE D-8 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

SUBMIT DRAWING OF TYPICAL ROLLER GUIDE SUPPORT, SUCH AS WOULD BE USED ON PENETRATION MK-2. (ANSWER) FIGURE GIVEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

GUIDE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + SUPPORT STRUCTURE + CONTAINMENT PENETRATION, PIPE

11-029457 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION D-7 - REFUELING CELL INTEGRITY AND CHEMICAL REACTION THEREIN AFTER ACCIDENT

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, 1 FIGURE, PAGE D-8 AND -9 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS MOST ADVERSE CHEMICAL REACTION CONSIDERED DURING REFUELING CELL DESIGN, AND ABILITY OF

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-029457 *CONTINUED*

REFUELING CELL TO CONTAIN RADIOACTIVE AND NONRADIOACTIVE MATERIALS WITHIN REFUELING CELL SHOULD SUCH AN ACCIDENT OCCUR. (ANSWER) CELL ENVIRONMENT TO BE MAINTAINED AT NOT MORE THAN 10 PPM O₂ OR H₂O. NO FIRES OR CHEMICAL REACTIONS CONSIDERED IN EVALUATING PRESSURE REQUIREMENTS. TO AVOID PRESSURIZING, A 20 INCH DIAMETER RUPTURE DISK IS PROVIDED TO RELIEVE AT DELTA P OF 4 PSI. SEE SECT. 7.2 OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CHEMICAL REACTION + FIRE + REACTOR, LMCR + ACCIDENT, REFUELING + REPORT, SAR + CONTAINMENT INTEGRITY + PRESSURE RELIEF + SEFOR (RE) + PRESSURE, INTERNAL + RESPONSE TO NRC QUESTION

11-029458 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION D-8 - CONSEQUENCES OF LINE BREAK ON CLOSED-LOOP SYSTEMS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

4 PAGES, PAGE D-9 THRU -12 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FROM TABLE VII-1 OF FDSAR, IT IS NOTED THAT THERE ARE IN EXCESS OF 60 INNER CONTAINMENT PENETRATIONS RANGING IN SIZE FROM 30-INCHES IN DIAMETER TO 1/4-INCH IN DIAMETER THAT DO NOT HAVE ISOLATION VALVES IN THE LINES BECAUSE THEY FORM CLOSED LOOPS AND ARE THUS CONSIDERED AN INTERNAL OR EXTERNAL EXTENSION OF THE CONTAINMENT. DISCUSS CONSEQUENCES SHOULD A LINE BREAK AND EXTENT TO WHICH THESE INTERNAL AND EXTERNAL CLOSED SYSTEMS HAVE BEEN PROTECTED FROM (A) FALLING OBJECTS, (B) MISSILES, (C) SEISMIC DISTURBANCES, (D) WIND LOADS AND TORNADOES, (E) FIRES, (F) OTHER CREDIBLE MISHAPS. (ANSWER ON PGS D-9 THRU -12.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + FIRE + REACTOR, LMCR + MISSILE GENERATION AND PROTECTION + CONTAINMENT ISOLATION + REPORT, SAR + CONTAINMENT INTEGRITY + SEFOR (RE) + EARTHQUAKE + RESPONSE TO NRC QUESTION + DESTRUCTIVE WIND + VALVES

11-029460 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION D-10 - BRAZED JOINTS OF COPPER REFRIGERANT LINES

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE D-14 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

REFRIGERANT LINES ARE COPPER TUBING BRAZED TO BOTH END CAPS OF PENETRATION SLEEVE PASSING THROUGH THE OUTER CONTAINMENT BARRIER. DISCUSS ADEQUACY OF BRAZED JOINTS TO WITHSTAND CONCURRENT PRESSURE, TEMPERATURE, AND PIPE REACTION LOADS EXPECTED UNDER ACCIDENT CONDITIONS. (ANSWER) PIPING IS BRAZED TO OUTER CLOSURE PLATE OF PENETRATION SLEEVES AND TO A CLOSURE PLATE ATTACHED THROUGH A BELLOWS JOINT TO INNER END OF PENETRATION SLEEVES. PIPE REACTIONS RELIEVED BY BELLOWS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION + AUXILIARY COOLING SYSTEM + PIPES AND PIPE FITTINGS

11-029461 ALSO IN CATEGORIES 9 AND 18 DATED 1968

RESPONSE TO QUESTION D-11 - INSTRUMENT LEAD PENETRATION

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE D-14 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

SLIDING FRICTION IS THE SOLE MEANS OF HOLDING SHIELDING PLUGS IN PLACE WHEN INSTRUMENT LEADS PASS THROUGH THE PENETRATION. DISCUSS RELATIONSHIP OF THE REFUELING CELL PRESSURE AND FORCE REQUIRED TO SLIDE PLUGS OUT. (ANSWER) WHEN IN USE, UTILITY WALL SLEEVE PLUG WOULD BE REPLACED, WITH CONFIGURATION HELD IN PLACE BY BATTING. NO DESIGN AT PRESENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION + ELECTRICAL CONDUCTION + CONTAINMENT PENETRATION, ELECTRICAL + INSTRUMENTS, MISC.

11-029462 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION D-12 - MULTIPLE QUAD-RING SEALS OF MANIPULATOR

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE D-15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FRICTION LOAD OF MULTIPLE QUAD-RING SEALS IS SOLE MEANS OF HOLDING MANIPULATOR IN PLACE UNTIL THE SLAVE ARM CONTACTS THE CELL WALL. DISCUSS (A) RELATIONSHIP OF REFUELING CELL PRESSURE AND THE FORCE REQUIRED TO MOVE THE MANIPULATOR OUTWARD AND (B) MEANS OF ATTACHMENT OF THE SLAVE ARM AND ITS LOAD CARRYING CAPABILITY WHEN IT CONTACTS THE REFUELING CELL WALL. (ANSWER) SEAL HAS BEEN

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-029462 *CONTINUED*

PROOF TESTED AT 2000 LB AXIAL LOAD WITHOUT SLIPPAGE (EQUIVALENT TO 25 PSIG DELTA P ACROSS CELL BARRIER).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

HOT CELL + REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + SEAL + REMOTE MANIPULATING AND VIEWING + RESPONSE TO NRC QUESTION

11-029463

ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION D-13 - CONNECT DISCONNECT FITTINGS OF BREATHING AIR SYSTEM

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, 1 FIGURE, PAGE D-15 AND -16 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

SUPPLY DRAWINGS OF CONNECT-DISCONNECT FITTINGS UTILIZED IN BREATHING AIR SYSTEM TO PERMIT AN EVALUATION TO BE MADE OF POSSIBILITY OF AIR BEING RELEASED IN INNER CONTAINMENT. (ANSWER) VENDOR DATA SHEETS SHOWN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + PERSONNEL PROTECTIVE DEVICE + REPORT, SAR + AIR + SEFOR (RE) + SYSTEM DESCRIPTION + SEAL + RESPONSE TO NRC QUESTION + VALVES

11-029464

ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION D-14 - CONTAINMENT LEAKAGE TESTS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

4 PAGES, 1 TABLE, PAGE D-17 THRU -20 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE PLANS FOR INNER AND OUTER CONTAINMENT LEAKAGE TESTS. (ANSWER ON PGS D-17 THRU -20.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + TEST, LEAK RATE + CONTAINMENT INTEGRITY + TEST, LEAK LOCATION + SEFOR (RE) + RESPONSE TO NRC QUESTION

11-029465

ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION D-15 - ISOLATION SYSTEMS FOR CONTAINMENT PENETRATIONS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE D-20 THRU -46 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

TABULATE ISOLATION SYSTEMS FOR CONTAINMENT PENETRATIONS, INCLUDING FUNCTIONAL LOCATION, PIPE AND VALVE SIZES, DESIGN CONDITIONS, AND REQUIRED FUNCTIONAL PERFORMANCE. (ANSWER) 25 TABULATION SHEETS GIVEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION

11-029467

ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION F-1 - REACTOR VESSEL SPECIFICATIONS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

6 PAGES, PAGE F-1 THRU -6 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

INDICATE REVISIONS OR EXPANSIONS OF REACTOR VESSEL SPECIFICATIONS. WERE COMPLETE REACTOR VESSEL CALCULATIONS PERFORMED FOR COMPLETE OPERATIONAL MODE OF VESSEL, INCLUDING ANY NEW OR REVISED TRANSIENTS. WAS COMPLIANCE WITH ASME CODE - SECTION III OBTAINED. (ANSWER) SPECS. WERE REVISED 3 TIMES. FINAL IS GIVEN IN APPENDIX A, SUPPLEMENT 16 OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + REACTOR, LMCR + PRESSURE VESSELS + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION

11-029468

ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION F-2 - O-RING FLEXING MARGINS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE F-6 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-029468 *CONTINUED*

HAS YOUR EVALUATION VERIFIED THAT O-RING FLEXING MARGINS ARE SUFFICIENT TO ENSURE A GOOD SEAT AT THE MAXIMUM CLOSURE FLANGE DEFLECTIONS. (ANSWER) VESSEL HEAD AND FLANGES DESIGNED FOR 1500 PSIG. MAX. FLANGE DEFLECTIONS ARE LESS THAN 1 MIL. SPRINGBACK OF SEAL IS 5 AND 6 MILS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + PRESSURE VESSELS + REPORT, SAR + SEFOR (RE) + FLANGE + SEAL + RESPONSE TO NRC QUESTION

11-029469 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION F-3 - BOLTS IN REACTOR VESSEL AFTER ACCIDENT

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, 1 TABLE, PAGE F-6 THRU -8 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FOR BOLTS HOLDING REACTOR VESSEL CUTER HEAD, CENTER HEAD, AND BOTTOM SHIELD PLUG WHICH ARE TO ABSORB ENERGY OF DESIGN BASIS ACCIDENT BLAST BY PLASTIC DEFORMATION (STRETCHING), EVALUATE CRITICAL VELOCITY AND COMPARE IT WITH DEFORMATION VELOCITY OF BOLTS. (ANSWER ON PAGES F-6 THRU - 8.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + DESIGN CRITERIA + REACTOR, LMCR + PRESSURE VESSELS + REPORT, SAR + FASTENER + SEFOR (RE) + RESPONSE TO NRC QUESTION + STRUCTURAL INTEGRITY

11-029470 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION F-4 - PRESSURE VESSEL DESIGN

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE F-9 AND -10 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS DESIGN BASIS FOR CYLINDRICAL SECTION BETWEEN REACTOR VESSEL SUPPORT FLANGE AND HEAD FLANGE AND CYLINDRICAL SECTION OF REACTOR VESSEL BELOW HEAD FLANGE. INCLUDE MAXIMUM OPERATING AND ACCIDENT LOADING CONDITIONS, COMPONENT AND NET STRESSES, AND DESIGN MARGINS FOR THOSE REGIONS. DISCUSS EFFECT OF IMPACT FORCES ON DESIGN MARGINS FOR ABOVE REGIONS FOR REACTOR VESSEL HEAD DROP ACCIDENT. CONSIDER CASES FOR ASYMMETRIC IMPACT OF (A) CENTER HEAD, (B) OUTER HEAD, AND (C) COMBINED CENTER HEAD AND OUTER HEAD. CONSIDER EFFECT ON DESIGN MARGINS OF CORE SUPPORT STRUCTURE FOR CASE WHERE CENTER IMPACTS ON VESSEL SHIELD SECTION ATTACHED TO CENTER HEAD DETACHES AND FALLS AT ANGLE ONTO TOP OF CORE. (ANSWER ON PAGES F-9 AND -10.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + DESIGN CRITERIA + REACTOR, LMCR + PRESSURE VESSELS + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + SUPPORT STRUCTURE

11-029477 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION G-6 - MAINTAINANCE OF CONTAINMENT GAS QUALITY

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, PAGE G-17 THRU -19 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE AND EVALUATE THE INSTRUMENTATION AND PROCEDURES WHICH ENSURE THAT THE DESIRED INERT GASES ARE THE ONLY ONES INTRODUCED INTO CONTAINMENT. (ANSWER ON PGS G-17 THRU -19.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + CONTAINMENT AIR COOLING + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + CONTAINMENT ATMOSPHERE, INERT

11-029487 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION I-1 - SEAL RING

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE I-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

SINCE ENTIRE WEIGHT OF SHROUD AND ITS INTERNALS IS TRANSFERRED TO REACTOR VESSEL WALL THROUGH SEAL RING, DISCUSS CAPABILITY OF TWELVE 3/4 UNC BOLTS TO PREVENT SHROUD AND CORE FROM DROPPING IN EVENT SEAL RING SHOULD FAIL. SURVEILLANCE AND QUALITY CONTROL PROGRAM THAT HAS BEEN OR WILL BE FOLLOWED TO ENSURE THAT RING WILL NOT FAIL AND MEANS WHEREBY IT WILL BE PROMPTLY DETECTED SHOULD IT FAIL, HOW RESIDUAL WELDING STRESSES IN SEAL RING WERE ESTABLISHED IN ORDER TO DETERMINE ITS MAXIMUM WORKING STRESS. (ANSWER) STRESS ON BOLT IS 6100 PSI. CODE ALLOWS 8500. SEAL FAILURE ALLOWS FLOW BYPASS COKE, WHICH IS EASILY DETECTED. RING WAS FORGED - NO WELDS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

3

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-029487 *CONTINUED*

FAILURE, COMPONENT + FABRICATION + REACTOR, LMCR + REPORT, SAR + STRESS + THERMAL MECHANICAL EFFECT + WELDING + FASTENER + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION + CORE COMPONENTS + STRUCTURAL INTEGRITY + QUALITY ASSURANCE + SURVEILLANCE PROGRAM + SYSTEM CAPABILITY

11-029488 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION I-2 - OUTLET SLEEVE INTEGRITY

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE I-2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT DESIGN PROVISIONS HAVE BEEN MADE TO ENSURE THAT MOUNT FAILURE, AND SUBSEQUENT COOLANT BYPASS DUE TO OUTLET SLEEVE DISPLACEMENT, IS A REMOTE POSSIBILITY. DISCUSS CONSEQUENCES OF FAILURE OF SLEEVE MOUNTING. HOW WOULD SUCH A FAILURE BE DETECTED. (ANSWER) FAILURE REQUIRES SHEARING OF 3/4-IN.-OD PIN. NO LOADS ON PIN. FAILURE DETECTED BY SUDDEN INCREASE IN COOLANT FLOWS. CONSEQUENCES ARE FLOW THROUGH CORE REDUCED. MAX. TEMP. STILL SAFELY BELOW BOILING POINT OF SODIUM. SCRAM OCCURS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, COMPONENT + ACCIDENT, CONSEQUENCES + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + CORE COMPONENTS

11-029489 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION I-3 - FASTENERS FOR VESSEL INTERNALS. BOLT LOADS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE I-3 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

IT HAS BEEN NOTED THAT MANY REACTOR VESSEL INTERNALS ARE TO BE ASSEMBLED BY MEANS OF THREADED FASTENERS, MADE FROM SIMILAR MATERIALS AND WITHOUT LUBRICANTS. IT IS PRESUMED THAT STIPULATED TORQUE VALUES FOR VARIOUS FASTENERS IMPLIED A REQUIRED BOLT LOAD. DISCUSS ANTICIPATED VARIATION IN BOLT LOADS FOR GIVEN TORQUE WHEN THREADED MEMBERS ARE BOTH CLEAN, UNLUBRICATED STAINLESS STEEL TYPE 304. CITE EXPERIENCE UPON WHICH ABOVE INFORMATION IS BASED. (ANSWER) VARIATION IN BOLT LOAD WITHIN +/- 25% OF NOMINAL. LOAD VARIATION BASED ON GE TESTS. ALL FASTENERS LOCKED BY PINS, LOCKWIRE, AND WELDING, AND DO NOT DEPEND ON LOAD TO PREVENT UNFASTENING.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + STRESS + STEEL, STAINLESS + FASTENER + SEFOR (RE) + RESPONSE TO NRC QUESTION + CORE COMPONENTS + STRUCTURAL INTEGRITY + LUBRICATION

11-029490 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION I-4 - CONSEQUENCES OF LOOSE PIECES IN PRIMARY COOLANT SYSTEM, PIPING INTEGRITY, ETC

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, PAGE I-3 THRU -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CONSIDERING LARGE NUMBER OF SMALL PIECES SPOT, PLUG, OR TACK WELDED IN PLACE WITHIN CORE ASSEMBLY AND LARGE NUMBER OF BOLTS, NUTS AND PINS, DISCUSS QUALITY ASSURANCE PROGRAM TO MINIMIZE LOSING SUCH PARTS. DISCUSS CONSEQUENCES ASSOCIATED WITH PRESENCE OF LOOSE PIECES IN PRIMARY COOLANT SYSTEM. IF LOOSE PIECES CAN SIGNIFICANTLY AFFECT CORE HEAT TRANSFER, PIPING OR VESSEL INTEGRITY, OR OPERABILITY OR DEVICES WHICH HAVE SAFETY-RELATED FUNCTION (E.G., CHECK VALVES), DISCUSS POSSIBLE NEED FOR DESIGN REMEDIAL MEASURES AND/OR DEVELOPMENT OR SURVEILLANCE MEASURES. (ANSWER) ALL FASTENERS SECURED BY LOCKWIRES OR WELDING. ALSO DISCUSSES PROBABILITY AND CONSEQUENCES OF LOOSE PIECES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, COMPONENT + ACCIDENT, CONSEQUENCES + REACTOR, LMCR + REPORT, SAR + FASTENER + SEFOR (RE) + RESPONSE TO NRC QUESTION + CORE COMPONENTS + OBSTRUCTION + VALVES

11-029491 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION I-5 - INCONEL INTEGRITY

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

4 PAGES, 2 FIGURES, PAGE I-5 THRU -8 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

AVAILABLE TEST DATA ON PERFORMANCE OF INCONEL 718 IN ENVIRONMENT TYPICAL OF THAT EXPECTED IN SEFOR DO NOT APPEAR COMPLETE ENOUGH TO VERIFY THAT PARTS CONSTRUCTED OF THIS MATERIAL (AND ASSOCIATED WELDS) WILL MAINTAIN THEIR INTEGRITY FOR DESIGN LIFE OF CORE. DISCUSS MEASURES TO BE TAKEN TO ASCERTAIN CONTINUED SERVICEABILITY OF PARTS MADE OF INCONEL 718 DURING COURSE OF SEFOR TEST PROGRAM. (ANSWER) SEAL RING FAILURE WILL BE DETECTED BY INCREASED COOLANT FLOW. TORQUE RODS FOR CORE CLAMP TIGHTENED ONLY TO 1/4 OF 0.2% YIELD STRENGTH. MAX. STRESS FROM SEVERE TRANSIENT IS ONLY 1/3 OF YIELD STRENGTH.

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-029491 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

INTEGRITY + REACTOR, LMCR + REPORT, SAR + TENSILE PROPERTY + WELDING + EMBRITTLEMENT + FASTENER + SEFOR (RE) +
STRESS STRAIN DATA + RESPONSE TO NRC QUESTION + INCONEL + CORE COMPONENTS + RADIATION EFFECT

11-029492 ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION I-6 - FAILURES OF AUXILIARY SODIUM INLET LINE CHECK VALVE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, PAGE I-9 THRU -11 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FOR THE AUXILIARY SODIUM INLET LINE CHECK VALVE, DISCUSS TYPE OF FAILURES CONSIDERED IN ITS
DESIGN, MEANS BY WHICH THESE FAILURES WILL BE DETECTED, AND CONSEQUENCES FOLLOWING EACH FAILURE.
BY WHAT MEANS AND HOW FREQUENTLY WILL OPERABILITY OF CHECK VALVE BE VERIFIED. (ANSWER ON PGS I-9
THRU -11.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + FASTENER + TEST, SYSTEM OPERABILITY + SEFOR (RE) + RESPONSE TO NRC QUESTION +
AUXILIARY COOLING SYSTEM + FAILURE MODE ANALYSIS + VALVES

11-029493 ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION I-7 - MARMON COUPLINGS, INTEGRITY
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE I-11 AND -12 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

INLET AND OUTLET AUXILIARY SODIUM LINES, AND COOLANT OVERFLOW LINE INSIDE REACTOR VESSEL, UTILIZE
MARMON TYPE COUPLINGS. DISCUSS CONSIDERATIONS GIVEN TO POSSIBILITY OF COUPLINGS BECOMING LOOSE,
AND DESCRIBE THE CONSEQUENCES OF SUCH A FAILURE. BY WHAT MEANS CAN COUPLING FAILURE BE DETECTED.
(ANSWER ON PGS I-11 AND -12.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + FAILURE + REACTOR, LMCR + REPORT, SAR + FASTENER + SEFOR (RE) + FLANGE + SEAL +
RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM

11-02951d ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION M-9 - LEAK RATE VS TIME FOR CONTAINMENT BARRIERS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, 1 FIGURE, PAGE M-23 AND -24 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FOR THE CALCULATIONAL BASIS FOR RADIOLOGICAL EFFECTS, PROVIDE THE LEAK RATE VERSUS TIME FOR EACH
OF THE CONTAINMENT BARRIERS THAT WERE USED FOR THE DOSE CALCULATIONS PRESENTED IN TABLE D-8.
(ANSWER ON PGS M-23 AND -24.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT ANALYSIS + DOSE + FISSION PRODUCT RELEASE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO
NRC QUESTION + LEAK RATE + OFF SITE

11-029714 ALSO IN CATEGORY 17

DATED 1968

FERMI CHANGE 13 - WAIVED CONTAINMENT GROSS LEAKAGE TEST
U.S. ATOMIC ENERGY COMMISSION

1 PAGE, LETTER TO POWER DEVELOPMENT COMPANY, OCTOBER 21, 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--
COMMONWEALTH ASSOC.

WAIVES ANNUAL CONTAINMENT GROSS LEAKAGE TEST FOR CURRENT SHUTDOWN TO INVESTIGATE OCT. 66 FUEL
DAMAGE INCIDENT. TEST TO BE PERFORMED PRIOR TO RELOADING WHERE THE REACTOR IS LESS THAN \$15
SUBCRITICAL. ***BASIS IS FISSION PRODUCTS IN PRIMARY SODIUM TOTAL LESS THAN 1 CURIE, DECAY HEAT
IS LESS THAN ONE WATT PER SUBASSEMBLY.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM
CHARGE \$2.00)

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + REACTOR, FAST + FERMI (LMFBR) + TEST, LEAK RATE + TECHNICAL
SPECIFICATIONS + TEST INTERVAL + REACTOR, LMFBR

11-030604 ALSO IN CATEGORIES 12 AND 18

DATED 1967

SEFOR FSAR VOL. 1 - REACTOR DESCRIPTION
GENERAL ELECTRIC CO.

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-030609 *CONTINUED*

200 PAGES, FIGURES, TABLES, SEFOR FACILITY DESCRIPTION AND SAFETY ANALYSIS REPORT VOL. 1, JULY 21, 1967, DOCKET 50-231

CONTAINS SECTIONS ON SUMMARY DESCRIPTION, SITE, REACTOR, COOLANT SYSTEM, SHIELDING, CONTAINMENT, SAFEGUARDS WASTE CONTROL, AND INSTRUMENTATION. A 20-MWT SODIUM-COOLED FAST-BREEDER PU02-U23802 FUELED EXPERIMENTAL REACTOR TO GAIN EXPERIENCE WITH SUCH REACTORS AND PARTICULARLY THE DOPPLER COEFF. AND ITS EFFECT ON REACTOR RESPONSE. CONTROLLED BY 10 MOVABLE NICKEL REFLECTORS AND 18 FIXED B4C RODS. CONTAINMENT BY STEEL-LINED REINFORCED-CONCRETE BARRIER SURROUNDED BY WELDED A-212B STEEL VESSEL. SODIUM INLET TEMP., 700 F, OUTLET TEMP., 820 F AT 5000 GPM. DELTA P IN MAIN LOOP, 31-PSI. PUMP DISCHARGE, 55 PSIG. SAFEGUARDS INCLUDE CONTAINMENT INERTING SAFETY VESSEL OF STEEL AROUND REACTOR VESSEL, 4 COOLANT SYSTEMS, AND CORE DISPERSION ASSEMBLY TO PREVENT CRITICAL MASS IF CORE AND VESSEL DISSASSEMBLE.

AVAILABILITY - NKC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

CONTAINMENT DESIGN + CONTAINMENT, MULTIPLE + DOPPLER COEFFICIENT + DOPPLER EFFECT + REACTOR, LMCR + PRESSURE VESSELS + REACTOR DESCRIPTION + REACTOR STABILITY + REPORT, SAR + SITING, REACTOR + SODIUM + ARGON + NITROGEN + INSTRUMENT, CONTROL + WASTE MANAGEMENT + SEFOR (RE) + SHIELDING + SYSTEM DESCRIPTION + EXPERIMENT + CONTAINMENT ATMOSPHERE, INERT + CORE COMPONENTS + PLUTONIUM DIOXIDE + MAIN COOLING SYSTEM + R AND D PROGRAM + ENGINEERED SAFETY FEATURE + REACTOR, LMFR + FUEL ELEMENTS

11-030675 ALSO IN CATEGORY 17 DATED 1969
ULTRASONIC DETECTOR ON SKATES CHECKS NUCLEAR VESSELS
POWER ENGINEERING 73(1), P.68 (JANUARY 1969)

INCLUDED IN DESIGN OF A1 350-500 MW LMFR. PERMANENT TRACKS IN THE SPACE BETWEEN REACTOR VESSEL AND ITS JACKET. CAN BE FITTED WITH TV CAMERA, ULTRASONIC INSPECTION UNIT, WELDING, OR CLEANING HEADS (WEIGHT ABOUT 25 LB). DEVELOPED BY NORTH AMERICAN ROCKWELL FOR APPLICATION IN EXPERIMENTAL AIRCRAFT AND APOLLO USAGE.

REACTOR, BWR + PRESSURE VESSELS + REACTOR, PWR + REACTOR, POWER + WELDING + EXAMINATION + REMOTE MANIPULATING AND VIEWING + R AND D PROGRAM + REACTOR, LMFR + PIPES AND PIPE FITTINGS

11-030748 ALSO IN CATEGORY 9 DATED 1967
ELIAS P + GOLDFUSS G
A PROTOTYPE PORTABLE PUMP AND OXIDE-CLEANUP SYSTEM FOR EBR-11 ROTATING SHIELD PLUG SEALS
ARGONNE NATIONAL LABORATORY
ANL-7379 +. 23 PAGES, 9 FIGURES, AUGUST 1967

INVESTIGATED THE OCCASIONAL STICKING OF TWO, MOTOR-DRIVEN, ROTATING SHIELD PLUGS PREPARATORY TO EXCHANGE OF FUEL AND OR SPECIAL SUBASSEMBLIES CONTAINING IRRADIATION-TEST SPECIMENS. POTENTIAL SOURCES OF STICKING - ACCUMULATIONS OF OXIDE MATERIAL IN THE BISMUTH-TIN ALLOY ROTARY FREEZE SEAL ON EACH PLUG. THIS REPORT DESCRIBES THE MEASURES USED TO REDUCE OXIDATION, MANUAL TECHNIQUES USED FOR BATCH-TYPE REMOVAL OF OXIDE MATERIAL, AND A PORTABLE PUMP AND CLEANUP SYSTEM DESIGNED TO EXPEDITE REMOVAL OF OXIDE MATERIAL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DECONTAMINATION + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + OXIDATION + REACTOR, TEST + INSTRUMENT, TEMPERATURE + REACTOR, ERDA OWNED + SHIELDING + SEAL + PUMPS

11-030845 ALSO IN CATEGORIES 12 AND 17 DATED 1967
POLOMIK EE + RUSSELL CE + QUINN EP
FLUID FLOW AND VIBRATION TESTS ON 1/2-SCALE MODEL OF THE SEFOR REACTOR VESSEL
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.
GEAP-5440+. APRIL 1967, 100P.

TESTS WERE RUN ON THE FLUID FLOW AND VIBRATION CHARACTERISTICS IN A HALF-SCALE MODEL OF THE SEFOR (SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR) REACTOR VESSEL. WATER, AT TEMPERATURES UP TO 265 F, WAS USED TO SIMULATE COOLANT CONDITIONS IN THE SODIUM-COOLED REACTOR. FLOWS WERE VARIED FROM 10 TO 100% RATED REYNOLDS NUMBER IN THE REACTOR. DATA WAS OBTAINED ON RELATIVE CORE FLOW DISTRIBUTION, FLOW PATTERNS, PRESSURE LOSSES, AND VIBRATION CHARACTERISTICS OF THE CANTILEVER-MOUNTED CORE SHROUD. FLUID FLOW RESULTS PROVIDED INFORMATION NECESSARY FOR VERIFICATION OF FINAL DESIGN. VIBRATION RESULTS INDICATED NO SERIOUS EFFECTS OF AMPLITUDE OR STRESS IN THE FULL-SCALE REACTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + MOCKUP + PRESSURE VESSELS + TESTING + WATER + REACTOR, PULSED + SEFOR (RE) + FLOW DISTRIBUTION + VIBRATION ANALYSIS

11-031340 ALSO IN CATEGORY 7 DATED 1968
MOBERLY JW + BARLOW M + GARRISON MC + PLANTING PJ
INTERACTION OF LIQUID SODIUM WITH 304 STAINLESS STEEL
STANFORD RESEARCH INSTITUTE, MENLO PARK, CALIF.

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-031340 *CONTINUED*
TID-24602 +. 28 PAGES, 20 FIGURES, JANUARY 31, 1968

IN PARTICULAR IT HAS BEEN OBSERVED THAT - (1) AFTER EXPOSURE OF THE FOILS AT 1200 F, SEVERE GRAIN BOUNDARY ATTACK OCCURS. (2) SURFACE PRECIPITATES, THOUGHT NOT TO BE SODIUM OR SODIUM OXIDE, ARE OBSERVED ON THE FOILS. (3) THE PRECIPITATES, WHICH MAY CONSIST OF ONE OR MORE COMPOUNDS, ARE ALSO FORMED ALONG AND WITHIN THE GRAIN BOUNDARIES AND HAVE A DEPENDENCE ON ORIENTATION OF THE 304 STAINLESS STEEL GRAINS. (4) ELECTRON DIFFRACTION STUDIES HAVE NOT AS YET IDENTIFIED THE COMPOSITION OF THESE PRECIPITATES. (5) FINALLY, TRANSMISSION ELECTRON MICROSCOPY IS AN EXCELLENT TECHNIQUE FOR FULLY EVALUATING ANY CLEANING PROCEDURE FOR METAL FOILS AFTER LIQUID SODIUM EXPOSURE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.05 MICROFICHE

CHEMICAL REACTION + CLADDING + REACTOR, LMCR + SODIUM + REACTOR, TEST + STEEL, STAINLESS + CORROSION + COOLANT CHEMISTRY + ELECTRON MICROSCOPY + REACTOR, LMCR

11-031443 DATED 1968
COMPELLI FA + WITHOP A + LAURITZEN TA + BUSBUUM HJ
THE EFFECT OF FAST-NEUTRON IRRADIATION ON THE MECHANICAL PROPERTIES OF SOME AUSTENITIC STAINLESS STEELS
GE-SUNNYVALE, CALIF.
2 PAGES, 2 FIGURES, 1 REFERENCE, ANS TRANSACTIONS, 11(1), PAGES 145-146 (JUNE 1968)

ELEVATED-TEMPERATURE TENSILE PROPERTIES OF COMMERCIAL FE-CR-NI ALLOYS WERE EVALUATED SUBSEQUENT TO FAST FLUX IRRADIATION. SPECIMENS OF TYPE-304, -316L, -347, AND INCOLOY-800 IN THE MILL-ANNEALED CONDITION WERE EXPOSED AT 1000 TO 1200 F IN EBR-II AND ACCUMULATED FLUENCES TO 6.7×10^{21} N/CM² (E GREATER THAN 1 MEV). POSTIRRADIATION TESTING WAS CONDUCTED IN AIR AT 1300 F AT A STRAIN RATE OF 0.02 MIN. WITHIN THE LIMITED NEUTRON FLUENCE RANGE INVESTIGATED, ALL MATERIALS EXHIBITED DECREASES IN UNIFORM STRAIN WITH INCREASING FLUENCE. SIGNIFICANT INCREASES IN STRENGTH WERE NOTED FOR ALL MATERIALS. EXTRAPOLATION OF DATA SUGGESTS THAT RADIATION-INDUCED STRENGTHENING IS ANNEALED OUT AT 1600 F.

EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + STEEL, STAINLESS + HIGH TEMPERATURE + NEUTRON + PROPERTY, PHYSICAL + INCOLOY + RADIATION EFFECT

11-031470 DATED 1968
MURPHY WF + STROHM HE
TUBE BURST TESTS ON IRRADIATED EBR-II TYPE 304L STAINLESS-STEEL FUEL CLADDING
ARGONNE NATIONAL LABORATORY
8 PAGES, 7 FIGURES, 14 REFERENCES, NUCLEAR APPLICATIONS 4(4), PAGES 222-9 (APRIL 1968)

TYPE-304L STAINLESS-STEEL CLADDING OF EBR-II FUEL ELEMENTS IRRADIATED TO 10^{22} N/CM² (FASTER THAN 0.1 MEV) HAS GREATER RUPTURE STRENGTH AND MUCH LESS DUCTILITY THAN HAVE UNIRRADIATED CONTROL SPECIMENS. SHAPE OF CURVE OF RUPTURE STRENGTH VS LOCATION ALONG CLADDING IS DIRECTLY RELATED TO FAST-NEUTRON EXPOSURE AND TO IRRADIATION TEMP.

BURST PRESSURE + EBR 1 AND 2 (RE) + CLADDING + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR, LMCR + REACTOR, TEST + STEEL, STAINLESS + FAILURE, TUBING + RADIATION EFFECT

11-031526 DATED 1968
MOTEFF J + EDWARDS JA + SMITH JP
RADIATION EFFECTS ON FIRST-AND SECOND-STAGE CREEP OF INCOLOY-800
GENERAL ELECTRIC, CINCINNATI
2 PAGES, ANS TRANSACTIONS, 11(1), PAGES 148-149 (JUNE 1968)

STUDIED THE EFFECTS OF NEUTRON IRRADIATIONS ON THE PRIMARY- AND SECONDARY-STAGE CREEP PROPERTIES OF INCOLOY-800 AT A TEST TEMPERATURE OF 704 C. THE SPECIMENS WERE IRRADIATED AT A TEMPERATURE OF 540 C TO A FAST-NEUTRON (E GREATER THAN 1 MEV) FLUENCE OF ABOUT 2.4×10^{19} N/CM² IN THE EBR-II FACILITY. INFORMATION ON THE INSTANTANEOUS CREEP RATE AS A FUNCTION OF CREEP TIME WAS OBTAINED BY MEASURING THE STRAIN IN THE GAGE SECTION OF THE SPECIMEN USING A PRECISION CATHETOMETER. IN THE CASE OF THE IRRADIATED SPECIMEN AND FOR ALL THE STRESS LEVELS CONSIDERED, THE CREEP RATE WOULD RAPIDLY DECREASE DURING THE PRIMARY-STAGE PORTION AND APPROACH A STEADY SECOND-STAGE PORTION CREEP VALUE.

EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + CREEP BEHAVIOR + HIGH TEMPERATURE + NEUTRON + INCOLOY + RADIATION EFFECT

11-031972 DATED 1967
JONES RH + PROBERT PB
STEAM GENERATORS - SUMMARY OF THE STATE-OF-TECHNOLOGY
CONF-670413-AND SUPPL. +. 16 PAGES, PAPER 2 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO, CALIFORNIA, APRIL 10-12, 1967. HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY, 1967

ANALYZES VARIOUS TYPES OF STEAM GENERATORS FOR THE LIQUID METAL COOLED FAST BREEDER REACTOR. OPERATING CHARACTERISTICS OF THE ALCO/BLM, AI, AND B AND W PROTOTYPE STEAM GENERATORS ARE TABULATED. CROSS SECTIONS OF EACH ARE SHOWN.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + STEAM GENERATOR

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-031973

DATED 1967

CHAPELOT A + CHAUVIN M + DELISLE JP + DENIELJU G + GOLLION H + GRANDCOLAS J + REYNES L

MAIN TECHNICAL PROBLEMS BY THE CONSTRUCTION AND TESTING OF RAPSODIE

CEA, PARIS

CONF-670413-AND SUPPL. +. 13 PAGES, PAGES 6.1-13 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO, CALIF., APRIL 10-12, 1967. HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY, 1967

SUMMARIZES PROBLEMS ASSOCIATED WITH THE CONSTRUCTION AND TESTING OF RAPSODIE. DISCUSSES USE OF MOCKUPS. QUALITY AND DIMENSIONAL CONTROL ARE OUTLINED. PRESENTS MECHANICAL PROBLEMS ASSOCIATED WITH HIGH TEMPERATURE LOW PRESSURE PIPING OF LIQUID SODIUM. PRESSURE CONTROL OF THE ARGON COVER GAS SYSTEM IS DISCUSSED. THE FUEL HANDLING MACHINE AND ITS ASSOCIATED PROBLEMS ARE OUTLINED. SODIUM PURIFICATION IS DISCUSSED BRIEFLY.

FABRICATION + FRANCE + REACTOR, LMCR + SODIUM + TESTING + ARGON + HIGH TEMPERATURE + PRESSURE, INTERNAL + PIPES AND PIPE FITTINGS + FUEL ELEMENTS

11-031977

DATED 1967

BERGSTROM RN + CHITTENDEN WA + MCCLINTOCK DC

CONSTRUCTION AND DESIGN PROBLEMS OF SODIUM COOLED FAST REACTORS

SARGENT AND LUNDY

CONF-670413-AND SUPPL. +. 12 PAGES, PAGES 6.45-56 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO, CALIF., APRIL 10-12, 1967. HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY, 1967

GIVES REPRESENTATIVE EXAMPLES OF DESIGN DEVELOPMENTS FOR NA-COOLED FAST REACTORS, AND INCLUDES ANALYSIS OF COMPLEX IRREGULAR REINFORCED CONCRETE STRUCTURES WHICH CONTAIN MANY DISCONTINUITIES AND PENETRATIONS AND WHICH ARE DESIGNED TO WITHSTAND INTERNAL PRESSURES THAT ARE ACCOMPANIED BY SIGNIFICANT TEMPERATURE CHANGES.

DESIGN CRITERIA + FABRICATION + REACTOR, LMCR + SODIUM + HIGH TEMPERATURE + STEEL LINER + PRESSURE, INTERNAL + EXPANSION + PIPES AND PIPE FITTINGS + PENETRATIONS, PRIMARY CONTAINMENT

11-032543

ALSO IN CATEGORY 18

DATED 1968

WOODFIELD FW

FAST FLUX TEST FACILITY DEVELOPMENT MONTHLY PROGRESS REPORT, MAY 1968

BATTELLE NORTHWEST

BNWL-603 +. 46 PAGES, 11 FIGURES, 2 TABLES, JUNE 10, 1968

SODIUM-METHANE REACTION RATE IS PRIMARILY DEPENDENT ON GAS VOLUME. ***CR-51 WAS FOUND DOWNSTREAM OF THE IRRADIATED 304-SS PIECE IN THE TEST LOOP. ACTIVITY DROPOFF COINCIDED WITH STEP CHANGE IN VELOCITY. ***EBR-II 304-SS ROD THIMBLE VOLUME INCREASED AS MUCH AS 1.17% FOLLOWING 4.8 X 10¹² NVT AT 792 F. ***CYCLIC STRAIN ALLOWED BY ASME CODE NOT CONSERVATIVE FOR IRRADIATED 304-SS AT 1200 F BECAUSE THE MANSON-HALFORD RELATIONSHIP INDICATES SHORTER FATIGUE LIFE FOR CONDITIONS CAUSING FAILURE IN 30,000 OR MORE CYCLES.

AVAILABILITY - F.W. WOODFIELD, BATTELLE-NORTHWEST, RICHLAND, WASH.

CODES AND STANDARDS + REACTOR, FAST + REACTOR, LMCR + RADIATION DAMAGE + REACTOR, TEST + STEEL, STAINLESS + FFTF (TR) + FAST NEUTRON + CRUD + IRRADIATION TESTING + FAILURE, FATIGUE + R AND D PROGRAM + EXPANSION + REACTOR, LMFBR

11-032629

ALSO IN CATEGORIES 5 AND 17

DATED 1968

EXAMINATION OF MELTED FERMI FUEL

POWER REACTOR DEVELOPMENT COMPANY

55 PAGES, FIGURES, TABLES, APPENDIXES B AND C OF REPORT ON THE FUEL MELTING INCIDENT, DECEMBER 15, 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

AN UNMELTED ELEMENT SHOWED ABOUT 200 PPM HYDROGEN IN CLAD (WELL BELOW 5000 SPALLING LEVEL), HIGH VALUES FROM A PARTICULAR MELT INGOT. OTHERWISE NORMAL. ANOTHER ELEMENT THERMOCOUPLE READING HIGHER THAN A MELTED ELEMENT, WAS ONLY BADLY BOWED. *** MOLTEN FUEL ADHERED TO STEEL CLAD BUT DID NOT PENETRATE. HYDROGEN CONTENT REACHED 4000 PPM WITHOUT CRACKING. ONLY XE-KR FOUND IN COVER GAS. CS, SR, I, BA-LA FOUND IN SODIUM COOLANT. WHILE THERE WAS A COMMON HOLE IN THE WRAPPER CANS BETWEEN MELTED ELEMENT, NO EVIDENCE OF INTERCHANGE OF FUEL. FLOW TESTS AND HEAT TRANSFER ANALYSIS SHOWED ABOUT 3% NOMINAL FLOW IN TWO MELTED ELEMENTS, 30 AND 7% FLOW IN DAMAGED ADJACENT ELEMENTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + SODIUM + EMBRITTLEMENT + HYDRIDE + REPORT, OPERATIONS ANALYSIS + HEAT TRANSFER ANALYSIS + ZIRCALOY + FISSION PRODUCT RETENTION + EXAMINATION + FLOW BLOCKAGE + STRUCTURAL INTEGRITY + REACTOR, LMFBR

11-032816

ALSO IN CATEGORY 18

DATED 1968

CONTAINMENT DESIGN

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-7 +. 14 PAGES, SEFOR, PSAR, SUPPLEMENT 7, FEB. 14, 1968, DOCKET 50-231

A CONDENSATION OF THE MATERIAL IN THE PSAR SECTION VII ON CONTAINMENT DESIGN FOR THE PURPOSE OF

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-032816 *CONTINUED*
CLARITY BY ELIMINATING REPETITION.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

CONTAINMENT DESIGN + REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + REPORT, PSAR + REACTOR, LMFBK

11-032836 ALSO IN CATEGORIES 5 AND 9 DATED 1969
RYDEN J
NONDESTRUCTIVE TESTING OF SMALL DIAMETER, STAINLESS STEEL FUEL CLAD TUBING
BATTTELLE-NORTHWEST, RICHLAND
BNWL-SA-2275 + CNF-690311-1 +. 20 PAGES, FROM NATIONAL SPRING CONFERENCE OF THE AMERICAN SOCIETY FOR
NONDESTRUCTIVE TESTING, LOS ANGELES, 1969

DISCUSSES THE TECHNIQUES USED BY THE PACIFIC NORTHWEST LAB. TO INSPECT FFTF FUEL CLADDING FOR DEFECTS. DESCRIBES THE THREE DIFFERENT TESTERS USED TO INSPECT THIS TUBING. THEY ARE A SINUSOIDAL EDDY CURRENT TESTER, AN ULTRASONIC PULSE-ECHO TESTER, AND AN ULTRASONIC TESTER THAT USES A SEND-RECEIVE SYSTEM OF TRANSDUCERS. MICROGRAPHS, STRIP CHARTS, AND STATISTICAL DATA OBTAINED FROM THE INSPECTION OF 20,000 FEET OF TUBING ARE PRESENTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + TESTING + STEEL, STAINLESS + FFTF (TR) + INSTRUMENT CALIBRATION + TEST, NONDESTRUCTIVE + R AND D PROGRAM + TUBING

11-032843 ALSO IN CATEGORIES 7 AND 18 DATED 1969
BULBA E + UNDERHILL D
APPLICATION OF A PEBBLE BED HEAT-SINK TO THE FAST FLUX TEST FACILITY CLEANUP SYSTEM
HARVARD UNIVERSITY
NYU-841-17 +. 19 PAGES, 2 FIGURES, 9 REFERENCES, MARCH 1969

CONTAINS MATHEMATICAL EQUATIONS FOR GAS TEMPERATURES FOLLOWING SODIUM RELEASE AND BURNING INTO FFTF VENTILATION SYSTEM. ALLOWS FOR VARIABLE INLET GAS TEMPERATURE BUT DOES NOT ACCOUNT FOR SODIUM DEPOSITION PRIOR TO RELEASES. EXPERIMENTAL WORK ON THE LATTER NEEDED, ALSO ON GAS-FLOW SCHEMES AND ON BEST STRATEGY FOR PLACING SYSTEM IN OPERATION IN ACCIDENTS OF VARIOUS MAGNITUDES.

DEPOSITION + REACTOR, FAST + FIRE + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + SODIUM + REACTOR, TEST + VENTILATION SYSTEM + FFTF (TR) + HEAT SINK + MATHEMATICAL TREATMENT + R AND D PROGRAM + GAS + REACTOR, LMFBK + PACKED BED + THERMAL TRANSIENT

11-032880 ALSO IN CATEGORIES 2 AND 18 DATED 1968
RESPONSE TO QUESTION 3 - TORNADO - REFUELING CELL OPERATION
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 1 PAGES, PAGE 12 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

WHAT SAFETY PRECAUTIONS ARE PLANNED FOR ENSURING SAFE OPERATION IN THE REFUELING CELL WITH REGARD TO POTENTIAL TORNADO DAMAGE. ***ANSWER FOR DESIGN PROVISIONS, SEE PSAR, SUPPLEMENT 1, PP. 3-61 THRU 3-66, FDSAR, VOL. 1, PAGE 7-1, PAGE 7-9, PAGE 7-16, AND FDSAR SUPPLEMENT 17, ANSWER TO QUESTIONS A-11, D-2, AND D-8.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REFUELING + FUEL STORAGE + STORAGE CONTAINER + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + DAMAGE + DESTRUCTIVE WIND + REPORT, PSAR + REACTOR, LMFBK

11-032885 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 8 - OPERATION OF CONTAINMENT DOORS AND HATCHES
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 5 PAGES, PAGE 21 THRU 25 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DISCUSS DESIGN AND OPERATION OF DOORS AND HATCHES OF CONTAINMENT USED FOR PASSAGE OF EQUIPMENT AND PERSONNEL. ***ANSWER ON PAGES 21 THRU 25.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + REACTOR, FAST + REACTOR, LMCR + CONTAINMENT AIR LOCK + CONTAINMENT EQUIPMENT HATCH + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBK

11-032886 ALSO IN CATEGORY 18 DATED 1968
RESPONSE TO QUESTION 9 - ACTIONS REQUIRED TO CLOSE TYPE 3 SYSTEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-10 +. 1 PAGE, PAGE 26 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-032886 *CONTINUED*

ENUMERATE HOW MANY MORE OF THE 78 TYPE-3 CONTAINMENT PENETRATIONS ARE DEPENDENT UPON OPERATIONS CARRIED OUT BY OPERATOR IN ORDER TO QUALIFY AS A CLOSED SYSTEM. ***SEE COMMENT K-2 OF PSAR, SUPPLEMENT 17.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

11-032887 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION 10 - CONSEQUENCES OF TYPE 3 LINE FAILURES

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-20 +. 8 PAGES, PAGES 27 THRU 34 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

PROVIDE DETAILED DISCUSSION OF 51 PENETRATIONS RANGING IN SIZE FROM 1/2 TO 2 INCHES, WITH REGARD TO FUNCTION AND CONSEQUENCES OF FAILURES. ***ANSWER ON PAGES 27 THRU 34.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + PENETRATIONS, PRIMARY CONTAINMENT

11-032888 ALSO IN CATEGORIES 5 AND 18 DATED 1968

RESPONSE TO QUESTION 11 - DYNAMIC LOADS FROM FRED TESTS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-20 +. 6 PAGES, 1 TAB., PAGES 35 THRU 40 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DISCUSS DYNAMIC LOADINGS IMPOSED BY MAXIMUM PLANNED TRANSIENT ON THE CORE SUPPORT STRUCTURE AND REACTOR VESSEL. ***ANSWER ON PAGES 35 THRU 40.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + CONTAINMENT VESSEL LOADING + REACTOR, FAST + REACTOR, LMCR + PRESSURE VESSELS + REACTOR TRANSIENT + STRESS + THERMAL MECHANICAL EFFECT + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + CORE COMPONENTS + REACTOR, LMFBR

11-032894 ALSO IN CATEGORIES 6 AND 9 AND 18 DATED 1968

RESPONSE TO QUESTION 17 - DESIGN OF PRESSURE VESSELS IN FRED

GENERAL ELECTRIC

50231-10 +. 2 PAGES, 1 FIGURE, PG. 53 AND 54 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

DESCRIBE BLOWOUT PATCH AND ACCUMULATOR ON THE EXCURSION DEVICE. *** ANSWER ON PAGES 53 AND 54.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + INSTRUMENTS, MISC.

11-032905 ALSO IN CATEGORIES 5 AND 18 DATED 1969

RESPONSE TO QUESTION 1 - SMALL LEAKS IN MAIN COOLANT SYSTEM

GENERAL ELECTRIC

50231-26 +. 3 PAGES, PAGES 1 THRU 3 OF SEFOR, PSAR, SUPPLEMENT 22, JAN. 24, 1969, DOCKET 50-231

DISCUSS CONSEQUENCES AND SYSTEM RESPONSE TO SMALL LEAKS IN THE MAIN COOLANT SYSTEM. *** ANSWER ON PAGES 1 THRU 3.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + MAIN COOLING SYSTEM + LEAK + REACTOR, LMFBR + PIPES AND PIPE FITTINGS

11-032921 ALSO IN CATEGORIES 6 AND 18 DATED 1968

RESPONSE TO QUESTION 6 - CORE COMPONENT LOADINGS DURING TRANSIENT TESTS

GENERAL ELECTRIC, SAN JOSE

50231-21 +. 5 PAGES, 1 FIGURE, 1 TABLE, PAGES 57 THRU 62 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL INFORMATION ON DYNAMIC LOADINGS IMPOSED ON CORE SUPPORT STRUCTURE DURING PROMPT TRANSIENT TEST PROGRAM. *** ANSWER ON PAGES 57 THRU 62.

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-032921 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

11-032924 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION 9 - SURVEILLANCE PROGRAM - REACTOR VESSEL
GENERAL ELECTRIC, SAN JOSE

50231-21 +. 2 PAGES, 1 TABLE, PAGES 67 AND 68 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL INFORMATION ON REACTOR VESSEL AND INTERNALS SURVEILLANCE PROGRAM. *** ANSWER ON PAGES 67 AND 68 PLUS PSAR SUPPLEMENT 16, SECT. 7.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + PRESSURE VESSELS + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + CORE COMPONENTS + SURVEILLANCE PROGRAM + REACTOR, LMFBR

11-032925 ALSO IN CATEGORY 18 DATED 1968

RESPONSE TO QUESTION 10 - VESSEL - THERMAL INSULATION

GENERAL ELECTRIC, SAN JOSE

50231-21 +. 3 PAGES, 1 FIGURE, PAGES 69 THRU 72 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL DETAILS ON DESIGN AND INSTALLATION OF THERMAL INSULATION ON CORE SUPPORT SHROUD. DISCUSS POTENTIAL CONSEQUENCES WITH REGARD TO FLOW BLOCKAGE IF ONE OF PLATES SHOULD COME LOOSE. *** ANSWER ON PAGES 69 THRU 72.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + PRESSURE VESSELS + THERMAL MECHANICAL EFFECT + REACTOR, PULSED + SEFOR (RE) + THERMAL INSULATION + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

11-032926 ALSO IN CATEGORIES 5 AND 18 DATED 1968

RESPONSE TO QUESTION 11 - SODIUM IN CORE - EFFECT ON MHA

GENERAL ELECTRIC, SAN JOSE

50231-21 +. 1 PAGE, PAGE 73 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

DISCUSS EFFECT OF SODIUM IN CORE REGION ON MHA ANALYSIS. *** ANSWER ON PAGE 73.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + REACTOR STABILITY + SODIUM + ACCIDENT, HYPOTHETICAL + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + FUEL ELEMENTS

11-032986 DATED 1969

LAURITZEN T + WITHOP A + WOLFF UE

SMELLING OF AUSTENITIC STAINLESS STEELS UNDER FAST NEUTRON IRRADIATION AT ELEVATED TEMPERATURES

GENERAL ELECTRIC CO., SUNNYVALE, CALIF. + GENERAL ELECTRIC CO., PLEASANTON CALIF.

4 PAGES, 1 FIGURE, 2 TABLES, 5 REFERENCES, NUCLEAR ENGINEERING AND DESIGN, 9(2), PAGES 265-268 (FEBRUARY 1969)

A DISCUSSION OF STUDIES ON VOLUME-INCREASE PHENOMENA OBSERVED IN HIGHLY IRRADIATED AUSTENITIC STAINLESS STEELS IS GIVEN. VOLUME INCREASES ARE RELATED TO THE PRODUCTION OF POLYHEDRAL CAVITIES. THE LIKELY SOURCE OF SWELLING IS A COMBINATION OF THE EFFECTS OF TWO REACTIONS - THE HELIUM-PRODUCING (N,ALPHA) REACTION, AND THE VACANCY-PRODUCING DISPLACEMENT REACTION. IT IS BELIEVED THAT HELIUM ATOMS SO GENERATED STABILIZE THE CAVITY NUCLEI.

EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + RADIATION DAMAGE + REACTOR, TEST + STEEL, STAINLESS + HIGH TEMPERATURE + HELIUM + NEUTRON + INCOLOY

11-033016 ALSO IN CATEGORY 17 DATED 1969

CHANGE 14 - PRIMARY SHIELD TANK ATMOSPHERE

U.S. ATOMIC ENERGY COMMISSION

5 PAGES, LETTER TO M.C. BECKMAN, PRDC, MARCH 26, 1969, DOCKET 50-16 TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

ALLOWS MAINTENANCE OPERATION WITH AIR IN PRIMARY SHIELD TANK FOR 10,000 HR OR LESS TO REPLACE BOOT SEAL. PROVIDES FOR GRAPHITE PURGING. *** DRL SAFETY EVALUATION CONSIDERS BORON LOSS FROM GRAPHITE SHIELDING, SHIELD TANK COLLAPSE, AND LEAKAGE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-033016 *CONTINUED*
REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + PRESSURE VESSELS + TECHNICAL SPECIFICATIONS + SEAL + CONTAINMENT ATMOSPHERE, INERT + SAFETY EVALUATION + REACTOR, LMFBR

11-033336 DATED 1968
SCHMIDT G
A METHOD FOR CALCULATING THE DESIGN STRENGTH OF FUEL CANS FOR LIQUID-METAL-COOLED FAST REACTORS
KERNFORSCHUNGSZENTRUM, KARLSRUHE (WEST GERMANY). INSTITUT FUER REAKTORENTWICKLUNG
EUR-FNR-570 + KFK-808 +. 145 PAGES, REF., JULY 1968

GIVES METHOD FOR CALCULATING THE SIZE OF THE NECESSARY FISSION GAS PLENUM AS A FUNCTION OF WALL THICKNESS AND BURN-UP FOR THE FUEL ROD CANS OF LIQUID-METAL-COOLED OXIDE-FUELED FAST REACTORS FOR GIVEN MATERIAL PROPERTIES (ESPECIALLY CREEP BEHAVIOR) AND TEMPERATURE AND POWER DISTRIBUTION AT THE INDIVIDUAL ROD. THE TECHNIQUE IS APPLIED TO VARIOUS EXAMPLES BY MEANS OF DIGITAL COMPUTER PROGRAMS, AND THE RESULTS ENABLE CONCLUSIONS TO BE DRAWN CONCERNING OPTIMUM FUEL-ELEMENT DESIGN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

COMPUTER PROGRAM + EURATOM + FABRICATION + CLADDING + CREEP PROPERTY + GERMANY + FISSION PRODUCT RETENTION + KARLSRUHE + CREEP + REACTOR, LMFBR + FUEL ELEMENTS

11-033446 ALSO IN CATEGORY 18 DATED 1968
LMFBR PROGRAM PLAN. VOLUME 6 - CORE DESIGN
LMFBR PROGRAM OFFICE, ARGONNE NATIONAL LAB.
WASH-1106 +. 400 PAGES, FIGURES, TABLE, AUGUST 1968

CONTENTS - SUMMARIES OF CORE DESIGN SYNTHESIS, TECHNOLOGY, UTILIZATION OF FACILITIES WITH MAJOR PORTION OF REPORT DEVOTED TO DESCRIPTION OF THE TASKS IN THE TECHNICAL PROGRAM. APPENDICES DISCUSS CHARACTERISTICS OF CORE AND FUEL DESIGNS, REQUIREMENTS AND CAPABILITIES FOR IRRADIATION TESTING, CURRENT PROJECTS RELATED TO CORE-DESIGN SYNTHESIS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, BREEDER + REACTOR, FAST + SODIUM + REACTOR COOLANT + DESIGN STUDY + IRRADIATION TESTING + CORE COMPONENTS + R AND D PROGRAM + REACTOR, LMFBR + JACOBS + FUEL ELEMENTS

11-033555 ALSO IN CATEGORIES 2 AND 18 DATED 1968
LMFBR PROGRAM PLAN. VOL. 2 - PLANT DESIGN
ARGONNE NATIONAL LAB., LMFBR PROGRAM OFFICE
WASH-1102 +. 197 PAGES, 8 FIGURES, 17 TABLES, AUGUST 1968

DISCUSSES DEVELOPMENT OF PLANT CONCEPTS AND TECHNOLOGY FOR PLANT DESIGN. GIVES DETAILS OF OVERALL PLAN. MAIN TOPICS - (1) PLANT DESIGN DEVELOPMENT STUDIES, (2) GENERAL STUDIES - SITING, ETC., (3) DEMONSTRATION PLANT PROGRAM, AND (4) COMPILATION, ASSESSMENT, AND DISSEMINATION OF DESIGN INFORMATION

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ECONOMICS + DESIGN STUDY + INFORMATION RETRIEVAL + R AND D PROGRAM + REACTOR, LMFBR + JACOBS

11-033953 ALSO IN CATEGORY 18 DATED 1967
WOODFIELD FW
FAST FLUX TEST FACILITY DEVELOPMENT MONTHLY PROGRESS REPORT, NOVEMBER 1967
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-639 +. 58 PAGES, DECEMBER 10, 1967

ONE OF A SERIES. TOPICS - COOLANT SYSTEMS DEVELOPMENT, EQUIPMENT, INSTRUMENTATION, METALS AND MATERIALS, AND PROCESS AND MATERIALS SPECIFICATIONS. FISSION-PRODUCT DIFFUSION THROUGH 304 SS CLADDING AFTER 1052 HR AT 1200 F WAS LESS THAN 1 MIL. 304 SS IRRADIATED AT 1.7 X 10²² ND/NT AT 1000 F REVEALED MARKED INCREASE IN 0.2% OFFSET YIELD STRENGTH AND A DECREASE IN DUCTILITY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + METAL + REACTOR, TEST + FISSION PRODUCT TRANSPORT + STEEL, STAINLESS + FFTF (TR) + CORROSION + COOLANT CHEMISTRY + DESIGN STUDY + MATERIAL + IRRADIATION TESTING + BATTELLE NORTHWEST + EQUIPMENT + R AND D PROGRAM + REACTOR, LMFBR + RADIATION EFFECT

11-033965 ALSO IN CATEGORY 18 DATED 1968
NASH CR + PECKINPAUGH CL
FAST FLUX TEST FACILITY EVALUATION OF CONCEPTUAL FEATURES FOR THE REACTOR REFUELING SYSTEM
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-683 +. 37 PAGES, MARCH 1968

PRESENTS RESULTS OF EVALUATION OF ALTERNATIVE METHODS OF FUEL HANDLING. THE SODIUM-COOLED-CASH

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-033965 *CONTINUED*

CONCEPT APPEARS MOST PROMISING BECAUSE OF CAPACITY TO REMOVE DECAY HEAT, NO INTERFACE PROBLEMS ON REACTOR SYSTEM AS DOES A GAS COOLING SYSTEM, AND EXCELLENT HEAT TRANSFER PROPERTIES OF SODIUM WHICH ALLOW GREATER STRETCH CAPABILITY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + REFUELING + FUEL HANDLING MACHINE + REMOTE MANIPULATING AND VIEWING + R AND D PROGRAM + REACTOR, LMFBR

11-033977 ALSO IN CATEGORIES 5 AND 18 DATED 1967

DENIELOU G + ROZENHOLC M + STORRER F + VAUTREY + VELLENEUVE J

PHENIX - MAIN SAFETY CRITERIA - CONSEQUENCES FOR THE REACTOR DESIGN

CEA-CEN/CADARACHE/DRP, CEA-CEN/SACLAY/DRP, EURATOM-CEN/CADARACHE/DRP

12 PAGES, 3 FIGURES, 4 REFERENCES, PAPER PRESENTED AT THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

DISCUSSES INFLUENCE OF SAFETY CONSIDERATIONS ON THE DESIGN OF PHENIX PROTOTYPE. DISCUSSES CORE COOLING FOR VARIOUS ACCIDENTS, CONTAINMENT OF FISSION PRODUCTS, AND MEANS FOR DETECTING LOCAL REACTIVITY AND TEMP. MEASUREMENTS.

AVAILABILITY - F. STORRER, EURATOM-CEN/CADARACHE/DRP

DESIGN CRITERIA + FRANCE + SODIUM + DESIGN STUDY + SAFETY PRINCIPLES AND PHILOSOPHY + REACTOR, LMFBR

11-033991 ALSO IN CATEGORY 18 DATED 1967

FIELD JH + JOHNSON ML + NOVAK PE

EVALUATION OF THE EFFECT OF DESIGN AND OPERATING VARIABLES ON SEFOR FUEL AND FUEL CLADDING

GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.

GEAP-5309 +. 72 PAGES, REFERENCES, DECEMBER 1967

FOURTEEN CAPSULE EXPERIMENTS WERE IRRADIATION-TESTED TO INVESTIGATE DESIGN PARAMETERS AND EXPECTED OPERATING CONDITIONS ON SEFOR FUEL. TESTS WITH REFERENCE DESIGN CLADDING THICKNESS AND FUEL-CLADDING GAPS SUBJECTED TO EXPECTED STEADY-STATE AND POWER OSCILLATION CONDITIONS RESULTED IN NO CLADDING DISTORTION. SIMILAR SPECIMENS SUBJECTED TO TREAT TRANSIENTS TO NEAR FUEL MELTING EXPERIENCED MINOR CLADDING RIDGES AT FUEL PELLET INTERFACES. CLADDING MAINTAINED ITS INTEGRITY DURING ALL TESTS INCLUDING NON-STANDARD CONDITIONS OF OPERATION ABOVE MELTING OF FUEL, REDUCED FUEL-CLADDING GAPS, AND SODIUM INGRESS INTO FUEL PRIOR TO TRANSIENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + FUEL BURNUP + REACTOR, LMCR + REACTOR, SAFETY RESEARCH + TREAT (PRR) + IRRADIATION TESTING + REACTOR, PULSED + SEFOR (RE) + STRUCTURAL INTEGRITY + PERFORMANCE LIMIT + FUEL ELEMENTS

11-033993 ALSO IN CATEGORY 18 DATED 1968

SODIUM-COOLED REACTORS - FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. TWENTY-SIXTH QUARTERLY REPORT, FEB.-APR. 1968

GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIFORNIA

GEAP-5631 +. 121 PAGES, REFERENCES, JUNE 1968

ONE OF A SERIES OF REPORTS ON R-AND-D PROGRAM TO DEVELOP FAST REACTORS EMPLOYING CERAMIC FUELS (MIXED PLUTONIUM-URANIUM OXIDE). TOPICS - FUEL-BEHAVIOR STUDIES, PERFORMANCE EVALUATION, FAST-FLUX IRRADIATION TESTING, AND CLADDING DEVELOPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CERAMICS + CLADDING + FUEL BURNUP + OXIDE + PLUTONIUM + URANIUM + IRRADIATION TESTING + PERFORMANCE LIMIT + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

11-033994 ALSO IN CATEGORY 18 DATED 1968

SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. TWENTY-FIFTH QUARTERLY REPORT, NOV. 1967-JANUARY 1968

GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.

GEAP-5584 +. 112 PAGES, REFERENCES, MARCH 1968

ONE OF A SERIES OF REPORTS ON R-AND-D PROGRAM TO DEVELOP FAST REACTORS EMPLOYING CERAMIC FUELS (MIXED PLUTONIUM-URANIUM OXIDE). TOPICS - FUEL-BEHAVIOR STUDIES, PERFORMANCE EVALUATION, FAST-FLUX IRRADIATION TESTING, AND CLADDING DEVELOPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CERAMICS + CLADDING + FUEL BURNUP + OXIDE + PLUTONIUM + URANIUM + IRRADIATION TESTING + PERFORMANCE LIMIT + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-033995 ALSO IN CATEGORY 18 DATED 1967
SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. TWENTY-FOURTH QUARTERLY REPORT, AUGUST-OCTOBER 1967
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.
GEAP-5541 +. 135 PAGES, REFERENCES, DECEMBER 1967

ONE OF A SERIES OF REPORTS ON R-AND-D PROGRAM TO DEVELOP FAST REACTORS EMPLOYING CERAMIC FUELS (MIXED PLUTONIUM-URANIUM OXIDE). TOPICS - FUEL-BEHAVIOR STUDIES, PERFORMANCE EVALUATION, FAST-FLUX IRRADIATION TESTING, AND CLADDING DEVELOPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CERAMICS + CLADDING + FUEL BURNUP + OXIDE + PLUTONIUM + URANIUM + IRRADIATION TESTING + PERFORMANCE LIMIT + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

11-033996 ALSO IN CATEGORY 18 DATED 1966
SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 19, MAY-JULY 1966
GENERAL ELECTRIC COMPANY, SAN JOSE, CALIF.
GEAP-5196 +. 95 PAGES, REFERENCES, AUGUST 1966

ONE OF A SERIES OF REPORTS ON R-AND-D PROGRAM TO DEVELOP FAST REACTORS EMPLOYING CERAMIC FUELS (MIXED PLUTONIUM-URANIUM OXIDE). TOPICS - FUEL-BEHAVIOR STUDIES, PERFORMANCE EVALUATION, FAST-FLUX IRRADIATION TESTING, AND CLADDING DEVELOPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CERAMICS + CLADDING + FUEL BURNUP + OXIDE + PLUTONIUM + URANIUM + IRRADIATION TESTING + PERFORMANCE LIMIT + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

11-034453 DATED 1968
PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON SODIUM TECHNOLOGY AND LARGE FAST REACTOR DESIGN
ARGONNE NATIONAL LABORATORY
ANL-7520 (PT. 1) +. 579 PAGES, FIGURES, TABLES, REFERENCES, NOVEMBER 1968

THIS REPORT PRESENTS THE PROCEEDINGS OF THE FOURTH OF A SERIES OF TOPICAL CONFERENCES SPONSORED BY ARGONNE NATIONAL LABORATORY AND THE USAEC ON SPECIFIC TOPICS RELATED TO THE DEVELOPMENT OF FAST POWER-BREEDER REACTORS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + REACTOR, FAST + SODIUM + STEEL, STAINLESS + HIGH TEMPERATURE + SAFETY PRINCIPLES AND PHILOSOPHY + STRESS RUPTURE + REACTOR, LMFBR

11-034669 ALSO IN CATEGORIES 6 AND 18 DATED 1969
ROSSIN AD + GLASSNER A
REACTOR DEVELOPMENT PROGRESS REPORT
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
ANL-7553 +. 123 PAGES, FIGURES, TABLES, REFERENCES, FEB. 1969

ONE OF A SERIES OF SIMILAR REPORTS. TOPICS - LMFBR DEVELOPMENT STUDIES, FAST REACTORS, GENERAL DEVELOPMENT STUDIES, NUCLEAR SAFETY, FUEL-MELTDOWN STUDIES WITH TREAT, CORE STRUCTURAL SAFETY, AND CONTAINMENT RESPONSE TO HIGH-ENERGY EXCURSION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL MELTDOWN + REACTOR, SAFETY RESEARCH + REACTOR, RESEARCH + TREAT (PRR) + CONTAINMENT INTEGRITY + EXCURSION, LARGE + REACTOR, PULSED + PRESSURE, INTERNAL + CORE COMPONENTS + STRUCTURAL INTEGRITY + R AND D PROGRAM + REACTOR, LMFBR

11-034673 ALSO IN CATEGORIES 7 AND 18 DATED 1969
PATRIARCA P + RUCKER DJ
FUELS AND MATERIALS DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1968
OAK RIDGE NATIONAL LAB., TENN.
ORNL-4390 +. 308 PAGES, FIGURES, TABLES, REFERENCES, MAY 1969

ONE OF A SERIES OF SIMILAR REPORTS. TOPICS - FUELS AND CLADDINGS, COOLANTS, RADIATION DAMAGE TO STRUCTURAL MATERIALS, TECHNIQUES OF FABRICATION AND TESTING, ENGINEERING PROPERTIES OF REACTOR MATERIALS, DEVELOPMENT OF FAST REACTOR OXIDE FUELS, AND LMFBR FUELS, FISSION-GAS RELEASE, ZIRCONIUM METALLURGY, ALKALI-METAL CORROSION, TUNGSTEN METALLURGY, AND CLAD MATERIALS FOR SPACE HEAT SOURCES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-034673 *CONTINUED*

FABRICATION + CLADDING + REACTOR, FAST + FISSION GAS RELEASE + ZIRCONIUM + DRNL + CORROSION + TUNGSTEN + IRRADIATION TESTING + DIRECT ENERGY CONVERSION DEVICES + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

11-035871 ALSO IN CATEGORY 18 DATED 1967
PRELIMINARY STUDIES FOR A REFERENCE CONCEPTUAL DESIGN OF A 1000 MWE LMFBR
ARGONNE NATIONAL LABORATORY
CEND-322 +. 200 PAGES, FIGURES, TABLES, REFERENCES, DECEMBER 1967

DESCRIBES THE CONCEPTUAL DESIGN OF THE HEAT-GENERATION SYSTEM FOR A 1000-MWE LMFBR. THE CONCEPT IS BASED ON ESTIMATED TECHNOLOGY THAT MAY BE ATTAINABLE BETWEEN 1975 AND 1985. THE CORE SELECTED FOR ADDITIONAL DESIGN STUDY AND REFINEMENT IS CYLINDRICAL, HAS A TWO-ZONE SEED REGION, 2 FT HIGH AND ABOUT 9 FT IN DIAMETER, WITH A 1.5-FT AXIAL BLANKETS AND A 1-FT RADIAL BLANKET. THE FUEL IS MIXED U AND PU CARBIDES, CLAD IN THIN STAINLESS-STEEL TUBES. FOR REFUELING, AN UNDER-THE-PLUG SCHEME WAS ADOPTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

FUEL HANDLING + REACTOR DESCRIPTION + DESIGN STUDY + REFUELING + CORE COMPONENTS + R AND D PROGRAM + REACTOR, LMFBR + FUEL ELEMENTS

11-036465 ALSO IN CATEGORY 18 DATED 1967
EXPLOSION CONTAINMENT IN THE PROTOTYPE FAST REACTOR SINGLE TANK SYSTEM - FURTHER MODEL EXPERIMENTS
SAFEX/P.53 +. 29 PAGES, 20 FIGURES, 6 TABLES, 3 REFERENCES, DECEMBER 1967

FURTHER EXPLOSION TESTS ON 1/25TH SCALE MODEL OF SINGLE TANK PROTOTYPE DESCRIBED CHECKING EFFECTS OF DETONATION OF TNT. DAMAGE RESULTS DESCRIBED FOR ROUND 4, ASSEMBLY D, ROUND 5, ASSEMBLY D, AND ROUND 6 ASSEMBLY D.

AVAILABILITY - UNITED KINGDOM ATOMIC ENERGY AUTHORITY, 11 CHARLES II STREET, LONDON, S.W. 1, ENGLAND

REACTOR, BREEDER + CONTAINMENT + REACTOR, FAST + REACTOR, LMCR + PRESSURE VESSELS + UNITED KINGDOM + EXPLOSIVE, CONVENTIONAL + PFR (LMFR) + MODEL TESTING

11-036466 DATED 1965
EXPLOSION CONTAINMENT IN THE PROTOTYPE FAST REACTOR SINGLE TANK SYSTEM
SAFEX/P.67 +. 40 PAGES, 33 FIGURES, 4 TABLES, 6 REFERENCES, AUGUST 8, 1965

DESCRIBES RESULTS OF FIRST TEST ON 1/16TH-SCALE MODEL. CHECKING EFFECTS OF DETONATION OF TNT. ROUND-10 RESULTS TABULATED.

AVAILABILITY - UNITED KINGDOM ATOMIC ENERGY AUTHORITY, 11 CHARLES II STREET, LONDON S.W.1, ENGLAND

REACTOR, BREEDER + CONTAINMENT + REACTOR, FAST + REACTOR, LMCR + UNITED KINGDOM + EXPLOSIVE, CONVENTIONAL + PFR (LMFR) + MODEL TESTING

11-036467 ALSO IN CATEGORY 18 DATED 1968
STATIC TESTS ON 1/16TH SCALE MODEL OF PFR VAULT ROOF
SAFEX/P.68 +. 25 PAGES, 20 FIGURES, 2 REFERENCES, 1968

RESULTS GIVEN FOR TEST OF SINGLE-TANK MODEL MADE WITH REINFORCED CONCRETE AND BUILT TO CLOSELY RELATE ALL ASPECTS TO SCALED-DOWN QUANTITIES AND DIMENSIONS. DRAWING AND DESCRIPTION OF MODEL GIVEN. TESTS SHOWED MODEL VAULT ROOF WILL SUPPORT 59 TONS (15,000 TONS FULL SCALE) AT A DEFLECTION OF 0.53 INCH (8.5 INCHES FULL SCALE).

AVAILABILITY - UNITED KINGDOM ATOMIC ENERGY AUTHORITY, 11 CHARLES II STREET, LONDON, S.W. 1, ENGLAND

REACTOR, BREEDER + CONTAINMENT + REACTOR, FAST + REACTOR, LMCR + UNITED KINGDOM + PFR (LMFR) + MODEL TESTING + STRUCTURAL ANALYSIS, DYNAMIC

11-036468 ALSO IN CATEGORIES 5 AND 8 DATED 1966
EXPLOSION CONTAINMENT IN THE PROTOTYPE FAST REACTOR SINGLE TANK SYSTEM
SAFEX/P.69 +. 30 PAGES, 25 FIGURES, 3 TABLES, 3 REFERENCES, JANUARY 1966

DESCRIBES RESULTS OF SECOND TEST ON 1/16TH-SCALE MODEL, CHECKING EFFECTS OF DETONATION OF TNT. RESULTS OF ROUND 11 AGREED WITH THOSE OF ROUND 10. ROUND-11 TEST SHOWED LIMITED DAMAGE.

AVAILABILITY - UNITED KINGDOM ATOMIC ENERGY AUTHORITY, 11 CHARLES II STREET, LONDON, S.W. 1, ENGLAND

REACTOR, BREEDER + CONTAINMENT + REACTOR, FAST + REACTOR, LMCR + UNITED KINGDOM + EXPLOSIVE, CONVENTIONAL + PFR (LMFR) + MODEL TESTING

11-036469 DATED 1966

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-036469 *CONTINUED*

THE STRENGTH OF A PFR TYPE FUEL SUB-ASSEMBLY WRAPPER SUBJECTED TO AN INTERNAL EXPLOSION
SAFEX/P.71 +. 15 PAGES, 9 FIGURES, 3 TABLES, FEBRUARY, 1966.

A POTENTIAL HAZARD IS REDUCED FLOW THROUGH FUEL WRAPPER CAUSED BY PARTIAL BLOCKAGE. LOCAL BOILING AND BURNOUT COULD LEAD TO EJECTION OF LIQUID SODIUM FROM THE WRAPPER AND MELTING OF FUEL. LIQUID SODIUM COMING IN CONTACT WITH LIQUID FUEL COULD CAUSE SODIUM VAPOUR EXPLOSION. IF A WRAPPER COULD CONTAIN THE LARGEST CREDIBLE EXPLOSION WITHOUT DISTORTION, THERE WOULD BE NO HAZARD TO REMAINDER OF CORE. THIS REPORT DESCRIBES EXPLOSION TESTS ON WRAPPERS AND DAMAGE RESULTS. WRAPPERS WERE RUPTURED BY THE EXPLOSIONS. ENERGY RELEASE IN SODIUM VAPOUR EXPLOSION NEEDS TO BE BETTER DEFINED. AVAL: UNITED KINGDOM ATOMIC ENERGY AUTHORITY, 11 STREET, LONDON, S.W. 1, ENGLAND

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + SODIUM + TEST, DESTRUCTIVE + UNITED KINGDOM + EXPLOSIVE, CONVENTIONAL + METAL, LIQUID + PFR (LMFR) + CORE COMPONENTS + FUEL ELEMENTS

11-036473 ALSO IN CATEGORY 7

DATED 1969

PROPOSAL ON VENTED CONTAINMENT FOR A LIQUID METAL FAST BREEDER REACTOR (LMFBR)
ARGONNE NATIONAL LABORATORY

ANL-7410 +. 445 PAGES, PGS. 216-323, TABLES, FIGURES, REFERENCES, JANUARY 1969

THIS IS A PROPOSAL FOR ALLEVIATING ACCIDENT CONSEQUENCES BY VENTING THE SECONDARY CONTAINMENT ATMOSPHERE THROUGH A HEAT SINK FILTERING MEDIA TO A STANDBY CONTROL VOLUME. ENOUGH OF A STUDY HAS BEEN MADE TO ASSURE THAT REASONABLY PRACTICAL DESIGN OF HEAT SINK-FILTER AND AUXILIARY CONTAINMENT BUILDING CAN BE OBTAINED WHICH WILL RESULT IN A SIGNIFICANT REDUCTION IN DESIGN PRESSURE FOR A WHOLE CONTAINMENT SYSTEM. FURTHER DETAILED STUDY OF THE TWO HEAT SINKS WHICH WERE ASSUMED, COUPLED WITH REFINEMENTS IN THE CURRENT NUMERICAL DYNAMIC ANALYSIS, IS NECESSARY BEFORE A SPECIFIC DESIGN CAN BE REALIZED AND ITS PERFORMANCE PROVED. EXPERIMENTAL STUDIES OF BEHAVIOR OF SAND-GRAVEL BEDS WITH VARIABLE INLET GAS TEMPERATURES ARE RECOMMENDED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + CONTAINMENT, PRESSURE VENTING + EXPLOSION + PLUTONIUM + HEAT SINK + REACTOR, LMFBR

11-036616 ALSO IN CATEGORY 18

DATED 1969

COLLINS CG + MUTEFF J + CHANDLER BA

EVALUATION OF THE POTENTIAL OF SELECTED ALLOYS FOR USE AS A FUEL CLADDING MATERIAL IN AN LMFBR
GENERAL ELECTRIC CO., CINCINNATI, OHIO. MISSILE AND SPACE DIV.

GEMP-573(REV.) +. 312 PAGES, FIGURES, TABLES, REFERENCES, JUNE 1969

THE EVALUATION IS BASED ON REVIEWS OF THE DATA ON THE SODIUM CORROSION RESISTANCE, NUCLEAR CHARACTERISTICS, IRRADIATION EFFECTS, AND CREEP OR CREEP-RUPTURE PROPERTIES. THE MATERIALS CONSIDERED INCLUDE THE ALLOYS OF IRON, NICKEL, COBALT, VANADIUM, NIOBIUM, MOLYBDENUM, TANTALUM, AND TUNGSTEN. IT IS CONCLUDED THAT THE ALLOYS OF IMMEDIATE POTENTIAL ARE THE AUSTENITIC STEELS AND THE Fe-Cr-Co-Ni ALLOYS, AND THAT THOSE WITH DEVELOPMENT POTENTIAL OFFERING HIGHER CREEP STRENGTHS INCLUDE THE NICKEL ALLOYS WITH LESS THAN ABOUT 50% NICKEL, AND THE COBALT, VANADIUM, MOLYBDENUM, AND NIOBIUM ALLOYS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

ALLOY + COBALT + CLADDING + METAL + MOLYBDENUM + SODIUM + NICKEL + STEEL, STAINLESS + IRON + MANITOBA + TANTALUM + CORROSION + TUNGSTEN + REVIEW + VANADIUM + CREEP + REACTOR, LMFBR + DUCTILITY + FUEL ELEMENTS

11-036627 ALSO IN CATEGORY 17

DATED 1969

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT APRIL-MAY 1969

ARGONNE NATIONAL LABORATORY

ANL-7577 +. 206 PAGES, FIGURES, TABLES, JULY 7, 1969

EBK-2 OPERATIONS - A SMALL SODIUM FIRE OCCURRED WHILE A SAMPLE WAS BEING TAKEN FROM THE PRIMARY SYSTEM. A STEADY DRIPPING OF SODIUM FROM THE SAMPLING PUMP WAS THE CAUSE. MARK-1A FUEL PINS SHORTENED DURING IRRADIATION - CAUSE IS BEING INVESTIGATED. ***LMFBR DEVELOPMENTS ARE SUMMARIZED. ***NUCLEAR SAFETY - DISCUSSES REACTOR KINETICS STUDIES, TREAT OPERATIONS, CHEMICAL REACTIONS AND ENERGY, AND ENGINEERED SAFETY FEATURES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CHEMICAL REACTION + REACTOR, FAST + FIRE + REACTOR, LMCR + METAL WATER REACTION + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR KINETICS + REACTOR, RESEARCH + SODIUM + REACTOR, TEST + TREAT (PRR) + IRRADIATION TESTING + REACTOR, PULSED + R AND D PROGRAM + ENGINEERED SAFETY FEATURE + REACTOR, LMFBR + FUEL ELEMENTS

11-036779 ALSO IN CATEGORY 17

DATED 1969

MATTHEWS KR + HENRY KF

LOCATION AND REPAIR OF A LEAK IN THE DOWNREAY FAST REACTOR PRIMARY CIRCUIT

10 PAGES, 13 FIGURES, JOURNAL OF THE BRITISH NUCLEAR ENERGY SOCIETY, 8(3), PAGES 168-177 (JULY 1969)

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-036779 *CONTINUED*

DESCRIBES METHODS USED TO DETECT, LOCATE, AND REPAIR THE LEAK. THE NAK COOLANT WAS DISCHARGED COMPLETELY, AND A HELIUM-GAS PRESSURE TEST STARTED. LEAK WAS SMALL, SO, TO INCREASE LEAK RATE, EACH OF THE 24 HEAT EXCHANGERS WAS MOVED 4 IN. TO OPEN THE LEAK AND OBTAIN A BIGGER LEAK RATE TO FACILITATE LOCATION. THIS ASSOCIATED LEAK WITH CIRCUIT 10A OR 10B. A BUBBLE-NOISE TECHNIQUE WITH MICROPHONES CLAMPED ON PIPES WAS USED TO LOCATE LEAK MORE PRECISELY. A CRACK IN A FAULTY WELD WAS FOUND. DETAILED DESCRIPTION OF REPAIR IS GIVEN.

REACTOR, FAST + MAINTENANCE AND REPAIR + FAILURE, PIPE + REACTOR, TEST + UNITED KINGDOM + WELDS + NAK + FAILURE, FABRICATION ERROR + TEST, LEAK LOCATION + METAL, LIQUID + DOUNREAY (LMFBR) + MAIN COOLING SYSTEM + LEAK + OPERATING EXPERIENCE + REACTOR, LMFBR

11-036825

DATED 1969

OAKLEY UJ

FFTF REACTOR VESSEL CONCEPT EVALUATION SUMMARY

BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-1011 +. 200 PAGES, FIGURES, TABLES, JULY 1969

TWO CONCEPTS EMBODYING PROPOSED DESIGN OF THE FFTF REACTOR PRESSURE WERE COMPARATIVELY EVALUATED FOR SELECTION OF THE CONFIGURATION DEMONSTRATING THE GREATER OVERALL ADVANTAGE. THE TWO CONCEPTS, THE ANNULAR INLET AND PIPED BOTTOM INLET, DIFFER PRIMARILY IN LOCATION OF INLET NOZZLES, METHOD OF VESSEL WALL COOLING, AND CONTAINMENT OF SYSTEM INLET PRESSURE. PRINCIPAL AREAS OF THE STUDY INVOLVED METHODS OF FABRICATION, EASE OF INSPECTION, THERMAL BALANCE, INDIVIDUAL DESIGN FEATURES, ACCIDENT CONTAINMENT CAPABILITIES, SHOCK AND VIBRATION RESPONSES, HEAT TRANSPORT AND EMERGENCY COOLING FACTORS, AND COST. RESULTS OF THE DETAILED EVALUATIONS WERE CONCLUDED, FROM THE STANDPOINT OF OVERALL ADVANTAGES, TO FAVOR THE PIPED-BOTTOM-INLET CONCEPT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + PRESSURE VESSELS + CONTAINMENT, SHOCK GENERATION AND PROTECTION + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + NOZZLE + EMERGENCY COOLING

11-036957

ALSO IN CATEGORIES 5 AND 8

DATED 1969

UKAEA SAFETY RESEARCH BULLETIN ISSUE NUMBER 4

UKAEA, AUTHORITY HEALTH & SAFETY BRANCH, RISLEY
UKAEA SAFETY RESEARCH BULLETIN ISSUE NO. 4 +. 50 PAGES, 62 REFERENCES, 1969

PROGRESS REVIEW OF UKAEA WORK IN NUCLEAR REACTOR SAFETY. SUMMARIZED ARE STUDIES PERTAINING TO THE SAFETY OF WATER COOLED REACTORS, LIQUID METAL FAST BREEDER REACTORS, AND THE FRACTURE OF CONCRETE AND STEEL PRESSURE VESSELS. INCLUDED IN THE WATER REACTOR STUDIES ARE COOLANT DEPRESSURIZATION, NATURAL CONVECTION HEAT TRANSFER, HEAT TRANSFER DURING DRYOUT CONDITIONS, BUBBLE GROWTH, ALUMINUM-WATER INTERACTION, AND RELIABILITY OF EMERGENCY CORE-COOLING SYSTEMS. INCLUDED IN THE LIQUID METAL FAST BREEDER REACTOR STUDIES ARE HEAT TRANSFER AND PRESSURE GENERATION FROM CORE DEBRIS FOLLOWING A MELTDOWN, CONTAINMENT OF RADIOACTIVITY, CONSEQUENCES OF A SODIUM FIRE, AND PRESSURE EFFECTS DUE TO A SODIUM FIRE.

AVAILABILITY - HER MAJESTY'S STATIONERY OFFICE, P.O. BOX 569, LONDON S.E.1, ENGLAND

ACCIDENT ANALYSIS + PRESSURE VESSELS + THERMAL ANALYSIS + WATER + CONCRETE, PRESTRESSED + CORE MELTDOWN + REACTOR, LWR + METAL, LIQUID + HEAT TRANSFER ANALYSIS + BLOWDOWN + BUBBLE + EMERGENCY COOLING + PRESSURE TRANSIENT + REACTOR, LMFBR + UKAEA + JACOBS

11-037182

ALSO IN CATEGORY 6

DATED 1967

1000 MWE LIQUID METAL FAST BREEDER REACTOR FOLLOW-ON STUDY - REPORT OF TASKS II AND III. A REFERENCE CONCEPTUAL DESIGN OF A 1000 MWE LMFBR. VOLUME II. STATIC DESIGN AND PERFORMANCE ANALYSES

COMBUSTION ENGINEERING, INC., WINDSOR, CONNECTICUT
CEND-337(VOL.2) +. 121 PAGES, FIGURES, TABLES, REFERENCES, 1967

THE 1000 MWE LIQUID METAL FAST BREEDER REACTOR FOLLOW-ON STUDY IS AN EXTENSION OF 1000 MWE DESIGN STUDIES PERFORMED PREVIOUSLY FOR THE ATOMIC ENERGY COMMISSION. (1) THE OBJECTIVE OF THE FOLLOW-ON PROGRAM IS TO PROVIDE INFORMATION WHICH WILL BE OF ASSISTANCE TO THE AEC IN THE FURTHERANCE OF THE LIQUID METAL FAST BREEDER REACTOR (LMFBR) PROGRAM PLAN. THE SCOPE OF THE WORK FOR COMBUSTION ENGINEERING IS IN THE AREA OF THE HEAT GENERATION SYSTEM OF THE NUCLEAR STEAM SUPPLY SYSTEM. THE OBJECTIVE OF THIS STUDY OF THE HEAT GENERATION SYSTEM HAS BEEN TO IDENTIFY THE RESEARCH AND DEVELOPMENT PROGRAMS WHICH ARE REQUIRED TO ACHIEVE THE SELECTED DESIGN BY 1980, OR AS SPECIFIED BY THE CONTRACTOR FOR A YEAR BETWEEN 1975 AND 1985.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + DESIGN STUDY + REFUELING + SYSTEM DESCRIPTION + REACTOR, LMFBR

11-038785

DATED 1967

BOEHM H + SCHMEIDER H

STRESS RUPTURE AND CREEP BEHAVIOR OF AUSTENITIC CR-NI STEELS IN THE PRESENCE OF NA

ARGONNE NATIONAL LABORATORY
ANL-TRANS-594 +. 14 PAGES, REFERENCES, 1967

CATEGORY 11
CONTAINMENT OF NUCLEAR FACILITIES

11-038785 *CONTINUED*

STUDIED THE INFLUENCE OF LIQUID SODIUM ON THE CREEP- AND STRESS-RUPTURE PROPERTIES OF TWO
AUSTENITIC STEELS AT 700 C. THE TESTS WERE CARRIED OUT ON SODIUM FILLED TUBES (X8CRNiMOVNB 1613
AND NICK 3020) UNDER UNIAXIAL STRESS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + SODIUM + STEEL + STRESS + STEEL, STAINLESS + CREEP BEHAVIOR + EMBRITTLEMENT + CORROSION +
STRESS RUPTURE

11-039017

DATED 1965

KNILEY DJ

LARGE SGR CALANDRIA VESSEL

ATOMICS INTERNATIONAL

NAA-SR-MEMO-11324 + 71 PAGES, APRIL 21, 1965

A LIMITED EXPERIMENTAL STRESS ANALYSIS WAS CONDUCTED ON A TYPICAL SECTION OF THE CALANDRIA VESSEL
UPPER STAYED HEAD. INTERIOR AND PERIPHERAL SECTIONS OF THE HEAD WERE INCORPORATED INTO THE TEST
FIXTURE TO PERMIT DETERMINATION OF STRESS AND DEFLECTION CHARACTERISTICS AT THE PERIPHERAL AND
INTERIOR LOCATIONS. DUE TO TIME LIMITATIONS, ONLY A MINIMUM OF TESTING WAS POSSIBLE. NO
ELEVATED TEMPERATURE TESTING WAS PERFORMED AND TESTING WAS LIMITED TO THE TEST SECTION
CORRESPONDING TO A TYPICAL INTERIOR REGION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + PRESSURE VESSELS + STRESS ANALYSIS + CREEP BEHAVIOR + PRESSURE,
INTERNAL + STRESS STRAIN DATA

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-004440 DATED 1964
KASPIC WA
EXPERIMENTAL TESTING OF CORE COMPONENT HANDLING EQUIPMENT AND HANDLING TECHNIQUES FOR HALLAM NUCLEAR POWER
FACILITY
ATOMICS INTERNATIONAL
91 PAGES, 41 FIGURES, 13 REFERENCES- SEPTEMBER 15, 1964, NAA-SR-7785- OTS

SATISFACTORY OPERATION OF THE CORE COMPONENT HANDLING EQUIPMENT AND HANDLING TECHNIQUES FOR THE HALLAM NUCLEAR POWER FACILITY WAS ACCOMPLISHED AND CONFIRMED BY A PROGRAM OF TESTING. THE TEST RESULTS INDICATED THAT THE OPERATIONAL TECHNIQUES WERE SAFE, AND THAT THE EQUIPMENT IS DEPENDABLE THROUGHOUT ITS EXPECTED RANGE OF OPERATIONS.

CONTAINMENT, PRESSURE VENTING + FUEL HANDLING + HALLAM (LMR) + REACTOR, LMCR

12-005262 ALSO IN CATEGORIES 10 AND 17 DATED 1965
LIQUID-METAL-COOLED REACTORS - EBR-II - OPERATIONS
ARGONNE NATIONAL LABORATORY
2 PAGES- ANL-7017 (REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT), PAGES 1 AND 2 OF CHAPTER I- MARCH 18, 1965- CFSTI

DURING A SITE-WIDE POWER FAILURE, THE 400-KW DIESEL GENERATOR STARTED, LOADED NORMALLY, AND THEN STOPPED. THIS FAILURE OF THE DIESEL WAS ACCOMPANIED BY A LOSS OF ANNUNCIATION IN THE CONTROL ROOM. INVESTIGATION ALSO REVEALED THAT THE DIESEL CANNOT BE RESTARTED AFTER A FAILURE OF THIS KIND UNTIL THE DIESEL-CONTROL SWITCH IN THE CONTROL ROOM HAS BEEN PLACED IN OFF MOMENTARILY AND BACK TO STANDBY. LACK OF THIS INFORMATION PREVENTED MANUAL RESTARTING OF THE DIESEL DURING THE POWER OUTAGE, CAUSES AND REMEDIAL ACTION.

CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + EMERGENCY POWER, ELECTRIC + FAILURE + REACTOR, LMCR

12-005263 ALSO IN CATEGORIES 9 AND 17 DATED 1965
LIQUID-METAL-COOLED REACTORS - EBR-II - EXAMINATION OF FAILED OSCILLATOR ROD AND THIMBLE
ARGONNE NATIONAL LABORATORY
2 PAGES- ANL-7017 (REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT), PAGES 2 AND 3 OF CHAPTER I- MARCH 18, 1965- CFSTI

THE FAILED OSCILLATOR ROD AND THIMBLE WAS EXAMINED, AND 27 BALLS FROM A BALL BEARING WERE FOUND. THE OSCILLATOR ROD WAS JAMMED BY 19 BALLS IN THE LOWER PORTION OF THE GUIDE SLEEVE. VERY HEAVY SCORING WAS OBSERVED IN TWO OF THE THREE LOWER GUIDES ON THE OSCILLATOR ROD AND IN THE CORRESPONDING GUIDE SLOTS OF THE GUIDE TUBE.

EBR 1 AND 2 (RE) + FAILURE + REACTOR, LMCR + OSCILLATOR, REACTIVITY + FAILURE, SCRAM MECHANISM

12-005807 ALSO IN CATEGORIES 6 AND 17 DATED 1965
EBR-II
ARGONNE NATIONAL LABORATORY
6 PAGES, 2 TABLES- MARCH 1965, ANL-7028, SECTION I, PP. 1-6 OF ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT- CFSTI

EBR-II REINVESTIGATION OF THE FAILURE OF THE DIESEL GENERATOR. CONTROL ROD CALIBRATION, SMALL LEAKS IN SEAL OF PERSONNEL AIR LOCK, OTHER MINOR FAILURES.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + CONTROL ROD WORTH + EBR 1 AND 2 (RE) + FAILURE + REACTOR, FAST + REACTOR, LMCR + CONTAINMENT AIR LOCK + GENERATOR, DIESEL

12-006308 DATED 1960
LEEDY WS
LIQUID METAL FUEL REACTOR EXPERIMENT - INVESTIGATION OF PRECLEANING AND DECONTAMINATION METHODS
THE BABCOCK AND WILCOX COMPANY
32 PAGES, 9 FIGURES, 4 TABLES- FEBRUARY, 1960, BAW-1094- CFSTI

IN THE EXPERIMENTAL LIQUID METAL FUEL REACTOR (LMFRE), THE STEEL SURFACE EXPOSED TO THE FUEL SOLUTION MUST BE CLEANED PRIOR TO SERVICE, AND RADIOACTIVE CONTAMINANTS MUST BE REMOVED FROM THE SYSTEM AFTER SERVICE. TO DETERMINE EFFECTIVE PROCEDURES, CROLOY 2-1/4 SPECIMENS CONTAINING AN ARTIFICIALLY INDUCED OXIDE SCALE WERE TREATED WITH VARIOUS SOLVENTS, AFTER WHICH THE CORROSION RATES WERE DETERMINED. IN ADDITION, BISMUTH SOLUBILITY IN VARIOUS SOLVENTS WAS DETERMINED, AND FISSION PRODUCT AND CORROSION PRODUCT SOLUBILITY WAS INVESTIGATED.

DECONTAMINATION + REACTOR, LMCR

12-006488 ALSO IN CATEGORY 17 DATED 1962
MUMFORD J
LEAK CHECK OF MODERATOR ELEMENTS IN THE HNPFC REACTOR
ATOMICS INTERNATIONAL
10 PAGES, 1 REFERENCE- DECEMBER 15, 1962, NAA-SR-MEMO-7837- CFSTI

AN INDUCTION PROBE WAS USED TO TRAVERSE REACTOR PROCESS CHANNELS THUS, LEAK CHECKING THE THREE

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-006488 *CONTINUED*

MODERATOR ELEMENTS SURROUNDING PROCESS CHANNEL. FIFTY-EIGHT PROCESS CHANNELS WERE TRAVERSED TO CHECK THE MAXIMUM NUMBER OF MODERATOR ELEMENTS (129). TWELVE ELEMENTS COULD NOT BE CHECKED SINCE THE PROCESS CHANNELS AROUND THESE ELEMENTS ARE NOT ACCESSIBLE THROUGH THE LOADING FACE SHIELD. THE INDUCTION COIL CONSISTED OF A PRIMARY COIL AND TWO SECONDARY COILS, WITH AN OSCILLATOR CONNECTED TO THE PRIMARY COIL. THE MAGNITUDE OF THE INDUCED VOLTAGE IN THE SECONDARY COILS WAS AN INDICATION OF THE CONDUCTIVITY OF THE SURROUNDINGS. THE SECONDARY COILS WERE CONNECTED TO A SANBURN RECORDER AND THE SIGNAL RECORDED AS THE PROBE TRAVERSED EACH CHANNEL. EXAMINATION OF THE RECORDED TRACES INDICATE THAT THERE WERE NO LEAKY MODERATOR ELEMENTS IN THE REACTOR.

FAILURE, COMPONENT + CONTAINMENT, PRESSURE VENTING + CLADDING + GRAPHITE + HALLAM (LMR) + REACTOR, LMCR + SODIUM + TEST, COMPONENT + TEST, PREOPERATIONAL

12-006817 ALSO IN CATEGORIES 5 AND 11 DATED 1965

BOULANGER JR

SNAP-10A--PIPING STRESS ANALYSIS

ATOMICS INTERNATIONAL

30 PAGES- JANUARY 22, 1965, NAA-SR-MEMO-9817- CFSTI

THIS REPORT CONTAINS THE STRESS ANALYSIS OF THE SNAP 10A PIPING SYSTEM. THIS SYSTEM IS MADE UP OF VARIOUS SIZES AND SHAPES OF STAINLESS STEEL TUBING WELDED TOGETHER INTO A NAK CONTAINMENT NETWORK. STAINLESS STEEL IS USED BECAUSE OF ITS STABILITY AT THE OPERATING TEMPERATURES AND ITS COMPATIBILITY WITH THE REMAINDER OF THE SYSTEM. BACKGROUND INFORMATION REGARDING THE CONFIGURATION AND MATERIALS USED, THE LOADS AND THEIR SOURCES AND THE ANALYTICAL VERIFICATIONS ARE PROVIDED. THIS ANALYSIS VERIFIES THE STRUCTURAL INTEGRITY OF THE SYSTEM DURING BOTH THE SNAP 10A THERMAL AND VIBRATION ENVIRONMENTS.

INTEGRITY + REACTOR, LMCR + SNAP 10A (SR) + REACTOR, SPACE + STEEL, STAINLESS + STRESS ANALYSIS + MAIN COOLING SYSTEM + PIPES AND PIPE FITTINGS

12-007550 ALSO IN CATEGORIES 5 AND 7 DATED 1965

BIERY JC + KENNEDY RM

MATHEMATICAL AND EXPERIMENTAL ANALYSIS OF TEMPERATURE DISTRIBUTIONS IN STORED IRRADIATED LAMPRE CAPSULES

LOS ALAMOS SCIENTIFIC LABORATORY

18 PAGES, 14 FIGURES, 12 REFERENCES- CHEMICAL ENGINEERING PROGRESS SYMPOSIUM SERIES 61(57)- 322-339 (1965)- HEAT TRANSFER-BOSTON, AMERICAN INSTITUTE OF CHEMICAL ENGINEERS

TEMPERATURE PROFILES IN STORED IRRADIATED LAMPRE (LOS ALAMOS MOLTEN PLUTONIUM REACTOR EXPERIMENT) FUEL CAPSULES WERE SIMULATED BOTH MATHEMATICALLY AND EXPERIMENTALLY. THE MATHEMATICAL ANALYSIS PRODUCED SOLUTION EQUATIONS FROM THE FOUR COUPLED SECOND-ORDER DIFFERENTIAL EQUATIONS BY AN ITERATIVE GENERATION PROCEDURE. THE GENERATED EQUATIONS WERE THEN USED IN AN ITERATIVE COMPUTER PROGRAM WHICH PRODUCED THE CONVERGENT TEMPERATURE PROFILES BY A CALCULATION AND CORRECTION TECHNIQUE. THE ITERATIVELY GENERATED EQUATIONS, COUPLED WITH THE COMPUTER PROGRAM, PRODUCED TEMPERATURES WHICH AGREED VERY SATISFACTORILY WITH THE EXPERIMENTAL SIMULATION RESULTS.

REACTOR, BREEDER + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + FUEL STORAGE + HEAT TRANSFER, CONDUCTION + HEAT GENERATION, INTERNAL + FUEL ELEMENTS

12-007670 ALSO IN CATEGORIES 6 AND 17 DATED 1965

LIQUID-METAL-COOLED REACTORS

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

87 PAGES, TABLES, FIGURES- ANL-7105, OCTOBER 20, 1965, PP. 1-35 OF ARGONNE NATIONAL LABORATORY REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, SEPTEMBER, 1965

EBR-11 - FUEL IRRADIATED PAST 0.8 ATOM PERCENT BURNUP SHOWS AN ACCELERATING RATE OF VOLUME INCREASE WITH BURNUP. FUEL VOLUME INCREASES OVER 8 PERCENT WILL REQUIRE PLASTIC STRAIN OF CLADDING. FUEL WILL BE EXAMINED AFTER EACH OPERATING CYCLE. CP-5 URANIUM DIOXIDE FUEL WAS OPERATED BRIEFLY WITH A CENTERLINE TEMPERATURE OF 2400 DEGREES CENTIGRADE WITH CP-5 AT MAXIMUM (4.2 MW) POWER. CLADDING WAS SEPARATED FROM FUEL WITH TANTALUM RODS IN ARGON ATMOSPHERE. TUNGSTEN - RHENIUM THERMOCOUPLES MEASURED THE CENTERLINE TEMPERATURE.

CRITICAL ASSEMBLY + DOPPLER EFFECT + EBR 1 AND 2 (RE) + FUEL BURNUP + FUEL INTEGRITY + IN PILE EXPERIMENT + INSTRUMENT, IN CORE + REACTOR, LMCR + MEASUREMENT + FFTF (TR) + IN PILE EXPERIMENT + IRRADIATION TESTING + ZPPK (CAF)

12-010356 ALSO IN CATEGORIES 11 AND 18 DATED 1965

SAUVAGE M + GUMUCHIAN M

1000-MWE REACTOR DESIGN STUDY OF A HANDLING CELL

COMMISSARIAT A L ENERGIE ATOMIQUE, CADARACHE, FRANCE, CENTRE D ETUDES NUCLEAIRES

OKNL-TR-988 + PNK/SEPR/65.008 +. 41 PAGES, FEBRUARY 18, 1965, JCL, \$4.60 FS, \$1.43 MF

THIS REPORT REPRESENTS A SUPPLEMENTARY CHAPTER TO THE 1000 MW REACTOR DESIGN AS PRESENTED IN GENEVA (CEA REPORT 2554). IT DESCRIBES A DESIGN MAKING USE OF A CELL INSTEAD OF ROTATING PLUGS TO CARRY OUT HANDLING. THE PURPOSE OF A STUDY OF THIS VARIANT IS TO PROPOSE A REACTOR DESIGN TO SERVE AS A BASIS FOR A MORE-DETAILED STUDY WITHIN THE SCOPE OF A DESIGN PROJECT. THUS, THE PROBLEMS POSED BY THE REACTOR BLOCK, STORAGE, HANDLING AND THE HANDLING CELL, HAVE BEEN EXAMINED, WITH THE STRESS PLACED ON THE LAST TWO POINTS, WHICH PRESENT THE GREATEST DIFFICULTIES. MUCH REMAINS UNKNOWN ABOUT THE CONTAMINATION OF THIS CELL BY SODIUM AT THE TIME OF HANDLING. METHODS ARE PROPOSED TO LIMIT THIS CONTAMINATION. THE PRINCIPAL ADVANTAGES ARE THE FOLLOWING - (A)

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-010356 *CONTINUED*

HANDLING TAKES PLACE WITHIN SIGHT. THIS VISIBILITY IN CERTAIN CASES IS DIRECT THROUGH PORT HOLES PROVIDED IN THE CELL, OR INDIRECT BY MEANS OF PERISCOPES OR TELEVISION CAMERAS. (B) SINCE THIS CELL IS LEAK-PROOF AND FILLED WITH ARGON, THE PROBLEMS OF TIGHTNESS BETWEEN REACTOR AND CELL ARE CONSIDERABLY SIMPLIFIED. (C) THIS CELL ALSO PERMITS HANDLING OF CONTAMINATED EQUIPMENT WITHOUT REQUIRING THE USE OF AUXILIARY, HIGHLY SHIELDED PROTECTIVE EQUIPMENT. (D) FINALLY, THE CELL PLAYS THE ROLE OF A TIGHT CHAMBER WHICH CAN RETAIN ACTIVE PRODUCTS, ITS STRUCTURE PERMITS IT TO OFFER CONTAINMENT OF A REACTOR EXPLOSION.

CONTAINMENT + DESIGN CRITERIA + REACTOR, FAST + HOT CELL + REACTOR, LMCR + MAINTENANCE AND REPAIR

12-011424 ALSO IN CATEGORIES 6 AND 9 DATED 1966

HORST KM

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. THIRD QUARTERLY REPORT, OCTOBER 1964 - JANUARY 1965

GENERAL ELECTRIC, SAN JOSE

GEAP-4799 +. 150 PAGES, FEBRUARY 1965, CFSTI \$4.00 CY, \$1.00 MN

THIS REPORT IS ONE OF A SERIES OF SUCH REPORTS ON THE FOLLOWING SUBJECTS - PLANNING, EXPERIMENTAL PROGRAM IN ZPR-III, FUEL DEVELOPMENT, INSTRUMENTATION, CORE DYNAMICS, REACTIVITY CONTROL, FUEL HANDLING, AND CORE DESIGN

REACTOR, BREEDER + CRITICAL ASSEMBLY + DOPPLER COEFFICIENT + REACTOR, FAST + FUEL HANDLING + HEAT TRANSFER + REACTOR, LMCR + INSTRUMENT, TEMPERATURE + TEST, PHYSICS + SEFOR (RE) + RESEARCH CONTRACT + INSTRUMENTS, MISC. + FUEL ELEMENTS

12-015934 ALSO IN CATEGORIES 7 AND 8 DATED 1966

SAVAGE HW

HISTORY OF THE AEC SODIUM COMPONENTS DEVELOPMENT PROGRAM. VOLUME I

OAK RIDGE NATIONAL LABORATORY

ORNL-3955 (VOL. 1) +. 401 PAGES, DECEMBER 1966

THIS HISTORY WILL BE OF GREAT INTEREST TO THOSE WHO HAVE HAD A PART IN THE DEVELOPMENT OF LIQUID-METAL COMPONENTS TECHNOLOGY DURING THE LAST EIGHT TO NINE YEARS, AND WE SINCERELY HOPE THAT IT WILL SERVE AS AN ACCURATE REFLECTION OF THE CONTRIBUTION OF THE SODIUM COMPONENTS DEVELOPMENT PROGRAM WORKING GROUP IS MAKING TO THAT DEVELOPING TECHNOLOGY. THIS VOLUME IS A SUMMARY OF DEVELOPMENT PROGRAMS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + FERMI (LMFBR) + REACTOR, LMCR + METAL WATER REACTION + SODIUM + TESTING + OPERATING EXPERIENCE SUMMARY + EQUIPMENT DESIGN + MAIN COOLING SYSTEM + PUMPS + INSTRUMENTS, MISC. + HEAT EXCHANGERS

12-015935 ALSO IN CATEGORIES 7 AND 8 DATED 1966

SAVAGE HW

HISTORY OF THE AEC SODIUM COMPONENTS DEVELOPMENT PROGRAM. VOLUME II

OAK RIDGE NATIONAL LABORATORY

ORNL-3955 (VOL. 2) +. 568 PAGES, DECEMBER 1966

THIS HISTORY WILL BE OF GREAT INTEREST TO THOSE WHO HAVE HAD A PART IN THE DEVELOPMENT OF LIQUID-METAL COMPONENTS TECHNOLOGY DURING THE LAST EIGHT TO NINE YEARS, AND WE SINCERELY HOPE THAT IT WILL SERVE AS AN ACCURATE REFLECTION OF THE CONTRIBUTION OF THE SODIUM COMPONENTS DEVELOPMENT PROGRAM WORKING GROUP IS MAKING TO THAT DEVELOPING TECHNOLOGY. THIS VOLUME CONSISTS OF REPORTS OF SUBCOMMITTEE MEETINGS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, LMCR + METAL WATER REACTION + SODIUM + TESTING + OPERATING EXPERIENCE SUMMARY + EQUIPMENT DESIGN + MAIN COOLING SYSTEM + PUMPS + INSTRUMENTS, MISC. + HEAT EXCHANGERS

12-015936 ALSO IN CATEGORIES 7 AND 8 DATED 1966

SAVAGE HW

HISTORY OF THE AEC SODIUM COMPONENTS DEVELOPMENT PROGRAM. VOLUME III

OAK RIDGE NATIONAL LABORATORY

ORNL-3955 (VOL. 3) +. 484 PAGES, TABLES, DECEMBER 1966

THIS HISTORY WILL BE OF GREAT INTEREST TO THOSE WHO HAVE HAD A PART IN DEVELOPING LIQUID-METAL COMPONENTS TECHNOLOGY DURING THE LAST EIGHT TO NINE YEARS, AND WE SINCERELY HOPE THAT IT WILL SERVE AS AN ACCURATE REFLECTION OF THE CONTRIBUTION OF THE SODIUM COMPONENTS DEVELOPMENT PROGRAM WORKING GROUP IS MAKING TO THAT DEVELOPING TECHNOLOGY. THIS VOLUME CONSISTS OF QUESTIONNAIRES SENT TO MEMBERS AND ALTERNATES, AND REPLIES. ALSO INCLUDED ARE THE MONTHLY COORDINATION REPORTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, LMCR + METAL WATER REACTION + SODIUM + TESTING + OPERATING EXPERIENCE SUMMARY +

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-015936 *CONTINUED*
EQUIPMENT DESIGN + MAIN COOLING SYSTEM + PUMPS + INSTRUMENTS, MISC. + HEAT EXCHANGERS

12-017102 ALSO IN CATEGORY 17 DATED 1966
HUTTER E + ELIAS P + VEITH DJ + WARE WR
EBR-II FUEL-UNLOADING MACHINE—DESIGN AND PERFORMANCE CHARACTERISTICS
ARGONNE NATIONAL LAB., ILL.
ANL-7201 +. 91 PAGES, 42 FIGURES, 6 REFERENCES, JUNE 1966

DESCRIBES THE FUEL-HANDLING SYSTEM, THE MACHINE, AND ITS OPERATION FOR THE SODIUM-COOLED REACTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + EBR 1 AND 2 (KE) + REACTOR, FAST + REACTOR, LMCR + OPERATION + FUEL HANDLING MACHINE +
EQUIPMENT DESIGN

12-018209 ALSO IN CATEGORIES 5 AND 11 DATED 1967
HARTY H + BATCH JM + PEDERSEN LT + WHEELER RG + FOX JC
ENGINEERING DEVELOPMENT DEPARTMENT QUARTERLY REPORT, APRIL-JUNE 1966
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-367 +. 76 PAGES, 18 FIGURES, 2 TABLES, APRIL 1967

PROGRESS REPORT COVERING SUCH TOPICS AS THE FAST FLUX TEST FACILITY, HEAT TRANSFER STUDIES, CROSS-
FLOW MIXING BETWEEN PARALLEL FLOW, BOILING-STABILITY STUDIES, IRRADIATION DAMAGE TO REACTOR
METALS, UPSTREAM BOILING BURNOUT, THE CSE, IODINE FORMS IN STEAM CONDENSATE, AND DECONTAMINATION
OF COATINGS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,
U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

HEAT TRANSFER, BOILING + BURNOUT HEAT FLUX + IODINE + REACTOR, LMCR + PLUTONIUM + STEAM + AEROSOL PROPERTIES +
CONTAINMENT R AND D + CSE + SAMPLING + HEAT TRANSFER EXPERIMENT

12-019567 DATED 1966
MUENCHOW HO + CHASE WL
AN INVESTIGATION OF TUBE WASTAGE CAUSED BY SMALL LEAKS OF WATER IN SODIUM-HEATED STEAM GENERATORS
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN
SM-85/24 +. 21 PAGES, FIGURES, TABLES, PRESENTED AT THE IAEA SYMPOSIUM ON ALKALI METAL COOLANTS, CORROSION
STUDIES AND SYSTEM OPERATING EXPERIENCE, VIENNA, 28 NOVEMBER - 2 DECEMBER 1966

DESCRIBES THE TEST EQUIPMENT, INITIAL TEST RESULTS, AND THE BEGINNING OF A PROGRAM FOR STUDYING
THE MECHANISMS OF METAL REMOVAL BY CORROSION, EROSION, OR OTHER PROCESSES IN A SODIUM-HEATED
STEAM GENERATOR. GENERAL CONCLUSIONS ARE (1) LARGE QUANTITIES OF HYDROGEN COULD BE VENTED, THUS
REDUCING THE SHELL-DESIGN REQUIREMENTS, AND (2) SIMPLE DESIGN INNOVATIONS MAY BE MADE TO THE TUBE
BUNDLES TO PREVENT TUBES ADJACENT TO A SMALL LEAK FROM EXPERIENCING HIGH REACTION TEMPERATURES
AND THUS PROPAGATING THE LEAK INTO A CHAIN REACTION.

AVAILABILITY - INTERNATIONAL ATOMIC ENERGY AGENCY (NEW YORK OR GENEVA)

CHEMICAL REACTION + DESIGN CRITERIA + REACTOR, LMCR + SODIUM + TESTING + WATER + CORROSION + EROSION +
FAILURE, TUBING + STEAM GENERATOR + HEAT EXCHANGERS

12-021245 ALSO IN CATEGORIES 6 AND 7 DATED 1966
REPORT OF WORK IN THE 4TH QUARTER OF 1966
COMMISSARIAL A L-ENERGIE ATOMIQUE, CADARACHE, FRANCE
EURFNR-267 +. 89 PAGES, 1966

THIS REPORT IS ONE OF A SERIES OF REPORTS ON WORK PERFORMED UNDER THE UNITED STATES-EURATOM FAST
REACTOR EXCHANGE PROGRAM. TOPICS COVERED ARE - CRITICALITY STUDIES, METALLURGY, RADIATION
EFFECTS, RADIATION SHIELDING AND PROTECTION, REACTOR CONTROL, REACTOR ENGINEERING, REACTOR FUELS,
AND REACTOR SAFETY.

AVAILABILITY - MICROCARD EDITIONS, INC., ACCOUNTING AND SHIPPING DEPT., WEST SALEM, WISCONSIN 54669

REACTOR, LMCR + ACCIDENT, REACTIVITY + REACTOR DYNAMICS + SODIUM + REACTOR CONTROL + CRITICALITY EXPERIMENT +
FISSION PRODUCT RETENTION

12-024373 DATED 1967
FREED E WJ + NEWCOMB L + KENNEDY RS
STATIC AND SLIDING CONTACT BEHAVIOR OF MATERIALS IN SODIUM ENVIRONMENTS AT ELEVATED TEMPERATURES
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-12,446 +. 64 PAGES, FIGURES, TABLES, REFERENCES, SEPTEMBER 29, 1967

TESTS ARE IN PROGRESS TO DETERMINE STATIC AND SLIDING CONTACT BEHAVIOR CHARACTERISTICS OF SELECTED
CANDIDATE MATERIAL COMBINATIONS IN LIQUID AND VAPOR SODIUM ENVIRONMENTS WITH INERT COVER GAS AT

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-024373 *CONTINUED*

ELEVATED TEMPERATURES. MOST STATIC CONTACT TESTS SHOWED THAT SELF-WELDING OF MATERIALS IN SODIUM TENDED TO PROGRESS WITH TIME. COMPOSITION AND HOT HARDNESS OF SPECIMENS AFFECTED RESULTS MORE THAN DID VARIATIONS IN SODIUM. SLIDING CONTACT TESTS INDICATED THAT METALS SHOULD BE HEAT-TREATED TO THEIR MAXIMUM PRACTICAL HARDNESS. MATERIAL COMBINATIONS OF HARD VS HARD TENDED TO BE MORE WEAR-RESISTANT THAN HARD VS SOFT. MATERIAL BUILDUP INSTEAD OF WEAR ON SLIDING CONTACT SURFACES WAS OBSERVED IN A NUMBER OF TESTS. SUBSTANTIAL DIFFERENCES IN WEAR BEHAVIOR WERE OBSERVED IN MATERIALS OF DIFFERENT COMPOSITIONS WITH SOMEWHAT SIMILAR HOT HARDNESS CHARACTERISTICS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00 COPY, \$0.65 MICROFICHE

ALLY + REACTOR, BREEDER + FAILURE + REACTOR, FAST + REACTOR, LMCR + SODIUM + TITANIUM + WELDING + PROPERTY, PHYSICAL

12-024471 ALSO IN CATEGORY 13 DATED 1967

BROWN CL + LAWRENCE LA

FFTF CRITICALITY SAFETY SYSTEM 72 - UNDERWATER EXAMINATION FACILITY FEASIBILITY OF USING A SOLUBLE POISON FOR CRITICALITY CONTROL

BATTELLE-NORTHWEST, RICHLAND, WASHINGTON

BNWL-594 +. 15 PAGES, 7 REFERENCES, SEPT. 1967

STUDY TO DETERMINE AMOUNT OF SOLUBLE SODIUM BORATE REQUIRED TO ENSURE CRITICALITY SAFETY. MINIMUM CONCENTRATION CALCULATED - 3 G B/LITER (0.59 G B-10/LITER). TWO METHODS RECOMMENDED TO ENSURE THAT BORON IS PRESENT ABOVE REQUIRED MINIMUM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

BORON + CRITICALITY SAFETY + REACTOR, FAST + REACTOR, LMCR + POISON, SOLUBLE + FUEL STORAGE + REACTOR, TEST + FFTF (TR) + CRITICALITY EXPERIMENT + REACTOR + RADIOCHEMICAL PLANT SAFETY + WATER TREATMENT + HANFORD SITE + FUEL REPROCESSING

12-025266 ALSO IN CATEGORY 13 DATED 1967

KETZLACH N

NUCLEAR SAFETY SRE CORE III FUEL ELEMENT STORAGE

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-MEMO-12472 +. 7 PAGES, MAY 17, 1967

THE NUCLEAR SAFETY OF A SINGLE-PLANE ARRAY OF SRE CORE-III FUEL ELEMENTS, SPACED ON 12-INCH CENTERS, HAS BEEN EVALUATED. THE ARRAY IS LESS THAN 45% OF CRITICAL, BASED ON THE MASS/UNIT-AREA PRINCIPLE FOR THE FUEL ROD SIZES OF INTEREST. THIS IS INDEPENDENT OF THE DEGREE OF WATER FLOODING. A FULLY FLOODED ARRAY OF THE ASSEMBLY OF ELEMENTS SPACED AS STATED ABOVE WOULD HAVE A K-EFF OF LESS THAN 0.6. AN UNMODERATED ASSEMBLY OF FUEL OF THE GIVEN ENRICHMENT HAVING A NEUTRON-REFLECTION EQUIVALENT TO WATER WOULD REQUIRE AT LEAST THREE TIMES AS MUCH FUEL FOR CRITICALITY, INDEPENDENT OF SPACING BETWEEN ELEMENTS OR THE THICKNESS OF WATER REFLECTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + FUEL STORAGE + URANIUM CARBIDE + SRE (RE) + STORAGE CONTAINER + FUEL REPROCESSING + FUEL ELEMENTS

12-027283 ALSO IN CATEGORY 7 DATED 1968

KUNKEL WP + POLLOCK BD + GUON J + ZWETZIG GB + SILBERBERG M + BERGER S

LMFBR FISSION PRODUCT CONTAMINATION AND CONTROL STUDIES - A STATUS REPORT

ATOMICS INTERNATIONAL

AI-AEC-12687 +. 76 PAGES, 24 FIGURES, 13 TABLES, 29 REFERENCES, JUNE 30, 1968

THE STATUS IS PRESENTED OF STUDIES AND EXPERIMENTS BEING CONDUCTED AT ATOMICS INTERNATIONAL TO DESCRIBE THE DISTRIBUTION OF FISSION PRODUCTS AND OTHER CONTAMINATION SOURCES RELEASED TO THE COOLANT OF THE LIQUID METAL FAST BREEDER REACTOR. EXPERIMENTAL RESULTS DERIVED FROM STUDIES OF CESIUM AND SODIUM IODIDE VAPORIZATION FROM SODIUM SOLUTIONS AND CESIUM COLD TRAPPING STUDIES ARE PRESENTED. THE FISSION PRODUCT BEHAVIOR LOOP WHICH WILL BE USED FOR A MORE COMPREHENSIVE SERIES OF INVESTIGATIONS OF FISSION PRODUCT BEHAVIOR IN SODIUM IS ALSO DESCRIBED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DEPOSITION + REACTOR, FAST + FISSION PRODUCT RELEASE + REACTOR, LMCR + SODIUM + FISSION PRODUCT TRANSPORT + AI + RADIOACTIVITY RELEASE + EXPERIMENT

12-029002 ALSO IN CATEGORIES 5 AND 9 DATED 1968

DAVISON DF + ROACH PF

AN EXPERIMENTAL CONTINUOUS-INDICATION PLUGGING METER FOR IMPURITY MONITORING IN LIQUID ALKALI METALS

UNITED KINGDOM ATOMIC ENERGY AUTHORITY

TKG REPORT 1640 (R) +. 7 PAGES, FIGURES, TABLES, REFERENCES, 1968

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-029002 *CONTINUED*

A CONTINUOUS READING PLUGGING METER WAS DEMONSTRATED THAT MONITORED SODIUM SATURATION TEMPERATURE 125 TO 280 C. THE LARGEST CYCLE VARIATION IN THE TEST TRACE WERE ± 5 C AT 125 C. UNDER STABLE CONDITIONS SATURATION TEMPERATURE COULD BE READ TO ± 2 C AT 125 AND ± 1 C AT 280.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + SODIUM + UNITED KINGDOM + COOLANT CHEMISTRY + INSTRUMENT, FLOW + INSTRUMENT, SURVEILLANCE + IMPURITY

12-029133

DATED 1968

EICHELBERGER RL

THE SOLUBILITY OF OXYGEN IN LIQUID SODIUM - A RECOMMENDED EXPRESSION

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

AI-AEC-12685 +. 21 PAGES, 3 FIGURES, 1 TABLE, 19 REFERENCES, NOVEMBER 1, 1968

THE SOLUBILITY OF OXYGEN IN SODIUM AS REPORTED IN NINE PUBLICATIONS FROM EIGHT LABORATORIES WAS REVIEWED, AND CRITERIA ESTABLISHED FOR THE SELECTION OF DATA TO BE USED TO DERIVE A RECOMMENDED SOLUBILITY CURVE. THE RECOMMENDED CURVE, BASED ON 107 SOLUBILITY DETERMINATIONS FROM FIVE LABORATORIES, IS DESCRIBED BY AN EQUATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

OXYGEN + SODIUM + THEORETICAL INVESTIGATION + MATHEMATICAL TREATMENT + REACTOR, LMFBF

12-029179

DATED 1967

SCHLECHTENDAHLE E

EJECTING SODIUM FROM REACTOR COOLANT CHANNELS

KERNFORSCHUNGSZENTRUM, KARLSRUHE, GER.

5 PAGES, NUKLEONIK, 10, PAGES 270-274 (NOV. 1967) (IN GERMAN)

THE PHENOMENOLOGICAL MODEL FOR REACTOR COOLANT NA BOILING AND EJECTION IS DESCRIBED. EQUATIONS USED FOR THE COOLANT EXPULSION, HEAT TRANSFER, AND VAPORIZATION ARE GIVEN. TIME CONSTANTS FOR TRANSIENT VAPORIZATION ARE DETERMINED. BUBBLE FORMATION IN THE NA COOLANT IS SHOWN AS A FUNCTION OF TIME. FUEL TEMPERATURE AND SODIUM FILM TEMPERATURE-DEPENDENCE ARE SHOWN ALSO.

HEAT TRANSFER + SODIUM + REACTOR COOLANT + REACTOR + BUBBLE + REACTOR, LMFBF

12-029494

ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION J-1 - DIRECT RELEASE TO ATMOSPHERE

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE J-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THERE ARE SITUATIONS CITED IN THE FD-SAR THAT NORMALLY INERTED REGIONS (IN AND/OR AR) MAY BE VENTED DIRECTLY TO THE ATMOSPHERE, RATHER THAN BEING DIRECTED TO THE VOLUME LIMITED GAS STORAGE TANKS. DISCUSS BASIS FOR THIS DIRECT RELEASE TO THE ATMOSPHERE, INCLUDING ACTIVITY LIMITS PLACED ON THIS RELEASE AND METHOD BY WHICH ACTIVITY CONTENT IS DETERMINED. (ANSWER) SEE QUESTION M-11 OF THIS SUPPLEMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

AIRBORNE RELEASE + REACTOR, LMCR + REPORT, SAR + SOURCE, CONTINUOUS + ARGON + NITROGEN + STACK + COVER GAS + SEFOR (RE) + RESPONSE TO NRC QUESTION

12-029495

ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION J-2 - COVER GAS ISOLATION VALVE SYSTEM

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE J-1 AND -2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS BASIS FOR SELECTING ISOLATION VALVE SYSTEM ASSOCIATED WITH COVER GAS IN SODIUM EXPANSION TANK FOR SECONDARY COOLANT SYSTEM. (ANSWER) SECONDARY COOLANT SYSTEMS ARE CLOSED LOOPS, BUT SINCE COVER GAS IS PROVIDED ON FEED AND BLEED BASIS IT IS NECESSARY TO ISOLATE GAS LINES TO CLOSE OUTSIDE OF CONTAINMENT PART OF LCOP.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + COVER GAS + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + VALVES

12-029514

ALSO IN CATEGORY 18

DATED 1968

RESPONSE TO QUESTION M-5 - COMBINATION FILTER FOR GASEOUS WASTE

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-029514 *CONTINUED*

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

15 PAGES, 1 FIGURE, PAGE M-6 THRU -9 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE A DESCRIPTION OF THE COMBINATION FILTER THROUGH WHICH GASEOUS WASTES ARE DISCHARGED TO THE ATMOSPHERE. ALSO, PROVIDE THE DESIGN SPECIFICATIONS FOR THESE FILTERS, THE FILTER TESTING PROCEDURES, AND THE PROPOSED FREQUENCY OF TESTING. (ANSWER ON PGS M-6 THRU -9 PLUS 1 FIGURE AND 6 PAGES OF FILTER SPECIFICATIONS.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

AIRBORNE RELEASE + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + TEST, FILTER + WASTE DISPOSAL, GAS + FILTER CHARACTERISTICS + FILTER, COMBINATION + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION

12-029527

DATED 1968

SEIM OS

LARGE VALVES FOR LIQUID-METAL-COOLED REACTORS

11 PAGES, REACTOR AND FUEL PROCESSING TECHNOLOGY, VOL. 11, NO. 3, PG. 127-137, JUNE 1968

VALVE-DESIGN REQUIREMENTS WERE SUMMARIZED ON BASIS OF FAST REACTOR CONCEPTUAL DESIGNS. SERVICE CONDITIONS INCLUDE NORMAL OPERATING TEMPERATURE TO 1140 F, TRANSIENTS TO 1200 F, AND PRESSURES TO 150 PSIG. FUTURE LIQUID-METAL-COOLED FAST-BREEDER REACTORS WILL ALMOST CERTAINLY INCLUDE CHECK VALVES APPROXIMATELY 30 IN. IN DIAMETER, DESIGNED FOR 20-YEAR LIFE IN PRIMARY-LOOP SERVICE AND FLOW-CONTROL VALVES 12 TO 20 IN. IN DIAMETER FOR 20-YEAR LIFE IN SECONDARY LOOPS. DEVELOPMENT PROBLEMS ARE REVIEWED. LIMITED VALVE EXPERIENCE DISCUSSED.

BIBLIOGRAPHY + REACTOR, FAST + EQUIPMENT + COMPONENTS + REACTOR, LMFBF + VALVES

12-029533

DATED 1968

SKLADZHEN SB

TOWARD AUTOMATIC AND CONTINUOUS PLUGGING METERS

6 PAGES, REACTOR AND FUEL PROCESSING TECHNOLOGY, VOL. 11, NO. 3, PG 138-143, JUNE 1968

A NUMBER OF TECHNIQUES AND DEVICES - IN ADDITION TO PLUGGING METERS - ARE BEING DEVELOPED TO MONITOR IMPURITY LEVELS IN SODIUM. AT THIS STAGE IT IS WORTHWHILE TO NOTE TWO CENTRAL FACTS - (1) OUR KNOWLEDGE OF THE CHEMISTRY OF SODIUM AND ITS IMPURITIES IS FAR FROM COMPLETE, AND (2) THERE IS NOT YET A TRULY SATISFACTORY DEVICE FOR MONITORING IMPURITIES IN SODIUM. HOWEVER, IN MY OPINION, UNTIL OTHER IN-LINE PURITY-MEASURING METHODS MATCH THE SIMPLICITY, RELIABILITY, AND ACCURACY OF PLUGGING METERS, PLUGGING METERS WILL CONTINUE TO BE USED TO MONITOR THE PURITY OF LIQUID METALS.

REACTOR, LMCR + SODIUM + CORROSION + COOLANT PURIFICATION SYSTEM + EQUIPMENT + COMPONENTS + REACTOR, LMFBF

12-029865

ALSO IN CATEGORIES 1 AND 18

DATED 1968

LMFBF PROGRAM PLAN. VOL. 3 - COMPONENTS

ARGONNE NATIONAL LAB., ILL.

WASH-1103 +. 437 PAGES, FIGURES, TABLES, AUGUST 1968

TWELVE TASK AREAS INCLUDE (1) COMPONENT REQUIREMENTS, (2) DEVELOPMENT OF QUALITY AND RELIABILITY ASSURANCE PROGRAMS, (3) COMPONENT MATERIALS REQUIREMENTS, (4) DEVELOPMENT OF FABRICATION AND NDT PROCESSES, (5) FACILITIES FOR COMPONENT DEVELOPMENT. SPECIFIC COMPONENT DEVELOPMENT PROGRAMS ARE (6) STEAM GENERATOR, (7) CORE HANDLING EQUIPMENT (8) COOLANT TRANSPORT AND STORAGE SYSTEMS, (9) REACTOR VESSEL AND RELATED COMPONENTS, (10) REACTIVITY CONTROL DRIVES, (11) LIQUID METAL PUMPS, (12) HEAT EXCHANGERS. ***EACH SECTION BRIEFLY STATES THE GOALS, INTERACTION WITH OTHER PROJECTS, CURRENT STUDIES, AND GIVES A SCHEDULE OR FLOW SHEET FOR THE EFFORT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FABRICATION + PRESSURE VESSELS + RELIABILITY, COMPONENT + TEST, NONDESTRUCTIVE + FUEL HANDLING MACHINE + PROPERTY, PHYSICAL + QUALITY ASSURANCE + MAIN COOLING SYSTEM + R AND D PROGRAM + STEAM GENERATOR + REACTOR, LMFBF + PUMPS + PIPES AND PIPE FITTINGS + JACOBS + HEAT EXCHANGERS + CONTROL ROD DRIVE MECHANISMS

12-030255

ALSO IN CATEGORIES 5 AND 10 AND 18 DATED 1968

RESPONSE TO QUESTION 4 - EMERGENCY COOLING

NORTHERN STATES POWER COMPANY

2 PAGES, PP. 21-22 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

EMERGENCY COOLING IS THAT FOLLOWING LOSS OF SECONDARY HEAT SINKS AND TOTAL LOSS OF SITE POWER. ***REFERS TO SUPPLEMENT 19, QUESTION 2, WHERE SAME QUESTION ANSWERED. A 50-KW AUXILIARY POWER UNIT HAS BEEN PROVIDED FOR AUX. PC PUMP, AND ALSO POWERS A GAS BLOWER TO COOL THIS PUMPS WINDINGS. LATTER GAS BLOWER SYSTEM IS DESCRIBED HERE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

EMERGENCY POWER, ELECTRIC + ACCIDENT, LOSS OF POWER + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + AUXILIARY COOLING SYSTEM + REACTOR, LMFBF + PUMPS

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-030609 ALSO IN CATEGORIES 11 AND 18 DATED 1967
SEFOR FSAR VOL. I - REACTOR DESCRIPTION
GENERAL ELECTRIC CO.
200 PAGES, FIGURES, TABLES, SEFOR FACILITY DESCRIPTION AND SAFETY ANALYSIS REPORT VOL. 1, JULY 21, 1967,
DUCKET 50-231

CONTAINS SECTIONS ON SUMMARY DESCRIPTION, SITE, REACTOR, COOLANT SYSTEM, SHIELDING, CONTAINMENT, SAFEGUARDS WASTE CONTROL, AND INSTRUMENTATION. A 20-MWT SODIUM-COOLED FAST-BREEDER PU02-U23802 FUELED EXPERIMENTAL REACTOR TO GAIN EXPERIENCE WITH SUCH REACTORS AND PARTICULARLY THE DOPPLER COEFF. AND ITS EFFECT ON REACTOR RESPONSE. CONTROLLED BY 10 MOVABLE NICKEL REFLECTORS AND 18 FIXED B4C RODS. CONTAINMENT BY STEEL-LINED REINFORCED-CONCRETE BARRIER SURROUNDED BY WELDED A-212B STEEL VESSEL. SODIUM INLET TEMP., 700 F, OUTLET TEMP., 820 F AT 5000 GPM. DELTA P IN MAIN LOOP, 31-PSI. PUMP DISCHARGE, 55 PSIG. SAFEGUARDS INCLUDE CONTAINMENT INERTING SAFETY VESSEL OF STEEL AROUND REACTOR VESSEL, 4 COOLANT SYSTEMS, AND CORE DISPERSION ASSEMBLY TO PREVENT CRITICAL MASS IF CORE AND VESSEL DISSASSEMBLE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

CONTAINMENT DESIGN + CONTAINMENT, MULTIPLE + DOPPLER COEFFICIENT + DOPPLER EFFECT + REACTOR, LMCR + PRESSURE VESSELS + REACTOR DESCRIPTION + REACTOR STABILITY + REPORT, SAR + SITING, REACTOR + SODIUM + ARGON + NITROGEN + INSTRUMENT, CONTROL + WASTE MANAGEMENT + SEFOR (RE) + SHIELDING + SYSTEM DESCRIPTION + EXPERIMENT + CONTAINMENT ATMOSPHERE, INERT + CORE COMPONENTS + PLUTONIUM DIOXIDE + MAIN COOLING SYSTEM + R AND D PROGRAM + ENGINEERED SAFETY FEATURE + REACTOR, LMFR + FUEL ELEMENTS

12-030664 ALSO IN CATEGORY 13 DATED 1968
WINEGARDNER WK
CONSIDERATIONS ON THE MEDIA FOR DECAY STORAGE OF FUEL ELEMENTS
BATTELLE MEMORIAL INSTITUTE
BNWL-909 +. 34 PAGES, 23 REFERENCES, NOVEMBER 1968

PREPARED IN RESPONSE TO FFTF DIVISION REQUEST FOR INFORMATION AND RECOMMENDATIONS CONCERNING THE DECAY STORAGE OF IRRADIATED FUEL ELEMENTS. SUMMARIZES INFORMATION OF STORAGE TECHNIQUES. DISCUSSES INFORMATION RELATED TO FUEL CLADDING CORROSION PROBLEMS THAT MAY RESULT FROM DECAY STORAGE. CONTAINS RECOMMENDATIONS CONCERNING STORAGE MEDIA.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + FUEL STORAGE + REACTOR, TEST + FFTF (TR) + CORROSION + REACTOR + FUEL REPROCESSING

12-030845 ALSO IN CATEGORIES 11 AND 17 DATED 1967
POLOMIK EE + RUSSELL CE + QUINN EP
FLUID FLOW AND VIBRATION TESTS ON 1/2-SCALE MODEL OF THE SEFOR REACTOR VESSEL
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.
GEAP-5440+. APRIL 1967, 100P.

TESTS WERE RUN ON THE FLUID FLOW AND VIBRATION CHARACTERISTICS IN A HALF-SCALE MODEL OF THE SEFOR (SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR) REACTOR VESSEL. WATER, AT TEMPERATURES UP TO 265 F, WAS USED TO SIMULATE COOLANT CONDITIONS IN THE SODIUM-COOLED REACTOR. FLOWS WERE VARIED FROM 10 TO 100% RATED REYNOLDS NUMBER IN THE REACTOR. DATA WAS OBTAINED ON RELATIVE CORE FLOW DISTRIBUTION, FLOW PATTERNS, PRESSURE LOSSES, AND VIBRATION CHARACTERISTICS OF THE CANTILEVER-MOUNTED CORE SHROUD. FLUID FLOW RESULTS PROVIDED INFORMATION NECESSARY FOR VERIFICATION OF FINAL DESIGN. VIBRATION RESULTS INDICATED NO SERIOUS EFFECTS OF AMPLITUDE OR STRESS IN THE FULL-SCALE REACTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + MOCKUP + PRESSURE VESSELS + TESTING + WATER + REACTOR, PULSED + SEFOR (RE) + FLOW DISTRIBUTION + VIBRATION ANALYSIS

12-030879 ALSO IN CATEGORY 13 DATED 1968
BALKWILL JK
MECHANICAL ELEMENTS OPERATING IN SODIUM AND OTHER ALKALI METALS. VOL. I LITERATURE SURVEY
LIQUID METAL ENGINEERING CENTER
LMCE-60-5 +. 124 PAGES, 17 TABLES, 77 REFERENCES, 1968

SEARCHED LITERATURE REPORTING INFORMATION PUBLISHED SINCE 1950 ON MECHANICAL DEVICES WITH RUBBING SURFACES OPERATING IN ALKALI METALS. SODIUM ENVIRONMENTS ARE EMPHASIZED, WITH DATA ALSO INCLUDED FROM POTASSIUM, NAK, AND LITHIUM EXPERIMENTS. THE CHARACTERISTICS OF POTENTIALLY USEFUL MATERIALS THAT ARE IMPORTANT IN LIQUID METAL APPLICATIONS ARE DISCUSSED. MATERIAL COMPATIBILITY AND FRICTION/WEAR TESTING PROGRAMS CONDUCTED SINCE 1950 ARE DESCRIBED. EXPERIMENTAL AND OPERATING EXPERIENCE WITH FULL-SIZE COMPONENTS IS PRESENTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

ALLOY + BIBLIOGRAPHY + REACTOR, LMCR + SODIUM + POTASSIUM + CORROSION + MATERIAL + LITHIUM + EXPERIMENT +

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-030879 *CONTINUED*
SURVEY + OPERATING EXPERIENCE

12-031003 ALSO IN CATEGORY 7 DATED 1968
CLOUGH WS
THE PARTITION OF IODINE BETWEEN LIQUID SODIUM AND THE GAS PHASE AT 500C
UNITED KINGDOM ATOMIC ENERGY AUTHORITY, HARWELL
AERE-R-5848 +. 16 PAGES, 6 FIGURES, 1 TABLE, 8 REFERENCES, AUGUST 1968

THE EFFECT OF INCREASING THE IODINE CONCENTRATION IS TO DECREASE THE VOLATILITY OF IODINE
RELATIVE TO THAT OF SODIUM. AT THE HIGHEST IODINE CONCENTRATION STUDIED THE IODINE IS ABOUT 30%
LESS VOLATILE THAN AT THE LOWEST IODINE CONCENTRATION LEVELS.

AVAILABILITY - SCIENTIFIC ADMINISTRATION OFFICE, ATOMIC ENERGY RESEARCH ESTABLISHMENT, HARWELL, DIDCOT,
BERKSHIRE, ENGLAND

FISSION GAS RELEASE + IODINE + REACTOR, LMCR + SODIUM + REACTOR + FUEL REPROCESSING + REACTOR, LMFR

12-031068 ALSO IN CATEGORY 7 DATED 1968
LMFR PROGRAM PLAN. VOL. 7 - FUELS AND MATERIALS
ARGONNE NATIONAL LAB., ILL.
WASH-1107 +. 457 PAGES, FIGURES, TABLES, REFERENCES, AUGUST 1968

THE FUELS AND MATERIALS ELEMENT OF THE PROGRAM PLAN COVERS RESEARCH AND DEVELOPMENT OF FUELS,
CLADDINGS, OTHER MATERIALS, AND FUEL-SYSTEM CONCEPTS FOR LMFRS. INCLUDED IS ACQUISITION OF DATA
ON INTRINSIC PROPERTIES OF MATERIALS AND ON THE NORMAL, OFF-NORMAL, AND TRANSIENT-CONDITION
PERFORMANCE OF FUEL SYSTEMS FOR SPECIFIC REACTOR MISSIONS. THE MAJOR EMPHASIS IS ON THE URANIUM-
PLUTONIUM CYCLE. A MAJOR INTERFACE OF THIS ELEMENT IS WITH THE CORE DESIGN ELEMENT, WHICH HAS
RESPONSIBILITY FOR DEVELOPMENT OF FUEL-ELEMENT DESIGNS. OTHER IMPORTANT INTERFACES ARE WITH THE
COMPONENTS, SODIUM TECHNOLOGY, AND FUEL RECYCLE ELEMENTS OF THE PROGRAM PLAN. THE FUELS AND
MATERIALS ELEMENT WILL PROVIDE PROPERTY DATA, INFORMATION, AND TECHNICAL EVALUATIONS TO THESE
ELEMENTS IN AREAS WITHIN THEIR SCOPE OF RESPONSIBILITY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ALLOY + CARBIDE + CERAMICS + DESIGN CRITERIA + FABRICATION + CLADDING + OXIDE + PLUTONIUM + TESTING + THERMAL
ANALYSIS + URANIUM + STEEL, STAINLESS + NITRIDE + PROPERTY, PHYSICAL + REACTOR, LMFR + JACOBS + FUEL ELEMENTS

12-031467 DATED 1968
ROBIN M + VAUTREY L
HANDLING FACILITY FOR A NUCLEAR REACTOR FUEL RECHARGING INSTALLATION
COMMISSARIAT A L ENERGIE ATOMIQUE
BRITISH PATENT 1,119,426 +. JULY 10, 1968

DESCRIBES A FUEL LOADING MACHINE FOR A LIQUID NA-COOLED FAST NUCLEAR REACTOR. THE MACHINE
CONSISTS OF A MAIN BEARING BEAM, TURNABLE ROTATED AROUND A VERTICAL AXIS, A MAIN CARRIAGE WHICH
IS MOVABLE ON THE TURNABLE, AND A CONNECTED ARM AND GRIPPERS. THE ARM IS FOLDED ALONG THE
TURNABLE WHEN THE LOADING MACHINE IS INTRODUCED OR WITHDRAWN FROM THE CORE VESSEL. THE MACHINE
IS EXTERNALLY DRIVEN AND ENTERS THE CORE VESSEL BY A HERMETIC DOOR.

AVAILABILITY - THE PATENT OFFICE, 25 SOUTHAMPTON BUILDING, LONDON 2, ENGLAND (\$0.49 CENTS/COPY)

REACTOR, FAST + REACTOR, LMCR + SODIUM + UNITED KINGDOM + REACTOR + FUEL HANDLING MACHINE + EQUIPMENT + PATENT

12-031505 ALSO IN CATEGORY 7 DATED 1968
GOLDSMITH P + WELLS AC + COX LC
AEROSOL EMISSION FROM CHROMIUM ALLOYS WHEN HEATED IN AIR AND IN CARBON DIOXIDE/1% CARBON MONOXIDE
UNITED KINGDOM ATOMIC ENERGY AUTHORITY, HARWELL, BERKSHIRE
AERE-R-5426 +. 20 PAGES, 5 FIGURES, 4 TABLES, 6 REFERENCES, SEPTEMBER 1968

WHEN 80-20 NICKEL CHROMIUM ALLOY AND 20-23 STAINLESS STEEL ARE HEATED IN AIR, A COPIOUS SUPPLY OF
SUBMICRON PARTICLES IS EMITTED FROM THE HOT ALLOYS AT TEMPERATURES OVER 850 C. THESE PARTICLES
HAVE BEEN FOUND TO BE COMPOSED OF A MIXTURE OF CHROMIUM TRIOXIDE AND CHROMIUM SESQUIOXIDE IN THE
RATIO 2-1. AT TEMPERATURES OF 1200 C AND BELOW, WELL OVER 90% OF THE EMISSION IS DUE TO THE
PRODUCTION OF THIS SUBMICRON AEROSOL. AT 1250 C IN DRY AIR, AS MUCH AS 62% OF THE CHROMIUM LOSS
FROM THE HEATED ALLOYS WAS IN THE FORM OF LARGE PARTICLES OF SCALE. MOISTURE INHIBITED THIS
SCALING MECHANISM BUT ENHANCED THE SUBMICRON AEROSOL PRODUCTION RATE.

AVAILABILITY - SCIENTIFIC ADMINISTRATION OFFICE, ATOMIC ENERGY RESEARCH ESTABLISHMENT, HARWELL, DIDCOT,
BERKSHIRE, ENGLAND

AEROSOL + ALLOY + REACTOR, LMCR + AEROSOL PRODUCTION + CHROMIUM + NICKEL + STEEL, STAINLESS + HIGH TEMPERATURE
+ FUEL REPROCESSING + REACTOR, LMFR

12-032271 DATED 1968
ANDERSON JW + NICHOLS JR
ENCAPSULATION OF PLUTONIUM AND OTHER RADIOACTIVE MATERIALS IN WELDED CONTAINERS

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-032271 *CONTINUEU*
LOS ALAMOS LABORATORY
LA-3937 +. 10 PAGES, 10 FIGURES, 1 TABLE, APRIL 1968

IN CONNECTION WITH THE CHARGING AND SEALING OF PLUTONIUM ALLOY INTO FUEL CAPSULES FOR LAMPRE (LOS ALAMOS MOLTEN PLUTONIUM REACTOR EXPERIMENT), PRODUCTION EQUIPMENT AND A PROCEDURE HAVE BEEN DEVELOPED THAT HAVE GENERAL APPLICATION. RADIOACTIVE OR OTHER NOXIOUS MATERIALS ARE LOADED INTO CYLINDRICAL CONTAINERS WITHOUT CONTAMINATING THE OUTSIDE OF THE CONTAINERS, AN INERT ATMOSPHERE IS PROVIDED, AND THE CONTAINERS ARE WELDED SHUT. THE EQUIPMENT CONSISTS OF A WELDING FIXTURE, ATMOSPHERE CONTROLS, AND A POWER SUPPLY. THE PROCEDURE DEPENDS ON CAREFUL HANDLING AND THE CONVENIENCE OF THE EQUIPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + PLUTONIUM + REACTOR, ERDA OWNED + REACTOR + LAMPRE 1 (RE) + EXPERIMENT + FUEL ELEMENTS

12-032303 ALSO IN CATEGORY 17 DATED 1968
CONTAMINATION FOUND IN WASHED PIPE INTERIORS
CONSUMERS POWER COMPANY
10 PAGES, HALLAM MONTHLY RETIREMENT REPORT NO. 28, NOVEMBER 1968, DOCKET 115-3, TYPE--LMR, MFG--A.E., AE--BECHTEL

DEMOLITION BEGAN. A NOV. 1 COMPLIANCE SURVEY FOUND 1 MRAD/HR AT CONTACT INSIDE WASHED SCRAP PRIMARY PIPING THAT HAD BEEN SURVEYED AT THE ENDS OF THE CUT SEGMENTS. RADIATION LEVELS WERE^{TO} BE MEASURED INSIDE THE PIPE, SO A SURVEY METER WAS MADE TO FIT ON A LONG POLE. PRIMARY 14.-IN. PIPING WAS FOUND TO READ 0.1 TO 1.8 MRAD/HR, HIGHEST AT WELDS NEAR TEES OR ELBOWS, ALSO AT SODIUM OXIDE DEPOSITS AT PIPE REDUCTIONS. PRIMARY PIPE WILL BE SEGREGATED AND REMONITORED BEFORE DISPOSAL.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

DECONTAMINATION + HALLAM (LMR) + REACTOR, LMCR + SODIUM + COMPLIANCE + REACTOR, POWER + CONTAMINATION + DECOMMISSIONING + PIPES AND PIPE FITTINGS

12-032391 ALSO IN CATEGORY 6 DATED 1969
MAGNUSON DW
CRITICALITY CALCULATIONS FOR LIGHT WATER REACTOR AND LIQUID METAL FAST BREEDER REACTOR FUELS
UNION CARBIDE CORPORATION, Y-12 PLANT, TENN.
Y-DR-9 +. 50 PAGES, 11 FIGURES, 8 TABLES, 6 REFERENCES, MARCH 4, 1969

CRITICALITY CALCULATIONS WERE PERFORMED OVER THE RANGE OF MODERATION TO SHOW CRITICAL SIZES AND MASSES FOR DISCHARGE FUEL COMPOSITIONS ANTICIPATED IN LWRS AND LMFBRs. THE STUDIES WERE DIRECTED TOWARD THE GENERATION OF CALCULATED DATA FOR CRITICALITY CONTROL IN PRELIMINARY EQUIPMENT DESIGNS FOR FUEL-RECYCLE PROCESSING PLANTS.

AVAILABILITY - D.W. MAGNUSON, OAK RIDGE Y-12 PLANT, P.O. BOX Y, OAK RIDGE, TENN. 37830

CRITICALITY SAFETY + DESIGN CRITERIA + PLUTONIUM + REACTOR, LWR + MATHEMATICAL TREATMENT + URANIUM-238 + REACTOR, LMFBR

12-032392 DATED 1969
JACKSON PM
ANNULAR INLET FTF HYDRAULIC FEATURE MODEL
BATTTELLE-NORTHWEST, WASHINGTON
BNWL-952 +. 28 PAGES, 8 FIGURES, FEBRUARY 1969

THE FTF ANNULAR INLET CONCEPT IS AN ARRANGEMENT FOR INTRODUCING REACTOR COOLANT THROUGH THE ANNULAR ZONE FORMED BETWEEN THE REACTOR VESSEL AND THE REACTOR CORE VESSEL. THIS CONCEPT ELIMINATES THE NEED FOR PIPING TO THE SUPPLY PLENUM AND PROVIDES A ZONE FOR INLET COOLANT MIXING (STUDIED USING A ONE-TENTH SCALE MODEL WITH FOUR EQUALLY SPACED INLET PIPES LOCATED PERPENDICULARLY TO THE VERTICAL AXIS). THE DIRECT IMPINGEMENT OF THE COOLANT STREAM AGAINST THE REACTOR VESSEL WALL PROVED TO BE AN UNDESIRABLE CONDITION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + FLOW THEORY AND EXPERIMENTS + REACTOR, LMCR + TESTING + FTF (TR) + EXPERIMENT + EQUIPMENT + FLOW, ANNULAR + COOLANT QUALITY

12-032396 DATED 1968
MILLER WE + BERNSTEIN GJ + LENC JR + PAUL RC
PREPARATION OF FIRST MARK 1A FUEL ALLOY FOR EBR-II
ARGONNE NATIONAL LABORATORY
ANL-7474 +. 19 PAGES, 7 FIGURES, 8 TABLES, 7 REFERENCES, AUGUST 1968

DESCRIBES PREPARATION OF FUEL PINS (DESIGNATED MARK-1A) FOR THE SECOND CORE LOADINGS OF THE EBR-

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-032396 *CONTINUED*

11. THIS ALLOY FUEL CONSISTED OF 200 KG OF U-5 W/O FISSION FUEL CONTAINING 52.2 +/- 1/2 W/O U-235. SOME DIFFERENCES IN EQUIPMENT AND PROCEDURES WERE USED FROM THAT EMPLOYED IN PREPARATION OF THE FIRST CORE AS A RESULT OF LARGER BATCH SIZES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + FABRICATION + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + URANIUM + URANIUM-235 + FUEL, METALLIC TYPE + FUEL ELEMENTS

12-032697

DATED 1969

GURLAND SH + HECKER TP + LOTTIG RA

PERFORMANCE AND EVALUATION OF TWO LIQUID-METAL PUMPS FOR SODIUM-POTASSIUM SERVICE
NATIONAL AERONAUTICS AND SPACE ADMINISTRATION, LEWIS RESEARCH CENTER

NASA-TM-X-1765 +. 15 PAGES, 8 FIGURES, 3 TABLES, 6 REFERENCES, APRIL 1969

TWO IDENTICAL HERMETICALLY-SEALED MOTOR-DRIVEN CENTRIFUGAL PUMPS OPERATED FOR 1483 HOURS IN NAK LOOPS, ONE AT 1150 F (895 K) AND THE OTHER AT 450 F (505 K). THE PUMPS INCORPORATE NAK-LUBRICATED TILTING-PAD JOURNAL AND THRUST BEARINGS AND WERE DESIGNED TO WITHSTAND A NUCLEAR RADIATION ENVIRONMENT. THE INDUCTION MOTORS WERE CANNED INSIDE THE PUMP HOUSINGS TO ELIMINATE THE NEED FOR ABSOLUTE SEALS BETWEEN THE PUMP AND THE MOTOR. BOTH PUMPS OPERATED SUCCESSFULLY FOR THE 1483 HOURS WITHOUT MAINTENANCE. THERE WERE NO PARAMETRIC EXCURSIONS OUTSIDE THE ENVELOPES CONSIDERED SAFE FOR 10,000 OR MORE HOURS OF OPERATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + SODIUM + REACTOR, SPACE + TESTING + POTASSIUM + SNAP (SR) + EQUIPMENT + PUMPS

12-032699

ALSO IN CATEGORY 7

DATED 1969

BREHM WF + KOVACEVICH EA + SHANNON DW

RADIOISOTOPE TRANSPORT AND DEPOSITION IN FLOWING SODIUM

BATTELLE MEMORIAL INSTITUTE

BNWL-969 +. 24 PAGES, 7 FIGURES, 2 TABLES, 10 REFERENCES, FEBRUARY 1969

STUDIED TRANSPORT AND DEPOSITION OF RADIOISOTOPES RELEASED TO FLOWING SODIUM FROM CORROSION OF IRRADIATED STAINLESS STEEL. QUANTITATIVE MEASUREMENTS OF DEPOSITION CAN BE MADE WITH SENSITIVE COUNTING TECHNIQUES. THE RESULTS OF THE STUDY INDICATE THAT CR-51 AND CO-60 ARE TRANSPORTED PRIMARILY BY ISOTOPIC EXCHANGE AND DO NOT RAPIDLY MIGRATE TO LOWER TEMP. REGIONS OF LOOP SYSTEMS. HOWEVER, MN-54 IS RELEASED IN LARGER QUANTITIES TO DEPOSIT THROUGHOUT THE TEST LOOPS. PREFERENTIAL DEPOSITION SITES ARE (1) IMMEDIATELY DOWNSTREAM FROM THE SOURCE, (2) AT MINIMUM TEMPERATURE LOCATIONS, AND (3) IN OXYGEN COLD TRAPS. THESE RESULTS INDICATE MN-54 IS TRANSPORTED BOTH BY EXCHANGE AND BY SOLUTION- AND- PRECIPITATION. DEPOSITED MN-54 MIGRATED RAPIDLY TO THE COLD TRAP, DEPOSITED CR-51 AND CO-60 MIGRATED SLOWLY BY EXCHANGE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DEPOSITION + REACTOR, LMCR + SODIUM + FISSION PRODUCT TRANSPORT + STEEL, STAINLESS + CORROSION + TRANSPORT PROPERTY + REACTOR, LMCR

12-032813

ALSO IN CATEGORY 18

DATED 1968

PRIMARY SHIELD DESIGN AND EVALUATION

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-8 +. 50 PAGES, FIGURES, TABLES, SEFOR, PSAR, SUPPLEMENT 8, FEBRUARY 1968, DOCKET 50-231

DESCRIBES CORE AND REFLECTOR SHIELD USED TO ATTENUATE RADIATION TO LEVEL WHERE CONCRETE CAN BE USED WITHOUT DAMAGE. SHIELD MATERIALS ARE STEEL, BORON CARBIDE, AND SERPENTINE. TOPICS INCLUDE FABRICATION, INSTALLATION, FIGURES, AND APPENDIX A CONTAINS TEMP.-DISTRIBUTION CALCULATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CONCRETE + REACTOR, FAST + REACTOR, LMCR + RADIATION DAMAGE + STEEL + THERMAL MECHANICAL EFFECT + REACTOR, PULSED + SEFOR (RE) + SHIELDING + SYSTEM DESCRIPTION + REPORT, PSAR + SYSTEM CAPABILITY + REACTOR, LMCR

12-032814

ALSO IN CATEGORY 18

DATED 1968

SHIELDING NUCLEAR DESIGN CALCULATIONS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-9 +. 175 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 9, FEBRUARY 1968, DOCKET 50-231

DESCRIBES CALCULATIONS OF NEUTRON AND GAMMA SOURCES AND ATTENUATION. DOSE RATES DURING AND AFTER OPERATION WERE CALCULATED TO SPECIFY SHIELD THICKNESSES AND COMPOSITIONS. ENERGY ABSORPTION RATES WERE CALCULATED TO PROVIDE COOLING REQUIREMENTS. DESCRIPTION OF SHIELD INCLUDED. A MORE DETAILED DESCRIPTION IS GIVEN IN SUPPLEMENT 8.

AVAILABILITY - NKC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-032814 *CONTINUED*

REACTOR, FAST + REACTOR, LMCR + SOURCE, RADIATION + RADIATION SAFETY AND CONTROL + REACTOR, PULSED + SEFOR (RE) + SHIELDING + MATHEMATICAL STUDY + THEORETICAL INVESTIGATION + REPORT, PSAR + MATHEMATICAL TREATMENT + REACTOR, LMFBR

12-032853

ALSO IN CATEGORY 5

DATED 1968

PADILLA A

ANALYSIS OF COOLANT MIXING IN THE FFTF FUEL BUNDLE

BATTELLE NORTHWEST, RICHLAND, WASHINGTON

BNWL-SA-1688 + CONF-680832-3 +. 32 PAGES, 17 FIGURES, 1 TABLE, 8 REFERENCES, MARCH 14, 1968, FROM 10TH NATIONAL HEAT TRANSFER CONFERENCE, PHILADELPHIA, PA.

PRESENTS SOME OF THE COOLANT-MIXING STUDIES IN THE FUEL BUNDLE WHICH WERE CARRIED OUT DURING THE CONCEPTUAL STAGE IN THE DESIGN OF THE LTR. THE PURPOSE OF THESE STUDIES WAS TO DETERMINE THE TEMPERATURE AND FLOW DISTRIBUTION IN THE FUEL BUNDLE, PROVIDE A TECHNICAL BASIS FOR INSTRUMENTATION OF THE DRIVER SUBASSEMBLY, AND IDENTIFY THERMAL-HYDRAULIC PROBLEM AREAS IN THE DESIGN OF THE DRIVER ASSEMBLY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + TEMPERATURE GRADIENT + FLOW DISTRIBUTION + EXPERIMENT + COOLANT QUALITY

12-032876

DATED 1968

BROWN RW

ANALYTICAL AND EXPERIMENTAL STUDIES OF DOUBLE-PIPE COUNTER-FLOW LIQUID METAL HEAT EXCHANGERS

ARGONNE NATIONAL LABORATORY

ANL-7479 +. 140 PAGES, FIGURES, TABLES, 10 REFERENCES, AUGUST 1968

PRESENTS AN ANALYTICAL AND EXPERIMENTAL STUDY OF HEAT TRANSFER IN COUNTERCURRENT, LIQUID METAL, DOUBLE-PIPE HEAT EXCHANGERS. THE MATHEMATICAL TREATMENT IS BASED ON AN EXACT SOLUTION OF THE BASIC ENERGY EQUATION RATHER THAN ON THE TRADITIONAL PRACTICE OF ASSUMING THE HEAT TRANSFER COEFFICIENTS OF INDIVIDUAL CHANNELS TO BE UNIFORM AND EQUAL TO A PREDETERMINED VALUE. THE SOLUTION PRESENTED HEREIN IS ACTUALLY AN EXTENSION OF THE CLASSICAL GRAETZ PROBLEM TO BOUNDARY CONDITIONS APPROPRIATE TO HEAT EXCHANGERS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

LIQUID FUEL + REACTOR, LMCR + EXPERIMENT + EQUIPMENT + MATHEMATICAL TREATMENT + REACTOR, LMFBR + HEAT EXCHANGERS

12-032903

ALSO IN CATEGORIES 5 AND 18

DATED 1969

SIMULTANEOUS FAILURES - LOCA AND SAFEGUARD

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231 +. 7 PAGES, 1 FIGURE, SEFOR, PSAR, SUPPLEMENT 23, FEB. 20, 1969, DOCKET 50-231.

DESCRIBES MODIFICATIONS MADE TO MITIGATE CONSEQUENCES OF TWO SIMULTANEOUS FAILURES, I.E., A FAILURE IN THE PRIMARY COOLANT SYSTEM FOLLOWED BY A FAILURE OF ONE OF THE SYSTEMS OR COMPONENTS (ENGINEERED SAFEGUARDS FEATURES). GIVES SUMMARY OF PROTECTION AND INHERENT DESIGN FEATURES OF THESE SAFEGUARDS THAT ASSURE OPERABILITY IN THE ENVIRONMENT IN WHICH THEY ARE REQUIRED TO OPERATE FOLLOWING A PRIMARY SYSTEM FAILURE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + MODIFICATION + REACTOR, PULSED + SEFOR (RE) + REPORT, PSAR + FAILURE, SIMULTANEOUS + SYSTEM OPERABILITY IN ACCIDENT + ENGINEERED SAFETY FEATURE + REACTOR, LMFBR

12-032904

ALSO IN CATEGORIES 5 AND 18

DATED 1969

RESPONSE TO AEC QUESTION

GENERAL ELECTRIC

50231-26 +. 7 PAGES, SEFOR, PSAR, SUPPLEMENT 22, JANUARY 24, 1969, DOCKET 50-231

SUPPLEMENTARY INFORMATION GIVEN TO PREVIOUSLY ANSWERED QUESTIONS CONCERNING THE CONSEQUENCES OF COOLANT SYSTEM LEAKS AND RESULTING SYSTEM CHARACTERISTICS (SEE COMMENTS 1 IN SUPPLEMENT 18 AND PAGE 53 OF SUPPLEMENT 19), SPECIFIC TOPICS AND SMALL LEAKS IN MAIN COOLANT SYSTEM, AND EVALUATION OF EQUIPMENT NEEDED TO RECOVER FROM A GUILLotine PIPE FAILURE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

NRC QUESTION + ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + INCIDENT, RECOVERY FROM + REACTOR, PULSED + SEFOR (RE) + EQUIPMENT + REPORT, PSAR + REACTOR, LMFBR

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-032906 ALSO IN CATEGORIES 5 AND 18 DATED 1969
RESPONSE TO QUESTION 2 - RECOVERY EQUIPMENT FOR LOCA PIPE FAILURE
GENERAL ELECTRIC
50231-26 +. 4 PAGES, PG. 4 THRU 7 OF SEFOR, PSAR, SUPPLEMENT 22, JAN. 24, 1969, DOCKET 50-231

DISCUSS EQUIPMENT NEEDED TO RECOVER FROM A GUILLOTINE PIPE FAILURE *** ANSWER ON PAGES 4 THRU 7.
SEE ALSO COMMENT 1 IN SUPPLEMENT 18.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + INCIDENT, RECOVERY FROM + REACTOR, PULSED + SEFOR
(RE) + RESPONSE TO NRC QUESTION + EQUIPMENT + REPORT, PSAR + REACTOR, LMFBR

12-032931 DATED 1969
GUNSON WE + GEORGE JA + HURNER HP + GOESER DK
SYSTEMS CONSIDERATIONS IN THE DESIGN OF SODIUM PUMPS FOR LIQUID-METAL FAST-BREEDER REACTORS
WESTINGHOUSE ELECTRIC CORP., MADISON, PA.
8 PAGES, 7 FIGURES, 1 TABLE, 5 REFERENCES, NUCLEAR APPLICATIONS 6(4), PAGES 279-86 (APRIL 1969)

EVALUATES SYSTEMS CONSIDERATIONS INFLUENCING THE DESIGN OF VERTICAL, CENTRIFUGAL, AND FREE SURFACE
PUMPS FOR LIQUID-METAL FAST BREEDER REACTORS. SYSTEMS WERE ANALYZED TO GUIDE IN SELECTIONS FOR
PUMPS WITH FLOW RANGES OF 10,000 TO 200,000 GAL/MIN AND HEAD RANGES FROM 100 TO 450 FT FOR BOTH
THE SEA-OF-SODIUM AND PIPED REACTOR ARRANGEMENTS. THE SYSTEM ANALYSES DEFINED THE PARAMETRIC
RELATIONSHIPS BETWEEN THE PUMP, THE COOLANT SYSTEM, AND THE OVERALL PLANT REQUIREMENTS.

DESIGN CRITERIA + REACTOR, FAST + SODIUM + EQUIPMENT + REACTOR, LMFBR + PUMPS

12-033050 DATED 1969
REEMSNYDER DC + LUNNAN WS + WEIGEL C
PERFORMANCE AND CAVITATION DAMAGE OF AN AXIAL-FLOW PUMP IN 1500 F (1089 K) LIQUID SODIUM
NATIONAL AERONAUTICS AND SPACE ADMINISTRATION, LEWIS RESEARCH CENTER
NASA-TN-D-5138 +. 24 PAGES, 11 FIGURES, 4 TABLES, 12 REFERENCES, APRIL 1969

A LOW-HEAD-RISE AXIAL-FLOW PUMP WAS OPERATED IN LIQUID SODIUM AT TEMPS TO 1500 F (1089 K) FOR 558
HR TO INVESTIGATE PUMP PERFORMANCE AND BLADE CAVITATION DAMAGE IN ALKALI METALS. THE RESEARCH
MOTOR HAD NINE INSERTABLE BLADES OF THREE DIFFERENT MATERIALS, A BLADE-TIP DIFFUSION FACTOR OF
0.40, AND A HUB-TIP-RADIUS RATIO OF 0.77. OVERALL PERFORMANCE WAS OBTAINED, AND A 200-HR
CAVITATION ENDURANCE TEST WAS CONDUCTED IN 1500 F (1089 K) SODIUM AT 3450 RPM. POST-TEST BLADE
INSPECTION INDICATED THAT THERE WAS SLIGHT CAVITATION DAMAGE AND THAT THE NI-CR-BASE SUPERALLOY
(RENE 41) WAS MORE CAVITATION-DAMAGE RESISTANT TO HIGH-TEMPERATURE SODIUM THAN EITHER 316 OR 318
STAINLESS STEEL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

FAILURE, EQUIPMENT + REACTOR, LMCR + SODIUM + REACTOR, SPACE + TESTING + SNAP (SR) + EQUIPMENT + PUMPS

12-033051 ALSO IN CATEGORIES 1 AND 9 DATED 1968
LIQUID METAL ENGINEERING CENTER - SEMIANNUAL TECHNICAL PROGRESS REPORT, JULY-DECEMBER 1968
LIQUID METAL ENGINEERING CENTER
LMEC-69-1 +. 281 PAGES, 87 FIGURES, 17 TABLES, REFERENCES, DECEMBER 1968

SUBJECTS COVERED - VALVES AND SODIUM TECHNOLOGY, PUMP STUDIES, PIPING STUDIES, LMFBR TEST AND
FACILITY PLANNING, CODES AND STANDARDS FOR LMFBR AND RELATED ACTIVITIES (REACTOR VESSEL,
CIRCULATING COLD TRAP, MECHANICAL FREE-SURFACE SODIUM PUMPS, ELECTRO-MAGNETIC PUMPS, SODIUM
PIPING, VALVES, AND QUALITY), MAINTENANCE AND MALFUNCTION ANALYSIS, COMBINED UNIT REACTOR VESSEL
STUDY, QUALITY ASSURANCE, OPERATION OF SODIUM COMPONENTS TEST INSTALLATION, GENERAL OPERATING
ACTIVITIES, LIQUID METAL INFORMATION CENTER, SODIUM PUMP TEST FACILITY, SODIUM PUMP SEAL TEST
PROGRAM, BEAMING TEST PROGRAM, STEAM GENERATOR AND INTERMEDIATE HEAT EXCHANGER PERFORMANCE
EVALUATION, INSTRUMENTATION TESTING, EVALUATION, CALIBRATION, AND STANDARDS, OPERATION AND
MAINTENANCE OF THE LCTL, CETS, AND HNPFP PUMP TEST LOOP, MODIFICATIONS TO THE LCTL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + CODES AND STANDARDS + PRESSURE VESSELS + SODIUM + TESTING + STRESS ANALYSIS + EQUIPMENT + STEAM
GENERATOR + PUMP, MAGNETIC + REACTOR, LMFBR + PUMPS + VALVES + PIPES AND PIPE FITTINGS + INSTRUMENTS, MISC.

12-033056 ALSO IN CATEGORIES 5 AND 8 DATED 1967
POPLAVSKY VM + ARISTARKOV NN + BAGDASAROV YE + BLAGOVOLIN SM + VYBLOV NF + PINKHASIK MS
SAFETY PROBLEMS WITH SINGLE WALL SODIUM/WATER HEAT EXCHANGERS FOR FAST REACTORS
UKAEA, DOUNREAY EXPERIMENTAL REACTOR ESTABLISHMENT
CONF-670916-4 +. 26 PAGES, 4 FIGURES, 6 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST
REACTORS, SEPTEMBER 19-22, 1967

SUMMARY OF RUSSIAN WORK REGARDING SODIUM-WATER REACTIONS DUE TO A TUBE RUPTURE IN A SINGLE-WALL
STEAM GENERATOR. DISCUSSION INCLUDED EXPERIMENTAL WORK AND OTHER FACTORS REGARDING THIS REACTION

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-033056 *CONTINUED*

INFLUENCING THE DESIGN OF THE BN-350 AND THE BOR STEAM GENERATORS. FACTORS CONSIDERED WERE SHOCK, HYDRODYNAMIC PHENOMENA IN THE SODIUM LOOP INCLUDING STRUCTURAL LOADING, THERMAL EFFECTS, EFFICIENCY OF THE REACTION PRODUCT EVACUATION SYSTEM, AND DETERIORATION OF THE STRUCTURAL COMPONENTS.

CHEMICAL REACTION + METAL WATER REACTION + SODIUM + STEAM + WATER + PRESSURE RELIEF + USSR + METAL, ALKALI + METAL, LIQUID + EQUIPMENT DESIGN + STEAM GENERATOR + PRESSURE PULSE + REACTOR, LMFBR + HEAT EXCHANGERS

12-034696 ALSO IN CATEGORY 13

DATED 1969

BOYD CL + MEYERS DJ + RAMSEY EB

THE FFTF FUEL EXAMINATION FACILITY

BATTELLE MEMORIAL INSTITUTE, PACIFIC NORTHWEST LABORATORY, RICHLAND, WASHINGTON

16 PAGES, 3 FIGURES, 1 TABLE, 10 REFERENCES, PP. 3-18 OF PROCEEDINGS OF THE 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

THE FUEL EXAMINATION FACILITY OF THE FAST FLUX TEST FACILITY (FFTF) WILL SERVE THE OPERATIONS STAFF AND EXPERIMENTERS WITH A RELIABLE MEANS, UNDER CONTROLLED CONDITIONS, FOR PERFORMING PROMPT NONDESTRUCTIVE EXAMINATION OF EXPERIMENTS AND OTHER COMPONENTS IRRADIATED IN THE FTR. EXAMINATION DATA WILL BE OBTAINED FROM SELECTED EXPERIMENTS THAT REQUIRE REACTOR REINSERTION, AND FROM EXPERIMENTS OR OTHER CORE COMPONENTS WHOSE IRRADIATION IS COMPLETE.

AVAILABILITY - C.L. BOYD + D.J. MEYERS + E.B. RAMSEY, BATTELLE MEMORIAL INSTITUTE PACIFIC NORTHWEST LABORATORY, P.O. BOX 999, RICHLAND, WASHINGTON 99352

BIBLIOGRAPHY + REACTOR, FAST + HOT CELL + REACTOR, LMCR + REACTOR, TEST + VENTILATION SYSTEM + FFTF (TR) + DESIGN STUDY + REACTOR + IRRADIATION TESTING + GAMMA + EXAMINATION + REMOTE MANIPULATING AND VIEWING + ALPHA FACILITIES + CORE COMPONENTS + FUEL ELEMENTS

12-034698

DATED 1969

SUTHERLAND JD + MARKLEY KA + RITZ WL

FUEL-HANDLING SYSTEM DESIGN FOR THE WESTINGHOUSE 1000-MW(E) LMFBR STUDY

WESTINGHOUSE ELECTRIC CORPORATION, ADVANCED REACTORS DIVISION, BOX 158, MADISON, PENNSYLVANIA 15663

7 PAGES, 2 FIGURES, 1 TABLE, 8 REFERENCES, PP. 39-45 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

A HOT-CELL REFUELING SYSTEM WAS SELECTED FOR THE WESTINGHOUSE 1000-MW(E) LMFBR PLANT STUDY. IN THIS SYSTEM, ALL REACTOR REFUELING OPERATIONS ARE CARRIED OUT REMOTELY INSIDE AN ARGON-FILLED HOT CELL. TO PREPARE THE REACTOR FOR REFUELING, A SHIELD PLUG IS REMOVED FROM THE REACTOR. FUEL IS THEN TRANSFERRED UNDER SODIUM BETWEEN THE REACTOR AND AN ADJACENT FUEL STORAGE TANK THROUGH A HORIZONTAL FUEL TRANSFER PORT CONNECTING THE TWO VESSELS.

AVAILABILITY - J.D. SUTHERLAND/R.A. MARKLEY/W.L. RITZ, WESTINGHOUSE ELECTRIC CORP., ADVANCED REACTORS DIV., BOX 158, MADISON, PA. 15663

FUEL HANDLING + HOT CELL + ARGON + DESIGN STUDY + REMOTE MANIPULATING AND VIEWING + CONTAINMENT ATMOSPHERE, INERT + MAINTENANCE, REMOTE + REACTOR, LMFBR

12-034705

DATED 1969

CULIN RV

FFTF INERT-GAS SELECTION STUDY

BATTELLE NORTHWEST LABORATORY, PACIFIC NORTHWEST LABORATORY, P.O. BOX 999, RICHLAND, WASHINGTON 99352

16 PAGES, 1 TABLE, 30 REFERENCES, PP. 116-131 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

THE FFTF WILL BE A PRIME TEST FACILITY FOR FUELS AND MATERIALS FOR THE UNITED STATES LIQUID METAL FAST BREEDER REACTOR PROGRAM. CONCEPTUAL DESIGN OF THE FFTF INCLUDES A RANGE OF APPLICATIONS THAT WILL REQUIRE INERT GAS AS A COVER GAS FOR LIQUID-METAL SYSTEMS OR AS A PROTECTIVE ATMOSPHERE IN ENCLOSURES FOR SYSTEMS CONTAINING RADIOACTIVE SODIUM. BECAUSE OF THE WIDE RANGE OF APPLICATIONS AND THEIR DIVERSE REQUIREMENTS, EXTENSIVE CONCEPTUAL STUDY EFFORT WAS REQUIRED TO SELECT THE APPROPRIATE GAS OR GASES FOR EACH APPLICATION.

AVAILABILITY - R.V. CULIN, BATTELLE NORTHWEST LABORATORY, PACIFIC NORTHWEST LABORATORY, P.O. BOX 999, RICHLAND, WASHINGTON 99352

BIBLIOGRAPHY + DESIGN CRITERIA + REACTOR, FAST + REACTOR, LMCR + SODIUM + REACTOR, TEST + ARGON + MOLTEN FUEL + FFTF (TR) + DESIGN STUDY + CONTAINMENT ATMOSPHERE, INERT

12-034706

ALSO IN CATEGORIES 3 AND 13 AND 17 DATED 1969

JOHNSON KP + MATTE J

OPERATING EXPERIENCE WITH THE FERMI FUEL HANDLING CASK CAR

POWER REACTOR DEVELOPMENT COMPANY, MONROE, MICHIGAN + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN

15 PAGES, 9 FIGURES, PP. 135-149 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

THE ENRICO FERMI ATOMIC POWER PLANT WAS PROVIDED WITH A CASK CAR FOR TRANSPORTING SPENT FUEL BETWEEN THE REACTOR BUILDING AND THE FUEL AND REPAIR BUILDING. THIS CAR, A MULTI-STATION, SELF-CONTAINED UNIT, EXHIBITED A VARIETY OF PROBLEMS AND DEFECTS IN EARLY OPERATION. ANALYSIS OF THESE PROBLEMS AND THE AVAILABLE SOLUTIONS REQUIRED TO ACHIEVE SATISFACTORY PERFORMANCE UNDER

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-034706 *CONTINUED*

DESIGNED OPERATING CONDITIONS LED TO CONSIDERATION OF REPLACING THE CASK CAR WITH A NEW FACILITY.

AVAILABILITY - K.P. JOHNSON, POWER REACTOR DEVELOPMENT COMPANY, P. O. BOX 725, MONROE, MICHIGAN 48161

REACTOR, BREEDER + FAILURE, EQUIPMENT + REACTOR, FAST + FERMI (LMFBR) + FUEL HANDLING + REACTOR, LMCR + SHIPPING CONTAINER + TRANSPORTATION AND HANDLING + FUEL HANDLING MACHINE + OPERATING EXPERIENCE + REACTOR, LMFBR

12-034707

ALSO IN CATEGORY 3

DATED 1969

WILLIAMS RD + HALLINAN GJ + KIRKBY RC

THE TECHNICAL EVOLUTION OF THE FERMI FUEL TRANSPORT FACILITY

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF. + POWER REACTOR DEVELOPMENT COMPANY, MONROE, MICHIGAN
11 PAGES, 8 FIGURES, 4 REFERENCES, PP. 150-160 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

THE ENRICO FERMI ATOMIC POWER PLANT IS EQUIPPED WITH A FUEL TRANSPORT FACILITY WHICH FUNCTIONS TO WITHDRAW FULL SUBASSEMBLIES AND SPECIAL CORE COMPONENTS FROM EITHER THE FUEL-STORAGE OR REACTOR TRANSFER ROTOR, AND TO TRANSPORT AND DEPOSIT THE SUBASSEMBLY OR COMPONENT INTO THE OPPOSITE ROTOR. THIS PAPER DISCUSSES THE EVOLUTION OF THE FUEL TRANSPORT FACILITY FROM THE ORIGINAL MULTI-BARRELED MACHINE TO THE PRESENT SINGLE-BARREL MACHINE. THE DECISION TO PURCHASE THE NEW MACHINE WAS MADE AFTER REVIEW OF THE MAINTENANCE REQUIREMENTS NECESSITATED BY DESIGN DEFICIENCIES IN THE ORIGINAL MACHINE.

AVAILABILITY - R.O. WILLIAMS, JR., ATOMICS INTERNATIONAL, P.O. BOX 309, CANOGA PARK, CALIFORNIA 91304

REACTOR, BREEDER + DESIGN CRITERIA + REACTOR, FAST + FERMI (LMFBR) + FUEL HANDLING + REACTOR, LMCR + TRANSPORTATION AND HANDLING + REACTOR, POWER + FUEL HANDLING MACHINE + REMOTE MANIPULATING AND VIEWING + OPERATING EXPERIENCE + REACTOR, LMFBR

12-034708

ALSO IN CATEGORY 13

DATED 1969

DAVIS JR

EBR-II FUEL HANDLING EQUIPMENT

ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO

13 PAGES, 10 FIGURES, PP. 161-173 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

OVER SEVEN YEARS OF EXPERIENCE HAS BEEN GAINED WITH THE EBR-II FUEL HANDLING SYSTEM, AND SINCE MAY, 1963, FUEL HANDLING HAS BEEN CONDUCTED IN A HIGH-TEMPERATURE SODIUM ENVIRONMENT. NORMAL PROBLEMS WERE EXPERIENCED, BUT THE HANDLING OF FUEL DURING THIS PERIOD CAN BE CONSIDERED QUITE SUCCESSFUL. MODIFICATIONS TO INCREASE THE PERFORMANCE AND RELIABILITY OF THE FUEL HANDLING EQUIPMENT HAVE BEEN A CONTINUING EFFORT SINCE THE ORIGINAL SYSTEM CHECKOUT.

AVAILABILITY - JOE R. DAVIS, ARGONNE NATIONAL LAB., P.O. BOX 2528, IDAHO FALLS, IDAHO 83401

REACTOR, BREEDER + CONTROL SYSTEM + REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + SODIUM + REACTOR, TEST + HIGH TEMPERATURE + STORAGE CONTAINER + EQUIPMENT + FUEL REPROCESSING + MAINTENANCE, REMOTE + OPERATING EXPERIENCE + PYROCHEMICAL

12-034784

ALSO IN CATEGORY 9

DATED 1969

ASTLEY ER

FAST FLUX TEST FACILITY - PERIODIC TECHNICAL REPORT - OCTOBER 1968 THROUGH FEBRUARY 1969

BATTELLE NORTHWEST, RICHLAND, WASHINGTON

BNWL-941 +. 200 PAGES, FIGURES, TABLES, MAY, 1969

COVERS PROGRAM MANAGEMENT, AND FFTF OPERATIONS, PLANT DESIGN, COMPONENTS, INSTRUMENTATION AND CONTROLS, SODIUM TECHNOLOGY, CORE DEVELOPMENT AND DESIGNS, SAFEGUARDS, AND CLADDING DEVELOPMENT

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151
\$3.00 COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + REACTOR, LMCR + SODIUM + REACTOR, TEST + FFTF (TR) + INSTRUMENT, CONTROL + CORROSION + COOLANT CHEMISTRY + DESIGN STUDY + CORE COMPONENTS + SAFETY ANALYSIS + FUEL ELEMENTS

12-035044

DATED 1969

LEE WT

CORROSION OF TYPE 18/18/2 STAINLESS STEEL IN 1200 DEGREE F. STATIC SODIUM

LMCR-69-11 +. 35 PAGES, 13 FIGURES, 2 TABLES, 12 REFERENCES, MAY 15, 1969

TO DETERMINE THE DEGREE OF CORROSION OR OTHER DEGRADATION THAT TYPE 18-18-2 STAINLESS STEEL (CR-NI-SI) WOULD EXPERIENCE IN A SODIUM ENVIRONMENT, 18 COUPONS WERE IMMERSUED IN 1200 F STATIC SODIUM FOR PERIODS UP TO 5060 HR. IN ADDITION, 12 TYPE 304 STAINLESS STEEL TABS WERE SIMULTANEOUSLY EXPOSED FOR COMPARISON. THE AVERAGE OXYGEN IMPURITY IN THE SODIUM WAS ABOUT 30 PPM. THE RESULTS OF THESE PRELIMINARY TESTS SHOW THAT TYPE 18-18-2 STAINLESS STEEL IS SUFFICIENTLY CORROSION-RESISTANT IN 1200F SODIUM TO WARRANT MORE THOROUGH EVALUATION.

SODIUM + TESTING + STEEL, STAINLESS + CORROSION + HIGH TEMPERATURE + EXPERIMENT + FUEL REPROCESSING + REACTOR, LMFBR

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-035195
COCHRAN JT
SEFOR FUEL ASSEMBLY SODIUM FLOW TESTS
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIFORNIA
GEAP-5466 +. 50 PAGES, SEPTEMBER 1968

DATED 1968

REPORT DESCRIBES RESULTS OF EXPERIMENTS IN SODIUM WITH A CORE CHANNEL AND ITS ASSOCIATED FUEL FOR USE IN THE SEFOR REACTOR CORE. ALSO INCLUDED IN THE EXPERIMENT WAS A DEVELOPMENT MODEL OF THE IN-CORE FLOW METER USED IN THE INSTRUMENTED FUEL ASSEMBLIES. A SUBSIDIARY TEST WAS PERFORMED TO EVALUATE THE GALLING EFFECTS IN SODIUM WHICH RESULT FROM THE RELATIVE MOTION BETWEEN THE TIGHTENER ROD AND THE FUEL SUPPORT SPRINGS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + SODIUM + TESTING + REACTOR, PULSED + SEFOR (RE) + VIBRATION + CORE COMPONENTS + FUEL ELEMENTS

12-035196
COCHRAN JT
SEFOR FUEL ROD SODIUM THERMAL SHOCK TESTS
U. S. ATOMIC ENERGY COMMISSION
GEAP-5467 +. 70 PAGES, AUGUST 1968

DATED 1968

REPORT DESCRIBES THE RESULTS OF THE THERMAL SHOCK EXPERIMENTS WITH MODEL FUEL AND EXTENSION RODS OF THE TYPES PROPOSED FOR USE IN THE SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR CORE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + SODIUM + TESTING + THERMAL MECHANICAL EFFECT + REACTOR, PULSED + SEFOR (RE) + FUEL ELEMENTS

12-035266
HARRIS RA
PRELIMINARY ANALYSIS OF POSTULATED MAXIMUM ACCIDENTS FOR THE FFTF - SUPPLEMENT 1 - ADDITIONAL CALCULATIONS FOR SODIUM-IN CONDITIONS
BATTELLE NORTHWEST, RICHLAND, WASHINGTON
BNWL-760 - SUPPLEMENT 1 +. 50 PAGES, 13 FIGURES, 3 TABLES, 15 REFERENCES, DECEMBER 1968

DATED 1968

PRESENTS A REEVALUATION OF THE CORE DISASSEMBLY CALCULATIONS REPORTED IN BNWL-760 FOR THE PARTICULAR CASE OF SODIUM-IN CONDITIONS. THE VALIDITY OF THE SODIUM-IN EQUATION OF STATE USED IN THE REFERENCED DOCUMENT IS REEXAMINED, AND A NEWLY DEVELOPED EXPRESSION, BASED ON EXPERIMENTAL DATA IN THE TEMPERATURE DOMAIN OF INTEREST, IS PRESENTED. THE EFFECT OF SODIUM VAPOR PRESSURE AS A DISASSEMBLY FORCE AND THE EFFECTS OF SODIUM COMPRESSIBILITY ARE PARAMETRICALLY EXAMINED. THE ANALYSIS INDICATES THAT THE DISASSEMBLY CALCULATIONS REPORTED IN BNWL-760 FOR SODIUM-IN CONDITIONS ARE PROBABLY QUITE PESSIMISTIC.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + SODIUM + REACTOR, TEST + ACCIDENT, HYPOTHETICAL + FFTF (TR) + URANIUM DIOXIDE

12-035363
REVIEW OF FUEL MODELING WORK AT ORNL
OAK RIDGE NATIONAL LABORATORY, TENNESSEE
CF-69-4-11 +. 39 PAGES, 13 FIGURES, 1 TABLE, 47 REFERENCES, APRIL 7, 1969

DATED 1969

ANALYTICAL PREDICTIONS AND EXPERIMENTAL DATA ARE COMPARED FOR IRRADIATION EXPERIMENTS WITH STAINLESS STEEL CLAD (U,Pu)O₂ FUEL PINS. THE STATUS OF THE PROFIL AND FMODEL CODES FOR PREDICTING FUEL ROD PERFORMANCE IS DISCUSSED WITH APPLICATIONS.

COMPUTER PROGRAM + COMPARISON, THEORY AND EXPERIENCE + STEEL, STAINLESS + URANIUM DIOXIDE + IRRADIATION TESTING + PLUTONIUM DIOXIDE + REACTOR, LMFBR + FUEL ELEMENTS

12-036504
BROOKS BP
FFTF VALVE EVALUATION
ATOMIC INTERNATIONAL, INC.
NAA-SR-MEMO-12282 +. 23 PAGES, FEBRUARY 1, 1967

DATED 1967

THE DISCUSSIONS CONTAINED IN INTERNAL AT DOCUMENTS ISSUED DURING 1966 PERTAINING TO THE PRIMARY SERVICE RATING OF VALVES FOR THE VARIOUS FFTF SERVICE CONDITIONS ARE EVALUATED. A BIBLIOGRAPHY OF DOCUMENTS REPRESENTING AN ACCUMULATION OF LIQUID METAL VALVE DATA AND USER EXPERIENCE IS PRESENTED. BASED ON A TABULATION OF VALVES UTILIZED IN THE SEVERAL LIQUID METAL SYSTEMS SHOWN IN BNWL-CC-400, RECOMMENDATIONS ARE MADE OF VALVE CONCEPTS FOR THE REQUIRED SERVICE CONDITIONS. REFERENCE TO SPECIFIC TEST OR USER DATA IS NOTED TO CORROBORATE THE RECOMMENDATIONS.

CATEGORY 12
PLANT SAFETY FEATURES - REACTOR

12-036504 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE/STAT

REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + HIGH TEMPERATURE + TECHNICAL SPECIFICATIONS + R
AND D PROGRAM + VALVES

12-036795

DATED 1969

KELLER DL

PROGRESS ON DEVELOPMENT OF MATERIALS AND TECHNOLOGY FOR ADVANCED REACTORS DURING JANUARY THROUGH MARCH, 1969
BATTELLE MEMORIAL INSTITUTE, COLUMBUS, OHIO
BMI-1862 +. 101 PAGES, FIGURES, TABLES, REFERENCES, APRIL 1969

REPORT COVERS WORK ON (1) URANIUM-PLUTONIUM NITRIDE FUELS FOR LMFBR APPLICATIONS, (2) RADIATION
EFFECTS ON FAST REACTOR CLADDING AND STRUCTURAL MATERIALS, AND (3) CREEP OF REACTOR FUEL
MATERIALS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CLADDING + PLUTONIUM + URANIUM + REACTOR + EXPERIMENT + CREEP + REACTOR, LMFBR + RADIATION EFFECT + FUEL
COOLANT INTERACTION + FUEL ELEMENTS

12-037429

DATED 1969

RUTHER WE + CLAAR TD + MATRAS S

THE ATTACK ON THE CLADDING OF EBR-II DRIVER FUEL ELEMENT C-249-16

ARGONNE NATIONAL LABORATORY

ANL-7537 +. 19 PAGES, 22 FIGURES, 3 TABLES, MARCH 1969

AN INVESTIGATION WAS MADE TO DETERMINE THE CAUSE OF GRAIN-BOUNDARY ATTACK IN SELECTIVE AREAS OF
THE TYPE 304L STAINLESS STEEL CLADDING OF A DRIVER FUEL ELEMENT REMOVED FROM EBR-II. A
HYPOTHESIS THAT IMPROPER CLEANING TECHNIQUES MAY HAVE CAUSED THE DAMAGE WAS REJECTED AFTER
VARIATIONS OF NORMAL CLEANING METHODS FAILED TO DUPLICATE THE PHENOMENON. CHEMICAL AND ELECTRON-
MICROPROBE ANALYSES, ALONG WITH METALLOGRAPHIC WORK, INDICATED THAT AN INCREASE IN CARBON
CONCENTRATION WAS ASSOCIATED WITH THE AREAS OF SELECTIVE ATTACK. FURTHER TESTING SHOWED THAT THE
MICROSTRUCTURE OF A STAINLESS STEEL SURFACE THAT HAD BEEN INTENTIONALLY CONTAMINATED WITH A
CARBONACEOUS SUBSTANCE BEFORE EXPOSURE TO SODIUM WAS ESSENTIALLY THE SAME AS THE MICROSTRUCTURE
OF THE CLADDING THAT HAD BEEN EXPOSED IN EBR-II.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + CLADDING + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + TESTING + STEEL, STAINLESS +
CHEMICAL ANALYSIS + CORROSION + FUEL ELEMENTS

12-037838

ALSO IN CATEGORIES 5 AND 17

DATED 1968

HIGH-TEMPERATURE LIQUID-METAL TECHNOLOGY REVIEW

BROOKHAVEN NATIONAL LABORATORY

BNL-50115(PR-32) +. 88 PAGES, 8 FIGURES, REFERENCES, APRIL 1968

THE PURPOSE OF THE REVIEW IS TO PROVIDE UP-TO-DATE INFORMATION ON THE VARIOUS RESEARCH AND
DEVELOPMENT PROGRAMS IN THE UNITED STATES IN THE FIELD OF HIGH-TEMPERATURE LIQUID-METAL
TECHNOLOGY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + REACTOR, BREEDER + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + THERMODYNAMICS + REACTOR
COOLANT + CORROSION + METAL, LIQUID + MASS TRANSFER + PROPERTY, PHYSICAL + REVIEW + R AND D PROGRAM

12-037850

ALSO IN CATEGORIES 5 AND 6

DATED 1967

DESCAMPS C + EGLEME M + LEE J + MICHEL A

ANALYSIS OF THERMAL AND HYDRAULIC BEHAVIOR OF SODIUM DURING A PROMPT POWER INSERTION IN A FUEL ASSEMBLY

COMMISSARIAT A L'ENERGIE ATOMIQUE, PARIS

CONF-670916 +. 17 PAGES, 14 REFERENCES, SEPT. 1967

DURING THE STUDY AND THE DESIGN OF AN IN-PILE NA LOOP FOR FUEL ELEMENT RUPTURE TEST, THE FUEL
DISPERSION AND SUBSEQUENT NA BEHAVIOR AFTER A PROMPT POWER INSERTION HAD TO BE EVALUATED. THE
FUEL HEATING, THE CLADDING RUPTURE, THE FUEL DISPERSION IN THE NA AND THE SUDDEN NA BOILING WERE
STUDIED. THE EXPERIMENTAL MEANS ARE DESCRIBED WHICH COULD CONTROL THE PROPOSED THEORETICAL MODEL.

AVAILABILITY - C. DESCAMPS, ET. AL., COMMISSARIAT A L'ENERGIE ATOMIQUE, PARIS, FRANCE

FAILURE, FUEL ELEMENT + HYDRAULIC ANALYSIS + IN PILE EXPERIMENT + REACTOR, LMCR + ACCIDENT, REACTIVITY +
SODIUM + THERMAL ANALYSIS + ANALYTICAL MODEL + ACCIDENT MODEL + FAILURE, CLADDING + BOILING

CATEGORY 13
PLANT SAFETY FEATURES - NONREACTOR

13-005315 DATED 1960
KLIMA BB
SRE FUEL ELEMENTS REPROCESSING - SAFETY REPORT
OAK RIDGE NATIONAL LABORATORY
37 PAGES, 16 FIGURES, 9 TABLES- ORNL-CF-59-11-122, REV 1, FEBRUARY 5, 1960- OTS

AN ANALYSIS HAS BEEN MADE OF THE STEPS TAKEN TO ASSURE SAFE HANDLING OF THE SRE FUEL ELEMENTS DURING REPROCESSING IN THE POWER REACTOR FUELS REPROCESSING PROGRAM, MECHANICAL PHASE, CELL COMPLEX IN BUILDING 3026.

ADMINISTRATIVE CONTROL + CRITICALITY SAFETY + EXPLOSION + FIRE + REACTOR, LMCR + REPORT, SAR + ORNL + SRE (RE) + FUEL REPROCESSING

13-023229 ALSO IN CATEGORY 3 DATED 1968
UNGER WE + CROUSE DJ + WATSON CD + IRVINE AR
LMFBR FUEL PROCESSING STUDIES STATUS REPORT FOR FEBRUARY 1968
OAK RIDGE NATIONAL LABORATORY
ORNL-CF-68-3-13 +. 21 PAGES, 4 FIGURES, 3 TABLES, MARCH 11, 1968

MONTHLY STATUS REPORT OF FAST BREEDER REACTOR AQUEOUS PROCESS DEVELOPMENT IN CHEMICAL TECHNOLOGY DIVISION. STUDIES UNDERWAY OF DISPOSAL OF SODIUM, BEHAVIOR AND REMOVAL OF IODINE, SOLVENT STABILITY UNDER RADIATION AND CHEMICAL DEGRADATION, REMOVAL OF NOBLE GASES, SHIPPING OF REACTOR FUEL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + IODINE + REACTOR, LMCR + NOBLE GAS + PLUTONIUM + SHIPPING CONTAINER + SODIUM + URANIUM + ORNL + RADIOCHEMICAL PROCESSING + EXPERIMENT + SOLVENT EXTRACTION PROCESS + FUEL REPROCESSING

13-024471 ALSO IN CATEGORY 14 DATED 1967
BROWN CL + LAWRENCE LA
FFTF CRITICALITY SAFETY SYSTEM 72 - UNDERWATER EXAMINATION FACILITY FEASIBILITY OF USING A SOLUBLE POISON FOR CRITICALITY CONTROL
BATTILLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-594 +. 15 PAGES, 7 REFERENCES, SEPT. 1967

STUDY TO DETERMINE AMOUNT OF SOLUBLE SODIUM BORATE REQUIRED TO ENSURE CRITICALITY SAFETY. MINIMUM CONCENTRATION CALCULATED - 3 G B/LITER (0.59 G B-10/LITER). TWO METHODS RECOMMENDED TO ENSURE THAT BORON IS PRESENT ABOVE REQUIRED MINIMUM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

BROWN + CRITICALITY SAFETY + REACTOR, FAST + REACTOR, LMCR + POISON, SOLUBLE + FUEL STORAGE + REACTOR, TEST + FFTF (TR) + CRITICALITY EXPERIMENT + REACTOR + RADIOCHEMICAL PLANT SAFETY + WATER TREATMENT + HANFORD SITE + FUEL REPROCESSING

13-025266 ALSO IN CATEGORY 12 DATED 1967
KETZLACH N
NUCLEAR SAFETY SRE CORE III FUEL ELEMENT STORAGE
ATUMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-MEMO-12472 +. 7 PAGES, MAY 17, 1967

THE NUCLEAR SAFETY OF A SINGLE-PLANE ARRAY OF SRE CORE-III FUEL ELEMENTS, SPACED ON 12-INCH CENTERS, HAS BEEN EVALUATED. THE ARRAY IS LESS THAN 45% OF CRITICAL, BASED ON THE MASS/UNIT-AREA PRINCIPLE FOR THE FUEL ROD SIZES OF INTEREST. THIS IS INDEPENDENT OF THE DEGREE OF WATER FLOODING. A FULLY FLOODED ARRAY OF THE ASSEMBLY OF ELEMENTS SPACED AS STATED ABOVE WOULD HAVE A K-EFF OF LESS THAN 0.6. AN UNMODERATED ASSEMBLY OF FUEL OF THE GIVEN ENRICHMENT HAVING A NEUTRON-REFLECTION EQUIVALENT TO WATER WOULD REQUIRE AT LEAST THREE TIMES AS MUCH FUEL FOR CRITICALITY, INDEPENDENT OF SPACING BETWEEN ELEMENTS OR THE THICKNESS OF WATER REFLECTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + FUEL STORAGE + URANIUM CARBIDE + SRE (RE) + STORAGE CONTAINER + FUEL REPROCESSING + FUEL ELEMENTS

13-027777 DATED 1968
GREENIER WS
SOME FACTORS AFFECTING THE CHOICE OF A SOLVENT EXTRACTION CONTACTOR FOR THE 1ST CYCLE REPROCESSING OF LMFBR FUELS
OAK RIDGE NATIONAL LAB., TENNESSEE
ORNL-TM-2290 +. 23 PAGES, 1 FIGURE, 9 TABLES, 18 REFERENCES, SEPT. 1968

THREE VERY DIFFERENT TYPES OF SOLVENT EXTRACTION CONTACTORS ARE BEING CONSIDERED FOR INCLUSION IN THE FIRST CYCLE LMFBR FUEL REPROCESSING CONCEPT. THE PULSED EXTRACTION COLUMN, THE SAVANNAH

CATEGORY 13
PLANT SAFETY FEATURES - NONREACTOR

13-027777 *CONTINUED*

RIVER CENTRIFUGAL CONTACTOR, AND THE ORNL STACKED-CLONE CONTACTOR HAVE BEEN COMPARED IN THIS REPORT USING A POSSIBLE LMFBR FLOWSHEET WITH TYPICAL FEED AND VARIOUS DESIGN AND OPERATING INFORMATION THAT IS AVAILABLE. CONTACTORS WERE SIZED FOR A REPROCESSING PLANT HAVING A CAPACITY OF 5 METRIC TONS (URANIUM + PLUTONIUM + FISSION PRODUCT) PER DAY. COMPARISONS WERE MADE ON THE BASIS OF APPROXIMATE SOLVENT RADIATION DOSAGE. IN BOTH THE CENTRIFUGAL CONTACTOR AND THE STACKED-CLONE CONTACTOR THIS WAS GREATER THAN 20% OF THAT IN THE PULSED COLUMN. ADDITIONAL COMPARISONS MUST BE MADE ON THE BASIS OF COST.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + RADIATION DAMAGE + COMPARISON, THEORY AND EXPERIENCE + EQUIPMENT + SOLVENT EXTRACTION PROCESS + FUEL REPROCESSING

13-028951

DATED 1968

LEWIS LC

THE PHYSICAL PROPERTIES OF NITRATE SOLUTIONS OF EBR-II FUEL ELEMENTS
IDAHO NUCLEAR CORP., NATIONAL REACTOR TESTING STATION, IDAHO FALLS, IDAHO
IN-1192 +. 16 PAGES, 14 FIGURES, 3 TABLES, 4 REFERENCES, AUGUST 1968

CONDUCTIVITY, VISCOSITY, AND DENSITY MEASUREMENTS WERE MADE ON HNO₃ SOLUTIONS SIMULATING THOSE ARISING FROM THE ELECTROLYTIC DISSOLUTION OF EBR-II NUCLEAR FUEL ELEMENTS. CONDUCTIVITY MEASUREMENTS WERE MADE AT 25 AND 90 C. SPECIFIC GRAVITY MEASUREMENTS WERE MADE AT 25 AND 96 C. VISCOSITY MEASUREMENTS WERE MADE AT 25 C. EQUATIONS RELATING VISCOSITY WITH COMPOSITION AND SPECIFIC GRAVITY WITH COMPOSITION AND TEMPERATURE WERE DERIVED. THE SOLUBILITY OF BORIC ACID IN THESE SOLUTIONS WAS ALSO DETERMINED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, 22151, \$3.00 COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + URANIUM + REACTOR, ERDA OWNED + EXPERIMENT + FUEL REPROCESSING + DISSOLUTION + NITRATE

13-030664

ALSO IN CATEGORY 12

DATED 1968

WINEGARDNER WK

CONSIDERATIONS ON THE MEDIA FOR DECAY STORAGE OF FUEL ELEMENTS

BATTELLE MEMORIAL INSTITUTE

BNWL-909 +. 34 PAGES, 23 REFERENCES, NOVEMBER 1968

PREPARED IN RESPONSE TO FFTF DIVISION REQUEST FOR INFORMATION AND RECOMMENDATIONS CONCERNING THE DECAY STORAGE OF IRRADIATED FUEL ELEMENTS. SUMMARIZES INFORMATION OF STORAGE TECHNIQUES. DISCUSSES INFORMATION RELATED TO FUEL CLADDING CORROSION PROBLEMS THAT MAY RESULT FROM DECAY STORAGE. CONTAINS RECOMMENDATIONS CONCERNING STORAGE MEDIA.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + FUEL STORAGE + REACTOR, TEST + FFTF (TR) + CORROSION + REACTOR + FUEL REPROCESSING

13-030879

ALSO IN CATEGORY 12

DATED 1968

BALKWILL JK

MECHANICAL ELEMENTS OPERATING IN SODIUM AND OTHER ALKALI METALS. VOL. I LITERATURE SURVEY

LIQUID METAL ENGINEERING CENTER

LMEC-68-5 +. 124 PAGES, 17 TABLES, 77 REFERENCES, 1968

SEARCHED LITERATURE REPORTING INFORMATION PUBLISHED SINCE 1950 ON MECHANICAL DEVICES WITH RUBBING SURFACES OPERATING IN ALKALI METALS. SODIUM ENVIRONMENTS ARE EMPHASIZED, WITH DATA ALSO INCLUDED FROM POTASSIUM, NAK, AND LITHIUM EXPERIMENTS. THE CHARACTERISTICS OF POTENTIALLY USEFUL MATERIALS THAT ARE IMPORTANT IN LIQUID METAL APPLICATIONS ARE DISCUSSED. MATERIAL COMPATIBILITY AND FRICTION/WEAR TESTING PROGRAMS CONDUCTED SINCE 1950 ARE DESCRIBED. EXPERIMENTAL AND OPERATING EXPERIENCE WITH FULL-SIZE COMPONENTS IS PRESENTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

ALLOY + BIBLIOGRAPHY + REACTOR, LMCR + SODIUM + POTASSIUM + CORROSION + MATERIAL + LITHIUM + EXPERIMENT + SURVEY + OPERATING EXPERIENCE

13-033337

DATED 1968

OCHSENFELD W

REPROCESSING FUEL ELEMENTS FOR FAST BREEDERS

KERNFORSCHUNGSZENTRUM, KARLSRUHE (WEST GERMANY)

EUR-FNK-581 + KFK-852 +. 14 PAGES, TAB., REF., SEPT. 1968

COVERS THE RECOVERY AND PURIFICATION OF PLUTONIUM, CERAMICS, AND NUCLEAR FUEL ACTINIDES SUCH AS NEPTUNIUM, AMERICIUM, AND CURIUM FROM FUEL ELEMENTS FOR FAST BREEDERS.

CATEGORY 13
PLANT SAFETY FEATURES - NONREACTOR

13-033337 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + REACTOR, BREEDER + REACTOR, FAST + PLUTONIUM + URANIUM + NEPTUNIUM + AMERICIUM + GERMANY + CURIUM + RADIOISOTOPE + EXPERIMENT + SOLVENT EXTRACTION PROCESS + KARLSRUHE + FUEL REPROCESSING + REACTOR, LMFBR + ACTINIDE

13-033428 ALSO IN CATEGORIES 3 AND 18 DATED 1968

LMFBR PROGRAM PLAN, VOLUME 8 - FUEL RECYCLE

LMFBR PROGRAM OFFICE, ARGONNE NATIONAL LAB.

WASH-1108 +. 350 PAGES, FIGURES, TABLES, AUGUST 1968

CONTENTS INCLUDE CHARACTERISTICS OF FUELS, FABRICATION, COSTS, PROCESSING OF CORE AND BLANKET, AND A THOROUGH DISCUSSION OF THE TASKS, INCLUDING COORDINATION OF RECYCLE DEVELOPMENTS, SYSTEM DEMONSTRATIONS, HANDLING AND SHIPPING, RECOVERY, FUEL PREPARATION AND FABRICATION, AND GENERAL STUDIES. APPENDICES INCLUDE FLOW SHEETS, SUB-TASKS, ECONOMICS, AND A COLLATION OF PROGRAM TASKS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, BREEDER + ECONOMICS + FABRICATION + REACTOR, FAST + RADIOCHEMICAL PROCESSING + RECOVERY PROCESS + FUEL REPROCESSING + R AND D PROGRAM + REACTOR, LMFBR + FUEL MANAGEMENT + JACOBS + FUEL ELEMENTS

13-034696 ALSO IN CATEGORY 12 DATED 1969

BOYD CL + MEYERS DJ + RAMSEY EB

THE FFTF FUEL EXAMINATION FACILITY

BATTELLE MEMORIAL INSTITUTE, PACIFIC NORTHWEST LABORATORY, RICHLAND, WASHINGTON

16 PAGES, 3 FIGURES, 1 TABLE, 10 REFERENCES, PP. 3-18 OF PROCEEDINGS OF THE 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

THE FUEL EXAMINATION FACILITY OF THE FAST FLUX TEST FACILITY (FFTF) WILL SERVE THE OPERATIONS STAFF AND EXPERIMENTERS WITH A RELIABLE MEANS, UNDER CONTROLLED CONDITIONS, FOR PERFORMING PROMPT NONDESTRUCTIVE EXAMINATION OF EXPERIMENTS AND OTHER COMPONENTS IRRADIATED IN THE FTR. EXAMINATION DATA WILL BE OBTAINED FROM SELECTED EXPERIMENTS THAT REQUIRE REACTOR REINSERTION, AND FROM EXPERIMENTS OR OTHER CORE COMPONENTS WHOSE IRRADIATION IS COMPLETE.

AVAILABILITY - C.L. BOYD + D.J. MEYERS + E.B. RAMSEY, BATTELLE MEMORIAL INSTITUTE PACIFIC NORTHWEST LABORATORY, P.O. BOX 999, RICHLAND, WASHINGTON 99352

BIBLIOGRAPHY + REACTOR, FAST + HOT CELL + REACTOR, LMCR + REACTOR, TEST + VENTILATION SYSTEM + FFTF (TR) + DESIGN STUDY + REACTOR + IRRADIATION TESTING + GAMMA + EXAMINATION + REMOTE MANIPULATING AND VIEWING + ALPHA FACILITIES + CORE COMPONENTS + FUEL ELEMENTS

13-034697 DATED 1969

WATSON CD + LOWRIE RS + GROENIER WS + WEST GA + CLINTON SD

HEAD-END PROCESSING OF SPENT LMFBR FUEL

OAK RIDGE NATIONAL LAB., TENN.

20 PAGES, 17 FIGURES, 2 TABLES, 19 REFERENCES, PP. 19-38 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

THE HEAD-END REPROCESSING STEPS AND EQUIPMENT FOR RECOVERING SPENT LMFBR FUEL AT A RATE OF FIVE METRIC TONS PER DAY WERE INVESTIGATED IN A CONCEPTUAL DESIGN STUDY. THE STEPS, WHICH CONSIST OF RECEIVING, CLEANING STORAGE, DISASSEMBLY, SINGULARIZATION, SHEARING, OXIDATION, AND LEACHING, WERE SCOPED AS TO THE PROBLEMS AND EQUIPMENT INVOLVED AND AS TO THE FEASIBILITY OF CARRYING OUT THE STEPS IN AN ALPHA-GAMMA-CONTAINED FACILITY. THE STUDY SHOWED THAT AN EXTENSIVE DEVELOPMENTAL AND EVALUATION PROGRAM WOULD BE REQUIRED TO ENSURE SATISFACTORY PERFORMANCE OF THE REMOTELY OPERABLE, HIGH-THROUGHPUT CONTINUOUS EQUIPMENT.

AVAILABILITY - C.D. WATSON, OAK RIDGE NATIONAL LAB., P.O. BOX X, OAK RIDGE, TENNESSEE 37830

BIBLIOGRAPHY + FUEL STORAGE + DESIGN STUDY + REMOTE MANIPULATING AND VIEWING + EQUIPMENT + SOLVENT EXTRACTION PROCESS + FUEL REPROCESSING + DISSOLUTION + REACTOR, LMFBR

13-034706 ALSO IN CATEGORIES 3 AND 12 AND 17 DATED 1969

JOHNSON KP + MATTE J

OPERATING EXPERIENCE WITH THE FERMI FUEL HANDLING CASK CAR

POWER REACTOR DEVELOPMENT COMPANY, MONROE, MICHIGAN + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN

15 PAGES, 9 FIGURES, PP. 135-149 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

THE ENRICO FERMI ATOMIC POWER PLANT WAS PROVIDED WITH A CASK CAR FOR TRANSPORTING SPENT FUEL BETWEEN THE REACTOR BUILDING AND THE FUEL AND REPAIR BUILDING. THIS CAR, A MULTI-STATION, SELF-CONTAINED UNIT, EXHIBITED A VARIETY OF PROBLEMS AND DEFECTS IN EARLY OPERATION. ANALYSIS OF THESE PROBLEMS AND THE AVAILABLE SOLUTIONS REQUIRED TO ACHIEVE SATISFACTORY PERFORMANCE UNDER DESIGNED OPERATING CONDITIONS LED TO CONSIDERATION OF REPLACING THE CASK CAR WITH A NEW FACILITY.

AVAILABILITY - K.P. JOHNSON, POWER REACTOR DEVELOPMENT COMPANY, P. O. BOX 725, MONROE, MICHIGAN 48161

REACTOR, BREEDER + FAILURE, EQUIPMENT + REACTOR, FAST + FERMI (LMFBR) + FUEL HANDLING + REACTOR, LMCR +

CATEGORY 13
PLANT SAFETY FEATURES - NONREACTOR

13-034706 *CONTINUED*

SHIPPING CONTAINER + TRANSPORTATION AND HANDLING + FUEL HANDLING MACHINE + OPERATING EXPERIENCE + REACTOR, LMFBR

13-034708 ALSO IN CATEGORY 12

DATED 1969

DAVIS JR

EBR-II FUEL HANDLING EQUIPMENT

ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO

13 PAGES, 10 FIGURES, PP. 161-173 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

OVER SEVEN YEARS OF EXPERIENCE HAS BEEN GAINED WITH THE EBR-II FUEL HANDLING SYSTEM, AND SINCE MAY, 1963, FUEL HANDLING HAS BEEN CONDUCTED IN A HIGH-TEMPERATURE SODIUM ENVIRONMENT. NORMAL PROBLEMS WERE EXPERIENCED, BUT THE HANDLING OF FUEL DURING THIS PERIOD CAN BE CONSIDERED QUITE SUCCESSFUL. MODIFICATIONS TO INCREASE THE PERFORMANCE AND RELIABILITY OF THE FUEL HANDLING EQUIPMENT HAVE BEEN A CONTINUING EFFORT SINCE THE ORIGINAL SYSTEM CHECKOUT.

AVAILABILITY - JOE R. DAVIS, ARGONNE NATIONAL LAB., P.O. BOX 2528, IDAHO FALLS, IDAHO 83401

REACTOR, BREEDER + CONTROL SYSTEM + REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + SODIUM + REACTOR, TEST + HIGH TEMPERATURE + STORAGE CONTAINER + EQUIPMENT + FUEL REPROCESSING + MAINTENANCE, REMOTE + OPERATING EXPERIENCE + PYROCHEMICAL

13-034711 ALSO IN CATEGORY 3

DATED 1969

MALECHA RF + COLEMAN LF + FINEMAN P + KELECUS A + SLAWECKI MA

EBR-II INTRASITE SPENT-FUEL SHIPPING CASKS

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

7 PAGES, 3 FIGURES, 1 REFERENCES, PP. 205-211 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

A CHANGE IN THE STATUS OF THE EXPERIMENTAL BREEDER REACTOR II (EBR-II) FROM AN EXPERIMENTAL REACTOR TO A NEUTRON IRRADIATION FACILITY NECESSITATED THE CONSTRUCTION OF TWO DIFFERENT CASKS. THE LEAD-SHIELDED CASKS ARE TO BE USED TO TRANSFER SPENT FUEL TO AN ON-SITE STORAGE AREA. ONE CASK HAS BEEN DESIGNED AND BUILT WHICH PROVIDES AN WATER-COOLED FUEL CAVITY. THE SECOND CASK IS DESIGNED FOR BOTTOM LOADING AND INCORPORATES AN INTEGRAL, TOTALLY ENCLOSED, RECIRCULATING AIR COOLING SYSTEM FOR THE FUEL.

AVAILABILITY - R.F. MALECHA, ARGONNE NATIONAL LABORATORY, 9700 S. CASS AVENUE, ARGONNE, ILLINOIS 60439

REACTOR, BREEDER + DESIGN CRITERIA + REACTOR, FAST + REACTOR, LMCR + SHIPPING CONTAINER + TRANSPORTATION AND HANDLING

13-036276

DATED 1969

UNGER WE + BLANCO RE + WATSON CD + CROUSE DJ + IRVINE AR

AQUEOUS PROCESSING OF LMFBR FUELS - PROGRESS REPORT - JUNE 1969 - NO. 4

ORNL-TM-2650

47 PAGES, FIGURES, TABLES, REFERENCES, JULY 1969

THIS REPORT CONTINUES A SERIES OUTLINING PROGRESS IN THE DEVELOPMENT OF AQUEOUS METHODS FOR THE REPROCESSING OF LMFBR FUELS. DEVELOPMENT WORK IS REPORTED ON PROBLEMS OF IRRADIATED FUEL TRANSPORT TO THE PROCESSING FACILITY, THE DISSOLUTION OF THE FUEL AND THE CHEMICAL RECOVERY OF THE PUO₂-UO₂ VALUES, THE CONTAINMENT OF VOLATILE FISSION PRODUCTS, AND PRODUCT PURIFICATION FOR SUBSEQUENT RECONSTITUTION INTO REACTOR FUEL. PERTINENT EXPERIMENTAL RESULTS ARE PRESENTED FOR THE INFORMATION OF THOSE IMMEDIATELY CONCERNED WITH THE FIELD. DETAILED DESCRIPTION OF EXPERIMENTAL WORK AND DATA ARE INCLUDED IN TOPICAL REPORTS AND IN THE CHEMICAL TECHNOLOGY DIVISION ANNUAL REPORT.

CRITICALITY SAFETY + WASTE MANAGEMENT + SHIELDING + SOLVENT EXTRACTION PROCESS + FUEL REPROCESSING + DISSOLUTION + REACTOR, LMFBR

CATEGORY 16
METEOROLOGICAL CONSIDERATIONS

16-029511 ALSO IN CATEGORIES 2 AND 18 DATED 1968
RESPONSE TO QUESTION M-2 - ON-SITE METEOROLOGICAL DATA
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE M-2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE ON-SITE METEOROLOGICAL DATA DESCRIBED IN TABLE III-5 OF THE FDSAR. (ANSWER) REPORT IN
FINAL STAGES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SITE CLIMATOLOGY + METEOROLOGY + SEFOR (RE) + RESPONSE TO NRC QUESTION

16-032915 ALSO IN CATEGORIES 2 AND 18 DATED 1968
AEC QUESTIONS - 11 TOPICS
GENERAL ELECTRIC
50231-21 +. 150 PAGES, 25 FIGURES, TABLES, SEFOR, PSAR, SUPPLEMENT 19, OCTOBER 1968, DOCKET 50-231

SEVENTY-THREE PAGES DEVOTED TO RESPONSES TO 11 AEC QUESTIONS ON CONTROL SYSTEM MODIFICATION,
AUXILIARY POWER, REFUELING ACCIDENT, THE EXCURSION DEVICE (FRED) LEAKS, TRANSIENT TEST LOADS,
NOZZLE FAILURE, FUEL-ROD PIN-HOLE LEAKS, SURVEILLANCE PROGRAM, THERMAL INSULATION, AND EFFECT OF
SODIUM IN THE CORE DURING MHA. APPENDIX (43 PAGES) CONSISTS OF SITE METEOROLOGICAL DATA.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

NRC QUESTION + CONTAINMENT VESSEL LOADING + CONTROL SYSTEM + EMERGENCY POWER, ELECTRIC + FAILURE, EQUIPMENT +
REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + ACCIDENT, REFUELING + SITE CLIMATOLOGY + ACCIDENT,
HYPOTHETICAL + REACTOR, PULSED + SEFOR (RE) + THERMAL INSULATION + NOZZLE + REPORT, PSAR + MAIN COOLING
SYSTEM + SURVEILLANCE PROGRAM + LEAK + REACTOR, LMFBR

16-034347 ALSO IN CATEGORY 7 DATED 1969
DICKSON CR + START GE + MARKEE EH
AERODYNAMIC EFFECTS OF THE EBR-II REACTOR COMPLEX ON EFFLUENT CONCENTRATION
AIR RESOURCES LABORATORIES, ENVIRONMENTAL SCIENCE SERVICES ADMINISTRATION
15 PAGES, 17 FIGURES, 2 TABLES, 9 REFERENCES, NUCLEAR SAFETY 10(3), PAGES 228-42 (MAY-JUNE 1969)

DISPERSION OF A TRACER RELEASED FROM THE DOWNWIND SIDE OF THE EXPERIMENTAL BREEDER REACTOR II (EBR-
II) WAS MEASURED IN STABLE AND UNSTABLE ATMOSPHERES. THE TRACER, URANINE DYE, WAS SAMPLED ON
FIVE CONCENTRIC SAMPLING ARCS BETWEEN 30 AND 600 M FROM THE RELEASE POINT. THE DIFFUSION RATE IN
THE EBR-II WAKE CONVERGED TO THE FREE ATMOSPHERE RATE BETWEEN 400 AND 600 M DOWNWIND OF THE
SOURCE POINT.

AEROSOL + AIRBORNE RELEASE + ATMOSPHERIC DIFFUSION + EBR 1 AND 2 (RE) + EFFLUENT + REACTOR, FAST + REACTOR,
LMCR + REACTOR, TEST + TESTING + TRACER, RADIOACTIVE + METEOROLOGY

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-002202 DATED 1964
KUCH L + LOEWENSTEIN W + MONSON H
ADDENDUM TO HAZARD SUMMARY REPORT EXPERIMENTAL BREEDER REACTOR-II (EBR-II)
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
ANL-5719(ADDENDUM) +. 307 PAGES, 44 TABLES, 108 FIGURES, 38 REFERENCES, JANUARY 7, 1964

ADDENDUM LIMITED TO MECHANICAL CHANGES SINCE 1957 REPORT AS RELATED TO REACTOR SAFETY (E.G.,
MANUAL INDEXING OF FUEL-HANDLING MACHINE CHANGED TO COMPUTER CARD). DRY CRITICALITY WAS ACHIEVED
SEPT. 30, 1961, AND REPORT DISCUSSES PROPOSED WET CRITICALITY AND POWER ESCALATION.

EBR 1 AND 2 (RE) + REACTOR, LMCR + REPORT, SAR + MODIFICATION

17-003407 ALSO IN CATEGORIES 6 AND 9 DATED 1963
KEATEN RW + PEARSON EN
REACTIVITY CALCULATIONS AND MEASUREMENTS AT THE SRE
ATOMICS INTERNATIONAL
33 PAGES, 5 TABLES, 10 FIGURES, 21 REFERENCES- ANS TOPICAL MEETING NUCLEAR PERFORMANCE OF POWER-REACTOR CORES,
SAN FRANCISCO, CALIFORNIA, SEPTEMBER 26-27, 1963- TID-7672- PP. 123-156

ORIGINAL CALCULATIONS USING STANDARD TWO-GROUP TECHNIQUES ARE COMPARED TO MEASURED VALUES OF
CRITICAL MASS, SODIUM WORTH, STATISTICAL WEIGHT, FLUX DISTRIBUTIONS, AND TEMPERATURE
COEFFICIENTS. A DISCREPANCY LARGER THAN THE EXPECTED CALCULATIONAL ERROR WAS REVEALED. REVISED
PHYSICS CALCULATIONS HAVE UTILIZED THORIUM AND U-235 RESONANCE INTERVALS CALCULATED BY A MONTE
CARLO CODE, AND THE RESULTS AGREE VERY WELL WITH THE MEASURED VALUES.

COMPUTER, ANALOG + COMPUTER, DIGITAL + REACTOR, LMCR + SODIUM COEFFICIENT + TEMPERATURE COEFFICIENT + SRE (RE)

17-003589 ALSO IN CATEGORIES 6 AND 9 DATED 1964
DARLEY DK + JENKINS OG + LARSON CL + MCCANN PH + MILLS RM + STIENS RR + WALTERS RJ
HALLAM NUCLEAR POWER FACILITY - REACTOR OPERATIONS ANALYSIS PROGRAM SEMIANNUAL PROGRESS REPORT NO. 2, MARCH 1,
1963 - AUGUST 31, 1963
ATOMICS INTERNATIONAL
162 PAGES, 23 TABLES, 51 FIGURES- MARCH 1, 1964, NAA-SR-9265

SECONDARY-TO-PRIMARY SODIUM LEAK WAS IN INTERMEDIATE HEAT EXCHANGER. IN SECONDARY SODIUM PUMP NO.
1, THERE WAS INTERFERENCE BETWEEN THE IMPELLER AND DIFFUSER. CONDENSATE LEAK IN STEAM DUMP WAS
TRACED TO A FAULTY GASKET. SODIUM FLOWMETER CALIBRATION LED TO OVERSHOTS IN POWER. THE THREE
SECONDARY PUMPS WERE BINDING. ONE CONTROL ROD THIMBLE HAD A VERTICAL CRACK APPROXIMATELY 6-1/2
IN. LONG. FURTHER REACTIVITY CHECKS AND CONTROL ROD CALIBRATIONS WERE MADE. THREE FAILURES OF
ZIRCALOY-2 THIMBLES LED TO REPLACE ALL COMPONENTS. CODE HAS BEEN DEVELOPED FOR THE COMPONENT
RELIABILITY PROGRAM.

CONTAINMENT, PRESSURE VENTING + CONTROL ROD BURNUP + FAILURE, EQUIPMENT + HALLAM (LMR) + REACTOR, LMCR +
FAILURE, PIPE + RELEASE RATE

17-003600 ALSO IN CATEGORY 6 DATED 1962
HALL DB
PLUTONIUM AS A FUEL FOR THE LOS ALAMOS MOLTEN PLUTONIUM REACTOR EXPERIMENT
LOS ALAMOS SCIENTIFIC LABORATORY
14 PAGES, 10 FIGURES, 6 REFERENCES- DECEMBER 1962, HW-75007- PP. 19.1-19.14- PROCEEDINGS PLUTONIUM AS A POWER
REACTOR FUEL, AMERICAN NUCLEAR SOCIETY TOPICAL MEETING, RICHLAND, WASHINGTON, SEPTEMBER 13-14, 1962

DESCRIPTION AND OPERATING EXPERIENCE WITH LAMPRE ARE PRESENTED.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR DESCRIPTION + PRESSURE DROP

17-003637 DATED 1964
BROWN KK
OPERATING HALLAM
6 PAGES, 4 FIGURES, 1 TABLE, 7 REFERENCES- NUCLEONICS, 22(9)- 41-46 (SEPTEMBER 1964)

REVIEWS OF EARLY AND RECENT OPERATION ARE PRESENTED, ALONG WITH A DESCRIPTION OF THE FACILITY AND
DESIGN DATA.

CONTAINMENT, PRESSURE VENTING + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR DESCRIPTION

17-003644 DATED 1964
STARR C
SODIUM COOLED THERMAL REACTORS
ATOMICS INTERNATIONAL
15 PAGES- UNITED NATIONS THIRD INTERNATIONAL CONFERENCE ON THE PEACEFUL USES OF ATOMIC ENERGY, GENEVA, AUGUST
31-SEPTEMBER 9, 1964. A/CONF. 28/P/206

THE DESIGN, DEVELOPMENT, AND OPERATIONS OF THE SODIUM REACTOR EXPERIMENT, HALLAM POWER REACTOR, A
LARGE ADVANCED SODIUM GRAPHITE REACTOR (SGR), AND SGR S IN GENERAL ARE REVIEWED.

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-003644 *CONTINUED*

CONTAINMENT, PRESSURE VENTING + FAILURE, FUEL ELEMENT + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + SRE (RE)

17-003645

DATED 1964

KOCH LJ + KERN FS (ANL) + WENSCH GW (USAEC) + BRANYAN CE + ALEXANDERSON EL (APDA)

SODIUM COOLED FAST BREEDER REACTORS

ARGONNE NATIONAL LABORATORY + U.S. ATOMIC ENERGY COMMISSION + ATOMIC POWER DEVELOPMENT ASSOCIATES
15 PAGES- UNITED NATIONS THIRD INTERNATIONAL CONFERENCE ON THE PEACEFUL USES OF ATOMIC ENERGY, GENEVA, AUGUST 31-SEPTEMBER 9, 1964. A/CONF. 28/P/207

A SUMMARY OF THE PROGRAM FOR DEVELOPMENT OF FAST REACTORS THAT WILL BE COMPATIBLE WITH ELECTRIC GENERATING SYSTEM REQUIREMENTS OF THE 1980 S IS PRESENTED, WITH SPECIAL EMPHASIS ON THE EXPERIMENTAL BREEDER REACTOR II (EBR-II), THE ENRICO FERMI ATOMIC POWER PLANT, AND MAJOR ENGINEERING CONSIDERATIONS RELATING TO LARGER, SOLID FUEL, HETEROGENEOUS, SODIUM-COOLED FAST POWER REACTORS. SODIUM FILLING AND PURIFICATION, FUEL HANDLING, PRIMARY SODIUM PUMP REMOVAL, LOW-POWER OPERATION, AND PREPARATION FOR POWER OPERATION ARE DESCRIBED FOR THE EBR-II. SODIUM SYSTEM OPERATION, COMPONENT REPAIR AND MODIFICATION, FUEL HANDLING, LOW-POWER OPERATION, AND PREPARATION FOR POWER OPERATION ARE DISCUSSED FOR THE FERMI REACTOR. DESIGN STUDIES FOR 1000-MW(E) FAST POWER REACTORS ARE DESCRIBED.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + EBR I AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + FUEL HANDLING + REACTOR, LMCR + REPORT, OPERATIONS

17-003970

DATED 1964

SMITH RJ + RALEIGH HD

U. S. POWER REACTORS. A BIBLIOGRAPHY OF SELECTED LITERATURE

DIVISION OF TECHNICAL INFORMATION, U. S. ATOMIC ENERGY COMMISSION
227 PAGES- MARCH 1964, TID-3313

AN ANNOTATED BIBLIOGRAPHY OF TOPICAL AND PROGRESS REPORTS AND OPEN LITERATURE IS PRESENTED ON THESE REACTORS- BIG ROCK POINT- BONUS- CAROLINAS-VIRGINIA- DRESDEN- ELK RIVER- EGCR- FERMI- FLORIDA WEST COAST- HALLAM- HUMBOLDT BAY- INDIAN POINT- PATHFINDER- PEACH BOTTOM- PIQUA- SAXTON- AND YANKEE.

BIBLIOGRAPHY + BIG ROCK POINT (BWR) + REACTOR, BWR + BONUS (ISR) + REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + CONTAINMENT, MULTIPLE + CONTAINMENT, PRESSURE SUPPRESSION + CONTAINMENT, PRESSURE VENTING + CVTR (PWR) + DRESDEN I (BWR) + EGCR (AGR) + ELK RIVER (BWR) + REACTOR, FAST + FERMI (LMFBR) + REACTOR, GCR + HALLAM (LMR) + REACTOR, HWR + HUMBOLDT BAY (BWR) + INDIAN POINT I (PWR) + REACTOR, LMCR + REACTOR, ORGANIC COOLED + PATHFINDER (ISR) + PEACH BOTTOM I (HTGR) + PIQUA (OCR) + REACTOR, PRESSURE TUBE + REACTOR, PWR + SAXTON (PWR) + REACTOR, INTERNAL SUPERHEAT + YANKEE ROWE (PWR)

17-004028

DATED 1964

PRICE JE

FINAL EVALUATION REPORT, HALLAM NUCLEAR POWER FACILITY, VOLUMES I-VI

ATOMIC INTERNATIONAL
SEPTEMBER 29, 1964, NAA-SR-9777, VOLS. I-VI- OTS

THIS DOCUMENT IS INTENDED TO PROVIDE FOR THE POWER INDUSTRY PERTINENT TESTING, START-UP AND OPERATING DATA ON THE HALLAM NUCLEAR POWER FACILITY. PROBLEMS ENCOUNTERED AND CORRECTIVE ACTIONS OR MODIFICATIONS PERFORMED PRIOR TO CULMINATION OF THE FORMAL TESTING PROGRAM ARE INCLUDED IN THE TEST COMPLETION REPORTS COMPRISING THE GREATER PART OF THIS DOCUMENT. EACH REPORT IS DIVIDED INTO FIVE SECTIONS, DELINEATING PURPOSE, METHOD, RESULTS, COMPLIANCE WITH CRITERIA, AND AN INDEPENDENT EVALUATION OF SYSTEM ADEQUACY.

CONTAINMENT, PRESSURE VENTING + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS

17-004301

DATED 1964

LEIPUNSKII AI + KAZACHKOVSKII OD + PINKHASIK MS + ARISTARKHOV NN + POIDO MS + KARPOV AV + LARIN YP + EFIMOV IA + KRASNOYAROV NV + TYMOSH BS + NIKULIN MP + ZHESTKOV DI + URALETS YA + GALKOV VI

5-YEAR OPERATION EXPERIENCE ON REACTOR BR-5

15 PAGES, 8 FIGURES, 3 TABLES, 2 REFERENCES- MAY 1964, A/CONF. 28/P/312- PRESENTED AT THIRD U. N. INTERNATIONAL CONFERENCE ON THE PEACEFUL USES OF ATOMIC ENERGY, GENEVA, AUGUST 31-SEPTEMBER 9, 1964- IAEA

PRINCIPAL RESULTS OF THE OPERATION OF THE SODIUM-COOLED FAST REACTOR BR-5 OF NOMINAL POWER 5000 KW ARE PRESENTED. REACTOR OPERATION AFTER THE DESIGN BURNUP OF 21 WITH SOME LOOSE FUEL ELEMENTS IS DESCRIBED.

REACTOR, FAST + FISSION PRODUCT RELEASE + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR KINETICS

17-005262

ALSO IN CATEGORIES 10 AND 12

DATED 1965

LIQUID-METAL-COOLED REACTORS - EBR-II - OPERATIONS

ARGONNE NATIONAL LABORATORY

2 PAGES- ANL-7017 (REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT), PAGES 1 AND 2 OF CHAPTER I- MARCH 18, 1965- CFSTI

DURING A SITE-WIDE POWER FAILURE, THE 400-KW DIESEL GENERATOR STARTED, LOADED NORMALLY, AND THEN

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-005262 *CONTINUED*

STOPPED. THIS FAILURE OF THE DIESEL WAS ACCOMPANIED BY A LOSS OF ANNUNCIATION IN THE CONTROL ROOM. INVESTIGATION ALSO REVEALED THAT THE DIESEL CANNOT BE RESTARTED AFTER A FAILURE OF THIS KIND UNTIL THE DIESEL-CONTROL SWITCH IN THE CONTROL ROOM HAS BEEN PLACED IN OFF MOMENTARILY AND BACK TO STANDBY. LACK OF THIS INFORMATION PREVENTED MANUAL RESTARTING OF THE DIESEL DURING THE POWER OUTAGE, CAUSES AND REMEDIAL ACTION.

CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + EMERGENCY POWER, ELECTRIC + FAILURE + REACTOR, LMCB

17-005263 ALSO IN CATEGORIES 9 AND 12 DATED 1965

LIQUID-METAL-COOLED REACTORS - EBR-II - EXAMINATION OF FAILED OSCILLATOR ROD AND THIMBLE

ARGONNE NATIONAL LABORATORY

2 PAGES- ANL-7017 (REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT), PAGES 2 AND 3 OF CHAPTER I- MARCH 18, 1965- CFSTI

THE FAILED OSCILLATOR ROD AND THIMBLE WAS EXAMINED, AND 27 BALLS FROM A BALL BEARING WERE FOUND. THE OSCILLATOR ROD WAS JAMMED BY 19 BALLS IN THE LOWER PORTION OF THE GUIDE SLEEVE. VERY HEAVY SCORING WAS OBSERVED IN TWO OF THE THREE LOWER GUIDES ON THE OSCILLATOR ROD AND IN THE CORRESPONDING GUIDE SLOTS OF THE GUIDE TUBE.

EBR 1 AND 2 (RE) + FAILURE + REACTOR, LMCB + OSCILLATOR, REACTIVITY + FAILURE, SCRAM MECHANISM

17-005264

DATED 1965

LIQUID-METAL-COOLED REACTORS - EBR-II

ARGONNE NATIONAL LABORATORY

10 PAGES- ANL-7017 (REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT), PAGES 1-10 OF CHAPTER I- MARCH 18, 1965- CFSTI

DURING THE MONTH OF FEBRUARY, EBR-II WAS RELOADED WITH ADDITIONAL FUEL IN PREPARATION FOR RESUMPTION OF POWER OPERATION. ALSO, SIX SUBASSEMBLIES THAT HAD BEEN IN THE REACTOR FOR A MAXIMUM OF ABOUT 0.25 A/D BURNUP, WERE REMOVED FOR DISASSEMBLY AND INSPECTION IN THE FUEL CYCLE FACILITY.

CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + FAILURE + REACTOR, LMCB + HEAT EXCHANGERS

17-005375

DATED 1964

SELECTED POWER REACTOR PROJECTS IN CANADA AND THE UNITED STATES OF AMERICA

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA

73 PAGES- NOVEMBER 1964, STI/DOC/10/36 (IAEA TECHNICAL REPORTS SERIES NO. 36)- IAEA

AS PART OF ITS ACTIVITIES IN CONNECTION WITH THE DEVELOPMENT OF NUCLEAR POWER, THE INTERNATIONAL ATOMIC ENERGY AGENCY HAS UNDERTAKEN A CONTINUING STUDY OF THE TECHNOLOGY AND ECONOMICS OF POWER REACTORS, WITH PARTICULAR REFERENCE TO THE NEEDS OF THE DEVELOPING COUNTRIES. INFORMATION ON THE PROGRESS MADE IN EIGHT POWER REACTOR PROJECTS, NAMELY THOSE OF BONUS, PATHFINDER, ELK RIVER, PIQUA, HALLAM, EXPERIMENTAL GAS-COOLED REACTOR (EGCR), HIGH-TEMPERATURE GAS-COOLED REACTOR (HTGCR) AND NUCLEAR POWER DEMONSTRATION (NPD), IS PRESENTED. DEVELOPMENTS DURING FY 63-64 ARE SHOWN, EMPHASIS BEING PLACED ON OPERATING EXPERIENCE IN THE CASE OF THOSE REACTORS WHICH HAVE BECOME CRITICAL.

BIG ROCK POINT (BWR) + REACTOR, BWR + CANADA + CONTAINMENT + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, LOW PRESSURE + ECONOMICS + ELK RIVER (BWR) + FUEL HANDLING + REACTOR, GCR + HALLAM (LMR) + HYDRAULIC EFFECT + IAEA + INSTRUMENT, NUCLEAR + REACTOR, LMCB + REPORT, OPERATIONS + REACTOR, ORGANIC COOLED + PATHFINDER (ISR) + PEACH BOTTOM 1 (HTGR) + PIQUA (OCR) + PRESSURE VESSELS + REACTOR, PRESSURE TUBE + REACTIVITY EFFECT + TRAINING + TRITIUM + URANIUM CARBIDE + VENTILATION SYSTEM + OPERATING EXPERIENCE SUMMARY + CRITICALITY EXPERIMENT + REACTOR STARTUP TESTING + CONTROL ROD DRIVE MECHANISMS

17-005807

ALSO IN CATEGORIES 6 AND 12

DATED 1965

EBR-II

ARGONNE NATIONAL LABORATORY

6 PAGES, 2 TABLES- MARCH 1965, ANL-7028, SECTION I, PP. 1-6 OF ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT- CFSTI

EBR-II REINVESTIGATION OF THE FAILURE OF THE DIESEL GENERATOR. CONTROL ROD CALIBRATION, SMALL LEAKS IN SEAL OF PERSONNEL AIR LOCK, OTHER MINOR FAILURES.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + CONTROL ROD WORTH + EBR 1 AND 2 (RE) + FAILURE + REACTOR, FAST + REACTOR, LMCB + CONTAINMENT AIR LOCK + GENERATOR, DIESEL

17-006310

DATED 1960

DURAND RE + JENKINS OG + JOHNSON LL + PEARSON EN + VORDERBRUEGGEN LE

SRE STANDARD OPERATING PROCEDURES

ATOMICS INTERNATIONAL

200 PAGES, FIGURES AND TABLES- JUNE 27, 1960, NAA-SR-MEMO-5326- CFSTI

PART I DESCRIBES GENERAL OPERATING INSTRUCTIONS, OPERATING LIMITS, AND USE OF THE MANUAL. PART II GIVES OPERATING INSTRUCTIONS FOR THE REACTOR, SODIUM HEAT TRANSFER AND SERVICE SYSTEMS, COOLING SYSTEM, HELIUM SYSTEM, NITROGEN AND DEHUMIDIFICATION SYSTEM, VENT AND BUILDING VENTILATION

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-006310 *CONTINUED*
SYSTEMS, LIQUID WASTE SYSTEM, EMERGENCY POWER SYSTEM, WASH CELLS, AND THE FUEL HANDLING MACHINERY.

ADMINISTRATIVE CONTROL + REACTOR, LMCR + SRE (RE)

17-006312 ALSO IN CATEGORIES 6 AND 8 DATED 1965
ADAMS RM + GLASSNER A
SODIUM AIR REACTION + FARET CRITICALS
ARGONNE NATIONAL LABORATORY
5 PAGES, 3 FIGURES, 1 TABLE, 2 REFERENCES- PAGES 11-15 FROM REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 19, 1965, ANL-7045- CFSTI

REGARDING EBR-II, CONTROL ROD CALIBRATION, DELAYED NEUTRON KINETICS, POWER COEFFICIENT, REACTIVITY GAIN BY FUEL REPLACEMENT. (FEW NUMERICAL VALUES GIVEN.) REGARDING FARET, FAIRLY DETAILED DISCUSSION OF TEMPERATURES AND PRESSURES CAUSED BY SODIUM FIRES. DATA ON REACTIVITY COEFFICIENTS, REACTIVITY WORTH AND PROMPT-NEUTRON LIFETIME.

REACTOR, BREEDER + CHEMICAL REACTION + CONTAINMENT, HIGH PRESSURE + CONTROL ROD WORTH + DELAYED NEUTRON + EBR 1 AND 2 (RE) + FIRE + REACTOR, LMCR + POWER COEFFICIENT + REACTIVITY COEFFICIENT + SODIUM + REACTOR, TEST + AIR + FFTF (TR) + PROMPT NEUTRON LIFETIME + CONTROL ROD CALIBRATION

17-006317 DATED 1964
PROCEEDINGS OF THE SODIUM COMPONENTS INFORMATION MEETING HELD IN PALO ALTO, CALIFORNIA, AUGUST 20-21, 1963
U.S. ATOMIC ENERGY COMMISSION, SAN FRANCISCO OPERATIONS OFFICE + GENERAL ELECTRIC COMPANY, SAN JOSE
306 PAGES, 14 FIGURES, 4 TABLES, 3 REFERENCES- MAY 1964, SAN-8002, (CONF-258)- PROCEEDINGS OF THE SODIUM COMPONENTS INFORMATION MEETING, PALO ALTO, CALIFORNIA, AUGUST 20-21, 1963- CFSTI

THE PURPOSE OF THIS MEETING WAS TO PRESENT THE STATUS OF CURRENT SODIUM COMPONENT DEVELOPMENT PROJECTS AND ENCOURAGE AN EXCHANGE OF OPERATING EXPERIENCE IN EXISTING SODIUM SYSTEMS. IN THIS LATTER REGARD, PARTICIPANTS WERE ENCOURAGED TO DISCUSS FREELY ANY OPERATING DIFFICULTIES AND TROUBLES ENCOUNTERED. IN ADDITION, PAPERS WERE GIVEN ON OTHER PROJECTS RELATING TO SODIUM TECHNOLOGY BUT WHICH ARE NOT NECESSARILY DIRECTED TOWARD A PARTICULAR COMPONENT OR OPERATING SYSTEM. IN ADDITION TO THE COMPONENT DEVELOPMENT WORK, SIGNIFICANT REACTOR EXPERIENCE WAS REPORTED BY- FERMI - FAILURE OF THE STEAM GENERATORS WITH A SODIUM-WATER REACTION- DAMAGE AND REPAIR OF THE CORE HOLDDOWN MECHANISM. SRE - POSITIVE POWER COEFFICIENT (SOLVED BY WIRE-RESTRAINTS ON FUEL PIN BOWING) AND TEMPERATURE OSCILLATION.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + CONTAINMENT, PRESSURE VENTING + EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + FUEL ELEMENT BOWING + FUEL HANDLING + GRAPHITE + HALLAM (LMR) + FAILURE, OPERATOR ERROR + REACTOR, LMCR + MAINTENANCE AND REPAIR + METAL WATER REACTION + SODIUM + OPERATING EXPERIENCE SUMMARY + SRE (RE) + MODIFICATION + HEAT EXCHANGERS

17-006357 ALSO IN CATEGORIES 6 AND 9 DATED 1964
DARLEY DK + JENKINS OG + LARSON CL + LEW DE + MILLS RM + PEARSON EN + RUBINSTEIN HJ + STIENS RR + TESZLER O + WALTERS RJ + WEAVER LM
NUCLEAR ANALYSIS
ATOMIC INTERNATIONAL
45 PAGES, 25 FIGURES, 7 TABLES, 3 REFERENCES- PAGES 9-53, SECTION III, FROM HALLAM NUCLEAR POWER FACILITY REACTOR OPERATIONS ANALYSIS PROGRAM SEMIANNUAL PROGRESS REPORT NO. 3, SEPTEMBER 1, 1963 - FEBRUARY 29, 1964- SEPTEMBER 15, 1964, NAA-SR-9799, CFSTI

REPLACEMENT OF ZIRCALOY-II CONTROL ROD THIMBLES WITH STAINLESS STEEL THIMBLES AND ADDITION OF 10 (UC) FUEL ELEMENTS. MODERATOR ELEMENT FAILURES LED TO SODIUM FLOODING OF THE GRAPHITE MODERATOR. EFFECT OF FUEL BURNUP IS NOT AS GREAT AS CURRENTLY USED. FUEL ELEMENT VARIABLE ORIFICES BECOMING STUCK. 20 REACTOR SCRAMS HAVE BEEN ANALYZED WITH RESPECT TO CAUSE, CONSEQUENCES, AND MEANS OF REDUCING SCRAM FREQUENCY.

FAILURE, COMPONENT + CONTAINMENT, PRESSURE VENTING + CONTROL SYSTEM + HALLAM (LMR) + REACTOR, LMCR + SCRAM, REAL + URANIUM CARBIDE + REPORT, OPERATIONS ANALYSIS

17-006457 DATED 1964
ADAMS RM + GLASSNER A
REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JUNE 1964
ARGONNE NATIONAL LABORATORY
102 PAGES, 35 FIGURES, 26 TABLES, 13 REFERENCES, JULY 21, 1964, ANL-6912- CFSTI

BORAX V AT 20 MWTH, MAXIMUM PLANT RADIATION LEVEL IS 155 MR/HR AT AIR EJECTOR CHARCOAL FILTER. CYCLIC NEUTRON FLUX INDICATION OBSERVED DURING HEATUPS. AN ACTUAL MOISTURE CARRYOVER DUE TO FEEDWATER CONTROLLER MALFUNCTION OCCURRED. REACTOR TRANSFER FUNCTION MEASUREMENTS MADE STEAM QUALITY MEASURED USING NA24 CARRYOVER. EBWR FUEL ELEMENT EXAMINED- SCALE WAS TENACIOUS, NO CRACKS OR WARPAGE NOTED EXTERNALLY. EBR II DELIBERATE FAILURE OF 13.8 KV ELECTRICAL SUPPLY AND INSTRUMENT AIR SYSTEMS.

REACTOR, BWR + BORAX, ALL (S-RR) + REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + EBWR (BWR) + REACTOR, FAST + FLUX DISTRIBUTION + FUEL INTEGRITY + REACTOR, LMCR + SCRAM, REAL + REACTOR, INTERNAL SUPERHEAT + TRANSFER FUNCTION + ELECTRIC POWER + POWER DISTRIBUTION + TEST, PLANT RESPONSE + COOLANT CHEMISTRY + INCIDENT, EQUIPMENT + INSTRUMENT, LIQUID LEVEL + IRRADIATION TESTING + FUEL ELEMENTS

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-006458
ADAMS RM + WEBER M
REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 1964
ARGONNE NATIONAL LABORATORY
113 PAGES, 32 FIGURES, 62 TABLES, 12 REFERENCES, JUNE 15, 1964, ANL-6904- CFSTI

DATED 1964

BORAX V TAKEN TO POWER AND CONVENTIONAL REACTOR MEASUREMENTS MADE. VESSEL OUTLET TEMP A FUNCTION OF WATER LEVEL BECAUSE OF HEAT TRANSFER. EFFECTS OF ROD POSITION ON POWER SPLIT AND NEUTRON INDICATORS. FAST APPROACH TO FULL POWER MADE BY VENTING STEAM TO ATMOSPHERE. RESISTENTS AND PH INVERSE TO POWER LEVEL- PARTIALLY DUE TO PURIFIED CONDENSATE USED AS FEEDWATER. REFERENCE MADE TO A DROPPED FUEL ELEMENT- TWO CRACKS.

REACTOR, BWR + BORAX, ALL (S-RR) + REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + REACTOR, FAST + FLUX DISTRIBUTION + FUEL HANDLING + REACTOR, LMCR + MAINTENANCE AND REPAIR + FAILURE, PIPE + REACTIVITY EFFECT + REACTOR, INTERNAL SUPERHEAT + TEMPERATURE COEFFICIENT + OPERATING EXPERIENCE SUMMARY + TEST, PLANT RESPONSE + COOLANT CHEMISTRY + INCIDENT, EQUIPMENT + REACTOR STARTUP EXPERIENCE

17-006488
MOWREY J
LEAK CHECK OF MODERATOR ELEMENTS IN THE HNPF REACTOR
ATOMICS INTERNATIONAL
10 PAGES, 1 REFERENCE- DECEMBER 19, 1962, NAA-SR-MEMO-7837- CFSTI

ALSO IN CATEGORY 12

DATED 1962

AN INDUCTION PROBE WAS USED TO TRAVERSE REACTOR PROCESS CHANNELS THUS, LEAK CHECKING THE THREE MODERATOR ELEMENTS SURROUNDING PROCESS CHANNEL. FIFTY-EIGHT PROCESS CHANNELS WERE TRAVERSED TO CHECK THE MAXIMUM NUMBER OF MODERATOR ELEMENTS (129). TWELVE ELEMENTS COULD NOT BE CHECKED SINCE THE PROCESS CHANNELS AROUND THESE ELEMENTS ARE NOT ACCESSIBLE THROUGH THE LOADING FACE SHIELD. THE INDUCTION COIL CONSISTED OF A PRIMARY COIL AND TWO SECONDARY COILS, WITH AN OSCILLATOR CONNECTED TO THE PRIMARY COIL. THE MAGNITUDE OF THE INDUCED VOLTAGE IN THE SECONDARY COILS WAS AN INDICATION OF THE CONDUCTIVITY OF THE SURROUNDINGS. THE SECONDARY COILS WERE CONNECTED TO A SANBORN RECORDER AND THE SIGNAL RECORDED AS THE PROBE TRAVERSED EACH CHANNEL. EXAMINATION OF THE RECORDED TRACES INDICATE THAT THERE WERE NO LEAKY MODERATOR ELEMENTS IN THE REACTOR.

FAILURE, COMPONENT + CONTAINMENT, PRESSURE VENTING + CLADDING + GRAPHITE + HALLAM (LMR) + REACTOR, LMCR + SODIUM + TEST, COMPONENT + TEST, PREOPERATIONAL

17-006555
EBR-II
ARGONNE NATIONAL LABORATORY
7 PAGES- APRIL 1965, ANL-7045- PAGES 1-7 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT- CFSTI

DATED 1965

THIS SECTION REPORTS TWO EBR II CONTROL RODS COULD NOT BE DISCONNECTED FROM DRIVES DURING A SHUTDOWN. OVERTRAVEL OF DRIVE CAM HAD BROKEN TWO HOLDING SCREWS. NO MENTION OF ANY IMPAIRED ABILITY TO SCRAM- APPARENTLY THIS IS A CAM OPERATED ONLY DURING DRIVE ROTATION TO UNLOCK ROD FROM DRIVE FOR ROD REPLACEMENT.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + FAILURE, EQUIPMENT + REACTOR, FAST + REACTOR, LMCR + MAINTENANCE AND REPAIR + OPERATING EXPERIENCE SUMMARY + CONTROL ROD DRIVE MECHANISMS

17-006561
EBR-II
ARGONNE NATIONAL LABORATORY
6 PAGES, 1 FIGURE, 2 TABLES- JUNE 25, 1965, ANL-7046- PP. 1-6, SECTION A, OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 1965- CFSTI

ALSO IN CATEGORIES 6 AND 9

DATED 1965

MEASUREMENT OF POWER COEFFICIENT- FISSION PRODUCT MONITOR- LEAK RATE TESTS, CONTROL ROD MALFUNCTION, REACTIVITY MONITORING DEVICES, FEEDBACK OF AXIAL FUEL EXPANSION.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + REACTIVITY EFFECT, EXPANSION + REACTOR, FAST + FISSION PRODUCT RELEASE + REACTOR, LMCR

17-006673
NUCLEAR SAFETY
ARGONNE NATIONAL LABORATORY
6 PAGES, 1 TABLE, 2 FIGURES- JULY 30, 1965, ANL-7071- PAGES 60-66 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JUNE 1965, CFSTI

ALSO IN CATEGORY 6

DATED 1965

COOLANT EXPULSION BY VAPORIZED COOLANT STUDIED WITH THEORETICALLY RESPECT TO SCALING LAWS. DISCUSSION OF PRESSURE TRANSDUCERS FOR EXPERIMENT ON THIS EXPULSION. TEST OF SAMPLES STUDIED IN TREAT. ADMINISTRATIVE REMARKS ON LARGE TREAT LOOP. FLOW MEASUREMENT IN TREAT INTEGRAL SODIUM LOOPS. REACTIVITY CALCULATIONS OF TREAT WITH EBR-II SUBASSEMBLY. APPARATUS FOR HEAT TRANSFER FROM HOT SPHERE MOVING THROUGH WATER AND PRELIMINARY RESULTS.

REACTOR, BREEDER + COMPUTER PROGRAM + CONTAINMENT, HIGH PRESSURE + EBR 1 AND 2 (RE) + REACTIVITY, EXCESS + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + REACTOR, TEST + THERMAL ANALYSIS + TREAT (PRR) + INSTRUMENT, FLOW

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-006726
DUNBAR FH

DATED 1964

A SUMMARY OF SRE SODIUM SYSTEMS EXPOSURE HISTORY
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA
23 PAGES- OCTOBER 22, 1964, NAA-SR-MEMO-10593- CFSTI

SINCE THE SRE WENT CRITICAL IN MARCH OF 1957, 64,250 CALENDAR HOURS HAVE ELAPSED. DURING THIS INTERVAL, THE REACTOR WAS AT A POWER NOTGT 15 MW(T) FOR 5,256 HOURS, NOTGT 10 MW(T) FOR 10,144 HOURS, AND NOTGT 5 MW(T) FOR 14,992 HOURS, CRITICAL (0.01% OR MORE OF FULL POWER) FOR 27,300 HOURS, AND DOWN FOR 36,950 HOURS. IN JULY OF 1959 SEVERAL FUEL ELEMENTS AND MODERATOR CANS WERE DAMAGED BECAUSE OF HIGH FUEL CHANNEL TEMPERATURES CAUSED BY FOULING. THE RECOVERY TOOK ABOUT 12 MONTHS AND INCLUDED THE REPLACEMENT OF 16 MODERATOR CANS AND THE LOADING OF THE SECOND SRE CORE (CORE II). THE INITIAL CRITICALITY OF CORE II WAS SEPTEMBER 9, 1960. SINCE APRIL OF 1957, THERE HAVE BEEN ABOUT 44,000 CUMULATIVE HOURS OF SODIUM FLOW IN THE PRIMARY SYSTEMS (CORE, MAIN PRIMARY, AND AUXILIARY PRIMARY) WITH AN APPROXIMATE RANGE OF 285 DEGREE F TO 1030 DEGREE F. SINCE MAY OF 1958, THERE HAVE BEEN ABOUT 25,000 CUMULATIVE HOURS OF SODIUM FLOW IN THE MAIN SECONDARY SYSTEM, WITH AN APPROXIMATE RANGE OF 300 DEGREE F TO 1000 DEGREE F. A GRAPHICAL PRESENTATION OF THE SRE OPERATING HISTORY IS GIVEN.

REACTOR, LMCR + REPORT, OPERATIONS + OPERATING EXPERIENCE SUMMARY + SRE (RE)

17-006914
MILLS RM

DATED 1964

DETECTION AND IDENTIFICATION OF FAILED MODERATOR ELEMENTS IN THE HNPFC CORE FROM OPERATING DATA
ATOMICS INTERNATIONAL
50 PAGES- OCTOBER 22, 1964, NAA-SR MEMO 10596- CFSTI

ANALYSIS OF HNPFC OPERATING DATA ESTABLISHED THAT A NUMBER OF MAJOR CORE ANOMALIES OCCURRING OVER THE EIGHT MONTHS FROM FEBRUARY THROUGH SEPTEMBER, 1964 HAVE BEEN CAUSED BY THE FAILURE AND SODIUM FLOODING OF NINE MODERATOR ELEMENTS RESULTING IN REACTIVITY LOSSES TOTALING ABOUT 151 CENTS. THREE PRINCIPAL MEANS OF DETECTION AND IDENTIFICATION OF THE FAILED MODERATOR ELEMENTS WERE USED- (1) A CONTINUOUS REACTIVITY HISTORY CALCULATION WHICH SHOWED A REACTIVITY LOSS WHENEVER A MODERATOR ELEMENT FAILED- (2) COMPARISONS OF RELATIVE POWER DISTRIBUTIONS ACROSS THE CORE WHICH SHOWED WIDESPREAD POWER DEPRESSIONS CENTERING AROUND THE FAILED MODERATOR ELEMENTS- AND (3) AN UNBALANCE, TILT, IN NUCLEAR POWER CHANNEL READING AROUND THE OUTSIDE EDGE OF THE CORE. THE FIRST TWO CALCULATIONS ARE PERFORMED ON A ROUTINE BASIS BY THE OPERATIONS ANALYSIS GROUP.

FAILURE, EQUIPMENT + FLUX DISTRIBUTION + GRAPHITE + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + STRESS + POWER DISTRIBUTION + OPERATING EXPERIENCE SUMMARY + INCIDENT + MODERATOR + REPORT, OPERATIONS ANALYSIS + FAILURE, CLADDING + REACTIVITY EFFECT, ANOMALOUS + INSTRUMENT, ABNORMAL INDICATION

17-006973

DATED 1965

BERGER S + DARLEY DK + DEBEAR WS + GLASS DN + JENKINS OG + LARSON CL + LEW DE + MILLS RM + PEARSON EN + PELHAM DD + PERRET JD + RUBINSTEIN HJ + TESZLER O + WALTERS RJ + WATROUS JD + WEAVER LM
HALLAM NUCLEAR POWER FACILITY REACTOR OPERATIONS ANALYSIS PROGRAM SEMIANNUAL PROGRESS REPORT NO. 4 FEBRUARY 29, 1964 - SEPTEMBER 30, 1964
ATOMICS INTERNATIONAL
151 PAGES, 81 FIGURES, 25 TABLES- MAY 15, 1965, NAA-SR-10743- CFSTI

SOME 30 PAGES DEVOTED TO MODERATOR ELEMENT FAILURE- EFFECTS ON REACTIVITY AND NUCLEAR INSTRUMENTATION, ETC. REACTIVITY VS TIME CURVES SHOW CHANGING RATE-OF-LOSS. SEVENTY PAGES DEVOTED TO HEAT TRANSFER SYSTEMS OPERATION.

FAILURE, COMPONENT + FLUX DISTRIBUTION + FUEL BURNUP + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + INSTRUMENT CALIBRATION + TEST, PLANT RESPONSE + COOLANT CHEMISTRY + MODERATOR + REPORT, OPERATIONS ANALYSIS + FLUX TILT + REACTIVITY EFFECT, ANOMALOUS + MAIN COOLING SYSTEM

17-006976

DATED 1965

NOVICK M + MCGINNIS FD + WHITHAM GK
EBR-I AND EBR-II OPERATING EXPERIENCE
ARGONNE NATIONAL LABORATORY, IDAHO
16 PAGES, 14 FIGURES, 5 TABLES, 13 REFERENCES- APRIL 28, 1965, ANS-100 (CONF-650417), PP. 25-40 OF ANS FAST REACTOR TECHNOLOGY NATIONAL TOPICAL MEETING, DETROIT, MICHIGAN, APRIL 26-28, 1965

SHORT DESCRIPTION OF EBR I AND REVIEW OF PLUTONIUM CORE MARK IV. SIMILAR TREATMENT FOR EBR II, WITH MORE ATTENTION TO INITIAL FILLING WITH SODIUM, STARTUP PHYSICS TESTS AND POWER RUNS, FAILURES IN PUMPS AND STEAM GENERATOR.

REACTOR, BREEDER + EBR I AND 2 (RE) + FAILURE, EQUIPMENT + REACTOR, FAST + REACTOR, LMCR + PLUTONIUM + REACTIVITY COEFFICIENT + REACTIVITY EFFECT + TRANSFER FUNCTION + OPERATING EXPERIENCE SUMMARY + COOLANT CHEMISTRY + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + MEASUREMENT, REACTIVITY + REACTOR STARTUP EXPERIENCE

17-006977

DATED 1965

ALEXANDERSON EL + BRANYAN CE + OLSON WR
OPERATING EXPERIENCE AT THE ENRICO FERMI ATOMIC POWER PLANT
POWER REACTOR DEVELOPMENT COMPANY + ATOMIC POWER DEVELOPMENT ASSOCIATES + DETROIT EDISON COMPANY
16 PAGES, 7 FIGURES, 3 TABLES, 7 REFERENCES- APRIL 28, 1965, ANS-100 (CONF-650417), PP. 41-56 OF ANS FAST REACTOR TECHNOLOGY NATIONAL TOPICAL MEETING, DETROIT, MICHIGAN, APRIL 26-28, 1965

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-006977 *CONTINUED*

A GENERAL REVIEW OF LOW POWER NUCLEAR TESTING (CONTROL ROD CALIBRATION, TEMPERATURE COEFFICIENT, ETC.), SAFETY SYSTEM PERFORMANCE, INSTRUMENT AND FUEL PERFORMANCE. REACTIVITY INCREASES AFTER HIGH ISOTHERMAL SYSTEM TEMPERATURES ARE INTERPRETED AS FUEL ELEMENT PHASE TRANSFORMATIONS. A SINGLE MODIFICATION CORRECTED A CIRCUIT IN WHICH ANY ONE PAIR OF DIODES (OUT OF 112 PAIRS) PREVENTED A SCRAM. A CONTROL ROD FAILED TO SCRAM AS SODIUM LEAKED INTO THE SCRAM MECHANISM AND FROZE.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + OSCILLATOR, REACTIVITY + TEMPERATURE COEFFICIENT + POWER DISTRIBUTION + OPERATING EXPERIENCE SUMMARY + CONTROL ROD CALIBRATION + FAILURE, SCRAM MECHANISM + FUEL HANDLING MACHINE + REACTIVITY EFFECT, ANOMALOUS + INSTRUMENTS, MISC.

17-006979

DATED 1965

KOCH LJ

FAST REACTOR FUEL HANDLING

ARGONNE NATIONAL LABORATORY

15 PAGES, 14 FIGURES, 1 TABLE- APRIL 28, 1965, ANS-100 (CONF-650417), PP. 333-347 OF ANS FAST REACTOR TECHNOLOGY NATIONAL TOPICAL MEETING, DETROIT, MICHIGAN, APRIL 26-28, 1965

COMPARES FUEL HANDLING MACHINERY OF EBR II, FERMI, AND FAST COMPACT REACTOR STUDY. BRIEF DISCUSSION OF OPERATING EXPERIENCE, TIME TO MOVE A FUEL ELEMENT, ETC.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + FAILURE, EQUIPMENT + REACTOR, FAST + FERMI (LMFBR) + FUEL HANDLING + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + FUEL HANDLING MACHINE

17-007108

ALSO IN CATEGORY 6

DATED 1965

PLUTONIUM RECYCLE REACTORS

ARGONNE NATIONAL LABORATORY

4 PAGES, 1 REFERENCE- JUNE 25, 1965, ANL-7046- PP. 63-66 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 1965- CFSTI

EBR II WAS OPERATED WITH ONE SINGLE UNCLAD FUEL PIN TO TEST FISSION PRODUCT MONITOR CALIBRATION. ANALYSIS OF ROD DROP MEASUREMENTS AT 45 MW SHOWS AT LEAST 16% OF THE REACTIVITY (APPROXIMATELY $.75 \times 10^{-4}$ DELTA K/K) IS RECOVERED IN LESS THAN 0.5 SEC, PROBABLY ASSOCIATED WITH AXIAL FUEL EXPANSION. METALLOGRAPHIC EXAMINATION OF CRACKED 304 SS CLADDING, WHICH HAD BEEN IN EBRW FOR 10 YEARS, REVEALED EACH GRAIN OF STEEL SURROUNDED BY BRITTLE CARBIDES, WHICH HAD PRECIPITATED FROM THE SOLID SOLUTION STAINLESS STEEL DURING THE 1150 DEGREE F STRESS RELIEF HEATING. FIVE SURFACE CRACKS WERE FOUND ON OTHER TESTS.

REACTOR, BWR + REACTOR, BREEDER + EBR 1 AND 2 (RE) + EBRW (BWR) + REACTOR, FAST + FUEL ELEMENT BOWING + REACTOR, LMCR + PRESSURE VESSELS + REACTIVITY EFFECT + ANL + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + MATERIAL

17-007167

DATED 1964

STICKING OF CONTROL ROD AND OSCILLATOR DRIVE MECHANISMS

ARGONNE NATIONAL LABORATORY

3 PAGES, 3 FIGURES- DECEMBER 21, 1964, ANL-6977, PAGES 28-30 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, NOVEMBER 1964- CFSTI

BALL BEARINGS WERE FOUND IN THE DRIVE SHAFTS AND SHAFTS WERE SCORED. MANY BALLS WERE NOT FOUND.

REACTOR, BREEDER + FAILURE, COMPONENT + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + CONTROL ROD SCRAM MECHANISM + INCIDENT, EQUIPMENT + INCIDENT, RECOVERY FROM

17-007435

ALSO IN CATEGORIES 6 AND 11

DATED 1965

EBR-II

ARGONNE NATIONAL LABORATORY

7 PAGES, 5 TABLES- AUGUST 24, 1965, ANL-7082, PP 1-7 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT JULY 1965, CFSTI, \$3.00 CY, \$0.75 MN

REGARDING EBR-II 10- SUBASSEMBLIES CONTAINING SPECIAL SURVEILLANCE FUEL PINS SWELLED FURTHER THAN NORMAL PINS AND WERE REPLACED. NORMAL PRODUCTION PINS ALSO SWELLED SOMEWHAT. OXIDES OF BISMUTH, TIN AND SODIUM FORMED IN THE SEAL TROUGH AND INTERFERED WITH ROTATION OF THE PLUGS AND THE TEMPERATURE DISTRIBUTION OF THE TROUGHS. PUMP M-1 FROZE TWICE, APPARENTLY BECAUSE OF SODIUM COMBUSTION IN SHAFT. CONCRETE OF PUMP SHIELD PLUGS GIVES OFF WATER. MINOR CONTROL ROD DRIVE TROUBLE. CONTAINMENT TESTING. CONTROL ROD WORTH AND POWER COEFFICIENT.

REACTOR, BREEDER + FAILURE, COMPONENT + CONTROL ROD WORTH + EBR 1 AND 2 (RE) + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR, LMCR + POWER COEFFICIENT + TEST, LEAK RATE + OPERATING EXPERIENCE SUMMARY + PUMPS + CONTROL ROD DRIVE MECHANISMS

17-007444

ALSO IN CATEGORIES 6 AND 8

DATED 1965

REACTOR KINETICS

ARGONNE NATIONAL LABORATORY

9 PAGES, 3 FIGURES- SEPTEMBER 21, 1965, ANL-7090, PP 69-75 OF REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, AUGUST 1965, CFSTI, \$3.00 CY

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-007444 *CONTINUED*

EBR-II- MARK I FUEL PIN FAILURE IN TREAT (FUEL VIGOROUSLY EXPELLED BY SODIUM VAPOR PRESSURE). INSTRUMENTATION FOR COOLANT EXPULSION STUDIES. DURING ACCIDENTS, CRITICAL FLOW MAY EXIST AT THE DISCHARGE END OF COOLANT CHEMICALS, RESULTING IN LARGE PRESSURE GRADIENTS. A PHYSICALLY MEANINGFUL WEIGHTING FUNCTION WAS DERIVED FOR SPACE-TIME REACTOR DYNAMICS. IRRADIATION OF METAL-WATER REACTION SAMPLES FOR TREAT. HFIR FUEL ELEMENT SAMPLES MELTED IN TREAT TRANSIENTS, RESULTING IN SOME METAL-WATER REACTIONS.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, GRAPHITE MODERATED + HYDRAULIC ANALYSIS + REACTOR, LMCR + SODIUM + SPACE DEPENDENT DYNAMICS + REACTOR, TEST + TREAT (PKR) + HFIR (FTR) + PRESSURE, INTERNAL

17-007670 ALSO IN CATEGORIES 6 AND 12 DATED 1965

LIQUID-METAL-COOLED REACTORS

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
87 PAGES, TABLES, FIGURES- ANL-7105, OCTOBER 20, 1965, PP. 1-35 OF ARGONNE NATIONAL LABORATORY REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, SEPTEMBER, 1965

EBR-II - FUEL IRRADIATED PAST 0.8 ATOM PERCENT BURNUP SHOWS AN ACCELERATING RATE OF VOLUME INCREASE WITH BURNUP. FUEL VOLUME INCREASES OVER 8 PERCENT WILL REQUIRE PLASTIC STRAIN OF CLADDING. FUEL WILL BE EXAMINED AFTER EACH OPERATING CYCLE. CP-5 URANIUM DIOXIDE FUEL WAS OPERATED BRIEFLY WITH A CENTERLINE TEMPERATURE OF 2400 DEGREES CENTIGRADE WITH CP-5 AT MAXIMUM (4.2 MW) POWER. CLADDING WAS SEPARATED FROM FUEL WITH TANTALUM RODS IN ARGON ATMOSPHERE. TUNGSTEN - RHENIUM THERMOCOUPLES MEASURED THE CENTERLINE TEMPERATURE.

CRITICAL ASSEMBLY + DOPPLER EFFECT + EBR 1 AND 2 (RE) + FUEL BURNUP + FUEL INTEGRITY + IN PILE EXPERIMENT + INSTRUMENT, IN CORE + REACTOR, LMCR + MEASUREMENT + FFTF (TR) + IN PILE EXPERIMENT + IRRADIATION TESTING + ZPPR (CAF)

17-007695 ALSO IN CATEGORIES 1 AND 6 AND 9 DATED 1965

OKRENT D

A LOOK AT FAST REACTOR SAFETY

ARGONNE NATIONAL LABORATORY
25 PAGES, 1 FIGURE, 2 TABLES, 131 REFERENCES- NUCLEAR SAFETY 6(4)- 317-342 (SUMMER, 1965)

DESIGN HISTORY (DOUNREAY REACTOR, EBR-II, ENRICO FERMI, PATTERNS AND TRENDS, SAFETY-RELATED DESIGN AND OPERATING EXPERIENCE). DIFFERENCES IN NUCLEAR SAFETY FOR LARGE-SODIUM-COOLED FAST REACTORS (DELAYED-NEUTRON FRACTION, DOPPLER COEFFICIENT, SODIUM COEFFICIENT AND VOID EFFECT, STABILITY, MAXIMUM ACCIDENT CALCULATIONS). LARGE-SODIUM-COOLED-FAST REACTOR DESIGNS AND ACCIDENT STUDIES, NON-NUCLEAR SAFETY CONSIDERATIONS, COOLANTS OTHER THAN SODIUM (STEAM, GAS, OTHER METAL COOLANTS), SAFETY PHILOSOPHY AND SAFETY RESEARCH, REFERENCES.

REACTOR, BREEDER + DELAYED NEUTRON + DOPPLER COEFFICIENT + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LARGE + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR STABILITY + SODIUM + SODIUM COEFFICIENT + STEAM + VOID COEFFICIENT + OPERATING EXPERIENCE SUMMARY + HELIUM + DOUNREAY (LMFBR) + SAFETY PRINCIPLES AND PHILOSOPHY

17-007708

DATED 1965

WHITHAM GK + LEMAN JD

RECENT PROBLEMS WITH EBR-II COMPONENTS

ARGONNE NATIONAL LABORATORY, IDAHO
2 PAGES, AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JACKSON LAKE LODGE, WYOMING, JULY 28-29, 1965, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 8, PAGES 11-12

EBR II WAS SHUT DOWN FROM OCTOBER 1964 TO MARCH 1965 BECAUSE OF TWO FAILURES - STICKING OF THREE CONTROL-ROD DRIVES AND A WATER-TO-AIR LEAK IN A STEAM GENERATOR. BALLS FROM A BALL BEARING HAD COME LOOSE.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + FAILURE, EQUIPMENT + REACTOR, FAST + REACTOR, LMCR + HEAT EXCHANGERS + CONTROL ROD DRIVE MECHANISMS

17-007709

DATED 1965

WOODRUFF RW

SRE CORE REPLACEMENT

ATOMICS INTERNATIONAL

1 PAGE, 1 REFERENCE, AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JACKSON LAKE LODGE, WYOMING, JULY 28-29, 1965, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 8, PAGE 12

SRE WILL BE REVISED AND UPDATED FOR USE AS A FUEL-TEST FACILITY. SINCE CORE REPLACEMENT NOW WOULD TAKE 40 DAYS, FUEL HANDLING IS BEING REVISED TO REDUCE FUTURE DOWNTIME.

FUEL HANDLING + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + REACTOR, TEST + SRE (RE)

17-007710

DATED 1965

NICHOLSON JO

DEMOLITION OF THE HEAT-TRANSFER SYSTEMS AT THE SODIUM REACTOR EXPERIMENT (IN PREPARATION FOR THE POWER EXPANSION PROGRAM)

ATOMICS INTERNATIONAL

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-007710 *CONTINUED*

2 PAGES, AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JACKSON LAKE LODGE, WYOMING, JULY 28-29, 1965, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 8, PAGES 12-13

BETA-GAMMA RATIO INCREASED FROM 8/1 TO 15/1 AFTER SODIUM REMOVED FROM PIPING. RADIOACTIVE COMPONENTS WERE GIVEN TREATMENT IN BUTYL-CELLOSOLVE, RATHER THAN BEING STEAM CLEANED FOLLOWING THE OIL BATH, THIS TO GIVE SLOW REACTION RATE. SEVEN WEEKS AND 16 MEN WERE USED- MAXIMUM EXPOSURE WAS 1.8 REMS.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + SRE (RE) + MODIFICATION + DECOMMISSIONING + POWER UPGRATING

17-007714

DATED 1965

NELSON CL + WHITHAM GK + BRYANT H + LEMAN D + GREENE JK
PLANT SYSTEMS TRAINING OF PERSONNEL FOR EBR-II
ARGONNE NATIONAL LABORATORY, IDAHO

15 PAGES, 3 REFERENCES, AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JACKSON LAKE LODGE, WYOMING, JULY 28-29, 1965, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 8, PAGE 16

TRAINING BEGAN 18 MONTHS BEFORE DRY CRITICAL, AND INCLUDED BOTH THEORY AND PRACTICE. LESSON PLANS WERE PREPARED TO COORDINATE TRAINING WHEN SHIFT CREWS WERE FORMED. AS STARTUP APPROACHED, TRAINING WAS DONE ON OVERTIME. THERE IS NOW A 40-MAN OPERATING CREW AND A 4-MAN TRAINING CREW, WHICH PREPARES LESSON PLANS. TWO SUPERVISORS SIGN EACH LESSON PLAN WHEN THEY BELIEVE THAT THE TRAINEE KNOWS THE MATERIAL. A FINAL ORAL AND WRITTEN EXAM IS ALSO USED. OPERATOR PERFORMANCE ON SHIFT CORRELATES WITH RESULTS OF THE EXAMINATION.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + TRAINING + REACTOR, ERDA OWNED

17-007720

ALSO IN CATEGORY 9

DATED 1965

MUELLER RE + BRANYAN CE + KLINKMAN AE
CONTROL-ROD CALIBRATIONS FOR THE ENRICO FERMI REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

2 PAGES, 1 TABLE- PAGES 21-22 OF TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JULY 28-29, 1965, JACKSON LAKE LODGE, WYOMING

LOW-POWER TESTING CALIBRATIONS WERE MADE TO CONFIRM DESIGN AND TO INVESTIGATE DIFFERENT CALIBRATION TECHNIQUES. UNSHADOWED AND SHADOWED INDIVIDUAL ROD WORTHS SHOWED MEASURED ROD WORTHS LESS THAN PREDICTED BASED ON CRITICAL-EXPERIMENT. AN INCONSISTENCY WAS DISCOVERED BETWEEN THE MEASURED INDIVIDUAL ROD WORTHS AND THE GANGED ROD WORTH. FUTURE POWER-CONTROL RODS WILL BE MEASURED BY SUBCRITICAL MULTIPLICATION.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + SHUTDOWN MARGIN + COMPARISON, THEORY AND EXPERIENCE + OPERATING EXPERIENCE SUMMARY + REACTOR STARTUP TESTING + CONTROL ROD CALIBRATION

17-007721

ALSO IN CATEGORY 9

DATED 1965

HORNE RE + KLINKMAN AE

A TECHNIQUE FOR DETERMINATION OF THE SAFETY-ROD INSERTION TIME FOR THE ENRICO FERMI REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

1 PAGE, 1 TABLE- PAGE 22 OF TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JULY 28-29, 1965, JACKSON LAKE LODGE, WYOMING

THE RELEASE OF THE SAFETY RODS PREVENTS USE OF CONVENTIONAL TIMING METHODS TO DETERMINE INSERTION TIME. A SIMPLE NUCLEAR METHOD OF MEASURING BOTH THE ROD WORTH AND THE INSERTION TIME HAS BEEN FOUND. A SANBORN TRACE OF THE REACTOR POWER AS A FUNCTION OF TIME, AFTER INITIATION OF A SCRAM IS ANALYZED TO OBTAIN THE DESIRED INSERTION TIME. IN ADDITION THE OUTPUT OF THE LATCH-OPEN MICROSWITCH AND THE LATCH MAGNET CURRENT WERE RECORDED ON THE SAME TRACE TO DETERMINE THE RODS TIME. THE TRACE OF THE ION-CHAMBER CURRENT WAS MANUALLY TRANSFORMED TO A DIGITAL FORM, AND THE SPACE-INDEPENDENT REACTOR KINETICS EQUATIONS WERE SOLVED BY A COMPUTER TO SHOW THE REACTIVITY AS A FUNCTION OF TIME AFTER SCRAM. THE INSERTION TIME DEFINED AS THE TIME BETWEEN RELEASE AND INSERTION OF 95% OF THE TOTAL ROD WORTH. THIS METHOD IS CONSIDERED TO BE ACCURATE TO 10 MSEC. MOST OTHER EXPERIMENTERS HAVE USED ROD-DROP EXPERIMENTS BY ANALYZING A LOW-SPEED RECORDING WITH THE PROMPT-JUMP APPROXIMATION. USING HIGH-SPEED RECORDING DOES NOT REQUIRE THIS APPROXIMATION.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + CONTROL ROD SCRAM MECHANISM + TEST, DROP

17-007728

ALSO IN CATEGORY 9

DATED 1965

ALEXANDERSON EL + HARRIGAN PM

ADDED SOURCE-RANGE DETECTORS FOR THE ENRICO FERMI ATOMIC POWER PLANT
POWER REACTOR DEVELOPMENT CO., DETROIT

2 PAGES, 1 FIGURE- PAGES 26-27 OF TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JULY 28-29, 1965, JACKSON LAKE LODGE, WYOMING

OPERATION AT LOWER POWER LEVELS AND PLANT FACTORS WOULD REQUIRE A STRONGER NEUTRON SOURCE, BUT ZIRCONIUM HYDRIDE WAS PLACED AROUND THE SOURCE RANGE DETECTORS. SOFTENING THE SPECTRUM INCREASED THE SHUTDOWN COUNT RATE ADEQUATELY (2/3 DECADE).

REACTOR, BREEDER + ECONOMICS + REACTOR, FAST + FERMI (LMFBR) + INSTRUMENT, NUCLEAR + REACTOR, LMCR + SOURCE, NEUTRON + OPERATING EXPERIENCE SUMMARY

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-007736

DATED 1965

KEATEN RW + MANNAS DA + OLSON WH

EVALUATION OF LONG-TERM OPERATION ON SRE SODIUM SYSTEMS

ATOMICS INTERNATIONAL, DIVISION OF NAA

2 PAGES- PAGES 32-33 OF TRANSACTION OF THE AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JULY 28-29, 1965, JACKSON LAKE LODGE, WYOMING

GENERAL CONDITION AFTER 9 YEARS IS GOOD, SHOWN BY EXAMINATION OF DISMANTLED PRIMARY AND SECONDARY SYSTEM. EROSION WAS MINIMAL, A HARD METALLIC LAYER (0.001 INCH) WAS FOUND IN SECONDARY COLD LEG. HEAT EXCHANGER OK, DESPITE MANY 200 DEGREE F SHOCKS. INOPERABLE CHECK VALVE WAS DUE TO DEPOSITS IN UNNECESSARILY SMALL CLEARANCES. SOME CARBON PLUGS IN HEAT EXCHANGER HAD BEEN DISSOLVED BY HIGH TEMPERATURE OPERATION LEAVING ONLY 10% OF TUBES PLUGGED.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + SODIUM + OPERATING EXPERIENCE SUMMARY + SRE (RE) + COOLANT CHEMISTRY + DECOMMISSIONING + MAIN COOLING SYSTEM + HEAT EXCHANGERS

17-007738

DATED 1965

DAVIDSON RH

MODERATOR ELEMENT FAILURES IN THE HALLAM NUCLEAR POWER FACILITY

CONSUMERS PUBLIC POWER DISTRICT, HALLAM, NEBRASKA

16 PAGES, 10 FIGURES, AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JACKSON LAKE LODGE, WYOMING, JULY 28-29, 1965, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 8, PAGE 35

SEVEN MODERATOR-CAN FAILURES WERE RAPIDLY DETECTED BY INSTRUMENTATION SHOWING FLUX TILT, LOCAL POWER DEPRESSION, AND REACTIVITY LOSS. FOR M51 FAILURE, A DELTA T REDUCTION OF 70 DEGREES F AND A LOSS OF 20 CENTS REACTIVITY RESULTED WITHIN 3 HOURS. SUBSEQUENT HOT-CELL EVALUATION SHOWED A CHARACTERISTIC STRESS FAILURE ON ALL ELEMENTS, ALLOWING SODIUM TO SOAK THE GRAPHITE, CAUSING NEUTRON POISON ACCUMULATION AND SWELLING.

FAILURE, COMPONENT + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + STRESS + MODERATOR + REACTIVITY EFFECT, ANOMALOUS

17-007743

DATED 1965

HANSEN AI + CARTER TJ

FISSION-PRODUCT AND CARBON RETENTION IN SRE COLD TRAPS

ATOMICS INTERNATIONAL

2 PAGES, AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JACKSON LAKE LODGE, WYOMING, JULY 28-29, 1965, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 8, PAGES 38-39

EIGHT COLD TRAPS, WHICH HAD BEEN USED TO REMOVE IMPURITIES, WERE EXAMINED FOR DISTRIBUTION AND KINDS OF IMPURITIES. SEVERAL RETENTION METHODS WERE FOUND, SUCH AS GETTERING ACTION OF CARBON AND SODIUM OXIDE.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + SRE (RE) + COOLANT CHEMISTRY

17-007746

DATED 1965

ALSO IN CATEGORY 11

WHITHAM GK + TEBB FJ + MCGINNIS FD

EBR-II REACTOR BUILDING LEAK TESTING

ARGONNE NATIONAL LABORATORY, IDAHO

2 PAGES, AMERICAN NUCLEAR SOCIETY CONFERENCE ON REACTOR OPERATING EXPERIENCE, JACKSON LAKE LODGE, WYOMING, JULY 28-29, 1965, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 8, PAGES 40-41

A 3-DAY LEAK TEST SHOWED LESS THAN 100 CU FT/DAY AT 36.2 PSIA, AFTER ALL OPERATIONAL DOORS INSTALLED. SINCE THEN, YEARLY TESTS HAVE BEEN ON MAJOR PENETRATIONS AND RANDOMLY CHOSEN ELECTRICAL PENETRATIONS. THEY PROPOSE TO ADOPT 8 OF 24 ELECTRICAL PENETRATIONS FOR TESTING, INSTEAD OF RANDOM TESTING, WHICH DISRUPTS OPERATIONS.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + TEST, LEAK RATE + CONTAINMENT LEAKAGE CONTROL

17-007848

DATED 1965

ALSO IN CATEGORY 6

SMITH RR + DOE CB + HAROLDSEN RO + MCGINNIS FD + NOVICK M

TERMINAL REPORT FOR THE MARK IV (PLUTONIUM) LOADING IN EBR-I

ARGONNE NATIONAL LABORATORY

94 PAGES, 45 FIGURES, 20 TABLES, 34 REFERENCES- APRIL 1965, ANL-6865- CFSTI, \$3.00 CY

DESCRIBES DESIGN OF VARIOUS SYSTEMS, CRITICAL LOADING AND OPERATIONS, BREEDING GAIN MEASUREMENTS STABILITY STUDIES, AND FISSION PRODUCT (IN COVER GAS) MONITOR DEVELOPMENT. A REACTIVITY ANOMALY WAS ASCRIBED TO VARIATION OF SUB-ASSEMBLY SPACING WITH TEMPERATURE.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + MEASUREMENT + PLUTONIUM + REACTOR STABILITY + TRANSFER FUNCTION + OPERATING EXPERIENCE SUMMARY

17-008052

DATED 1965

ALSO IN CATEGORY 9

MCCOOL WJ

EQUIPMENT FAILURE AT THE EXPERIMENTAL BREEDER REACTOR-II (EBR-II)

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-008052 *CONTINUED*
DIVISION OF OPERATIONAL SAFETY, AEC
2 PAGES- JULY 12, 1965, REACTOR OPERATING EXPERIENCE 65-13- AEC-DOS

TWO OF 11 CONTROL RODS FAILED TO SCRAM BECAUSE STEEL BALLS HAD CAUSED BINDING BETWEEN THE ROD SHAFTS AND THE REACTOR COVER. BALLS CAME FROM FAILED BALL-BUSHING, WHICH WAS FOUND TO BE A CARBON STEEL VERSION OF THE SPECIFIED STAINLESS-STEEL BALL BUSHING.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + MATERIAL + REACTOR, ERDA OWNED + FAILURE, SCRAM MECHANISM + CONTROL ROD DRIVE MECHANISMS

17-008056 DATED 1966
EXPLOSION IN EBR II CONTAINMENT VESSEL EQUIPMENT AIRLOCK
1 PAGE- ATOMIC ENERGY CLEARING HOUSE 12(1)- 22 (JANUARY 3, 1966)- PDR

A MINOR CHEMICAL EXPLOSION OCCURRED LATE DECEMBER 1965 IN THE EQUIPMENT AIR-LOCK BETWEEN THE EBR-II REACTOR BUILDING AND THE ATTACHED FUEL CYCLE FACILITY, CAUSING SOME DAMAGE TO THE AIR LOCK COVERS. NO INJURIES WERE SUSTAINED AND NO RADIOACTIVE MATERIAL WAS INVOLVED. THE REACTOR WAS NOT IN OPERATION AT THE TIME OF THE EXPLOSION, WHICH IS NOW BELIEVED TO HAVE BEEN CAUSED BY AN ACCUMULATION OF COMBUSTIBLE GAS FROM BATTERIES IN THE AIR-LOCK.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + EXPLOSION + REACTOR, FAST + REACTOR, LMCR + CONTAINMENT AIR LOCK + INCIDENT, EQUIPMENT + REACTOR, ERDA OWNED

17-008097 DATED 1965
EBR-II
ARGONNE NATIONAL LABORATORY
11 PAGES, 3 FIGURES, 3 TABLES, 1 REFERENCE- OCTOBER 1965, ANL-7115, REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, PP 1-11- CFSTI, \$3.00 CY, \$0.75 MN

THIS REPORT BRIEFLY MENTIONS MODIFICATIONS MADE ON THE EBR-II. THE REASON FOR CONTROL ROD DRIVE DISABILITY WAS DISCUSSED. OF MORE IMPORTANCE IS THE DISCUSSION OF A NEW OSCILLATOR ROD SYSTEM FOR MONITORING OF EBR-II REACTOR KINETICS.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + INSTRUMENT, IN CORE + REACTOR, LMCR + CONTROL ROD DRIVE MECHANISMS

17-008375 DATED 1962
MCCOOL WJ
HALLAM NUCLEAR POWER FACILITY CARBON TRAP CELL FIRE
DIVISION OF OPERATIONAL SAFETY, U. S. ATOMIC ENERGY COMMISSION
2 PAGES- DECEMBER 26, 1962, REACTOR OPERATING EXPERIENCE - 63-2- AEC-DOS

A SODIUM FIRE OCCURRED WHEN THE CARBON TRAP WAS OPENED FOR INSPECTION. THE MAINTENANCE PERSONNEL READ THE INLET AND OUTLET TEMPERATURES, AND OVERLOOKED THE TRAP PREHEAT OVEN INDICATIONS THAT THE SODIUM WAS STILL LIQUID. PROTECTIVE CLOTHING SAVED ONE INDIVIDUAL WHO WAS LITERALLY SPRAYED. THE FIRE WAS FINALLY EXTINGUISHED BY CLOSING THE CELL AND FLOODING WITH NITROGEN. NO RADIATION EXPOSURE OR DAMAGE RESULTED.

FIRE + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + PERSONNEL PROTECTIVE DEVICE + SODIUM + OPERATING EXPERIENCE SUMMARY + REACTOR, ERDA OWNED + INCIDENT, HUMAN ERROR

17-008825 DATED 1965
HALLAM NUCLEAR POWER FACILITY MONTHLY OPERATING REPORT NO. 33
CONSUMERS PUBLIC POWER DISTRICT
54 PAGES- APRIL 1965, TID-22013- CFSTI, \$3.00

AT THE END OF THE MONTH 133 OF THE 150 FUEL ELEMENTS HAD BEEN REMOVED FROM THE REACTOR, ALL 19 CONTROL RODS AND THIMBLES HAD BEEN REMOVED, AND SODIUM WAS DRAINED BELOW THE BOTTOM GRID PLATE OF THE REACTOR. TWENTY-TWO PROCESS TUBES REMAINED IN THE REACTOR, SEVENTEEN OF THESE CONTAINED FUEL ELEMENTS. IN ADDITION THE LOWER CUT-OFF PORTION OF TWO DUMMY ELEMENTS AND THE SOURCE THIMBLE REMAINED IN THE REACTOR. ALL THREE PRIMARY AND SECONDARY LOOPS HAD BEEN DRAINED TO THE FILL TANKS AND ALL THREE STEAM GENERATORS WERE STORED WITH INERT ATMOSPHERES. SECONDARY EXPANSION TANK NO. 3 HAD BEEN REMOVED FROM THE SYSTEM. EIGHT MODERATOR ELEMENTS HAD BEEN REMOVED INCLUDING S/N 110, A KNOWN RUPTURED ELEMENT.

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + SODIUM + DECOMMISSIONING

17-008867 DATED 1965
MCCARTHY WJ
UNNECESSARY CONTAINMENT ISOLATIONS AT FERMI DUE TO SYSTEM TEMPERATURE TRANSIENT
POWER REACTOR DEVELOPMENT CO., DETROIT, MICHIGAN
DOCKET 50-16 +. 1 PAGE- ATOMIC ENERGY CLEARING HOUSE 12(5)- 27 (JANUARY 31, 1965)

HYDROGEN VALUE IN STEAM GENERATOR INCREASES ON POWER REDUCTION DUE TO TEMPERATURE CHANGES, RATHER THAN STEAM LEAKAGE. THEY REQUEST TEMPORARY WAIVER OF TECH SPEC REQUIREMENTS FOR FURTHER TESTING, IN ORDER TO ELIMINATE UNNECESSARY CONTAINMENT ISOLATIONS.

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-008867 *CONTINUED*
REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + CONTAINMENT ISOLATION + OPERATING EXPERIENCE SUMMARY + HEAT EXCHANGERS

17-008984 DATED 1965
OPERATING HISTORY
WESTINGHOUSE ELECTRIC CORPORATION
WCAP-4321 +. 2 PAGES- PP. 10-12 OF EXAMINATION OF THE ENRICO FERMI SODIUM COLD TRAP, 64 FIGURES, 12 TABLES, 9 REFERENCES- NOVEMBER 1965- CFSTI, \$5.00 CY

DURING THE 1000 DEG F TEST IN JANUARY 1963, DECOMPOSITION OF THE ORGANIC BINDER FOR THE GRAPHITE MATERIAL IN THE REACTOR VESSEL ROTATING PLUG RELEASED CARBONACEOUS AND HYDROGENOUS MATERIALS TO THE COVER GAS. THE PLUGGING TEMPERATURE REACHED 425 DEG F WHILE THE REACTOR VESSEL WAS AT 445 DEG F.

FERMI (LMFBR) + REACTOR, LMCR + COOLANT CHEMISTRY + FAILURE, DESIGN ERROR + FUEL HANDLING MACHINE

17-008985 DATED 1965
EXAMINATION OF THE ENRICO FERMI SODIUM COLD TRAP
WESTINGHOUSE ELECTRIC CORPORATION
WCAP-4321 +. 195 PAGES, 64 FIGURES, 12 TABLES, 9 REFERENCES- NOVEMBER 1965- CFSTI \$5.00 CY

THE PRIMARY SYSTEM COLDTRAP WAS REMOVED PRIOR TO REACTOR CRITICALITY FOR EXAMINATION IN FEBRUARY 1963, AFTER 3 YEARS OF INTERMITTENT OPERATION. NONE OF THE PREVIOUSLY REMOVED TRAPS AT OTHER REACTORS HAD BEEN EXAMINED TO VERIFY DESIGN. A SAMPLING METHOD WAS DEVELOPED, AND TESTS REVEALED LESS THAN 17 PERCENT OF THE DESIGN OXYGEN REMOVAL CAPACITY USED. HIGH CONCENTRATIONS OF COPPER AND IRON WERE ALSO FOUND. PORTIONS OF THE MESH AND INNER WALL HAD BEEN DISPLACED, THOUGH NO FLOW MALDISTRIBUTION WAS FOUND.

FERMI (LMFBR) + REACTOR, LMCR + COMPARISON, THEORY AND EXPERIENCE + ANALYTICAL TECHNIQUE + COOLANT CHEMISTRY + COOLANT PURIFICATION SYSTEM

17-009107 ALSO IN CATEGORY 9 DATED 1964
HALL RJ + AVERY KE + MOORE EA
FUEL CLADDING FAILURE MONITOR FOR SODIUM REACTORS
ATOMICS INTERNATIONAL
NAA-SR-6463 +. 26 PAGES, 9 FIGURES, 1 TABLE- DECEMBER 1, 1964- CFSTI \$0.50

FILTERED HELIUM COVER GAS FOR THE SRE FLOWS PAST A GAMMA SCINTILLATION SPECTROMETER MONITORING XENON-133 PEAK. WHILE NO FUEL FAILED DURING CORE 11 OPERATION, THE BACKGROUND WAS A FACTOR OF 1000 BELOW SATURATION WHILE REMAINING HIGH ENOUGH TO EASILY CHECK THE INSTRUMENT RESPONSE.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + SRE (RE) + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + FAILURE, CLADDING

17-009420 DATED 1965
HANSEN AI
THE EFFECTS OF LONG-TERM OPERATION ON SRE SODIUM SYSTEM COMPONENTS
ATOMICS INTERNATIONAL
NAA-SR-11396 +. 50 PAGES, 23 FIGURES, 8 TABLES, REFERENCES, AUGUST 31, 1965, CFSTI \$3.00 CY

MAJOR COMPONENTS FROM THE MAIN HEAT TRANSFER SYSTEM OF THE SRE WERE DISASSEMBLED AND EVALUATED AFTER SEVEN YEARS OF OPERATION. A PARTIALLY SATURATED COLD TRAP CONTAINED FISSION PRODUCTS AND A LARGE NUMBER OF OTHER IMPURITIES IN ADDITION TO OXIDES. THE MAIN INTERMEDIATE HEAT EXCHANGER SHOWED NO MATERIAL DEFECTS NOR THERMAL DEFICIENCY, ALTHOUGH 24 OF THE 316 TUBES WERE BLOCKED BY CARBONACEOUS MATERIAL. THE CONTROL-ROD POISON RING EXPOSED TO THE HIGHEST INTEGRATED FLUX (3.3 TIMES 10 TO THE 20 NVT) REVEALED A DIAMETER GROWTH OF 0.56 PERCENT. NO MAJOR DEFECTS OR WEAKNESSES WERE DISCOVERED IN ANY OF THE COMPONENTS EXAMINED.

REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + SRE (RE) + COOLANT CHEMISTRY + MAIN COOLING SYSTEM + CONTROL RODS + HEAT EXCHANGERS

17-009794 ALSO IN CATEGORY 9 DATED 1965
SUNDERMAN ID
MONTHLY OPERATING REPORT NO. 38, SEPTEMBER, 1965
HALLAM NUCLEAR POWER FACILITY
TID-22454 +. 22 PAGES, 1 FIGURE, 6 TABLES, SEPTEMBER 1965, PDR

THE HALLAM OPERATING CONTRACT WITH AEC WAS TERMINATED SEPTEMBER 9, 1965. CONSUMERS POWER WILL CARETAKE THE FACILITY. REACTOR-BUILDING PRESSURE-DIFFERENTIAL SYSTEM MALFUNCTIONED DUE TO WATER IN THE CUPS AT THE END OF THE SENSING LINES. SIX MODERATOR-CLAD FAILURES WERE DUE TO LONG-TERM STRESS RUPTURE. THE SEVENTH OCCURRED BECAUSE THREE OTHER MODERATOR ELEMENT FAILURES CAUSED A MECHANICALLY PRODUCED GOUGE WHICH FAILED ON THE NEXT HEATUP, DUE TO THERMAL STRESS.

CONTAINMENT, PRESSURE VENTING + HALLAM (LMR) + REACTOR, LMCR + FAILURE, INSTRUMENT + MODERATOR + FAILURE, SEQUENTIAL

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-010050

DATED 1966

COCHRAN JL + MONSON HO + OWENS JE

OPERATING EXPERIENCE WITH LARGE SODIUM COOLED REACTOR PLANTS

CONSUMERS PUBLIC POWER, HALLAM + ARGONNE NATIONAL LABORATORY, ARGONNE + ATOMICS INTERNATIONAL, CANOGA PARK
48 PAGES, 17 FIGURES, 7 TABLES, 11 REFERENCES, APRIL 1966, PRESENTED AT THE AMERICAN POWER CONFERENCE 28TH
ANNUAL MEETING, APRIL 26-28, 1966, CHICAGO, ILLINOIS, APC, \$0.25 CY

FERMI STEAM GENERATORS FAILED TWICE, ONCE DUE TO STRESS CORROSION CRACKING INITIATED BY A CLEANING-
SOLUTION RESIDUE, ONCE DURING CLEAN-UP OPERATION DUE TO VIBRATION. THE SODIUM-WATER REACTION WAS
HANDLED BY THE PRESSURE-RELIEF SYSTEM. HALLAM SECONDARY EXPANSION TANK FAILED BECAUSE A HALIDE-
COATED WELDING ROD WAS USED FOR FIELD INSTALLATION OF AN INTERNAL Baffle.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR
+ OPERATING EXPERIENCE SUMMARY + STRESS CORROSION

17-010242

ALSO IN CATEGORY 8

DATED 1966

GOLMANN K + MINUSHKIN B

SODIUM TECHNOLOGY

UNITED NUCLEAR CORPORATION

TID-8541 +. 105 PAGES, 230 REFERENCES, 37 FIGURES, 14 TABLES, PP. 321-425 OF REACTOR TECHNOLOGY, SELECTED
REVIEWS--1965, L. E. LINK, EDITOR, ARGONNE NATIONAL LABORATORY, JANUARY 1966, CFSTI, \$4.50 CY, \$2.00 MN

THIS REVIEW COVERS SODIUM TECHNOLOGY THAT HAS BEEN DEVELOPED IN CONJUNCTION WITH THE DESIGN,
CONSTRUCTION, AND OPERATION OF LIQUID-METAL-COOLED REACTOR PLANTS. ALSO REVIEWED IS THE STATE OF
THE ART OF SODIUM TECHNOLOGY AS RELATED TO NEXT-GENERATION POWER PLANTS, WHICH ARE PROJECTED TO
HAVE A CAPABILITY OF PRODUCING STEAM AT 1050 F, WITH MAXIMUM LIQUID-METAL TEMPERATURES OF 1200 TO
1300 F. WITHIN THIS FRAMEWORK THE LIQUID METALS OF INTEREST ARE SODIUM AND SODIUM-POTASSIUM
(NAK). REVIEWED HERE ARE MATERIALS TECHNOLOGY, SUCH AS MATERIALS SELECTION, CORROSION, CONTROL
OF SODIUM PURITY, AND THE EXPERIENCE GAINED IN THESE AREAS WITH OPERATING PLANTS.

CONTAINMENT DESIGN + REACTOR, LMCR + OPERATION + SODIUM + NAK + CORROSION + METAL, LIQUID

17-010263

DATED 1961

FILLMORE FL

ANALYSIS OF SRE POWER EXCURSION OF JULY 13, 1959

ATOMICS INTERNATIONAL

NAA-SR-5898 +. 47 PAGES, 8 FIGURES, 9 TABLES, 8 REFERENCES, SEPTEMBER 15, 1961, CFSTI, \$1.00 CY

THE REACTOR ACTED ABNORMALLY AT 10 PERCENT, BUT A POWER INCREASE WAS STARTED WHICH CONTINUED
(DESPITE CONTROL-ROD INSERTION) TO ABOUT 25 PERCENT POWER, WHEN POWER RAPIDLY DROPPED (DUE TO
TRANSIENT SODIUM-BOILING DECREASING). THE REACTOR WAS BROUGHT BACK TO POWER AND INCREASED UNTIL,
DESPITE ROD INSERTION, A 7.5-SEC TRANSIENT TO 70 PERCENT POWER OCCURRED WHEN 10 FUEL CHANNELS
PROGRESSIVELY VOIDED SODIUM. REPORT PRESENTS ANALOG SIMULATION TO DETERMINE MAGNITUDES AND RATES
OF REACTIVITY CHANGES IN ORDER TO CHOOSE TRUE CAUSE FROM SEVERAL POSSIBLE. DOPPLER WAS SHOWN TO
BE TOO SMALL (AND FOLLOWED VOIDING) TO RESTRAIN ACCIDENT.

COMPUTER, ANALOG + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + REACTOR TRANSIENT + VOID COEFFICIENT + SRE
(RE) + INCIDENT

17-010622

DATED 1965

POWER REACTOR DEVELOPMENT COMPANY, ENRICO FERMI ATOMIC POWER PLANT REPORT FOR DECEMBER 1965

POWER REACTOR DEVELOPMENT COMPANY

EF-28 +. 14 PAGES, 2 FIGURES, DECEMBER 1965, DOCKET NO. 50-16

THIS ISSUE IN THE SERIES SEEMS ADEQUATE IN DEPTH AND COVERS OPERATIONS, HEALTH PHYSICS,
MAINTENANCE AND MODIFICATION, NONNUCLEAR TEST PROGRAMS, AND FACILITY CHANGES.

USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + OPERATING EXPERIENCE SUMMARY +
MODIFICATION + REPORT, OPERATIONS SUMMARY

17-010624

DATED 1966

ENRICO FERMI ATOMIC POWER PLANT REPORT FOR FEBRUARY 1966

POWER REACTOR DEVELOPMENT COMPANY

EF-30 +. 11 PAGES, FEBRUARY 1966, DOCKET NO. 50-16

THIS REPORT IS ONE OF A SERIES. IT SEEMS ADEQUATE IN DEPTH, AND COVERS OPERATIONS, HEALTH
PHYSICS, MAINTENANCE, AND MODIFICATION, NONNUCLEAR TEST PROGRAMS, AND FACILITY CHANGES.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + OPERATING EXPERIENCE SUMMARY +
MODIFICATION + REPORT, OPERATIONS SUMMARY

17-010986

DATED 1966

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-010986 *CONTINUED*

HYDROGEN EXPLOSION AT REACTOR FACILITY

DIVISION OF OPERATIONAL SAFETY, U. S. ATOMIC ENERGY COMMISSION
SERIOUS ACCIDENTS NO. 270 +. 4 PAGES, 5 FIGURES, JULY 15, 1966, DPI

AN EXPLOSION RESULTED FROM TEMPORARY STORAGE OF A FUEL-TRANSFER COFFIN IN AN AIRLOCK WITH THE EMERGENCY BATTERIES ACCIDENTALLY LEFT ON FULL CHARGE. THE HYDROGEN-AIR EXPLOSION OCCURRED WHEN THE DOOR WAS OPENED, PROBABLY FROM A SWITCH OR RELAY SPARK. THE PRESSURE (45 TO 70 PSI) BLEW THE ONE-TON AIRLOCK DOOR OFF ITS HINGES AND GENERATED MISSILES INSIDE THE CONTAINMENT.

REACTOR, BREEDER + EBR 1 AND 2 (RE) + EXPLOSION + REACTOR, FAST + HYDROGEN + REACTOR, LMCR + MISSILE GENERATION AND PROTECTION + FUEL HANDLING MACHINE

17-011040

ALSO IN CATEGORY 5

DATED 1966

FERMI POWER COEFFICIENT ANALYSIS

POWER REACTOR DEVELOPMENT COMPANY

2 PAGES, JULY 22, 1966, ATOMIC ENERGY CLEARING HOUSE, 12(32), PAGES 12-13 (AUGUST 8, 1966)

REVIEW OF DATA SHOWS THAT POWER COEFFICIENT IS LINEAR WITH POWER, AND TRANSFER-FUNCTION MEASUREMENTS SHOW REACTOR STRONGLY STABLE. PRINCIPLE CHANGES WERE - MORE-NEGATIVE CORE SODIUM EXPANSION EFFECT AND LESS-NEGATIVE CORE RADIAL EXPANSION EFFECT. NO CAUSE FOR CONCERN EXISTS, AS PREVIOUS SAFETY ANALYSIS USED POWER COEFFICIENTS AS LOW AS MINUS 0.15 CENT PER MEGAWATT, AND PRESENT COEFFICIENT IS MINUS 0.18.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + POWER COEFFICIENT + COMPARISON, THEORY AND EXPERIENCE

17-011138

DATED 1966

PETERS NT + AMOROSI A

FEATURES OF SECOND GENERATION SODIUM-COOLED FAST REACTORS

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

13 PAGES, 4 FIGURES, 2 TABLES, 1966 ANNUAL MEETING AMERICAN NUCLEAR SOCIETY, DENVER, COLORADO, JUNE 20-23, 1966, ANS TRANS. 9(1), PAGE 215

DISCUSSES FEATURES OF FERMI-I PLANT WHICH SHOULD BE RETAINED OR MODIFIED FOR THE DESIGN OF A FERMI-II 500-MW(e) PLANT. BRIEFLY CONSIDERS DESIGN OF FUEL AND CORE, VESSEL, FUEL-HANDLING SYSTEM, AND HEAT-TRANSPORT SYSTEM. POINTS OUT THAT THE INABILITY TO INSPECT PIPE HANGERS AND THE POSSIBLE ADVERSE EFFECT OF THE SECONDARY CONTAINMENT OF THE PRIMARY PIPING ON THE PIPE INTEGRITY LEADS TO CONSIDERATION OF AVOIDING THE NEED FOR SECONDARY CONTAINMENT.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + FAILURE, PIPE + OPERATING EXPERIENCE SUMMARY

17-011428

DATED 1964

OPERATING AUTHORIZATION AND TECHNICAL SPECIFICATIONS FOR FULL POWER OPERATION OF HALLAM

CONSUMERS PUBLIC POWER DISTRICT, LINCOLN, NEBRASKA

30 PAGES, FEBRUARY 5, 1964, DOCKET NO. 115-3, PDR

AUTHORIZATION REFERS TO OPERATION AT 256 MW(TH).

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + TECHNICAL SPECIFICATIONS

17-011605

DATED 1965

THOMPSON DH

VAPOR TRAPS PROTECT LIQUID-METAL SYSTEMS

COMBUSTION ENGINEERING INC.

12 PAGES, 11 FIGURES, 1 TABLE, 19 REFERENCES, POWER REACTOR TECHNOLOGY 8(4), PAGES 259-270, (FALL 1965)

DISCUSSED THE PRINCIPLES OF TRAPS FOR VAPORIZED METAL USED IN LIQUID-METAL SYSTEMS. DESCRIBED THE INERT-GAS BLANKET SYSTEMS AND VAPOR TRAPS USED AT SEVERAL REACTORS. DEVELOPMENT WORK ON A CONTINUOUS-HELICAL-FIN TYPE OF VAPOR TRAP APPEARED PROMISING BUT TERMINATION OF THE PROJECT PREVENTED COMPLETION OF THE EVALUATION TESTS. AN EFFICIENCY OF 99.8 PERCENT WAS ACHIEVED WITH A SODIUM-VAPOR TRAP RECENTLY DEVELOPED BY ARGONNE NATIONAL LABORATORY. RECOMMENDATIONS FOR DESIGN INCLUDE LOCATING THE TRAP NEAR THE VAPOR SOURCE OR ENSURING ADEQUATE HEAT CAPACITY FOR THE GAS PIPING, AND USING GRADUATED-DENSITY OR ZONE-DENSITY PACKINGS. A COMPREHENSIVE DEVELOPMENT PROGRAM WAS RECOMMENDED.

REACTOR, LMCR + CONDENSATION + NAK + VAPOR PRESSURE + METAL, LIQUID + COOLANT PURIFICATION SYSTEM

17-011666

DATED 1959

MCDONALD J + MARTEN W

A SIPHON BREAK AS A BLOCKING VALVE

ATOMICS INTERNATIONAL

NAA-SR-4050 +. 25 PAGES, 11 FIGURES, OCTOBER 15, 1959, CFSTI \$0.75 CY

REPORTS AN EXPERIMENT TO DETERMINE THE FEASIBILITY OF USING THE BREAKING OF A SIPHON AS A QUICK-ACTING MEANS FOR STOPPING SODIUM FLOW FOLLOWING LOSS OF PUMP POWER TO A 2-IN. PIPE SYSTEM WITH A HIGH-SPEED FREE-SURFACE CENTRIFUGAL PUMP WAS USED IN THIS INVESTIGATION. RUNS WERE MADE WITH SODIUM AT 500 AND 940 F, COVER GAS AT VARIOUS PRESSURES UP TO 10 PSIG, AND REYNOLDS NUMBERS UP TO

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-011666 *CONTINUED*

APPROXIMATELY 360,000. THE SIPHON-BREAK WAS ESTABLISHED AS AN EFFECTIVE METHOD FOR RAPID FLOW STOPPAGE - HOWEVER, A BRIEF REVERSAL OF FLOW FOLLOWS THE INITIAL FLOW STOPPAGE. AN EXPRESSION FOR THE FLOW TRANSIENT FOLLOWING THE BREAKING OF THE SIPHON WAS DERIVED WHICH AGREED REASONABLY WELL WITH EXPERIMENTAL RESULTS.

REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + VALVES

17-011671

DATED 1958

NEBIKER RP

ORIGIN OF WATER VAPOR IN THE GALLERIES

ATOMICS INTERNATIONAL

NAA-SR-MEMO-2434 +. 8 PAGES, JANUARY 23, 1958, CFSTI \$1.80 FS

EXPERIMENTS WERE CONDUCTED TO DETERMINE THE AMOUNT OF WATER CONDENSING IN THE GALLERIES AT SRE. THE AMOUNT WAS DEEMED GREAT ENOUGH TO CONSTITUTE A FIRE HAZARD IN EVENT OF A SODIUM LEAK. THE FOLLOWING RECOMMENDATIONS FOR CONTROL OF THE GALLERY ATMOSPHERE WATER CONTENT WERE MADE - (A) APPLY A SEALANT OR STAINLESS LINER ON THE GALLERY WALLS TO MINIMIZE ENTRY OF GROUND WATER THROUGH THE CONCRETE, (B) INSTALL A DEHUMIDIFIER THAT WILL PRECIPITATE THE MOISTURE FASTER THAN IT ENTERS, (C) CIRCULATE THE NITROGEN COVER GAS THROUGH A DESSICANT, (D) PROVIDE A DEVICE TO CONTINUALLY MEASURE THE MOISTURE CONTENT OF THE GALLERY ATMOSPHERE.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + METAL WATER REACTION + SRE (RE) + CONTAINMENT ATMOSPHERE, INERT + WATER VAPOR

17-011795

ALSO IN CATEGORY 7

DATED 1966

FERMI FUEL ELEMENT FAILURE, OCTOBER 4, 1966

POWER REACTOR DEVELOPMENT COMPANY

1 PAGE, NUCLEONICS WEEK, 7(41), [OCTOBER 13, 1966]

AT 34 MW(TH) DURING A STARTUP, THE CONTROL RODS SEEMED TO BE WITHDRAWN FARTHER THAN NORMAL. TWO CORE-OUTLET TEMPERATURES WERE HIGHER THAN NORMAL. COVER-GAS ACTIVITY INCREASED, THE REACTOR-BUILDING RADIATION MONITORS GAVE AN ISOLATION SIGNAL, AND THE REACTOR WAS SCRAMMED. AS OF OCT. 10, 1966, 22 CENTS REACTIVITY HAD BEEN LOST, AND THE COVER-GAS XENON WAS ABOUT EQUAL TO THAT EXPECTED FROM A CORE SUBASSEMBLY. REACTIVITY BEGAN TO BE LOST AT ABOUT 15 MW(TH).

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + INCIDENT

17-011945

ALSO IN CATEGORY 7

DATED 1966

MCCARTHY WJ

FERMI FUEL DAMAGE PRELIMINARY REPORT, OCTOBER 10, 1966

POWER REACTOR DEVELOPMENT COMPANY, DETROIT, MICHIGAN

2 PAGES, ATOMIC ENERGY CLEARING HOUSE 12(42), PAGES 9-10 (OCTOBER 17, 1966) DOCKET NO. 50-16

BRIEF REVIEW OF AN INCIDENT ON OCTOBER 5, 1966. ON RAISING THE POWER LEVEL TO 34 MW(TH), A REGULATING ROD SEEMED FARTHER OUT THAN NORMAL, ALSO TWO FUEL-OUTLET TEMPERATURES WERE ABOVE NORMAL. COVER-GAS ACTIVITY HAD INCREASED. SOME FUEL DAMAGE AND REDISTRIBUTION RESULTED, AND INVESTIGATIONS ARE CONTINUING.

REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + REACTOR, LMCR

17-011994

DATED 1965

ALEXANDERSON EL + BRANYAN CE + OLSON WR

OPERATING EXPERIENCE AT THE ENRICO FERMI ATOMIC POWER PLANT

POWER REACTOR DEVELOPMENT CO., DETROIT + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT + THE DETROIT

EDISON CO., DETROIT

ANS-100 +. 16 PAGES, 7 FIGURES, 3 TABLES, 7 REFERENCES, MARCH 1, 1965, DOCKET NO. 50-16, PAGES 41-56 OF FAST REACTOR TECHNOLOGY, AMERICAN NUCLEAR SOCIETY NATIONAL TOPICAL MEETING, DETROIT, MICHIGAN, APRIL 26-28, 1965

GENERAL DISCUSSION OF PLANT COMPONENTS, AS AFFECTED BY TEST-PROGRAM OPERATION. SHORT REPORT OF NUCLEAR TESTING BELOW 1 MW(TH). ONE SAFETY SYSTEM CIRCUIT WAS FOUND IN WHICH FAILURE OF ONE PAIR OF DIODES (OUT OF 117 PAIRS) RESULTED IN A CANNOT-SCRAM SITUATION. IN-CORE BF₃ DETECTORS DISCARDED AS INITIAL LOADING SHOWED THEM SUBJECT TO GAMMA INTERFERENCE. ONE SAFETY ROD FAILED TO SCRAM AS SODIUM LEAKED THROUGH A BELLOWS AND FROZE THE RELEASE MECHANISM. CONTAINMENT LEAKS FOUND IN FOUR NEOPRENE WASHER SEALS, (A FLEXIBLE-GASKET-TYPE SEAL IS NOW USED - WEATHERBAN). SLIGHT GAINS IN REACTIVITY SEEM TO FOLLOW PERIODS OF HIGH SODIUM TEMPERATURE, AS PERHAPS THE ZIRCONIUM CLADDING TAKES UP HYDROGEN. GAS-ENTRAINMENT REACTIVITY EFFECT NOT SEEN.

REACTOR, FAST + REACTOR, LMCR + REACTOR STARTUP EXPERIENCE + FAILURE, SCRAM MECHANISM + SAFE FAILURE CRITERION + REACTIVITY EFFECT, ANOMALOUS + CONTAINMENT LEAKAGE CONTROL + INSTRUMENT, STARTUP

17-012102

DATED 1966

STRAUSS SD

HALLAM - DEATH OF A CONCEPT

4 PAGES, 6 FIGURES, 2 TABLES, 14 REFERENCES, NUCLEONICS 24(10) PAGES 63-66 (OCTOBER 1966) DOCKET NO. 115-3

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-012102 *CONTINUED*

PRESENTS THE REASONS FOR TERMINATING THE PROGRAM. A STRONG ARGUMENT FOR TERMINATION WAS THE UPGRADED SODIUM REACTOR EXPERIMENT, WHICH REQUIRES A 200 F HIGHER SODIUM TEMPERATURE, MORE APPLICABLE TO THE REFERENCE 1000-MW(E) SODIUM-COOLED FAST BREEDER. THE CONTAINERS FOR THE GRAPHITE MODERATOR BLOCKS WERE THE MAJOR FACTOR RESPONSIBLE FOR TERMINATION. LOCALIZED REACTIVITY LOSSES AND POWER DEPRESSIONS INDICATED THAT THE MODERATOR CANS WERE BEING PENETRATED BY THE SODIUM. VENTING SNORKELS TO EQUALIZE PRESSURE IN AND OUTSIDE THE CANS HAD BEEN ELIMINATED, AND SCALLOPED VERTICAL EDGES WERE INTRODUCED. THESE CHANGES RESULTED IN EXCESSIVE STRESSES. THE OVERALL PLANT AVAILABILITY HAD BEEN ONLY 61 PERCENT. STATUS OF DISPOSITION OF PLANT IS GIVEN. OTHER PROBLEMS (AND SOLUTIONS) ARE DISCUSSED, SUCH AS INSTRUMENTATION NOISE, HEAT-EXCHANGER-TUBE FAILURE, MOISTURE CARRYOVER FROM EVAPORATOR TO SEPARATOR (HAPPENING STEAM-GENERATOR OPERATION), OXIDE CONTAMINATION OF THE SODIUM, AND JAMMING OF THE FUEL-CHANNEL-ORIFICE PLUGS.

FAILURE, COMPONENT + FAILURE, EQUIPMENT + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + CRUD + INSTRUMENT, ABNORMAL INDICATION

17-012303

ALSO IN CATEGORY 9

DATED 1963

CASHETTA J

NUCLEAR INSTRUMENT ANOMALIES DURING RISE-TO-POWER TESTING AT THE HALLAM NUCLEAR POWER FACILITY

ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA

NAA-SR-8714 +. 47 PAGES, JULY 13, 1963, CFSTI \$2.00 CY

SEVERAL PROBLEMS ASSOCIATED WITH THE NUCLEAR INSTRUMENTATION WERE DISCOVERED DURING THE RISE-TO-POWER TESTING PHASES OF THE HALLAM NUCLEAR POWER FACILITY. THE PROBLEMS RANGED IN SEVERITY FROM THE TYPE JUSTIFYING PREVENTATIVE MAINTENANCE TO THOSE DICTATING ESSENTIAL MODIFICATION. BECAUSE OF THE LIMITED TIME AVAILABLE FOR EXPERIMENTATION AND INVESTIGATION, IT WAS NOT POSSIBLE TO PERFORM A COMPLETE ANALYTICAL STUDY OF THE PROBLEMS. THE REPORT ENCOMPASSES THE REMEDIAL ACTION INITIATED TO PROVIDE SATISFACTORY OPERATION OF THE NUCLEAR INSTRUMENTATION.

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + INSTRUMENT, NUCLEAR + REACTOR, LMCR + TEST, PLANT RESPONSE + FAILURE, INSTRUMENT + MODIFICATION

17-012319

ALSO IN CATEGORY 11

DATED 1965

SELECTED OPERATING EXPERIENCES

POWER REACTOR DEVELOPMENT COMPANY

EF-28 +. 3 PAGES, POWER REACTOR DEVELOPMENT COMPANY, ENRICO FERMI ATOMIC POWER PLANT REPORT FOR DECEMBER 1965 PAGES 1-2 AND 10, DECEMBER 1965, DOCKET NO. 50-16

THE NUCLEAR TESTING AT LESS THAN 1 MW(TH) EXTENDED FROM AUGUST 1963 TO DECEMBER 1965. THE FIRST POWER INCREASE ABOVE 1 MW(TH) TOOK PLACE DECEMBER 29, 1965. THE NEW SOURCE-RANGE DETECTORS (INCLUDING TWO SURROUNDED BY ZIRCONIUM HYDRIDE) WORKED WELL. WHILE DAILY CHECKS OF ELECTRICAL PENETRATION LEAKAGE AT 5 PSIG WERE BELOW 50 SCF/DAY, TESTS AT 32 PSIG INDICATED 110.

USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

CONTAINMENT, HIGH PRESSURE + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + TEST, LEAK RATE + REPORT, OPERATIONS SUMMARY + INSTRUMENT, STARTUP RANGE

17-012341

ALSO IN CATEGORY 1

DATED 1966

INDUSTRY, AEC LEADERS CLASH OVER FAST BREEDER PROGRAM

ATOMIC ENERGY COMMISSION

3 PAGES, NUCLEONICS WEEK 7(45), PAGES 1-3, (NOVEMBER 9, 1966)

INDUSTRIAL REPRESENTATIVES INDICATED THAT THEY NEEDED SYSTEMS-ENGINEERING EXPERIENCE WITH SODIUM-COOLED REACTORS, WHICH THEY CANNOT GET FROM JUST ONE OPERATING REACTOR (EBR-2). SECOND, AN INTENSIVE PROGRAM ON SODIUM-REACTOR SAFETY DEVELOPMENT IS NEEDED. FINALLY, THE PROGRAM NEEDS TO BE ACCELERATED. AEC COMMISSIONER RAMEY INDICATED THAT SEVERAL 200- TO 500-MW(E) DEMONSTRATION BREEDER PLANTS WOULD NEED TO BE BUILT, REQUIRING A MASSIVE DEVELOPMENT PROGRAM FOR WHICH PRESENT TECHNOLOGY IS NOT READY.

REACTOR, FAST + REACTOR, LMCR + REVIEW

17-012383

ALSO IN CATEGORY 9

DATED 1966

HESS JW

DESIGN AND OPERATING EXPERIENCE WITH THE CONTROL AND SAFETY ROD DRIVE MECHANISMS FOR THE ENRICO FERMI FAST BREEDER REACTOR

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT

APDA-301 +. 221 PAGES, 42 FIGURES, 6 TABLES, JUNE 1, 1966, CFSTI \$6.00 CY, \$1.25 MN

TRACES THE EVOLUTION OF THE DESIGN AND POLICY FOR THE CONTROL-ROD DRIVES. TESTING AND OPERATING EXPERIENCE ARE SUMMARIZED. DURING THE FIRST YEAR OF REACTOR OPERATION, ONLY TWO SIGNIFICANT MALFUNCTIONS OCCURRED - FAILURE OF A LATCH BELLWIS ALLOWED SODIUM TO FREEZE ONE SAFETY ROD, AND OPENING A BLOCKING VANE TOO FAST WITH A ROTARY SOLENOID DISTURBED THE LIMIT-SWITCH SETTINGS, SOLVED BY OPENING THE VANE WITH A SMALL GEAR MOTOR. EXPERIENCE WITH THESE UNITS ARE EVALUATED AND SUGGESTIONS MADE FOR IMPROVEMENT. MANY PROBLEMS RESULTED FROM LACK OF OPERATING EXPERIENCE, AND THE MOST SERIOUS DIFFICULTIES CAME FROM NOT TESTING COMPONENTS OR PROTOTYPES IN HOT SODIUM BEFORE ATTEMPTING TO INSTALL THEM IN THE REACTOR. EXPERIENCE TO DATE INDICATES THAT CONSIDERATION SHOULD BE GIVEN TO DETAILING PROCEDURES FOR OPERATION, CALIBRATION, AND

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-012383 *CONTINUED*
TROUBLESHOOTING. 140 PAGES OF APPENDIXES PROVIDE DETAILED SPECIFICATIONS AND CALCULATIONS.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + CONTROL ROD, SHIM, SAFETY + CONTROL ROD SCRAM MECHANISM + DESIGN STUDY + MODIFICATION + TEST, CONTROL ROD DRIVE + FAILURE, SCRAM MECHANISM + CONTROL ROD DRIVE MECHANISMS

17-013236
PRDC REQUESTS CHANGE 11 FOR FERMI
POWER REACTOR DEVELOPMENT COMPANY
5 PAGES, NOVEMBER 10, 1966, DOCKET NO. 50-16

DATED 1966

PRDC REQUESTS 6 MONTHS AUTHORITY TO DISCONNECT ELECTRICAL LEADS TO MAIN MOTORS TO PREVENT A FALSE START FROM RAISING FUEL ELEMENTS OUT OF CORE, AS CORE HOLD-DOWN PLATE MUST BE RAISED FOR FUEL HANDLING. PONY MOTORS WOULD MAINTAIN DECAY-HEAT-REMOVAL FLOW.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + TECHNICAL SPECIFICATIONS + REFUELING + CORE COMPONENTS + PUMPS

17-013239
FERMI CHANGE 11 - MAIN PUMP DISCONNECTION WITH CORE HOLD DOWN PLATE RAISED
DIVISION OF REACTOR LICENSING
4 PAGES, NOVEMBER 21, 1966, DOCKET NO. 50-16, PDR

DATED 1966

AEC ALLOWS PRIMARY PUMP ELECTRICAL LEADS TO BE DISCONNECTED (WITH FLOW MAINTAINED BY PONY MOTORS) WITH THE HOLD-DOWN PLATE RAISED, TO PREVENT PUMP STARTING AND WASHING FUEL ELEMENTS OUT OF THE CORE DURING INVESTIGATION OF CORE MELTDOWN.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + TECHNICAL SPECIFICATIONS + REFUELING + CORE COMPONENTS + SAFETY EVALUATION + PUMPS

17-013314
AEC-DRL SAFETY EVALUATION OF FERMI REACTOR PROPOSED OPERATION AT 200 MWTH
DIVISION OF REACTOR LICENSING, USAEC
54 PAGES, 4 TABLES, 46 REFERENCES, JULY 9, 1965, DOCKET NO. 50-16

DATED 1965

THIS UPDATES DRL REVIEW OF OPERATION LESS THAN 1 MW BY REVIEWING OPERATING EXPERIENCE, SYSTEM TESTS, ETC. AREAS MORE COMPLETELY ANALYZED WERE - STEAM-GENERATOR-TUBE-LEAKAGE EXPERIENCE, FUEL AND REPAIR BUILDING, AND CONTAINMENT AND MISSILE PROTECTION. LEAKAGE OF STEAM-GENERATOR TUBES AND SEQUENTIAL FAILURE OF OTHER TUBES HAS OCCURRED AND BEEN INVESTIGATED.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

REACTOR, BREEDER + FAILURE, EQUIPMENT + REACTOR, FAST + FERMI (LMFBR) + LICENSE STATUS + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + FAILURE, INSTRUMENT + REVIEW + (MONTICELLO-BWR)

17-013315
OPERATING EXPERIENCE AND INCIDENTS AT FERMI REACTOR
DIVISION OF REACTOR LICENSING, USAEC
54 PAGES, 4 TABLES, 46 REFERENCES, JULY 9, 1965, AEC-DRL SAFETY EVALUATION OF FERMI REACTOR PROPOSED OPERATION AT 200 MWTH

DATED 1965

A SLIGHT REACTIVITY GAIN OCCURRED DUE TO LONGITUDINAL FUEL-PIN SHRINKAGE OR ZIRCONIUM HYDRIDE FORMATION. CLAD INDENTATIONS OCCUR AT SUPPORT AREAS. HIGH COVER-GAS PRESSURE PLUS CORRODED BELLAWS CAUSED SODIUM TO FREEZE AND ONE CONTROL ROD TO STICK. SAFETY RODS DROPPED 14 TIMES DUE TO INSUFFICIENT MAGNETISM. A TRANSISTOR FAILED IN AN ALARM MODULE. A RESISTOR FAILURE RENDERED ONE SOURCE-RANGE PERIOD SCRAM CHANNEL INEFFECTIVE FOR ONE DAY. CORE PRESSURE DROP WAS LOW, AND TESTS SHOWED THAT SAFETY RODS DID NOT FLOAT UNTIL 120 PERCENT DESIGN FLOW WAS REACHED. A FUEL ELEMENT WAS INSERTED INTO AN OCCUPIED STORAGE POSITION. PRIMARY-LOOP CHECK-VALVES HAVE CAUSED 3/8-IN. PIPE MOVEMENTS, AND NEW VALVES WERE DESIGNED. CARBON SHIELDING MATERIAL IN THE PRIMARY SHIELD PLUG EVOLVED GASES WHICH CONTAMINATED THE SODIUM AND MAY LEAD TO CARBURIZATION OF THE STAINLESS STEEL. A SINGLE STEAM-GENERATOR-TUBE LEAK, CAUSED BY VIBRATION, LED TO OTHER FAILURES DUE TO THINNING OF TUBE WALLS AS A RESULT OF SODIUM-WATER REACTIONS. THE FUEL-HANDLING-CASK CAR HAS HAD A LARGE NUMBER OF DIFFICULTIES THAT ARE CONTINUING, AND USE IS NOT AUTHORIZED FOR TRANSPORTING IRRADIATED FUEL UNTIL THOROUGH TESTING IS SATISFACTORY. WHILE THE DIESEL HAS OCCASIONALLY FAILED TO START ON TIME, ONLY SEVEN MOMENTARY POWER OUTAGES HAVE OCCURRED IN FIVE YEARS, WHEN ONE OF THE TWO FEEDERS WAS HIT BY LIGHTNING.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

REACTOR, BREEDER + EMERGENCY POWER, ELECTRIC + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + METAL WATER REACTION + REACTIVITY EFFECT + TEST, LEAK RATE + WELDS + OPERATING EXPERIENCE SUMMARY + FAILURE, INSTRUMENT + COOLANT CHEMISTRY + MODIFICATION + FAILURE, FATIGUE + FAILURE, SCRAM MECHANISM + FUEL HANDLING MACHINE + FAILURE, ADMINISTRATIVE CONTROL + REACTIVITY EFFECT, ANOMALOUS + SAFETY EVALUATION + CONTROL RODS

17-013334

DATED 1966

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-013534 *CONTINUED*
GRISWOLD AS + ULSON WR
POWER REACTOR DEVELOPMENT COMPANY, ENRICO FERMI ATOMIC POWER PLANT REPORT FOR JUNE 1966
POWER REACTOR DEVELOPMENT CO., DETROIT
PRDC-EF-34 +. 11 PAGES, JUNE 1966, DOCKET NO. 50-16

OPERATION AT 1/3 POWER (67 MWTH) SHOWED THAT THE NUCLEAR INSTRUMENT READ NEARLY 10% HIGH, AND SODIUM FLOW WAS LOW. VARIOUS OPERATING AND NOISE TESTS WERE RUN. A 4.8-KV CABLE FAULT INTERRUPTED POWER, AND THE DIESEL STARTED. PARTS OF THE TRANSFER TANK FROZE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + ACCIDENT, LOSS OF POWER + INSTRUMENT, ABNORMAL INDICATION + REPORT, OPERATIONS SUMMARY

17-013535 DATED 1966
ENRICO FERMI ATOMIC POWER PLANT REPORT FOR JULY 1966
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-35 +. 11 PAGES, JULY 1966

FOUR HEAT-BALANCE TESTS SHOW THAT THE NUCLEAR INSTRUMENT READ 12% HIGH AT 100 MWTH, AFTER CORRECTIONS MADE TO SODIUM FLOW INSTRUMENTS (WHICH HAD NEVER BEEN CALIBRATED IN HOT SODIUM). THE STEAM-DUMP-TO-CONDENSER SYSTEM WAS FOUND ADEQUATE FOR ONLY 50% POWER. TWO UNSCHEDULED SCRAMS DURING TESTING OCCURRED WHEN WATER BOILING TOO SOON IN STEAM GENERATORS CAUSED FLOW INSTABILITIES, AND THUS HIGH SODIUM TEMPERATURE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

HEAT TRANSFER, BOILING + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + INSTRUMENT, FLOW + SCRAM, SPURIOUS + INSTRUMENT, ABNORMAL INDICATION + REPORT, OPERATIONS SUMMARY

17-013536 DATED 1966
ENRICO FERMI ATOMIC POWER PLANT REPORT FOR AUGUST 1966
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-36 +. 11 PAGES, AUGUST 1966, DOCKET NO. 50-16

DURING THE 100-MWTH RUN, NO. 1 STEAM-GENERATOR COVER-GAS HYDROGEN CONTENT BEGAN INCREASING. INSPECTION SHOWED NUMEROUS MICROBRAZE WELD DEFECTS (OVERLAYS APPLIED IN LAST REPAIR). BRAZE WAS REMOVED, AND WELDED WITH 2 1/4 CR - 1 MO ROD. THREE TUBE-SHEET WELD LEAKS WERE FOUND. CERTAIN POSITIONS OF THE FUEL HANDLING-MACHINE LATCH MECHANISM WERE INTERFERED WITH BY THE LIMIT SWITCH AND DID NOT STOP ON BOTTOM. STREAMING PRODUCED A 2-REM/HR FIELD AROUND THE PRIMARY SODIUM SAMPLING STATION.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + WELDS + FUEL HANDLING MACHINE + SHIELDING + REPORT, OPERATIONS SUMMARY + FAILURE, TUBING + STEAM GENERATOR

17-013537 DATED 1966
FERMI AMENDMENT 4 - STORAGE OF SEALED PLUTONIUM FUEL SPECIMEN
DIVISION OF REACTOR LICENSING
8 PAGES, DECEMBER 2, 1966, DOCKET NO. 50-16

FERMI ALLOWED TO STORE 500 SEALED PLUTONIUM FUEL ELEMENT TEST SPECIMENS.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + PLUTONIUM + TECHNICAL SPECIFICATIONS + SAFETY ANALYSIS

17-013979 DATED 1966
FERMI CHANGE 11 - DISCONNECT POWER TO MAIN PUMPS DURING FUEL REMOVAL
DIVISION OF REACTOR LICENSING
4 PAGES, NOVEMBER 21, 1966, DOCKET NO. 50-16

AEC APPROVES DISCONNECTION OF MAIN-PUMP ELECTRICAL LEADS WHILE CORE HOLD-DOWN MECHANISM IS RAISED TO EXAMINE DAMAGED FUEL ELEMENTS. CORE COOLING MAINTAINED BY PONY MOTORS.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D.C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + TECHNICAL SPECIFICATIONS + SAFETY EVALUATION + PUMPS

17-014334 DATED 1965
HALFEN FJ
LSGR FAST SHUTDOWN PROCEDURE
ATOMICS INTERNATIONAL
NAA-SR-MEMO-11041 +. 45 PAGES, FEBRUARY 12, 1965

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-014334 *CONTINUED*

SEVERAL ADDITIONAL SHUTDOWN PROCEDURES WERE ANALYZED FOR POSSIBLE USE ON THE 200-MWE SGR. THESE SCHEMES ARE - (1) SEQUENCED ROD DROP, (2) ROD RUNDOWN AND PUMP SHUTDOWN DELAY, (3) SCRAM AND PUMP SHUTDOWN LEAD, (4) ROD DROP AND RUNDOWN OF RODS. ALL THESE SCHEMES ASSUME THAT THE PUMPS ARE SHUTDOWN SOME TIME PRIOR TO, DURING, OR SHORTLY AFTER CONTROL-ROD INSERTION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$2.00 COPY

CONTROL SYSTEM + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + CONTROL RODS + PUMPS

17-014676 ALSO IN CATEGORY 5 DATED 1966
NUCLEATE BOILING SHOWN TO HAVE SEVERAL REGIMES - LITERATURE SURVEY

LOS ALAMOS SCIENTIFIC LABORATORY
LA-3625-MS +. 1 PAGE, QUARTERLY STATUS REPORT ON ADVANCED REACTOR TECHNOLOGY (ART) FOR PERIOD ENDING OCTOBER 31, 1966, PAGE 13, NOVEMBER 1966

A FOUR-MONTH SURVEY OF THE LITERATURE ON BOILING HEAT TRANSFER AND TWO-PHASE FLOW WAS COMPLETED. THE MORE RECENT PUBLICATIONS REVEALED THAT RESEARCHERS HAVE TURNED TO A STUDY OF THE BASIC MECHANISM OF BOILING. THESE STUDIES HAVE SHOWN THAT THE CONVENTIONAL CONCEPT OF NUCLEATE BOILING UPON WHICH ALL PRIOR CORRELATIONS WERE BASED IS NOT CORRECT. THREE AND POSSIBLY FOUR REGIMES OF NUCLEATE BOILING HAVE BEEN DEMONSTRATED, WITH ONLY THE LOW-HEAT-FLUX RANGE CORRESPONDING TO THE IDEAS HELD PREVIOUSLY ABOUT NUCLEATE BOILING. THE NEW VIEWS AND THEIR CONSEQUENCES HAVE BEEN EXPLORED - A SUMMARY OF THE LITERATURE AND SPECIFIC RECOMMENDATIONS FOR FUTURE WORK HAVE BEEN PREPARED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICRONEGATIVE

HEAT TRANSFER, BOILING + REACTOR, LMCR + SODIUM + HEAT TRANSFER EXPERIMENT + NUCLEATE BOILING

17-014787 ALSO IN CATEGORY 8 DATED 1966

DOUGLAS RE
EFFECTS OF WATER LEAKAGE INTO TANKS CONTAINING SODIUM
ATOMIC INTERNATIONAL, CANOGA PARK
NAA-SR-MEMO-12239 +. 14 PAGES, NOVEMBER 10, 1966

ONE METHOD FOR DISPOSING OF THE HALLAM PRIMARY SODIUM IS TO BURY THE STORAGE TANKS WITHOUT PRIOR REACTION OF THE SODIUM. A TEST WAS PERFORMED TO DETERMINE THE EFFECTS OF GROUND WATER LEAKAGE INTO THE TANKS THROUGH PINHOLES OR CRACKS. A HALF QUART CAN WAS SUBMERGED AND VARIOUS SIZED HOLES DRILLED. RESULTS INDICATE THAT THE SODIUM-WATER REACTION WOULD TAKE PLACE AT A SELF-REGULATING RATE, AND NO EXCESSIVE INTERNAL PRESSURE INCREASE OR EXPLOSIVE CONDITION WOULD BE CREATED IN THE TANKS UNDER CONDITIONS SIMILAR TO THOSE IMPOSED FOR THE TEST.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

EXPLOSION + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + METAL WATER REACTION + SODIUM + WASTE DISPOSAL, TERRESTRIAL + DECOMMISSIONING

17-014804 ALSO IN CATEGORY 18 DATED 1966
ENRICO FERMI REACTOR- USE FOR IRRADIATION TESTING AND BACKGROUND INFORMATION

JOINT COMMITTEE ON ATOMIC ENERGY
270 PAGES, FIGURES, TABLES, APRIL 5, 1966, U.S. GOVERNMENT PRINTING OFFICE, 1966

37 PAGES OF TWO HOURS OF ORAL TESTIMONY. THE REST CONSISTS OF LETTERS, SUMMARIES OF FINANCES AND COSTS, ACKS, ORL, LEGAL OPINIONS, ETC. CONTENT IS A COMPLETE SUMMARY OF FERMI-REACTOR ADMINISTRATIVE HISTORY TO THE MELTDOWN OF OCT. 10, 1966. THE POINT WAS MADE THAT FAST-NEUTRON FUEL-CLADDING DAMAGE CAN BE OBTAINED QUICKER AND IN THE ACTUAL OPERATING CONDITIONS THROUGH FERMI USAGE.

AVAILABILITY - GOVERNMENT PRINTING OFFICE, WASHINGTON, D. C. 20402

REACTOR, BREEDER + CONGRESSIONAL ACTIVITY + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + IRRADIATION TESTING + FUEL ELEMENTS

17-014856 DATED 1965
HALLAM NUCLEAR POWER FACILITY MONTHLY OPERATING REPORT NO. 31, FEBRUARY 1965

CONSUMERS PUBLIC POWER DISTRICT, HALLAM
TID-21893 +. 79 PAGES, FEBRUARY 1965, DOCKET NO. 115-3

THIS REPORT IS ONE OF A SERIES. SINCE THE REACTOR IS BEING DISMANTLED, THE PERTINENT INFORMATION CONCERNS MAINTENANCE, CORE-ELEMENT TRANSFERS, RADIATION SAFETY AND CONTROL. SHORT SUMMARIES OF EACH DAYS WORK ARE GIVEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-014857

DATED 1965

HALLAM DIESEL CONTROLS PREVENT A START DURING TEST

CONSUMERS PUBLIC POWER DISTRICT, HALLAM

TID-21893 +. 1 PAGE FROM HALLAM NUCLEAR POWER FACILITY MONTHLY OPERATING REPORT NO. 31, FEBRUARY 1965, DOCKET NO. 115-3

A TEST OF THE DIESEL GENERATOR REVEALED THAT IT WOULD NOT START AUTOMATICALLY UPON POWER FAILURE AND THAT ITS BREAKER WOULD NOT CLOSE IN AFTER THE GENERATOR WAS STARTED MANUALLY. THE INTERLOCK CONTACTS ON BREAKER 52B-1 WERE BINDING ENOUGH TO PREVENT THE MAIN CONTACTS FROM OPENING FULLY. CONSEQUENTLY, THE DIESEL BREAKER WOULD NOT CLOSE TO PICK UP THE LOAD.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, COMPONENT + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + INSTRUMENT, INTERLOCK + GENERATOR, DIESEL

17-014858

DATED 1965

HALLAM NUCLEAR POWER FACILITY MONTHLY OPERATING REPORT NO. 32, MARCH 1965

CONSUMERS PUBLIC POWER DISTRICT, HALLAM, NEBR.

TID-21861 +. 77 PAGES, MARCH 1965, DOCKET NO. 115-3

THIS REPORT IS ONE OF A SERIES. SINCE THE REACTOR IS BEING DISMANTLED, THE PERTINENT INFORMATION CONCERNS MAINTENANCE, CORE-ELEMENT TRANSFERS, RADIATION SAFETY AND CONTROL. SHORT SUMMARIES OF EACH DAYS WORK ARE GIVEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

17-014859

DATED 1965

HALLAM NUCLEAR POWER FACILITY MONTHLY OPERATING REPORT NO. 35, JUNE 1965

CONSUMERS PUBLIC POWER DISTRICT, HALLAM

TID-22135 +. 52 PAGES, JUNE 1965, DOCKET NO. 115-3

THIS REPORT IS ONE OF A SERIES. SINCE THE REACTOR IS BEING DISMANTLED, THE PERTINENT INFORMATION CONCERNS MAINTENANCE, CORE-ELEMENT TRANSFERS, RADIATION SAFETY AND CONTROL. SHORT SUMMARIES OF EACH DAYS WORK ARE GIVEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

17-014917

ALSO IN CATEGORIES 5 AND 6

DATED 1967

DICKERMAN LE + ROBINSON LE + BLUMENTHAL B + STEWART R

BEHAVIOR OF TH-20 WT/PERCENT U FAST-REACTOR FUEL UNDER TRANSIENT HEATING TO FAILURE

ARGONNE NATIONAL LABORATORY

10 PAGES, 10 FIGURES, 2 TABLES, 17 REFERENCES, NUCLEAR APPLICATIONS 3(1) PAGES 9-17 (JANUARY 1967)

SODIUM-BUNDED TH-20 WT % U EBR II FUEL ELEMENTS WERE STUDIED IN TRANSIENT MELTDOWN EXPERIMENTS RANGING UP TO COMPLETE SAMPLE MELTDOWN, CONDUCTED IN TREAT. THE HIGH-MELTING THORIUM MATRIX SUPPORTED THE FUEL DURING TRANSIENT HEATING AT TEMPERATURES SIGNIFICANTLY BEYOND THE SOLIDUS. BECAUSE THE FUEL WAS STILL RELATIVELY INTACT AT THE TEMPERATURES OF CLADDING FAILURE, INTERNAL PRESSURE WAS NOT EFFECTIVE IN EJECTING FUEL UPON FAILURE. THE DOMINANT MECHANISM OF FUEL MOVEMENT WAS GRAVITY.

REACTOR, FAST + FUEL MELTDOWN + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + TREAT (PKR) + THERMAL TRANSIENT

17-016029

DATED 1964

PERRET JD

PERIOD SPIKES RELATED TO SRE OPERATING HISTORY

ATOMIC INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR MEMO-9523 +. 21 PAGES, FIGURES, REFERENCES, MARCH 27, 1964

SEVENTY-FOUR PERIOD SPIKES WERE RECORDED DURING SRE CORE-II OPERATION FROM OCTOBER 8, 1962, TO FEBRUARY 15, 1964. THE OCCURENCE, FREQUENCY, AND MAGNITUDE OF THESE EVENTS AS RELATED TO SRE OPERATIONS IS DOCUMENTED. THE OBSERVATIONS AND REASONING WHICH SUPPORT GAS-BUBBLE PASSAGE THROUGH THE CORE AS THE MOST PROBABLE CAUSE ARE PRESENTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + REACTIVITY EFFECT + SRE (RE) + REACTIVITY EFFECT, ANOMALOUS + REACTIVITY EFFECT, VOID

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-016032 DATED 1966
ADAMS RM + GLASSNER A
REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, NOVEMBER 1966
ARGONNE NATIONAL LABORATORY, ILL.
ANL-7279 +. 98 PAGES, (PAGES 86-95.) 26 FIGURES, 20 TABLES, DECEMBER 21, 1966

DISCUSSES LIQUID SUPERHEAT TESTS WITH SODIUM, FUEL-MELTDOWN STUDIES, TRANSIENT HEAT TRANSFER FROM SMALL HOT PARTICLES IN LIQUID SODIUM, AND HIGH-TEMPERATURE PHYSICAL PROPERTIES OF REACTOR CORE MATERIALS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

FUEL MELTDOWN + HEAT TRANSFER + LIQUID FUEL + REACTOR, LMCR + SODIUM + THERMAL ANALYSIS + HIGH TEMPERATURE + PROPERTY, PHYSICAL + COKE COMPONENTS + FUEL ELEMENTS

17-016033 DATED 1965
QUINN EP
VIBRATION OF SEFOR FUEL RODS IN PARALLEL FLOW
GENERAL ELECTRIC CO., SAN JOSE, CALIF. ATOMIC POWER EQUIPMENT DEPT.
GEAP-4966 +. 35 PAGES, TABLES, SEPT. 1965

TESTS WERE RUN WITH A FULL-SCALE SEFOR FUEL ROD ASSEMBLY TO DETERMINE THE ROD VIBRATION RESULTING FROM FLUID FLOW. THE MAXIMUM VIBRATION FOR A NORMAL ROD ASSEMBLY CORRESPONDING TO REACTOR CONDITIONS WAS FOUND TO BE 0.0001 IN. AT 67 CPS. LARGER VIBRATION AT LOWER FREQUENCIES WAS MEASURED FOR TWO OTHER ASSEMBLIES WITH A REDUCED NUMBER OF AXIAL SUPPORTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + HYDRAULIC EFFECT + REACTOR, LMCR + MEASUREMENT + SODIUM + SEFOR (RE) + VIBRATION + MAIN COOLING SYSTEM + FUEL ELEMENTS

17-016131 DATED 1964
STEARNS JD
CARBURIZATION POTENTIAL DETERMINATION FOR HNPf HOT TRAP SODIUM SAMPLES NOS. 18, 19, AND 20
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA
NAA-SR-MEMU-10123 +. 14 PAGES, JUNE 24, 1964

THE PROCEDURE AND RESULTS ARE GIVEN FOR DETERMINING THE CARBURIZATION POTENTIAL OF HNPf HOT-TRAP SODIUM AND THE EXAMINATION OF SODIUM FOR EVIDENCE OF GRAPHITE RELEASED FROM FAILED MODERATOR CANS. THE ANALYSIS OF SODIUM FOR DETECTING MODERATOR-CAN FAILURE IS NOT EFFECTIVE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

CARBON + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + SODIUM + CHEMICAL ANALYSIS + COOLANT CHEMISTRY + METAL, LIQUID + EXAMINATION + COOLANT QUALITY

17-016148 DATED 1967
NOVICK M + BACCA JP + CERUTTI BC + CISSEL DW
ROTATING PLUG SEALS
ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO
3 PAGES, 1 FIGURE, 2 TABLES, PAGES 7-9 FROM EBR-II OPERATING EXPERIENCE PRESENTED AT THE AMERICAN NUCLEAR SOCIETY MEETING, APRIL 10-12, 1967

THE SEALS CONTAIN BISMUTH-TIN-ALLOY, OF WHICH THE UPPER HALF IS FROZEN AND THE LOWER HALF IS MOLTEN. THE SEAL ALLOY IS MELTED COMPLETELY TO PERMIT ROTATION. THE TIME REQUIRED TO COMPLETE THE MELTING INCREASED FROM 6 HOURS TO 24, AND THE PLUGS WOULD BIND DURING ROTATION. INVESTIGATION REVEALED SIGNIFICANT QUANTITIES OF METALLIC OXIDES AS THE RESULT OF AIR EXPOSURE. REMOVAL OF THE OXIDIZED MATERIAL AND REPLACEMENT WITH NEW ALLOY CORRECTED THE PROBLEM.

AVAILABILITY - M. NOVICK, J. P. BACCA, B. C. CERUTTI AND D. W. CISSEL, ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO

REACTOR, BREEDER + EBR 1 AND 2 (RE) + FAILURE, EQUIPMENT + REACTOR, FAST + REACTOR, LMCR + MAINTENANCE AND REPAIR + OPERATING EXPERIENCE SUMMARY + FUEL HANDLING MACHINE + SHIELDING + SEAL + MAIN COOLING SYSTEM

17-016149 DATED 1967
NOVICK M + BACCA JP + CERUTTI BC + CISSEL DW
SECONDARY SODIUM SYSTEM
ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO
1 PAGE, 1 FIGURE, PAGE 4 FROM EBR-II OPERATING EXPERIENCE PRESENTED AT THE AMERICAN NUCLEAR SOCIETY MEETING, APRIL 10-12, 1967

THE CONTROL SYSTEM FOR TRACE HEATING THE SECONDARY SODIUM PIPING WAS MODIFIED TO USE SILICON-CONTROLLED RECTIFIERS IN PLACE OF MECHANICAL RELAYS. THE EXCESSIVE MAINTENANCE REQUIRED BY THE RELAYS WAS ELIMINATED.

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-016149 *CONTINUED*

AVAILABILITY - M. NOVICK, J. P. BACCA, B. C. CERUTTI AND D. W. CISSEL, ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + INSTRUMENT, PROCESS + REACTOR, LMCR + MAINTENANCE AND REPAIR + OPERATING EXPERIENCE SUMMARY + FAILURE, INSTRUMENT + RELAYS

17-016150

DATED 1967

NOVICK M + BACCA JP + CERUTTI BC + CISSEL DW

FUEL TRANSFER MECHANISM

ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO

1 PAGE, FROM PAGE 2 OF EBR-II OPERATING EXPERIENCE PRESENTED AT THE AMERICAN NUCLEAR SOCIETY MEETING, APRIL 10-12, 1967

INSPECTION OF THE FUEL-TRANSFER MECHANISM TO DETERMINE REASON FOR DIFFICULTIES WITH THE LOCKING-PIN ACTUATING ARM REVEALED THAT THE BELL CRANK HAD FAILED AND THAT AN ALUMINUM-BRONZE BUSHING WAS BADLY DETERIORATED. AN IMPROVED DESIGN ELIMINATED THE USE OF ALUMINUM-BRONZE. REMOVAL OF THE TRANSFER ARM FROM THE PRIMARY TANK INVOLVED RADIATION LEVELS OF THE ORDER OF 10 MR/HR.

AVAILABILITY - M. NOVICK, J. P. BACCA, B. C. CERUTTI AND D. W. CISSEL, ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO

REACTOR, BREEDER + EBR 1 AND 2 (RE) + FAILURE, EQUIPMENT + REACTOR, FAST + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + RADIATION SAFETY AND CONTROL + FAILURE, DESIGN ERROR + FUEL HANDLING MACHINE

17-016151

DATED 1967

NOVICK M + BACCA JP + CERUTTI BC + CISSEL DW

EBR-II OPERATING EXPERIENCE

ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO

20 PAGES, 10 FIGURES, 3 TABLES, 11 REFERENCES, EBR-II OPERATING EXPERIENCE PRESENTED AT THE AMERICAN NUCLEAR SOCIETY MEETING, APRIL 10-12, 1967

REPROCESSING AND REFABRICATION IN THE ADJACENT FUEL-CYCLE FACILITY OF ALL DRIVER FUEL WAS HIGHLY SUCCESSFUL. NO FAILURE OCCURRED WITH ANY OF THE 24,000 FUEL ELEMENTS THAT HAVE UNDERGONE IRRADIATION, OR WITH ANY MAJOR REACTOR COMPONENT. SOME OF THE FUEL IN THE REACTOR IS NOW IN ITS THIRD CYCLE OF OPERATION THROUGH THE FUEL-CYCLE FACILITY AND REACTOR. PERFORMANCE OF THE DRIVER FUEL WAS LESS THAN EXPECTED, RESULTING IN A LARGER REQUIREMENT FOR FUEL AND INITIATING AN IMPROVED DRIVER-FUEL DEVELOPMENT PROGRAM.

AVAILABILITY - M. NOVICK, J. P. BACCA, B. C. CERUTTI AND D. W. CISSEL, ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + FUEL BURNUP + FUEL INTEGRITY + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + FUEL REPROCESSING

17-016408

DATED 1967

FERMI CHANGE 12 TO ALLOW SODIUM DRAINING FOR REACTOR VESSEL INSPECTION.

UNITED STATES ATOMIC ENERGY COMMISSION, WASHINGTON, D. C.

6 PAGES, JUNE 16, 1967, DCKET NO. 50-16, CHANGE 12 TO OPERATING LICENSE

FERMI GRANTED FIVE SPECIFIC TECHNICAL SPECIFICATION CHANGES TO ALLOW SODIUM DRAINING FOR REACTOR-VESSEL INSPECTION.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + PRESSURE VESSELS + TECHNICAL SPECIFICATIONS + INCIDENT, RECOVERY FROM + EXAMINATION + SAFETY EVALUATION

17-016456

DATED 1966

DISCHINGER RH

PERFORMANCE EVALUATION OF HEAT EXCHANGERS FOR SODIUM-COOLED REACTORS. QUARTERLY TECHNICAL REPORT, OCTOBER 1, 1965-DECEMBER 31, 1965

UNITED NUCLEAR CORP., ELMSFORD, N. Y.

UNC-5142 +. 32 PAGES, TABLES, JANUARY 20, 1966

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS DISCUSSED ARE THE THERMAL-HYDRAULIC PERFORMANCE AND STRUCTURAL-PERFORMANCE EVALUATION OF HEAT EXCHANGERS FOR SODIUM-COOLED REACTORS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REACTOR, POWER + R AND D PROGRAM + HEAT EXCHANGERS

17-016534

ALSO IN CATEGORY 18

DATED 1967

ADAMS RM + GLASSNER A

EBR II POWER COEFFICIENT, AARR LOSS OF FLOW STUDIES

ARGONNE NATIONAL LAB., ARGONNE, ILLINOIS

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-016534 *CONTINUED*

ANL-7329 +. 103 PAGES, 28 FIGURES, 29 TABLES, 23 REFERENCES, FROM REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, APRIL 1967.

(PAGE 2-4) EBR-II STARTUP WITH STAINLESS-STEEL ASSEMBLIES REPLACING SOME DEPLETED URANIUM SHOWED A DECREASING POWER COEFFICIENT, BUT TESTS SHOWED THAT IT INCREASED ABOVE HALF POWER. (PAGE 75-82) AARR PRIMARY COOLANT SYSTEM BURSTING PRESSURE/STRAIN-ENERGY ARE CALCULATED AS 5000 PSI/524 MW SEC, AND THE VESSEL IS STRONGER. A 163-MW SEC NUCLEAR EXCURSION FOLLOWED BY A 1300-MW SEC METAL-WATER REACTION, WOULD GIVE A 4270-PSI VESSEL PRESSURE THAT EQUALIZES AT 3300 PSI. FLOW-COASTDOWN ANALOG-COMPUTER STUDY SHOWS PCNY MOTOR FLOW IS NECESSARY TO PREVENT HIGH FUEL-ELEMENT TEMPERATURES AT END OF LIFE. HOOKED-END BEAM-HOLE SEAL RINGS WERE ACCEPTABLY LEAK-RATE TESTED AT 55 PSI DELTA P.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + EBR 1 AND 2 (KE) + REACTOR, FAST + REACTOR, LMR + ACCIDENT, LOSS OF FLOW + METAL WATER REACTION + POWER COEFFICIENT + PRESSURE VESSELS + TEST, LEAK RATE + AARR (RR) + REACTOR, ERDA OWNED + BEAM HOLE + SEAL + REACTIVITY EFFECT, ANOMALOUS + REACTOR, FLUX TRAP + PERFORMANCE LIMIT

17-016765

DATED 1967

HALLAM CHANGE 5 - FACILITY RETIREMENT-PLAN

UNITED STATES ATOMIC ENERGY COMMISSION, WASHINGTON, D. C.
5 PAGES, FEBRUARY 8, 1967, DOCKET NO. 115-3

PROVIDES FOR DISMANTLING HALLAM, BASED ON NAA-SR-MEMO 12209 (RETIREMENT PLAN).

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMR + TECHNICAL SPECIFICATIONS + DECOMMISSIONING

17-016766

DATED 1966

HALLAM NUCLEAR POWER FACILITY RETIREMENT PLANT REVISION 4 (OCTOBER 10, 1966)

CONSUMERS PUBLIC POWER DISTRICT, COLUMBUS, NEBRASKA
NAA-SR-MEMO-12209 +. 36 PAGES, OCTOBER 28, 1966, DOCKET NO. 115-3

PROVIDES CRITERIA (DISPOSAL AND DECONTAMINATION) FOR VARIOUS SYSTEMS AND COMPONENTS, AND ASSIGNS WORK RESPONSIBILITIES TO UTILITY AND MANUFACTURER.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C. OR -- CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICRONEGATIVE

ADMINISTRATIVE CONTROL + DECONTAMINATION + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMR + DECOMMISSIONING

17-017093

DATED 1962

DEBEAK WS

CONSIDERATIONS RELATING TO OPERATION OF THE SRE WITH A DAMAGED REFLECTOR ELEMENT

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-MEMO-7152 +. 33 PAGES, 2 TABLES, MARCH 13, 1962

PROVIDES INFORMATION OF POSSIBLE EFFECTS OF A DAMAGED REFLECTOR CAN. EFFECTS ON NUCLEAR AND HEAT TRANSFER CHARACTERISTICS, AND MATERIAL PROPERTIES ARE CONSIDERED, PLUS CALCULATIONS AS TO THE EXPECTED CARBON INPUT TO THE PRIMARY SODIUM AND REMOVAL CAPABILITY OF THE HOT TRAP. EXPERIMENTAL DATA INDICATIVE OF BENEFICIAL PERFORMANCE FROM THE GOLD TRAP ARE INCLUDED, AS ARE OTHER OBSERVATIONS ON SYSTEM CLEANUP SINCE THE ADMISSION OF HYDROCARBON TO THE PRIMARY SODIUM IN JUNE 1959. INFORMATION PROVIDED ON SERVICE LIFE OF THE HOT TRAP UNDER SEVERAL OPERATING CIRCUMSTANCES AND ON MINIMUM CARBON CONCENTRATION WHICH CAN BE ACHIEVED WITH VARIOUS VALUES OF CARBON INPUT. TECHNIQUES FOR EARLY DETECTION OF FUTURE DAMAGE OR CHANGING SYSTEM CONDITIONS ARE SPECIFIED. (CONCLUSION) - EXISTING SODIUM PURIFICATION EQUIPMENT IS ADEQUATE, CONTINUED POWER OPERATION IS FEASIBLE AND TECHNICALLY SOUND.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, COMPONENT + REACTOR, GRAPHITE MODERATED + REACTOR, LMR + SRE (RE) + COOLANT CHEMISTRY + REPORT, OPERATIONS ANALYSIS + REFLECTOR + COOLANT PURIFICATION SYSTEM + REACTOR PHYSICS + FUEL COOLANT INTERACTION

17-017102

ALSO IN CATEGORY 12

DATED 1966

HUTTER E + ELIAS P + VEITH DJ + WARE WK

EBR-II FUEL-UNLOADING MACHINE--DESIGN AND PERFORMANCE CHARACTERISTICS

ARGONNE NATIONAL LAB., ILL.
ANL-7201 +. 91 PAGES, 42 FIGURES, 6 REFERENCES, JUNE 1966

DESCRIBES THE FUEL-HANDLING SYSTEM, THE MACHINE, AND ITS OPERATION FOR THE SODIUM-COOLED REACTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS,

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-017102 *CONTINUED*

U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + OPERATION + FUEL HANDLING MACHINE + EQUIPMENT DESIGN

17-017131

ALSO IN CATEGORY 6

DATED 1967

WOODRUFF RW

EFFECT OF DISSOLVED HELIUM IN SRE

ATOMICS INTERNATIONAL, SAN DIEGO, CALIFORNIA

CONF-670-713 +. 2 PAGES, 4 REFERENCES, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 10, PAGE 5 AND 6, (JULY 1967), CONFERENCE ON REACTOR OPERATING EXPERIENCE, REACTOR OPERATIONS DIVISION OF THE AMERICAN NUCLEAR SOCIETY, ATLANTIC CITY, JULY 23-26, 1967

SOLUBILITY OF INERT COVER GASES (HE OR AR) WILL BE A DESIGN PROBLEM IN LIQUID-METAL-COOLED REACTORS. SRE TRANSIENTS (20 F AND 25% ON NEUTRON FLUX) OCCURRED ABOUT DAILY AND LASTED 1 SEC - THE CORE TRANSIENT TIME. COVER-GAS BUBBLES WERE THOUGHT TO ACCUMULATE AT A HEAT-EXCHANGER OUTLET UNTIL ENOUGH VOLUME CHOKED FLOW AND SWEEPED OUT A LARGE BUBBLE. SMALL BUBBLES WERE NOT TRAPPED BUT PASSED CONTINUOUSLY THROUGH THE CORE, PROBABLY ACCOUNTING FOR THE NOISE-SPECTRUM DEVIATION FROM A WHITE SPECTRUM. BLEED LINES AT TRAPS, AND DE-ENTRAINMENT AT REACTOR INLETS SHOULD BE USED. ARGON WOULD HAVE SIGNIFICANTLY REDUCED THE VOLUME OF THE GAS TRANSPORTED.

REACTOR, LMCR + NOISE + SRE (RE) + HELIUM + REACTOR, ERDA OWNED + REACTIVITY EFFECT, ANOMALOUS + COOLANT QUALITY

17-017164

DATED 1967

WHITHAM GK + SPALDING TR

EBR-II REFUELING EXPERIENCE

ARGONNE NATIONAL LAB., ARGONNE, ILLINOIS

CONF-670-713 +. 1 PAGE, ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 10, PAGE 32, (JULY 1967), CONFERENCE ON REACTOR OPERATING EXPERIENCE, REACTOR OPERATIONS DIVISION OF THE AMERICAN NUCLEAR SOCIETY, ATLANTIC CITY, JULY 23-26, 1967

DISCUSSES GENERAL EXPERIENCE. THE GRIPPER NEEDS FREQUENT MAINTENANCE AND CLEANING. REMOTE OPERATION OF THE MECHANISM IS BY A DIGITAL COMPUTER AND PUNCHED CARDS.

COMPUTER PROGRAM + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + REACTOR, ERDA OWNED + FUEL HANDLING MACHINE

17-017167

ALSO IN CATEGORY 5

DATED 1967

FRIEDLAND AJ

A STUDY OF THE PROBABILITY OF A FERMI SUBASSEMBLY HAVING OVERHEATED BECAUSE OF A MIGRATORY PLUG

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN

CONF-670-713 +. 17 PAGES, 8 FIGURES, 1 TABLE, 5 REFERENCES, ABSTRACT IN ANS TRANSACTIONS, SUPPLEMENT TO VOLUME 10, PAGE 33, (JULY 1967), PAPER PRESENTED AT THE CONFERENCE ON REACTOR OPERATING EXPERIENCE, REACTOR OPERATIONS DIVISION OF THE AMERICAN NUCLEAR SOCIETY, ATLANTIC CITY, JULY 23-26, 1967

REACTOR OPERATING HISTORY WAS USED TO ESTIMATE THE PROBABILITY DISTRIBUTION OF PLUG SIZE, AND THESE TO DETERMINE PROBABILITY OF OVERHEATING. THIS APPROACH COULD BE GENERALLY APPLIED. SMALL-DIAMETER PLUGS ARE UNLIKELY TO CAUSE MELTDOWN, WHILE LARGE PLUGS WOULD NOT HAVE GONE THROUGH 11 OPERATING PERIODS WITHOUT CAUSING A MELTDOWN. THE MOST PROBABLE PLUG DIAMETER IS 1.5 TIMES THE GRID-PLATE-HOLE DIAMETER AND GIVES A 0.045 PROBABILITY OF OVERHEATING. THE OVERALL PROBABILITY OF OVERHEATING WAS CALCULATED AS 15%.

REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + ANALYTICAL MODEL + ACCIDENT, PROBABILITY OF + COOLANT CHEMISTRY + FLOW BLOCKAGE + MATHEMATICAL TREATMENT + FUEL COOLANT INTERACTION

17-017395

DATED 1965

CHALOT A + GINIER R + SAUVAGE M

FIRST EXPERIMENTAL RESULTS ON THE ENGINEERING MODEL OF RAPSODIE

COMMISSARIAT A L-ENERGIE ATOMIQUE, CADARACHE, FRANCE. CENTRE D-ETUDES NUCLEAIRES

EURFNR-49F + CEA-R-2557 +. 109 PAGES, FIGURES, JUNE, 1964

DESCRIBES EXPERIMENTS ON ENGINEERING MODEL AND ON SODIUM FACILITY IN HR-1 LABORATORY HALL, SUCH AS - (1) DIFFICULTIES ENCOUNTERED DURING ERECTION AND ASSEMBLY OF THE MODEL, INCLUDING PROBLEMS WITH NUCLEAR CLEANNESS, INTERVENTIONS INSIDE LEAKPROOF VESSEL AFTER ACCIDENTAL SODIUM ENTRY, MINOR DIFFICULTIES INVOLVING BASE PLATES AND ROTATING PLUGS, RESULTS OF FIRST EXPERIMENTS (LOADING OF SUBASSEMBLIES, PREHEATING, THERMAL SHOCKS, ETC), (2) TESTS ON TWO PHOTOTYPE CONTROL-ROD-DRIVE MECHANISMS AT THE SODIUM FACILITY, WHICH LED TO DESIGN OF DEFINITIVE DRIVE MECHANISM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

FRANCE + REACTOR, LMCR + MOCKUP + OUT OF PILE EXPERIMENT + OPERATING EXPERIENCE SUMMARY

17-017479

ALSO IN CATEGORY 9

DATED 1966

SMITH RK + DOE CB

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-017479 *CONTINUED*
CLADDING FAILURE SIMULATION TESTS IN EBR-II
ARGONNE NATIONAL LAB., ILL.
ANL-7067 +. 24 PAGES, 17 FIGURES, 1 TABLE, REFERENCES, DECEMBER 1966

TESTS, USING A SINGLE UNCLAD FUEL ROD, ON (1) CWD (CHANGED WIRE DETECTOR, BASED ON DETECTION OF FISSION GAS IN THE CORE COVER GAS), AND (2) FERD (FUEL ELEMENT RUPTURE DETECTOR), BASED ON DETECTION OF DELAYED-NEUTRON-EMITTING FISSION PRODUCTS IN PRIMARY COOLANT. COUNTING RATES WERE MEASURED WITH NORMAL CONDITIONS AND WITH A SINGLE UNCLAD MARK-1A FUEL PIN IN THE CORE. SIGNAL-TO-NOISE RATIOS ARE 12 FOR CWD AND 6.7 FOR FERD. FERD HAS A MUCH SHORTER TIME RESPONSE (20 SEC) BUT WILL NOT USUALLY RESPOND TO FAILURES IN THE FUEL-ROD GAS PLENUM. SINCE CWD IS SENSITIVE TO BOTH BOND-TYPE AND GAS-TYPE FAILURES, THE SYSTEMS COMPLEMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + REACTOR, LMCR + TEST, PROOF + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + REPORT, OPERATIONS ANALYSIS + REACTOR, ERDA OWNED

17-017500 ALSO IN CATEGORY 18 DATED 1967
ANOMALOUS SWELLING OF EBR-II MARK 1A FUEL
ARGONNE NATIONAL LAB., ARGONNE, ILL.
ANL-7317 +. 5 PAGES, 1 TABLE, PAGES 8-12 FROM REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MARCH 1967

WHILE EARLIER 48% ENRICHED FUEL SWELLED 6-7% AT 1.2 A/O FUEL BURNUP, THE 1A 52% ENRICHED FUEL IS UNEXPECTEDLY SWELLING TWICE THAT AMOUNT, UNLESS THE FUEL CONTAINS REMELTED REJECT PINS AND PIN ENDS. VARIOUS MEASUREMENTS TAKEN, AND IRRADIATION STUDIES BEGUN IN NOV. 1966 ARE SUMMARIZED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + EBR 1 AND 2 (RE) + FUEL BURNUP + REACTOR, LMCR + REACTOR, ERDA OWNED + EXPANSION + FUEL ELEMENTS

17-018032 DATED 1967
FERMI FLOW RESTRICTION FOUND
2 PAGES, NUCLEONICS WEEK 8(38), (SEPTEMBER 21, 1967)

A BORESCOPE INSERTED THRU THE GRID PLATE INTO THE DRAINED REACTOR VESSEL SHOWED A FLAT OBJECT (8 X 4 IN.) LYING ON THE MELTDOWN PLATE. IT WAS NOT RECOGNIZED AS A REACTOR COMPONENT AND POSSIBLY LEFT BY MISTAKE DURING CONSTRUCTION. THE OBJECT MUST BE COMPRESSED OR CUT UP TO PASS THRU THE 1.625-IN.-DIAM GRID HOLES, OR MUST BE TAKEN THROUGH A TORTUROUS 50-FT COOLANT INLET PATH. A FULL-SCALE MODEL IS BEING BUILT TO AID IN RETRIEVAL.

FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + INCIDENT + REMOTE MANIPULATING AND VIEWING + FLOW BLOCKAGE + FAILURE, INSTALLATION ERROR

17-018595 DATED 1965
CRCWN PL
SODIUM PUMP RELIABILITY DEMONSTRATION
ATOMICS INTERNATIONAL, CANOGA PARK
NAA-SR-MEMO-11485 +. 22 PAGES, REFERENCES, JULY 1, 1965

THE DESIGN, CONSTRUCTION, AND OPERATION OF LARGE SODIUM-COOLED REACTORS REQUIRE INCREASED RELIABILITY AND CONFIDENCE IN THE ABILITY OF SODIUM PUMPS TO INSTALL AND OPERATE WITHOUT EXCESSIVE DEVELOPMENT AND SHAKEDOWN TIME. EXPERIENCE AT HNPf INDICATES THAT THE DEVELOPMENT OF SUCH PUMPS IS STILL IN ITS EARLY STAGES. OPERATING HISTORY IS DISCUSSED, AND, CONTINUED OPERATION, AS A MEANS OF COMPLETING THE DEVELOPMENT CYCLE AND INCREASING RELIABILITY DEMONSTRATION, IS EVALUATED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + SODIUM + RELIABILITY, COMPONENT + PUMPS

17-018606 DATED 1967
PECKINPAUGH CL
SRE-PEP PUMP PERFORMANCE
ATOMICS INTERNATIONAL, CANOGA PARK
NAA-SR-12409 +. 41 PAGES, 16 FIGURES, 2 TABLES, 9 REFERENCES, JUNE 6, 1967

FREE-SURFACE CENTRIFUGAL PUMPS (2 MAIN-SODIUM-SYSTEM AND 2 AUXILIARY-SYSTEM PUMPS) WERE INSTALLED IN PLACE OF THE ORIGINAL FREEZE-SEAL PUMPS, WHICH HAD ENTRAINED GAS IN COOLANT AND HAD EXPERIENCED OIL LEAKS. NEW PUMPS, BASED ON HALLAM EXPERIENCE, ARE ALSO MORE RELIABLE. DESCRIBES PUMPS IN DETAIL AND MODIFICATIONS MADE TO LIMIT OIL LEAKS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-018606 *CONTINUED*
REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + SRE (KE) + COOLANT CHEMISTRY + MODIFICATION + RELIABILITY,
SYSTEM + TEST, PREOPERATIONAL + PUMPS

17-018681 DATED 1965
SMITH FA
HALLAM OPERATION IN RETROSPECT
11 PAGES, FIGURES, TABLES, REFERENCES, POWER REACTOR TECHNOLOGY 8(4) PAGES 279-89 (FALL 1965)

REVIEWS THE NORMAL OPERATION, DIFFICULT COMPONENT PROBLEMS, AND UNRESOLVED PROBLEMS. DIFFICULT PROBLEMS DISCUSSED INCLUDE THOSE FROM NUCLEAR INSTRUMENTATION, PLANT PROTECTIVE INSTRUMENTS, CONTROL-ROD THIMBLES, THROTTLE VALVES, HEAT EXCHANGERS, STEAM GENERATORS, AND GAS ENTRAINMENT IN SODIUM EXPANSION TANK. THE UNRESOLVED PROBLEMS INCLUDE CRACKS IN SODIUM EXPANSION TANK, PLUGGING OF OXIDE COLD TRAPS, UNDERSTANDING AND MEASURING CHEMICAL EFFECTS OF CARBON IN SODIUM, VENTING AND FILLING INERT GAS FOR COVER-GAS SYSTEM, FUEL-CHANNEL VARIABLE-COOLANT-FLOW ORIFICES, AND MODERATOR CLADDING LEAKAGE.

FAILURE, EQUIPMENT + FAILURE + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + PLANT PROTECTIVE SYSTEM + FAILURE, INSTRUMENT + FAILURE, FABRICATION ERROR + FAILURE, DESIGN ERROR + FAILURE, CLADDING + COOLANT QUALITY + VALVES + HEAT EXCHANGERS + CONTROL ROD DRIVE MECHANISMS

17-018682 DATED 1966
KELLER LE + DAVIDSON RH
HALLAM NUCLEAR POWER FACILITY OPERATING EXPERIENCE
CONSUMERS PUBLIC POWER DISTRICT + TENNESSEE VALLEY AUTHORITY
7 PAGES, FIGURES, TABLES, NUCLEAR SAFETY 7(2) PAGES 236-42 (WINTER 65-66)

BRIEFLY DESCRIBES THE FACILITY AND OUTLINES THE OPERATING HISTORY. TUBE VIBRATION IN THE HEAT EXCHANGERS RESULTED IN ONE TUBE BREAKING. PROBLEMS WITH THE MAIN SODIUM CIRCULATING PUMPS INCLUDED FREEZE-UP BY SODIUM VIBRATION DUE TO IMPROPER SHIMMING OF THRUST BEARING, BINDING OF THE PUMPS BECAUSE OF WARPAGE DUE TO UNEVEN TEMPERATURE AND CLOGGING FROM DEBRIS LEFT IN THE SYSTEM DURING FABRICATION, AND SEAL LEAKAGE CAUSED BY IMPROPER OPERATION. CLADDING FAILURE ON THE GRAPHITE MODERATOR ELEMENTS RESULTING IN SWELLING UP TO 5% DUE TO ABSORPTION OF SODIUM. SPACING IN THE CORE WAS DESIGNED FOR ONLY 2% EXPANSION. THIS CLADDING FAILURE RESULTED IN THE FACILITY SHUTDOWN.

FAILURE, EQUIPMENT + FAILURE + GRAPHITE + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + FAILURE, OPERATOR ERROR + REACTOR, LMCR + SODIUM + REACTOR, POWER + OPERATING EXPERIENCE SUMMARY + MODERATOR + FAILURE, DESIGN ERROR + FAILURE, CLADDING + FAILURE, ADMINISTRATIVE CONTROL + VIBRATION + FAILURE, INSTALLATION ERROR + MAIN COOLING SYSTEM + FAILURE, TUBING + HEAT EXCHANGERS

17-018857 ALSO IN CATEGORY 5 DATED 1967
PHILLIPS JL
DOUNREAY FAST REACTOR MODIFICATIONS
UNITED KINGDOM ATOMIC ENERGY AUTHORITY, DOUNREAY
4 PAGES, 6 FIGURES, NUCLEAR ENGINEERING 12(128) PAGES 40-43 (OCTOBER 18, 1967)

MODIFICATIONS WERE NECESSARY BECAUSE OF COOLANT-FLOW INSTABILITIES IN THE OUTER BREEDER ELEMENTS AS A RESULT OF OPPOSING CONVECTIVE AND BUOYANCY-INDUCED FLOWS. THERMAL EFFECTS CAUSED EXCESSIVE FUEL SWELLING IN 1/3 OF THE OUTER BREEDER ELEMENTS TO SUCH AN EXTENT THAT MAJOR MODIFICATIONS WERE NECESSARY TO DISCHARGE MOST OF THEM. THE MODIFICATIONS, TOOLING, AND METHODS ARE DESCRIBED.

REACTOR, BREEDER + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR, LMCR + MAINTENANCE AND REPAIR + UNITED KINGDOM + MODIFICATION + FAILURE, DESIGN ERROR + DOUNREAY (LMFBR) + FLOW STABILITY + MAINTENANCE, REMOTE

17-018946 ALSO IN CATEGORY 6 DATED 1967
KLICKMAN AE + CALLEN RC
ANOMALOUS REACTIVITY EFFECTS IN THE FERMI INCIDENT
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + POWER REACTOR DEVELOPMENT COMPANY
CONF-670 602 +. 1 PAGE, ANS TRANSACTIONS 10(1) PAGE 334, 1967 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, SAN DIEGO, CALIF., JUNE 11-15, 1967

REVIEWS THE REACTIVITY EFFECTS THAT OCCURRED DURING THE FUEL MELTDOWN OF OCT. 1966. WHEN REACTOR POWER REACHED 5.5 MW, REACTIVITY INSERTION WAS AT THE RATE OF 0.185 CENT/MIN FOR 27 MIN UNTIL POWER REACHED 18 MW. THIS WAS ATTRIBUTED TO OVERHEATING OF THE CORE. AT 18 MW, THE RATE ABRUPTLY CHANGED TO 1.17 CENTS/MIN FOR 18 MIN UNTIL THE POWER WAS 30 MW. SUPERIMPOSED ON THIS LATTER TREND WERE THREE TRANSIENTS. THIS TREND AND THE TRANSIENTS WERE ATTRIBUTED TO FUEL MOTION. THE TOTAL REACTIVITY LOSSSES WERE ATTRIBUTED TO FUEL MOTION. THE TOTAL REACTIVITY LOSS DETERMINED BY THE SHIM-AND REGULATING-ROD HISTORIES WAS 26 CENTS.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + REACTOR, LMCR + POWER COEFFICIENT + INCIDENT + REACTIVITY EFFECT, ANOMALOUS

17-018947 ALSO IN CATEGORY 7 DATED 1967
MUELLER RE + BEAUDRY RJ + FELDES JG
FISSION PRODUCT ACTIVITY ANALYSIS OF FERMI FUEL FAILURE

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-018947 *CONTINUED*

ATOMIC POWER DEVELOPMENT ASSOCIATED, INC.
CONF-670 602 +. 2 PAGES, 1 FIGURE, ANS TRANSACTIONS 10(1) PAGES 334-335, 1967 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, SAN DIEGO, CALIF., JUNE 11-15, 1967

DESCRIBES THE MEASUREMENTS, ANALYSIS TECHNIQUE, AND CONCLUSIONS FOR DETERMINING EXTENT OF FUEL FAILURE. MEASUREMENTS OF THE FISSION PRODUCT ACTIVITY IN THE COOLANT AND COVER GAS INDICATED 2% OF THE CORE MELTED, AND AGREED WITH THE REACTIVITY LOSS. OTHER OBSERVATIONS WERE - (1) NOBLE-GAS RELEASE WAS GREATER THAN THAT IN PREVIOUS EXPERIMENTS, 50% VS 10%, (2) PARTICULATE RELEASE WAS GREATER THAN EXPECTED, (3) CONSIDERABLE PLATEOUT OCCURRED, (4) SODIUM ANALYSIS IS LESS RELIABLE THAN COVER-GAS ANALYSIS.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FISSION PRODUCT RELEASE + FAILURE, FUEL ELEMENT + REACTOR, LMCR + ANALYTICAL TECHNIQUE + FISSION PRODUCT ACTIVITY, GROSS + INCIDENT + RADIOCHEMICAL ANALYSIS + MAIN COOLING SYSTEM

17-019481

DATED 1967

BUNTAIN RE + FRANKLIN HV + MARSH DD + COOPER LJ
HALLAM MONTHLY RETIREMENT REPORT
CONSUMERS PUBLIC POWER DISTRICT, HALLAM NUCLEAR POWER FACILITY, SHELDON STATION
17 PAGES, 2 FIGURES, 4 TABLES, JULY 1967, DOCKET 115-3

ONE OF A SERIES OF MONTHLY REPORTS DESCRIBING THE MOTHBALLING OF HALLAM. SECTIONS INCLUDE - (I) PLANT RETIREMENT HISTORY (ONE-PARAGRAPH DAILY-WORK SUMMARIES), (II) FUEL STATUS, (III) MAINTENANCE, (IV) DISASSEMBLY AND CANISTERING OF IRRADIATED FUEL, (V) RADIATION AND CONTAMINATION MONITORING.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

FUEL HANDLING + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

17-019555

ALSO IN CATEGORY 7

DATED 1962

HART RS
DISTRIBUTION OF FISSION PRODUCT CONTAMINATION IN THE SRE
ATOMICS INTERNATIONAL, A DIVISION OF NORTH AMERICAN AVIATION, INC., CANOGA PARK, CALIFORNIA
NAA-SR-6690 +. 27 PAGES, 4 FIGURES, 12 TABLES, 5 REFERENCES, MARCH 1, 1962

SUMMARIZES FISSION-PRODUCT RELEASE AND DISTRIBUTION DATA COLLECTED WHILE MODIFYING AND CLEANING THE REACTOR AFTER THE FUEL-MELTING INCIDENT IN JULY 1959 (SEE NAA-SR-4488). THE PRINCIPAL CONCLUSIONS DRAWN FROM THE DATA ARE - (1) VOLATILITY PREFERENCE IS INVOLVED IN THE RELEASE OF VARIOUS FISSION PRODUCTS. (2) ONLY XE AND KR WERE FOUND IN THE HELIUM COVER GAS. (3) THE IODINE RELEASE FRACTION WAS UNUSUALLY SMALL. (4) APPRECIABLE DEPOSITION OCCURRED THROUGHOUT THE PIPING SYSTEM, WITH MARKED SELECTIVITY. SR, CE, ZR-NB DEPOSITION WAS GREATER THAN CS OR I. (5) ALTHOUGH 5000 TO 10,000 CURIES WERE RELEASED, NO RADIOLOGICAL EMERGENCY OCCURRED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

FISSION GAS RELEASE + FISSION PRODUCT RELEASE + FISSION PRODUCT, IODINE + FISSION PRODUCT, VOLATILE + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + SURFACE CONTAMINATION + SRE (RE) + FISSION PRODUCT ACTIVITY, GROSS + INCIDENT + FAILURE, CLADDING + FISSION PRODUCT RETENTION + MAIN COOLING SYSTEM

17-019556

DATED 1967

CHAPELOT A + CHAUVIN M + DELISLE JP + DENIELOU G + GOLLION H + GRANDCOLAS J + REYNES L
SODIUM LEAKAGE DURING ARGON OVERPRESSURIZATION
FRANCE. COMMISSARIAT A L-ENERGIE ATOMIQUE
13 PAGES, 4 FIGURES, 4 REFERENCES, APRIL 1967, PAPER PRESENTED AT THE NATIONAL TOPICAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, APRIL 10-12, 1967, FAST REACTORS-OPERATING, UNDER CONSTRUCTION, PLANNED FOR CONSTRUCTION

DURING TESTING OF THE ARGON BLANKET SYSTEM, SODIUM LEAKED THROUGH A PLUNGING TUBE CAUSED BY AN OVERPRESSURE OF THE BLANKET. REVISIONS INCLUDED (1) EITHER SUPPRESSING THE PLUNGING TUBES OR BORING THEM TO EQUILIBRATE INTERNAL AND EXTERNAL PRESSURE, AND (2) THE MECHANICAL RELIEF VALVES WERE REPLACED BY A LARGER HYDRAULIC VALVE. RECOMMENDS AVOIDING USE OF PLUNGING TUBES IN SODIUM REACTORS. SUPPRESSION OF THE PLUNGING TUBES CAUSED LOSS OF COOLING CAPABILITY DURING TRANSFER OF A SUBASSEMBLY BETWEEN THE CORE AND THE FUEL-HANDLING MACHINE, AND CONSEQUENTLY THE SUBASSEMBLIES MUST BE KEPT IN THE VESSEL FOR HEAT DECAY BEFORE REMOVAL.

FRANCE + REACTOR, LMCR + SODIUM + MODIFICATION + TEST, PREOPERATIONAL + FAILURE, DESIGN ERROR + FUEL HANDLING MACHINE + AUXILIARY COOLING + LEAK

17-019557

DATED 1967

CHAPELOT A + CHAUVIN M + DELISLE JP + DENIELOU G + GOLLION H + GRANDCOLAS J + REYNES L
MAIN TECHNICAL PROBLEMS RAISED BY THE CONSTRUCTION AND TESTING OF RAPSODIE
FRANCE. COMMISSARIAT A L-ENERGIE ATOMIQUE
13 PAGES, 4 FIGURES, 4 REFERENCES, APRIL 1967, PAPER PRESENTED AT THE NATIONAL TOPICAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, APRIL 10-12, 1967, FAST REACTORS-OPERATING, UNDER CONSTRUCTION, PLANNED FOR CONSTRUCTION

DISCUSSES PROBLEMS ASSOCIATED WITH (1) THERMAL STRESSES AND MISALIGNMENT OF HIGH-TEMPERATURE

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-019557 *CONTINUED*

PIPING AND VESSELS, (2) ARGON COVER-GAS-SYSTEM TIGHTNESS, AND PRESSURE AND PURITY CONTROL, (3) FUEL HANDLING, (4) CONTROL-ROD-DRIVE-MECHANISM CLEARANCES, LUBRICATION, REPAIR, (5) SODIUM-PURIFICATION-SYSTEM VALVE FAILURES, COLD-TRAP PLUGGING.

FAILURE + FRANCE + FUEL HANDLING + REACTOR, LMCR + PRESSURE VESSELS + STRESS + OPERATING EXPERIENCE SUMMARY + THERMAL CONSIDERATION + COOLANT PURIFICATION SYSTEM + VALVES + PIPES AND PIPE FITTINGS + CONTROL ROD DRIVE MECHANISMS

17-019558

DATED 1967

WOODRUFF RW + DURAND RE + OWENS JE
SRE AND HNPf OPERATING AND MODIFICATION EXPERIENCE
ATOMICS INTERNATIONAL

17 PAGES, 16 FIGURES, 23 REFERENCES, APRIL 1967, PAPER PRESENTED AT THE NATIONAL TOPICAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, APRIL 10-12, 1967, FAST REACTORS-OPERATING, UNDER CONSTRUCTION, PLANNED FOR CONSTRUCTION

SUMMARIZES DIFFICULTIES EXPERIENCED AT BOTH PLANTS WITH SYSTEMS AND EQUIPMENT, INCLUDING STEAM GENERATORS, HEAT EXCHANGERS, SODIUM PUMPS, VALVES, AND FUEL-HANDLING MACHINES. DESCRIBES MAINTENANCE PROBLEMS AND EXPERIENCE ACQUIRED WHILE MODIFYING THE SRE FOR POWER UPGRATING FROM 20 TO 30 MWt. BRIEFLY DISCUSSES THE HNPf RETIREMENT AND PROGRAM FOR DISPOSAL OF RADIOACTIVE MATERIALS.

FAILURE + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + OPERATING EXPERIENCE SUMMARY + SRE (RE) + FUEL HANDLING MACHINE + DECOMMISSIONING + POWER UPGRATING + MAIN COOLING SYSTEM + PUMPS + VALVES + HEAT EXCHANGERS

17-019559

DATED 1966

BUSSY P + DENIELOU G + GAJAC G + GUILLET H + LABAT P + PUNTIER R + ROZENHOLC M + VAUTREY L + ZALESKI P
RAPSODIE DEVELOPMENT PROGRAM FROM DESIGN TO OPERATION
COMMISSARIAT A L-ENERGIE ATOMIQUE - FRANCE + GROUPEMENT ATOMIQUE ALSACIENNE ATLANTIQUE

21 PAGES, 11 FIGURES, 3 TABLES, APRIL 1966, PAPER PRESENTED AT THE NATIONAL TOPICAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, APRIL 10-12, 1967, FAST REACTORS-OPERATING, UNDER CONSTRUCTION, PLANNED FOR CONSTRUCTION

DESCRIBES IN DETAIL THE ORIGIN, RESEARCH, DESIGN WORK, CONSTRUCTION PROGRAM, AND STARTUP-TESTING PROGRAM FOR RAPSODIE. A FEW PROBLEMS WHICH AFFECTED THE SCHEDULE ARE BRIEFLY DISCUSSED, INCLUDING THE DISPLACEMENT OF THE REACTOR VESSEL DURING THE INITIAL SYSTEM HEATUP, PUMP SEIZURES, PIPING FAILURES, AND A VALVE FAILURE ALLOWING SODIUM TO OVERFLOW ABOVE THE PLUG OF THE REACTOR VESSEL.

FAILURE + FRANCE + REACTOR, LMCR + FAILURE, PIPE + SODIUM + REACTOR STARTUP TESTING + R AND D PROGRAM + PUMPS + VALVES

17-019561

DATED 1967

BERGSTROM RN + CHITTENDEN WA + MCCLINTOCK DC
CONSTRUCTION AND DESIGN PROBLEMS OF SODIUM COOLED FAST REACTORS
SARGENT AND LUNDY ENGINEERS, CHICAGO, ILL.

25 PAGES, 2 FIGURES, APRIL 12, 1967, PAPER PRESENTED AT THE NATIONAL TOPICAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, APRIL 10-12, 1967, FAST REACTORS-OPERATING, UNDER CONSTRUCTION, PLANNED FOR CONSTRUCTION

REVIEWS THE EFFECT ON REACTOR DESIGN DUE TO THE BOTHERSOME CHARACTERISTICS OF SODIUM, WHICH INCLUDE ITS AFFINITY FOR OXYGEN, FIRE-HAZARD POTENTIAL, SYSTEM PLUGGING BY Na₂O₂ IMPURITIES, RADIOACTIVITY, AND LOW MELTING POINT. THE ADVANTAGES AND DISADVANTAGES OF CONCRETE CONTAINMENT STRUCTURES AND THE PROBLEMS WITH STRESS ANALYSIS ARE DESCRIBED.

CONCRETE + CONTAINMENT + DESIGN CRITERIA + REACTOR, FAST + REACTOR, LMCR + SODIUM + THERMAL MECHANICAL EFFECT + MEASUREMENT, TEMPERATURE + MATERIAL + PROPERTY, PHYSICAL + MAIN COOLING SYSTEM + PIPES AND PIPE FITTINGS

17-019756

ALSO IN CATEGORY 7

DATED 1959

LUNDHOLM JG + JARRETT AA
SRE FUEL ELEMENT DAMAGE
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-4488 +. 153 PAGES, 49 FIGURES, 30 TABLES, REFERENCES, NOVEMBER 30, 1959

INTERIM REPORT ON THE ORIGIN, NATURE, AND CONSEQUENCES OF THE DAMAGE. CLADDING FAILURE ON 13 FUEL ELEMENTS RESULTED FROM THE FORMATION OF LOW-MELTING FE-U ALLOY, ACCELERATED BY LOCAL HIGH TEMPERATURES RESULTING FROM COOLANT-CHANNEL BLOCKAGE. BLOCKAGE WAS ATTRIBUTED TO DECOMPOSITION PRODUCTS OF TETRALIN (PUMP COOLANT), WHICH HAD LEAKED INTO THE PRIMARY SYSTEM THROUGH A SMALL HOLE WORN THROUGH THE WALL OF A FREEZE-SEAL GLAND OF A MAIN PUMP. TENTATIVE CONCLUSIONS INDICATE THAT ALL THE IODINE RELEASED FROM THE FUEL WAS RETAINED BY THE COOLANT. (SEE SUPPLEMENT FOR FINAL ANALYSIS).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, EQUIPMENT + FISSION PRODUCT, IODINE + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + SRE (RE) + INCIDENT + REPORT, OPERATIONS ANALYSIS + FAILURE, CLADDING + FISSION PRODUCT RETENTION + FLOW BLOCKAGE + MAIN COOLING SYSTEM

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-019757 ALSO IN CATEGORY 6 DATED 1959
LUNDHOLM JG + JARRETT AA
REACTIVITY ANOMALIES IN SRE
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-4488 +. 33 PAGES, FIGURES, TABLES, PAGES III-1 TO 18 AND IV-D-1 TO 15 FROM SRE FUEL ELEMENT DAMAGE,
NOVEMBER 30, 1959

DESCRIBES REACTIVITY EFFECTS RELATED TO FUEL-ELEMENT-DAMAGE INCIDENT. BETWEEN RUNS, FUEL ELEMENTS WERE WASHED TO REMOVE SURFACE DEPOSITS FROM TETRALIN. DURING RUN 9, A REACTIVITY OF 1/2% WAS ATTRIBUTED TO APPROXIMATIONS IN CALCULATIONS OF XE CORRECTION. IN RUN 11, A LOSS OF 1/4% WAS ATTRIBUTED TO THIMBLE CHANGES, AND, NEAR THE END OF RUN, A GAIN OF 1/3% WAS ATTRIBUTED TO PRELIMINARY CALCULATIONS. 3 DAYS LATER, THE REACTIVITY WAS NORMAL. FOR RUN 13, SHIM-ROD MOTION FOR 6 HR INDICATED A 0.3% INCREASE AND THEN A 0.1% INCREASE FOR THE NEXT 3 DAYS, ATTRIBUTED TO A TETRALIN (SUMP COOLANT) LEAK INTO THE PRIMARY SYSTEM. IN RUN 14, A LOSS OF 1.2% DURING THE FIRST 4 DAYS WAS LATER ATTRIBUTED TO A PARTED FUEL CLUSTER DROPPING FROM THE FUEL REGION AND TO GRADUAL FLOODING OF MODERATOR CANS WITH SODIUM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, FUEL ELEMENT + REACTOR, LMCR + COMPARISON, THEORY AND EXPERIENCE + XENON + SRE (RE) + FAILURE, ADMINISTRATIVE CONTROL + REACTIVITY EFFECT, ANOMALOUS

17-019758 ALSO IN CATEGORY 9 DATED 1959
LUNDHOLM JG + JARRETT AA
SRE POWER EXCURSION NOT HALTED BY FAST PERIOD
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-4488 +. 3 PAGES, 2 TABLES, PAGES III-11 TO 13 FROM SRE FUEL ELEMENT DAMAGE, NOVEMBER 30, 1959

AFTER STARTING A POWER INCREASE, THE LEVEL PERSISTED IN RISING FASTER THAN EXPECTED EVEN THOUGH CONTROL RODS WERE BEING SLOWLY INSERTED TO HOLD IT BACK. THREE 50-SEC POSITIVE-PERIOD TRANSIENTS OCCURRED, FOLLOWED BY A 7-1/2 SEC PERIOD. THE REACTOR WAS SCRAMMED. PEAK POWER WAS 24 MW. A SETBACK WAS NOT INITIATED AUTOMATICALLY AT A 10-SEC PERIOD BY A SWITCH ACTUATED BY A NOTCH IN A CAM, BECAUSE IT WOULD OPERATE WELL ONLY IF THE RATE OF CHANGE OF THE PERIOD WAS SLOW. THE CAM WAS MODIFIED TO OPERATE AT ALL RATES OF CHANGE. A POSSIBLE EXPLANATION FOR THE TRANSIENT IS THE PARTING AND DROPPING OF A LOWER SECTION OF A SINGLE ELEMENT OR THE DOPPLER EFFECT (SEE SUPPL. FOR FINAL ANALYSIS).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

DOPPLER EFFECT + FAILURE, FUEL ELEMENT + REACTOR, LMCR + REACTOR TRANSIENT + SRE (RE) + FAILURE, INSTRUMENT + REACTIVITY EFFECT, ANOMALOUS + INSTRUMENT, PERIOD + RECORDERS

17-019759 DATED 1959
LUNDHOLM JG + JARRETT AA
RECOVERY FROM FUEL MELTDOWN INCIDENT
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-4488 +. 7 PAGES, 3 TABLES, PAGES III-18 TO 24 FROM SRE FUEL ELEMENT DAMAGE, NOVEMBER 30, 1959

DISCUSSES METHOD OF REMOVAL AND INSPECTION OF FUEL ELEMENTS. BROKEN FUEL ELEMENTS WERE LODGING IN THE FUEL-HANDLING CASK, AND EFFORTS TO DISLODGE WERE CAUSING EXCESSIVE EXPOSURES. THE MODERATOR-HANDLING CASK WAS MODIFIED FOR THIS OPERATION. BRIEFLY DESCRIBES METHOD OF CANNING DAMAGED ELEMENTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, FUEL ELEMENT + REACTOR, LMCR + SRE (RE) + INCIDENT, RECOVERY FROM + FUEL HANDLING MACHINE

17-019762 DATED 1967
GUNBY AL
NONNUCLEAR PERFORMANCE OF SRE-PEP SYSTEMS
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-12408 +. 59 PAGES, 27 FIGURES, 3 TABLES, 13 REFERENCES, JUNE 30, 1967

DESCRIBES PREOPERATIONAL TESTS OF PLANT SYSTEMS MODIFIED FOR POWER UPGRATING FROM 20 TO 30 MW, PERFORMED JUST BEFORE DEACTIVATION. GAS ENTRAINMENT IN THE MAIN PRIMARY PUMP WAS CORRECTED BY ADJUSTING THE PUMP-LEVEL-CONTROL SYSTEM. VIBRATIONAL TEST OF PUMPS AND HEAT EXCHANGER WERE SATISFACTORY. THE SODIUM SERVICE VAULT COOLING SYSTEM HAD INSUFFICIENT CAPABILITY. AMPLIFIERS IN THE HEAT-TRANSFER PORTION OF THE PROTECTIVE INSTRUMENTATION WERE UNSTABLE. REQUIRED MODIFICATIONS BEFORE POWER OPERATION ARE GIVEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, EQUIPMENT + REACTOR, LMCR + SRE (RE) + FAILURE, INSTRUMENT + MODIFICATION + TEST, PREOPERATIONAL + TEST, SYSTEM OPERABILITY + POWER UPGRATING + VIBRATION + PUMPS + HEAT EXCHANGERS

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-019860 ALSO IN CATEGORY 5 DATED 1961
FILLMORE FL
SRE FUEL ELEMENT DAMAGE
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-4488 (SUPPL) +. 49 PAGES, 21 FIGURES, TABLES, 11 REFERENCES, JUNE 30, 1961

THE CAUSE OF THE FUEL-ELEMENT FAILURES WAS ATTRIBUTED TO FORMATION OF LOW-MELTING ALLOY WITH THE CLADDING AND TO THERMAL CYCLING AND EXPANSION OF THE FUEL SLUGS UNTIL THE CLADDING BURST. REACTIVITY CHANGES BEFORE AND DURING THE POWER EXCURSION WERE EXPLAINED. A NEGATIVE 0.06% REACTIVITY EXCURSION AT 1807 HR WAS ATTRIBUTED TO COLLAPSE OF VAPOR VOIDS WHICH HAD FORMED IN PARTIALLY PLUGGED CHANNELS. AT 1821 HR, A POSITIVE RISE OF 0.04% IN A 3-MIN RAMP WAS ATTRIBUTED TO INCREASED MODERATOR TEMPERATURE OR VOID FORMATION. AT 1825 HR, THE POSITIVE EXCURSION OF 0.3% REACTIVITY IN A 5- TO 10-SEC RAMP WAS ATTRIBUTED TO VOID FORMATION IN 10 FUEL CHANNELS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + REACTOR TRANSIENT + VOID COEFFICIENT + SRE (RE) + INCIDENT + TEMPERATURE REACTIVITY EFFECT + FAILURE, CLADDING + FLOW BLOCKAGE + REACTIVITY EFFECT, ANOMALOUS

17-019861 DATED 1961
RECOVERY FROM FUEL MELTDOWN
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-4488 (SUPPL) +. 8 PAGES, 4 FIGURES, FROM SRE FUEL ELEMENT DAMAGE. FINAL REPORT, JUNE 30, 1961

DESCRIBES (1) DAMAGED ELEMENT REMOVAL OPERATIONS, (2) CORE INSPECTION, SURVEY, AND CLEANUP, (3) REMOVAL AND REPLACEMENT OPERATIONS FOR DAMAGED MODERATOR ASSEMBLIES, (4) AND CLEANUP OF PIPING AND EQUIPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

DECONTAMINATION + FUEL MELTDOWN + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + SRE (RE) + INCIDENT, RECOVERY FROM + MAIN COOLING SYSTEM + PIPES AND PIPE FITTINGS

17-019862 ALSO IN CATEGORY 9 DATED 1961
MODIFICATIONS TO REACTOR
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-4488 (SUPPL) +. 14 PAGES, 9 FIGURES, FROM SRE FUEL ELEMENT DAMAGE, FINAL REPORT, JUNE 30, 1961

MODIFICATIONS TO AVOID A SECOND MAJOR FUEL-ELEMENT FAILURE INCLUDE ELIMINATING THE USE OF TETRALIN AS A SERVICE COOLANT, INCREASING CLEARANCE BETWEEN ELEMENT AND CHANNEL, RELOCATION OF THE ELEMENT ORIFICE PLATE, ADDING A STRUCTURAL MEMBER TO THE ROD CLUSTER, ATTACHING A STRAINER AT THE BOTTOM OF THE ELEMENT, INSTALLING A CONTINUOUS COVER-GAS MONITOR, INSTALLING A TRANSIENT MONITOR (ACTIVATED BY AN ABNORMAL SIGNAL SUCH AS A SCRAM) TO MONITOR FLUX, PERIOD, TEMPERATURES, FLOW, AND ROD POSITION. TWO RECORDERS WERE INSTALLED TO MEASURE FUEL-SLUG TEMPERATURES. THREE 20-PEN RECORDERS WERE INSTALLED TO MONITOR PROTECTIVE SYSTEM ACTUATIONS, AND AN ADDITIONAL PROTECTIVE SYSTEM WAS ADDED TO PREVENT A LARGE MISMATCH BETWEEN FLUX AND POWER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, FUEL ELEMENT + INSTRUMENT, PROTECTIVE + REACTOR, LMCR + SRE (RE) + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + MODIFICATION + FAILURE, CLADDING + INSTRUMENTS, MISC. + FUEL ELEMENTS + RECORDERS

17-020045 DATED 1963
LEIPUNSKII AI + KAZACHKOVSKII OD + PINKHASIK MS + ARISTARKHOV NN + KARPOV AV + LARIN EP + EFIMOV IA
OPERATING EXPERIENCE WITH THE BR-5 REACTOR
TRG-INF-SER-328 +. 14 PAGES, 4 FIGURES, 2 REFERENCES, 1963

REPLACEMENT OF A PRIMARY PUMP WAS IMPROVED BY PARTIAL DRAINAGE OF THE SYSTEM (RATHER THAN COMPLETE DRAINAGE) AND ALLOWING THE REMAINING NA TO SOLIDIFY, THUS PREVENTING LEAKAGE. PARTIAL DRAINING AND FREEZING REDUCED OXIDE CONTAMINATION. OTHER MAINTENANCE ALSO IMPROVED BY ALLOWING SOLIDIFICATION OF NA. DISCUSSES CORE-DISCHARGE OPERATIONS, METHODS OF INSPECTING FUEL ELEMENTS FOR FAILURES, AND PROCEDURE FOR DECONTAMINATING THE PRIMARY-SYSTEM EQUIPMENT AND PIPING.

AVAILABILITY - JOHN CRERAR LIBRARY, 35 WEST 33RD ST., CHICAGO, ILL. 60616, \$1.60 COPY

DECONTAMINATION + FUEL HANDLING + REACTOR, LMCR + MAINTENANCE AND REPAIR + SODIUM + OPERATING EXPERIENCE SUMMARY + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + USSR + MAIN COOLING SYSTEM

17-020467 DATED 1965
SODIUM-COOLED REACTORS PROGRAM. FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 13, OCTOBER 1964-JANUARY 1965
GENERAL ELECTRIC COMPANY, SAN JOSE, CALIF.
GEAP-4795 +. 92 PAGES, JANUARY 1965

ONE OF A SERIES OF MONTHLY PROGRESS REPORTS CONCERNED WITH PU-U OXIDE FUELS, INCLUDING IRRADIATION

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-020467 *CONTINUED*

TESTING, SODIUM-FUEL COMPATIBILITY, FUEL FABRICATION, FUEL PERFORMANCE EVALUATION, FUEL MELTING AND REDISTRIBUTION, FUEL BURNUP, STOICHIOMETRIC EFFECTS ON PHYSICAL PROPERTIES, REACTOR PHYSICS CALCULATIONS, AND ANALYSIS OF VARIOUS CORES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + FUEL BURNUP + REACTOR, LMCR + URANIUM OXIDE + PLUTONIUM OXIDE + CENTERLINE MELTING + IRRADIATION TESTING + PROPERTY, PHYSICAL + STOICHIOMETRY + REACTOR PHYSICS + R AND D PROGRAM + FUEL ELEMENTS

17-020468

DATED 1965

SODIUM-COOLED REACTORS PROGRAM. FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 14. FEBRUARY-APRIL 1967

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

GEAP-4860 +. 76 PAGES, MAY 1965

ONE OF A SERIES OF MONTHLY PROGRESS REPORTS CONCERNED WITH MIXED PU-U OXIDE FUELS, INCLUDING IRRADIATION TESTING, SODIUM-FUEL COMPATIBILITY, FUEL FABRICATION, FUEL PERFORMANCE EVALUATION, FUEL MELTING AND REDISTRIBUTION, FUEL BURNUP, STOICHIOMETRIC EFFECTS ON PHYSICAL PROPERTIES, REACTOR PHYSICS CALCULATIONS, AND ANALYSIS OF VARIOUS CORES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + FUEL BURNUP + REACTOR, LMCR + URANIUM OXIDE + PLUTONIUM OXIDE + CENTERLINE MELTING + IRRADIATION TESTING + PROPERTY, PHYSICAL + STOICHIOMETRY + REACTOR PHYSICS + R AND D PROGRAM + FUEL ELEMENTS

17-020469

DATED 1965

SODIUM-COOLED REACTORS PROGRAM. FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. FIFTEENTH QUARTERLY REPORT, MAY-JULY 1965

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

GEAP-4931 +. 39 PAGES, REFERENCES, AUGUST 1965

ONE OF A SERIES OF MONTHLY PROGRESS REPORTS CONCERNED WITH MIXED PU-U OXIDE FUELS, INCLUDING IRRADIATION TESTING, SODIUM-FUEL COMPATIBILITY, FUEL FABRICATION, FUEL PERFORMANCE EVALUATION, FUEL MELTING AND REDISTRIBUTION, FUEL BURNUP, STOICHIOMETRIC EFFECTS ON PHYSICAL PROPERTIES, REACTOR PHYSICS CALCULATIONS, AND ANALYSIS OF VARIOUS CORES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + FUEL BURNUP + REACTOR, LMCR + URANIUM OXIDE + PLUTONIUM OXIDE + CENTERLINE MELTING + IRRADIATION TESTING + PROPERTY, PHYSICAL + STOICHIOMETRY + REACTOR PHYSICS + R AND D PROGRAM + FUEL ELEMENTS

17-020470

DATED 1966

SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 18. FEBRUARY-APRIL 1966

GENERAL ELECTRIC COMPANY, SAN JOSE, CALIF.

GEAP-5158 +. 112 PAGES, REFERENCES, AUGUST 1966

ONE OF A SERIES OF MONTHLY PROGRESS REPORTS CONCERNED WITH MIXED PU-U OXIDE FUELS, INCLUDING IRRADIATION TESTING, SODIUM-FUEL COMPATIBILITY, FUEL FABRICATION, FUEL PERFORMANCE EVALUATION, FUEL MELTING AND REDISTRIBUTION, FUEL BURNUP, STOICHIOMETRIC EFFECTS ON PHYSICAL PROPERTIES, REACTOR PHYSICS CALCULATIONS, AND ANALYSIS OF VARIOUS CORES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + FUEL BURNUP + REACTOR, LMCR + URANIUM OXIDE + PLUTONIUM OXIDE + CENTERLINE MELTING + IRRADIATION TESTING + PROPERTY, PHYSICAL + STOICHIOMETRY + REACTOR PHYSICS + R AND D PROGRAM + FUEL ELEMENTS

17-020472

DATED 1967

SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 22. FEBRUARY-APRIL 1967

GENERAL ELECTRIC CO., SUNNYVALE, CALIF.

GEAP-5491 +. 102 PAGES, MAY 1967

ONE OF A SERIES OF MONTHLY PROGRESS REPORTS CONCERNED WITH MIXED PU-U OXIDE FUELS, INCLUDING IRRADIATION TESTING, SODIUM-FUEL COMPATIBILITY, FUEL FABRICATION, FUEL PERFORMANCE EVALUATION, FUEL MELTING AND REDISTRIBUTION, FUEL BURNUP, STOICHIOMETRIC EFFECTS ON PHYSICAL PROPERTIES, REACTOR PHYSICS CALCULATIONS, AND ANALYSIS OF VARIOUS CORES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + FUEL BURNUP + REACTOR, LMCR + URANIUM OXIDE + PLUTONIUM OXIDE + CENTERLINE MELTING + IRRADIATION TESTING + PROPERTY, PHYSICAL + STOICHIOMETRY + REACTOR PHYSICS + R AND D PROGRAM + FUEL ELEMENTS

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-020516
ORL ALLOWS DISMANTLING OF HALLAM
DIVISION OF REACTOR LICENSING, AEC
9 PAGES, NOVEMBER 3, 1967, DOCKET 115-3

DATED 1967

ORL AUTHORIZES CONSUMERS PUBLIC POWER TO DISMANTLE AND DECONTAMINATE THE HALLAM REACTOR FACILITY. HALLAM NUCLEAR POWER FACILITY RETIREMENT PLAN (NAA-SR-MEMO-12340, REVISED 5/20/67) IS TO BE FOLLOWED. ABOUT 310,000 CURIES WILL BE ENTOMBED IN THE BELOW-GRADE PART OF THE STRUCTURE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C. 20432

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + DECOMMISSIONING + SAFETY EVALUATION

17-020579 ALSO IN CATEGORY 10 DATED 1966
MCCARTHY JF
SHORT ACROSS BOTH 480V FEEDERS AT FERMI
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + POWER REACTOR DEVELOPMENT COMPANY + DETROIT EDISON COMPANY
APDA-CFE-1 +. 1 PAGE, PAGE 19 OF COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER
PLANT, AUGUST 1966

(PG 19) THE JUNE 29 POWER OUTAGE FAILED BOTH 480-VOLT FEEDS. ONE PHASE OF ONE PRIMARY SIDE OF THE UNGROUNDED DELTA-WOUND 4800/480-V TRANSFORMER HAD SHOWN PREVIOUS SPORADIC GROUNDING. THIS GROUNDS THE AFFECTED PHASE WITHOUT CURRENT FLOW AND RAISE THE OTHER PHASES FROM 2770 TO 4800 V. ON JUNE 29, THE RAISED VOLTAGE PROBABLY TRIGGERED A WEAK SPOT IN THE OTHER FEED, RESULTING IN A PHASE-TO-PHASE SHORT THROUGH GROUND, OPENING THE 4800-V CIRCUIT BREAKERS AND DESTROYING THE CABLES. DIESEL CARRIED THE DIESEL BUS LOADS.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

EMERGENCY POWER, ELECTRIC + FERMI (LMFBR) + ACCIDENT, LOSS OF POWER + FAILURE, SEQUENTIAL + FAILURE, SIMULTANEOUS + REPORT, OPERATIONS SUMMARY + REACTOR, LMCR

17-020638 ALSO IN CATEGORY 9 DATED 1966
MCCARTHY WJ + JENS WH
ENRICO FERMI FAST BREEDER REACTOR
POWER REACTOR DEVELOPMENT CO. + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
4 PAGES, NUCLEAR NEWS 10(11), PAGE 54-57, (NOVEMBER 1966)

SHORT DISCUSSION OF CURRENT STATUS, MOSTLY WHAT HAS BEEN LEARNED. AN INLET STRAINER, CORE-PLATE STANDOFF PINS, OR PROVISION FOR CROSS FLOW WOULD HELP ELIMINATE FUTURE MELTDOWNS. REACTIVITY RATE-OF-CHANGE INFORMATION AND A FASTER FISSION-PRODUCT MONITOR (DELAYED NEUTRONS AT CORE EXIT TO SUPPLEMENT PRESENT SENSITIVE COVER-GAS AS DETECTOR WITH 13-MIN TRANSPORT LAG) IS NEEDED. FEW PROVISIONS WERE MADE FOR INCIDENT RECOVERY, E.G., DIFFICULT ACCESS TO WHOLE CORE, NO PROVISION FOR DRAINING VESSEL OR FOR ON-SITE INSPECTION OF IRRADIATED FUEL.

FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + OPERATING EXPERIENCE SUMMARY + INCIDENT + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + INCIDENT, RECOVERY FROM + FUEL HANDLING MACHINE + REMOTE MANIPULATING AND VIEWING + FLOW BLOCKAGE + INSTRUMENT, OPERATING REACTIVITY

17-020665 DATED 1967
SUNDERMAN IO
HALLAM NUCLEAR POWER FACILITY MONTHLY OPERATING REPORT NO. 47.
CONSUMERS PUBLIC POWER DISTRICT, HALLAM, NEBR.
TID-23306 +. 15 PAGES, TABLES, JUNE 1966

ONE OF A SERIES OF PROGRESS REPORTS. SINCE THE REACTOR IS IN THE PROCESS OF BEING DISMANTLED, THE MORE APPROPRIATE INFORMATION IS IN MAINTENANCE, CORE-ELEMENT TRANSFERS, RADIATION SAFETY, AND CONTROL. SHORT SUMMARIES OF EACH DAYS WORK ARE GIVEN.

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

17-020727 DATED 1967
POWER REACTOR DEVELOPMENT COMPANY ENRICO FERMI ATOMIC POWER PLANT REPORT FOR SEPTEMBER 1967
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-49 + EF-49 +. 7 PAGES, SEPTEMBER 1967, DOCKET 50-16

BMI INVESTIGATIONS OF ELEMENT M-098 SHOWED 20% OF THE URANIUM IN THE LOWER AXIAL BLANKET AND 50% MOLTEN AT SOME TIME. A THIN FOREIGN OBJECT ABOUT 8 X 4 IN. WAS FOUND IN THE MELTDOWN PAN.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C. 20432

FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + REPORT, OPERATIONS + EXAMINATION + FLOW BLOCKAGE

17-020733 ALSO IN CATEGORY 5 DATED 1966
ANDERSON WF + ANDERSON WJ + HEISLER MP + MCKISSON RL + ZWENG DJ
SCTI SODIUM HEATER FAILURE

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-020733 *CONTINUED*

ATOMIC INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-MEMO-12090 +. 57 PAGES, 18 FIGURES, 2 TABLES, 12 REFERENCES, AUGUST 1, 1966

DURING CHECKOUT, A CIRCUMFERENTIAL TUBE CRACK RELEASED 50 GAL OF SODIUM. ADJACENT TUBES WERE DISTORTED BY REACTION AND ALSO BY KAOCAST. CAUSE WAS A LACK OF MOBILITY DURING THERMAL EXPANSION BY IMPROPER KAOCAST INSULATION, WHICH HAD BEEN POURED AS A MONOLITHIC SLAB INSTEAD OF AS A SERIES OF BLOCKS. CHLORIDE STRESS CORROSION EVIDENCE DISCUSSED. THE HEATER CAN BE REPAIRED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + THERMAL INSULATION + FAILURE, INSTALLATION ERROR + FAILURE, TUBING + HEAT EXCHANGERS

17-021081

DATED 1959

LEONARD WD

EFFECT OF SODIUM SPILLAGE AND LOSS OF NITROGEN COOLING IN IMX CELLS AND PIPE TUNNEL

ATOMIC INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-MEMO-3560 +. 14 PAGES, REFERENCES, MARCH 9, 1959

GIVES RESULTS OF THEORETICAL ANALYSIS TO DETERMINE THE MAXIMUM TEMPERATURES AND PRESSURES IN THE (HNPFI) INTERMEDIATE HEAT-EXCHANGER CELLS AND PIPE TUNNEL RESULTING FROM A LOSS OF NITROGEN COOLING OR SODIUM SPILLAGE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + FAILURE + HALLAM (LMR) + REACTOR, LMCR + SODIUM + THEORETICAL INVESTIGATION + HEAT EXCHANGERS

17-021216

DATED 1967

ENRICO FERMI MELTDOWN

4 PAGES, 5 FIGURES, NUCLEAR ENGINEERING 2(138), PAGE 843-846, (NOV. 1967)

THE CAUSE OF THE MELTDOWN IS ATTRIBUTED TO A FOREIGN BODY 8 TO 10 IN. LONG AND 3 TO 4 IN. WIDE, WHICH BLOCKED THE COOLANT FLOW IN SEVERAL FUEL CHANNELS. THE EXTENT OF DAMAGE TO FUEL ELEMENTS IS DESCRIBED. STATES THAT INFORMATION WAS AVAILABLE IN THE CONTROL ROOM, WHICH, IF IT HAD BEEN CLEARLY PRESENTED TO OPERATORS, THEY MIGHT HAVE ACTED IN TIME TO PREVENT THE MELTING. INCIDENT INDICATES (1) A STRAINER AT INLET PLENUM IS DESIRABLE, (2) A DELAYED-NEUTRON DETECTOR NEEDED FOR QUICKER INDICATIONS OF FAILURE, (3) DESIGN SHOULD ALLOW FOR ACCESS TO WHOLE CORE AND BLANKET UNDER UNUSUAL CONDITIONS, (4) VESSEL DRAIN SYSTEM NEEDED, AND (5) IMPROVED DETECTION TECHNIQUES.

FAILURE + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + OPERATING EXPERIENCE SUMMARY + INCIDENT + FAILURE, ADMINISTRATIVE CONTROL + FLOW BLOCKAGE

17-021335

DATED 1966

REIZY CE

SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 17, NOVEMBER 1965-JANUARY 1966

GENERAL ELECTRIC CO., SAN JOSE, CALIF. ADVANCED PRODUCTS OPERATION

GEAP-5098 +. 80 PAGES, FIGURES, TABLES, REFERENCES, FEB. 1966

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS INCLUDE IRRADIATION TESTING OF MIXED PU-U OXIDES, GAP-CONDUCTIVITY EXPERIMENTS, FUEL BURNUP, REACTOR PHYSICS AND CORE ANALYSIS, VENTED-FUEL DEVELOPMENT, AND CLADDING-COMPATIBILITY STUDIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

CLADDING + FUEL BURNUP + REACTOR, LMCR + SODIUM + URANIUM OXIDE + PLUTONIUM OXIDE + IRRADIATION TESTING + REACTOR PHYSICS + HEAT CONDUCTANCE, FUEL TO CLAD + R AND D PROGRAM + FUEL ELEMENTS

17-021350

DATED 1967

VENDRYES G + DENIELOU G

RAPSODIE, THE FRENCH FAST NEUTRON REACTOR

COMMISSARIAT A L-ENERGIE ATOMIQUE

3 PAGES, 1 FIGURE, INDUSTRIES ATOMIQUES 11(5/6), PAGES 49-51 (1967) IN FRENCH

INTEGRATING RAPSODIE WITHIN THE FAST-REACTOR PROGRAM IN FRANCE IS DISCUSSED. THE PRELIMINARY STUDIES AND THE CONSTRUCTION OF THE REACTOR ARE GIVEN. THE RESULTS OBTAINED DURING THE FINAL STAGE OF THE START-UP AND THOSE EXPECTED DURING OPERATION ARE OUTLINED. A BRIEF COMMENT ON THE PHENIX REACTOR IS INCLUDED.

REACTOR, BREEDER + FRANCE + REACTOR, LMCR + REACTOR STARTUP TESTING

17-021351

DATED 1967

CHAUVIN M + ROSENHOLC M

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-021351 *CONTINUED*
THE CONSTRUCTION OF RAPSODIE
COMMISSARIAT A L ENERGIE ATOMIQUE
7 PAGES, 4 FIGURES, 2 TABLES, INDUSTRIES ATOMIQUES 11(5/6), PAGES 53-59 (1967) IN FRENCH

THE MAIN CONSTRUCTION PHASES OF RAPSODIE, FROM THE DECISION TO ERECT THE REACTOR AT CADARACHE TO THE BEGINNING OF THE START-UP AND TO THE MOMENT WHEN THE REACTOR BECAME FIRST CRITICAL, ARE DESCRIBED. THE ESSENTIAL FEATURES OF RAPSODIE AND THE DIFFICULTIES ENCOUNTERED DURING ITS CONSTRUCTION ARE DISCUSSED.

REACTOR, BREEDER + FRANCE + REACTOR, LMCR + REACTOR DESCRIPTION

17-021352 DATED 1967
CHAPELOT A + LELAIT R
SOME ASPECTS OF THE DRIVE-CONTROL OF RAPSODIE
COMMISSARIAT A L-ENERGIE ATOMIQUE
11 PAGES, 5 FIGURES, INDUSTRIES ATOMIQUES 11(5/6), PAGES 60-70 (1967) IN FRENCH

SOME OF THE ASPECTS OF THE CONTROL SYSTEM AND THE GENERAL PRINCIPLES OF THE AUTOMATION DESIGN ARE DISCUSSED. THE SAFETY MECHANISMS APPLIED IN THE CONTROL OF COMPONENTS OF THE PLANT AND IN SAFETY OPERATIONS, HOWEVER, DEMAND MUCH MORE ELABORATE TECHNOLOGICAL STRUCTURES AND METHODS. THE IMPORTANCE OF CONVENTIONAL MEASURING AND REGULATION DEVICES IS STRESSED. SOME SPECIFIC CONTROL AND MEASURING SYSTEMS DESIGNED FOR OPERATION WITH SODIUM ARE DESCRIBED. A DATA-PROCESSING SYSTEM COMPLETES THE CLASSICAL INDICATING UNITS AND ASSUMES THE MORE SUBTLE CONTROL AND PROCESSING OPERATIONS.

REACTOR, BREEDER + CONTROL SYSTEM + FRANCE + REACTOR, LMCR + MEASUREMENT + REACTOR SAFETY SYSTEM + DATA PROCESSING + SYSTEM DESCRIPTION + CONTROL ROD DRIVE MECHANISMS

17-021353 DATED 1967
PONTIER R
THE TESTING AND STARTING UP OF RAPSODIE
COMMISSARIAT A L-ENERGIE ATOMIQUE
5 PAGES, 1 FIGURE, INDUSTRIES ATOMIQUES 11(5/6), PAGES 87-91 (1967) IN FRENCH

DESCRIBES THE STARTUP OF THE RAPSODIE. LISTS MAIN PHASES OF THE COURSE OF TESTS, PROBLEMS ENCOUNTERED, AND RESULTS.

REACTOR, BREEDER + FRANCE + REACTOR, LMCR + REACTOR STARTUP TESTING + TEST, PHYSICS

17-021736 DATED 1961
MANN LA
EBR-I OPERATING EXPERIENCE
4 PAGES, NUCLEAR SAFETY 3(2), PAGES 68-71 (DECEMBER 1961)

BRIEFLY DESCRIBES THE REACTOR. IN 1955, A PARTIAL MELTDOWN OCCURRED DURING AN EXPERIMENT WHEN AN OPERATOR MISTAKENLY ACTIVATED THE WRONG CONTROL RODS. OPERATION WITH MARK-I AND -II CORES INDICATED THAT A POSITIVE METAL TEMPERATURE COEFFICIENT RESULTED WHEN THE COOLANT FLOW CHANGED. THIS WAS CAUSED BY FUEL-ROD BOWING TOWARD THE CORE AXIS BECAUSE OF THE INCREASED FLUX AND FISSION RATE AS A FUNCTION OF NEARNESS TO THE AXIS. EXPERIMENTS SUBSTANTIATED THE EXPLANATION. MARK-III CORE DESIGN INCLUDED STIFFENING RIBS WHICH PREVENTED THE POSITIVE COEFFICIENT FROM BOWING THE FUEL RODS.

EBR 1 AND 2 (RE) + REACTOR, FAST + FUEL ELEMENT BOWING + FUEL MELTDOWN + FAILURE, OPERATOR ERROR + REACTOR, LMCR + REACTIVITY EFFECT + REACTOR, TEST + OPERATING EXPERIENCE SUMMARY + MODIFICATION + INCIDENT, HUMAN ERROR + FUEL ELEMENTS

17-021741 DATED 1961
DICKINSON RW
SRE OPERATING EXPERIENCE
8 PAGES, NUCLEAR SAFETY 2(3), PAGE 54-61, (MARCH 1961)

REVIEWS THE PROBLEMS AND FAILURES THAT HAVE OCCURRED IN 3-1/2 YEARS OF OPERATION, WHICH INCLUDE PUMP BINDING AND FREEZE-SEAL LEAKS, THERMAL STRATIFICATION OF SODIUM IN THE MAIN INTERMEDIATE HEAT EXCHANGER DUE TO POOR DESIGN, OPERATIONAL FAILURE OF BELLOW-SEALED VALVES, AND THERMAL-STRESS PROBLEMS IN THE OUTLET NOZZLE. DISCUSSES THE RADIATION LEVELS IN THE PRIMARY COOLING SYSTEM AND RECOVERY OPERATIONS AS RESULT OF FAILURE OF 13 FUEL ELEMENTS CAUSED BY FLOW BLOCKAGE (BY A FREEZE-SEAL COOLANT, TETRALIN, WHICH LEAKED FROM A FAILED THERMOCOUPLE WELL). THE KINETICS AND STABILITY OF THE REACTOR ARE BRIEFLY CONSIDERED.

FAILURE, EQUIPMENT + FAILURE, FUEL ELEMENT + REACTOR, LMCR + REACTOR KINETICS + REACTOR STABILITY + STRESS + THERMAL MECHANICAL EFFECT + OPERATING EXPERIENCE SUMMARY + SRE (RE) + INCIDENT, RECOVERY FROM + RADIOACTIVITY RELEASE + FAILURE, DESIGN ERROR + PUMPS + VALVES + PIPES AND PIPE FITTINGS + HEAT EXCHANGERS

17-021753 DATED 1967
ARGONNE NATIONAL LABORATORY. REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT. OCTOBER 1967
ARGONNE NATIONAL LAB., ARGONNE, ILLINOIS

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-021753 *CONTINUED*
ANS-7391 +. 185 PAGES, FIGURES, TABLES, NOV. 30, 1967

ONE OF A SERIES OF REPORTS COVERING EBR II, ZPR 3, ZPPR, AARR, LMFBR, ETC. OVER HALF CONCERNS THE LIQUID-METAL FAST-BREEDER PROGRAM. (EBR II) AN OSCILLATOR ROD WAS INSTALLED IN PLACE OF A CONTROL ROD. INITIAL TESTS INDICATED DIFFICULTIES (PREVIOUSLY EXPERIENCED) WITH RUBBING AT LOW FREQUENCIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCB + REPORT, OPERATIONS + OSCILLATOR, REACTIVITY + REACTOR, RESEARCH + CONTROL ROD DRIVE MECHANISMS

17-021968 ALSO IN CATEGORY 1 DATED 1968
AEC TESTIMONY TO JCAE ON FERMI MELTDOWN CAUSE
ATOMIC ENERGY COMMISSION, US
1 PAGE, ATOMIC ENERGY CLEARING HOUSE 14(7), PAGE 46 (FEBRUARY 12, 1968)

ACCORDING TO MILTON SHAW, THE FLOW-BLOCKING OBJECT WAS A PIECE OF A CONICAL FLOW GUIDE WHICH HAD VIBRATED LOOSE. REP. HOSMER NOTED THAT THE DESIGNER IN 1959 COVERED THE VESSEL BOTTOM WITH ZIRCONIUM SHEET, AND SAID PAUA HAD RECEIVED A MEMO INDICATING IT WAS EASIER TO PUT ZIRCONIUM SHROUDS OVER THE CORE THAN TO JUSTIFY NOT PUTTING IT IN TO THE ACRS. SHAW OBSERVED IT WAS A LAST-MINUTE DECISION HURRIEDLY MADE, NOT SHOWN IN THE FINAL CONSTRUCTION DRAWINGS. HOSMER ASKED IF ACRS HAD PEOPLE SO FRIGHTENED. SHAW REPLIED, NO, AND THIS WAS NOT AN EXCUSE NOT TO DO A THOROUGH ENGINEERING JOB.

CONGRESSIONAL ACTIVITY + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCB + ACRS + FLOW BLOCKAGE + QUALITY ASSURANCE

17-022456 ALSO IN CATEGORY 5 DATED 1967
EBR II MEASUREMENTS OF PROMPT POWER COEFFICIENT FOILED
ARGONNE NATIONAL LAB., ARGONNE, ILL.
ANL-7399 +. 7 PAGES, 5 FIGURES, 2 TABLES, PAGE 48-54 OF ARGONNE NATIONAL LABORATORY. REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, NOVEMBER 1967

TRANSFER FUNCTIONS WERE MEASURED. THE WORTH OF THE ROD USED IN ROD-DROP EXPERIMENTS CHANGES ABOVE THE 0.5-MW CALIBRATION LEVEL. DURING THE DROPPING OF THE ROD, IT SEEMS THAT ROD WORTH INCREASES ENOUGH TO CANCEL THE EFFECT OF PROMPT NEGATIVE FEEDBACK FOR 0.3 SEC. TOTAL FEEDBACK IS ABOUT 0.33 CENT. REACTIVITY DIFFERENCES BETWEEN FULL-POWER FLOW AND REDUCED-POWER FLOW CONDITIONS SEEM TO BE STEADILY INCREASING WITH FUEL BURNUP.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

CONTROL ROD WORTH + EBR 1 AND 2 (RE) + FUEL BURNUP + REACTOR, LMCB + POWER COEFFICIENT + TRANSFER FUNCTION + RESPONSE TIME + CONTROL ROD CALIBRATION + REPORT, OPERATIONS ANALYSIS + TEST, PHYSICS

17-022457 ALSO IN CATEGORY 9 DATED 1967
EBR 2 STEAM GENERATOR LEVEL CONTROLLER MALFUNCTION
ARGONNE NATIONAL LAB., ARGONNE, ILL.
ANL-7399 +. 1 PAGE, PAGE 94 OF ARGONNE NATIONAL LABORATORY. REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, NOVEMBER 1967, DECEMBER 1967

ON THE NOV. 12, 1967, STARTUP, AT 20 MW THE PRIMARY BULK-SODIUM TEMPERATURE BEGAN INCREASING AND COULD NOT BE CONTROLLED BY ADJUSTING SECONDARY FLOW. A MALFUNCTION OF THE STEAM-DRUM WATER-LEVEL CONTROLLER/ALARM ALLOWED THE DRUM TO BOIL DRY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, COMPONENT + EBR 1 AND 2 (RE) + CONTROLLER + REACTOR, LMCB + INCIDENT, EQUIPMENT + INSTRUMENT, LIQUID LEVEL + STEAM GENERATOR + INDEPENDENCE

17-022498 DATED 1958
EBR-1 CORE DISASSEMBLY AFTER MELTDOWN
ARGONNE NATIONAL LABORATORY
THIS IS A 13 MINUTE FILM REFERENCED ON PAGE 18 OF THE USAEC 16 MM FILM CATALOG (PROFESSIONAL LEVEL) FOR 1966-67, PRODUCED BY USAEC'S ARGONNE NATIONAL LABORATORY, FOR SALE BY BYRON MOTION PICTURES, AT \$42.31 PER PRINT, INCLUDING SHIPPING CASE, F.O.B. WASHINGTON, D. C.

PRESENTS SOME MAJOR ASPECTS OF THE REMOVAL AND SUBSEQUENT DISASSEMBLY OF THE CORE AFTER MELTDOWN, AND ILLUSTRATES HOT-LAB REMOTE-CONTROL TECHNIQUES USED TO SEPARATE AND RECOVER ENRICHED FUEL FROM THE BLANKET MATERIAL.

AVAILABILITY - AVAILABLE FOR LOAN (FREE) FROM USAEC HEADQUARTERS, FIELD LIBRARIES, AND ARGONNE NATIONAL LABORATORY, 9700 CASS AVE., ARGONNE, ILL. CLEARED FOR TELEVISION

REACTOR, BREEDER + EBR 1 AND 2 (RE) + HOT CELL + REACTOR, LMCB + CORE MELTDOWN + INCIDENT, RECOVERY FROM

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-022900 DATED 1966
SELECTED FERMII OPERATING EXPERIENCE - AUGUST 1966
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-36 +. 11 PAGES, DOCKET 50-16, AUGUST 1966

THIS REPORT IS ONE OF A SERIES. SEEMS ADEQUATE IN DEPTH. COVERS OPERATIONS, HEALTH PHYSICS, MAINTENANCE AND MODIFICATIONS, NONNUCLEAR TEST PROGRAM, AND FACILITY CHANGE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + MODIFICATION + REPORT, OPERATIONS SUMMARY + OPERATING EXPERIENCE

17-022902 DATED 1966
SELECTED FERMII OPERATING EXPERIENCES - DECEMBER 1966
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-40 +. 10 PAGES, DECEMBER 1966, DOCKET 50-16

THIS REPORT IS ONE OF A SERIES. SEEMS ADEQUATE IN DEPTH. COVERS OPERATIONS, HEALTH PHYSICS, MAINTENANCE AND MODIFICATIONS, NONNUCLEAR TEST PROGRAM, AND FACILITY CHANGE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM WASHINGTON, D. C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + MODIFICATION + REPORT, OPERATIONS SUMMARY + OPERATING EXPERIENCE

17-022903 DATED 1967
SELECTED FERMII OPERATING EXPERIENCES - FEBRUARY 1967
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-42 +. 10 PAGES, FEBRUARY 1967, DOCKET 50-16

THIS REPORT IS ONE OF A SERIES. SEEMS ADEQUATE IN DEPTH. COVERS OPERATIONS, HEALTH PHYSICS, MAINTENANCE AND MODIFICATIONS, NONNUCLEAR TEST PROGRAM, AND FACILITY CHANGE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + MODIFICATION + REPORT, OPERATIONS SUMMARY + OPERATING EXPERIENCE

17-022904 DATED 1967
SELECTED FERMII OPERATING EXPERIENCE - MARCH 1967
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-43 +. 11 PAGES, MARCH 1967, DOCKET 50-16

THIS REPORT IS ONE OF A SERIES. SEEMS ADEQUATE IN DEPTH. COVERS OPERATIONS, HEALTH PHYSICS, MAINTENANCE AND MODIFICATIONS, NONNUCLEAR TEST PROGRAM, AND FACILITY CHANGE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + MODIFICATION + REPORT, OPERATIONS SUMMARY + OPERATING EXPERIENCE

17-022905 DATED 1967
SELECTED FERMII OPERATING EXPERIENCE - APRIL 1967
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-44 +. 10 PAGES, APRIL 1967, DOCKET 50-16

THIS REPORT IS ONE OF A SERIES. SEEMS ADEQUATE IN DEPTH. COVERS OPERATIONS, HEALTH PHYSICS, MAINTENANCE AND MODIFICATIONS, NONNUCLEAR TEST PROGRAM, AND FACILITY CHANGE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + MODIFICATION + REPORT, OPERATIONS SUMMARY + OPERATING EXPERIENCE

17-022906 DATED 1967
SELECTED FERMII OPERATING EXPERIENCES - MAY 1967
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-45 +. 9 PAGES, DOCKET 50-16, AUGUST 30, 1967

THIS REPORT IS ONE OF A SERIES. SEEMS ADEQUATE IN DEPTH. COVERS OPERATIONS, HEALTH PHYSICS, MAINTENANCE AND MODIFICATIONS, NONNUCLEAR TEST PROGRAM, AND FACILITY CHANGE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + MODIFICATION +

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-022966 *CONTINUED*
REPORT, OPERATIONS SUMMARY + OPERATING EXPERIENCE

17-022967 DATED 1967
SELECTED FERMI OPERATING EXPERIENCES - JUNE 1967
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-46 +. 8 PAGES, DOCKET 50-16, AUGUST 30, 1967

THIS REPORT IS ONE OF A SERIES. SEEMS ADEQUATE IN DEPTH. COVERS OPERATIONS, HEALTH PHYSICS, MAINTENANCE AND MODIFICATIONS, NONNUCLEAR TEST PROGRAM, AND FACILITY CHANGE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + MODIFICATION + REPORT, OPERATIONS SUMMARY + OPERATING EXPERIENCE

17-022909 DATED 1967
SELECTED FERMI OPERATING EXPERIENCE - OCTOBER 1967
POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-50 +. 8 PAGES, DECEMBER 8, 1967, DOCKET 50-16

THIS REPORT IS ONE OF A SERIES. SEEMS ADEQUATE IN DEPTH. COVERS OPERATIONS, HEALTH PHYSICS, MAINTENANCE AND MODIFICATIONS, NONNUCLEAR TEST PROGRAM, AND FACILITY CHANGE.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + MODIFICATION + REPORT, OPERATIONS SUMMARY + OPERATING EXPERIENCE

17-022914 DATED 1967
MYD HD
CARBON REMOVAL FROM SRE SODIUM FOLLOWING PEP MODIFICATIONS
ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.
NAA-SR-12410 +. 73 PAGES, 19 FIGURES, 8 TABLES, NOV. 1, 1967

DESCRIBES AND EVALUATES CHEMICAL, MICROHARDNESS TRAVERSE, AND METALLOGRAPHIC METHODS FOR MEASURING SODIUM CARBURIZING EFFECTS ON 304 SS. AN ESTIMATED 43.3 LB OF CARBON WAS REMOVED, LEAVING ABOUT 1.5 LB. ADDITIONAL HOT-TRAPPING (AT 1200F) TIME TO REDUCE CONTENT TO GIVE CLADDING UNRESTRICTED SERVICE LIFE IS 2.7 MONTHS, AND TO PROTECT MAIN INTERMEDIATE HEAT EXCHANGER IS 0.8 TO 8 MONTHS. A SMALL BYPASS LOOP FOR EXPOSING SAMPLE SPECIMENS IS DESCRIBED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

CARBON + DECONTAMINATION + CLADDING + REACTOR, LMCR + STEEL + SRE (RE) + COOLANT CHEMISTRY + COOLANT PURIFICATION SYSTEM

17-023776 DATED 1968
FERMI RADIATION EMERGENCY PROCEDURES
POWER REACTOR DEVELOPMENT COMPANY
25 PAGES, FIGURES, FEBRUARY 14, 1968, DOCKET 50-16, TYPE--LMFBR, MFG.--APDA, AE--COMMONWEALTH ASSOC.

(LETTER TO DRL) REPLIES TO DRL LETTER OF DEC. 29, 1967, WITH FEB. 1968 UPDATINGS OF THE INSTRUCTIONS TO BE FOLLOWED IN THE EVENT OF FIRE (2 PG), NOTIFICATION OF RADIATION INCIDENTS (5 PG), AND RADIATION EMERGENCY PROCEDURE - OPERATING PROCEDURE 704 (25 PG). THE LATTER IS 5 PG OF GENERAL INSTRUCTIONS, 11 PG OF SPECIFIC INSTRUCTION FOR 4 CLASSES OF EMERGENCIES, AND 8 PG OF APPENDICES.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + EMERGENCY PROCEDURE

17-023946 ALSO IN CATEGORY 9 DATED 1967
BALL GL
RESULTS OF THE LOW-POWER NUCLEAR TEST PROGRAM FOR THE ENRICO FERMI FAST BREEDER REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-302 +. 128 PAGES, 29 FIGURES, 16 TABLES, REFERENCES, FEB. 1967

(PG 41) 1/M FUEL-LOADING PLOTS EXTRAPOLATED VERY CLOSE TO CRITICAL VALUE, HOWEVER WAS NOT USED FOR OFFICIAL PREDICTIONS BECAUSE LINEAR EXTRAPOLATION GAVE UNACCEPTABLE OVERESTIMATES. (PG 67) NEUTRON-DETECTOR GUIDE TUBES WERE FLUX-MAPPED (GAMMA AND NEUTRON) TO SET DETECTOR LOCATIONS. GAMMA PEAKS WERE FOUND NEAR SODIUM PIPES, AND NEUTRON PEAKS DUE TO BORON WINDOW AND TO TRANSITION DELTA. (PG 87) INDIVIDUAL SAFETY RODS WERE WORTH 15% LESS THAN PREDICTED, AND THE GROUP WAS SHADOWED BY 11%, GIVING AN \$8 WORTH. (PG 118) RELATIVELY LARGE INCREASES IN REACTIVITY FOLLOWING SHORT PERIODS WITH THE COOLANT SYSTEM ISOTHERMAL AT 700 F, ARE CALCULATED TO BE FROM PHASE-TRANSFORMATION SHRINKAGE (27 IN-HOURS) AND HYDRIDING OF THE CLAD (22 IN-HOURS).

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-023946 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + CONTROL ROD INTERACTION + CONTROL ROD WORTH + REACTIVITY EFFECT, EXPANSION + REACTOR, FAST
+ FERMI (LMFBR) + REACTOR, LMCR + INSTRUMENT CALIBRATION + REACTOR STARTUP TESTING + HYDRIDE + ZIRCALOY +
REACTIVITY EFFECT, ANOMALOUS + INSTRUMENT, ABNORMAL INDICATION

17-024647

DATED 1966

SUNDERMAN 10

HALLAM NUCLEAR POWER FACILITY MONTHLY OPERATING REPORT NO. 45

CONSUMERS PUBLIC POWER DISTRICT, HALLAM, NEBR.

TID-23023 +. 15 PAGES, APRIL 1966

THIS PROGRESS REPORT IS ONE OF A SERIES. SINCE THE REACTOR IS BEING DISMANTLED, THE MORE
APPROPRIATE INFORMATION RELATES TO MAINTENANCE, CORE-ELEMENT TRANSFERS, RADIATION SAFETY AND
CONTROL. INCLUDES SHORT SUMMARIES OF EACH DAYS WORK.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

17-025315

DATED 1968

HALLAM DECOMMISSIONING CONTINUES

CONSUMERS PUBLIC POWER DISTRICT

31 PAGES, FIGURES, TABLES, FROM HALLAM NUCLEAR POWER FACILITY MONTHLY RETIREMENT REPORT NO. 18, JANUARY 1968

WORK CONTINUED ON CUTTING OUT SECTIONS OF PIPE CONTAINING VARIOUS VALVES AND FITTING FOR EVENTUAL
REUSE. SODIUM LEFT UNDRAINABLE IN THE REACTOR VESSEL WILL BE CONTACTED WITH STEAM.

AVAILABILITY - USAEC, PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

17-025730

ALSO IN CATEGORY 18

DATED 1968

INFORMATION REQUESTED ON SEFOR EMERGENCY PLANS

USAEC, DIVISION OF REACTOR LICENSING

4 PAGES, MAY 27, 1968, LETTER TO K. P. COHEN FROM P. A. MORRIS, DOCKET 50-231

DRL HAS BEEN REVIEWING STATUS OF MEDICAL PLANS, INCLUDING FACILITIES AND PERSONNEL FOR TREATMENT
OF CONTAMINATED OR IRRADIATED PERSONNEL, AND REQUESTS INFORMATION ON (1) ON-SITE DECONTAMINATION
FACILITIES, (2) QUALIFICATIONS AND TRAINING OF LOCAL PHYSICIANS, (3) ARRANGEMENTS FOR
TRANSPORTING THE INJURED, (4) DISTANCE TO NEAREST HOSPITAL AGREEABLE TO ACCEPTING INJURED PEOPLE,
(5) QUALIFICATIONS AND TRAINING OF HOSPITAL PERSONNEL, (6) LIMITATIONS ON HOSPITAL AVAILABILITY,
(7) WRITTEN PLAN OF ACTION AT LOCAL HOSPITAL. ***SEPARATE LETTER INDICATES TRAINING SEMINAR THAT
LOCAL MEDICAL PERSONNEL CAN ATTEND.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

NRC QUESTION + REACTOR, FAST + REACTOR, LMCR + RADIATION INJURY TREATMENT + EMERGENCY PROCEDURE + REACTOR,
PULSED + SEFOR (RE) + RADIOLOGICAL ASSISTANCE + OPERATING LICENSE PROCESS

17-025901

DATED 1968

FERMI REMOVES ZIRCONIUM PLATE

POWER REACTOR DEVELOPMENT COMPANY

PROC-EF-55 +. 7 PAGES, POWER REACTOR DEVELOPMENT COMPANY ENRICO FERMI ATOMIC POWER PLANT REPORT FOR MARCH
1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

THE ZIRCONIUM PLATE WAS REMOVED FROM THE REACTOR ON MARCH 22. THE 200-MR/HR SURFACE RADIATION WAS
REDUCED TO 2 MR/HR BY DECONTAMINATION. DESPITE THE MOCKUP WORK, THE REMOTE-HANDLING TOOLS NEEDED
MINOR MODIFICATION. A PATCH CUT FROM PRIMARY PIPING WAS UNWETTED BY SODIUM, AND THE BETA
RADIATION (150 MR/HR) WAS REDUCED TO BACKGROUND BY DECONTAMINATION.

AVAILABILITY - USAEC, PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

CONTAINMENT + DECONTAMINATION + FERMI (LMFBR) + REPORT, OPERATIONS + BETA EMITTER + REMOTE MANIPULATING AND
VIEWING + FLOW BLOCKAGE + MAIN COOLING SYSTEM + REACTOR, LMCR + PIPES AND PIPE FITTINGS

17-025935

ALSO IN CATEGORY 18

DATED 1968

MACPHERSON RE

FAST REACTOR SAFETY CONFERENCE

OAK RIDGE NATIONAL LABORATORY, OAK RIDGE, TENNESSEE

8 PAGES, TABLE, REFERENCE, NUCLEAR SAFETY, 9(3), PAGES 202-209 (MAY-JUNE, 1968)

BRIEFLY SUMMARIZES THE SESSIONS COVERING (1) CORE MELTDOWN ACCIDENTS, (2) SODIUM BOILING AND
SUPERHEATING, (3) INCIDENTS IN REACTORS, AND (4) SAFETY. MALFUNCTIONS IN THE BRS WERE (1) FUEL-

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-025935 *CONTINUED*

ELEMENT LEAKS CAUSED BY HEATER SHORTING, PLUS AIR IN LEAKAGE CAUSING MASSIVE SODIUM OXIDE PLUGGING, AND (2) REACTIVITY PERTURBATIONS DUE TO ENTRAINMENT OF ARGON BY THE SODIUM STREAM. ***PLUGGING OF SUBASSEMBLIES BY OXIDE HAS BEEN DETECTED PRIMARILY BY THE EFFECT ON PRIMARY FLOW. A PANEL DISCUSSION SHOWED THAT ATTENTION IS SWITCHING FROM THE LARGE, LESS PROBABLE ACCIDENT TO THE SMALLER, MORE FREQUENT ACCIDENTS WHICH AFFECT PLANT USE.

ACCIDENT ANALYSIS + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + SODIUM + INCIDENT + SAFETY PRINCIPLES AND PHILOSOPHY + REVIEW + FLOW BLOCKAGE + COOLANT QUALITY

17-025979

DATED 1968

FERMI VESSEL RADIATION SURVEY

POWER REACTOR DEVELOPMENT COMPANY

PRDC-EF-54 +. 6 PAGES, FEBRUARY 1968, POWER REACTOR DEVELOPMENT COMPANY ENRICO FERMI ATOMIC POWER PLANT REPORT FOR FEBRUARY 1968

A SURVEY REVEALS THAT THE GAMMA FIELD IN THE REACTOR RANGES FROM 1500 R/HR AT THE CORE CENTER TO 1 R/HR HALFWAY UP INTO THE ROTATING SHIELD PLUG.

FERMI (LMFBR) + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + PRESSURE VESSELS + SURVEY, RADIATION + REACTOR, LMFBR

17-025956

DATED 1968

FERMI REPORTS CORROSION FAILURE OF LOW LEVEL WASTE PIPES

POWER REACTOR DEVELOPMENT COMPANY, DETROIT, MICHIGAN

3 PAGES, LETTER TO P.A. MORRIS FROM W.J. MCCARTHY, JUNE 4, 1968, DOCKET NO. 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

ON APRIL 30, A 2-IN.-DIA. CARBON-STEEL GAS-WASTE PIPE WAS FOUND CORRODED THROUGH, 4 FT BELOW GROUND. THE TWO GAS PIPES ARE IN AN 8-IN.-DIA. CARBON-STEEL CONTAINMENT PIPE DEEP UNDER GROUND. THE 8-IN.-DIA. PIPE IS BADLY CORRODED AND MUST BE PERMANENTLY REPAIRED. THE 2-IN. PIPE WAS UNPROTECTED WHERE IT LEFT THE 8-IN. ONE, BUT THE NEW 2-IN. PIPE IS NOW COVERED WITH BITUMASTIC. ***A LIQUID-WASTE PIPE WAS FOUND CORRODED THROUGH IN SEVERAL SPOTS DUE TO LOCAL BITUMASTIC FAILURE. ***SINCE NOV. 1966, THESE PIPES HAVE CARRIED ONLY LOW-LEVEL WASTE. THE FAR-BLDG. LIQUID-WASTE PIPE (SS IN A CARBON-STEEL CONTAINMENT PIPE) WILL BE INSPECTED.

AVAILABILITY - PUBLIC DOCUMENT ROOM

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + FAILURE, PIPE + WASTE HANDLING + REACTOR, POWER + CORROSION + REACTOR, LMFBR

17-026099

DATED 1968

FERMI FLOW BLOCKAGE OBJECT IDENTIFIED

POWER REACTOR DEVELOPMENT COMPANY, DETROIT, MICHIGAN

PRDC-EF-53 +. 9 PAGES, TABLES, DOCKET NO. 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

UNIDENTIFIED FOREIGN OBJECT HAS NOW BEEN IDENTIFIED AS A 1/6 SEGMENT OF ZIRCONIUM SHEET COVERING A CONICAL FLOW GUIDE UNDER THE CORE. SHOWN ON HSR FIGURES 102.2 AND 102.9. IT DOES NOT APPEAR ON ANY CONSTRUCTION PRINTS BUT WAS LOCATED FROM CONSTRUCTION PHOTOGRAPHS. THROUGH ALL THE LOWER-PLENUM PHOTOGRAPHY, NO ATTENTION WAS PAID TO THE FLOW GUIDE. BORESCOPE NOW FINDS 1 SEGMENT LOOSE AND A SCREW MISSING ON ANOTHER.

AVAILABILITY - USAEC, PUBLIC DOCUMENT ROOM

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + REPORT, OPERATIONS + FAILURE, ADMINISTRATIVE CONTROL + FLOW BLOCKAGE + REACTOR, LMFBR

17-026164

DATED 1967

HNPFC ENVIRONMENTAL MONITORING PROGRAM

CONSUMERS PUBLIC POWER DISTRICT

12 PAGES, 10 TABLES, TWENTH-FOURTH QUARTERLY REPORT, APRIL, MAY, JUNE 1967, DOCKET NO. 115-3, TYPE--PWR, MFG--B+W, AE--CON ED

PRESENTS DATA FROM ENVIRONMENTAL SAMPLES ACCUMULATED DURING APRIL, MAY, AND JUNE 1967. BRIEF DISCUSSIONS ARE GIVEN ON SAMPLING TECHNIQUES, ANALYTICAL CRITERIA, AND COUNTING CRITERIA.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

COUNTER + EFFLUENT + HALLAM (LMK) + REACTOR, LMCR + WATER + SAMPLING + AIR + ANALYTICAL TECHNIQUE + MONITOR, ENVIRONMENTAL + GROSS ALPHA + GROSS BETA + SEDIMENT

17-026203

DATED 1968

HALLAM ENVIRONMENTAL SURVEY PROGRAM-APRIL 67 THROUGH MARCH 68

CONSUMERS PUBLIC POWER DISTRICT, HALLAM, NEBRASKA

12 PAGES, LETTER TO P.A. MORRIS FROM R.E. BUNTAIN, JUNE 3, 1968, DOCKET NO. 115-3, TYPE--LMR, MFG--A.I., AE--BECHTEL

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-026203 *CONTINUED*

A SET OF NUMBERS (GROSS ALPHA AND GROSS BETA) FOR AIR-SAMPLE DATA AT PLANT SITE AND AT CITY OF HALLAM, AND FROM PLANT RAW-WATER SUPPLY AND RETENTION PONDS AND SOIL.

AVAILABILITY-USAEC, PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

EFFLUENT + HALLAM (LMR) + REACTOR, LMCR + WATER + SAMPLING + AIR + MONITOR, ENVIRONMENTAL + GROSS BETA + GROSS GAMMA + SOIL + SEDIMENT

17-026300

DATED 1967

REACTOR ENGINEERING DIVISION ANNUAL REPORT

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ANL-7290 +. 234 PAGES, FIGURES, TABLES, REFERENCES, APRIL 1967

TOPICS DISCUSSED ARE (1) EBR-II PLANT IMPROVEMENTS AND EXPERIMENTAL SUPPORT, (2) AAKR TITLE-1 DESIGN AND SUPPORTING RESEARCH, (3) EBWR REPAIRS AND REACTOR KINETICS, (4) NUCLEAR ROCKET PROGRAM, (5) FAST-REACTOR R AND D, (6) LIQUID-METAL HEAT-TRANSFER PROGRAM, (7) MAGNETOHYDRODYNAMICS, (8) REACTOR SAFETY RESEARCH PROGRAM COVERING COOLANT DYNAMICS, REACTOR CONTROL AND STABILITY, FUEL MELTDOWN STUDIES, AND CONTAINMENT, AND (9) FEASIBILITY STUDY OF ASSS, USING 10,000-MW SODIUM-COOLED FAST BREEDER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + CONTAINMENT + EBR 1 AND 2 (RE) + REACTOR, FAST + FLOW THEORY AND EXPERIMENTS + FUEL MELTDOWN + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR STABILITY + REACTOR, RESEARCH + AARR (RR) + REACTOR CONTROL + NUCLEAR ROCKET + REVIEW + R AND D PROGRAM

17-026301

DATED 1967

SPENT FUEL MONITOR - SAFEGUARD AGAINST LOADING ERROR

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ANL-7290 +. 3 PAGES, FIGURE, PAGES 13-15 OF THE REACTOR ENGINEERING DIVISION ANNUAL REPORT, APRIL 1967

BRIEFLY DESCRIBES A GAMMA SCANNING DEVICE TO DIFFERENTIATE BETWEEN A SPENT AND REPROCESSED ELEMENT FOR EBR-II DURING REFUELING OPERATIONS. GAMMA CHAMBER WILL BE RAISED OR LOWERED 60 IN. TO OBTAIN RADIATION PROFILE (THAT FOR A SPENT UNIT WILL BE TYPICALLY COSINE IN SHAPE). EVERY UNIT IS TO BE SCANNED BEFORE TRANSFER INTO REACTOR. TOO HIGH A FUEL BURNUP COULD LEAD TO FUEL FAILURE. CONCEPTUAL DESIGN COMPLETED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + EBR 1 AND 2 (RE) + FUEL BURNUP + FUEL HANDLING + REACTOR, LMCR + SODIUM + INSTRUMENT, FUEL SCANNING + REACTOR, ERDA OWNED + REFUELING + GAMMA + SYSTEM DESCRIPTION + RADIATION MONITORS + FUEL ELEMENTS

17-026398

DATED 1963

KOPCHINSKI NL

REMOVAL OF A MAIN HEAT TRANSFER SYSTEM THROTTLE VALVE AT THE HALLAM NUCLEAR POWER FACILITY

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-8492 +. 17 PAGES, 9 FIGURES, MAY 25, 1963

REPORTS THE REMOVAL AND REPAIR OF A PRIMARY-LOOP THROTTLE VALVE FOR REFERENCE IN EVENT SUCH IS REPEATED IN FUTURE. VALVE BECAME INOPERABLE, AND DEEP SCORE MARKS WERE NOTED ON STEM. NO REASON FOR BINDING GIVEN. NEW VALVE STEM INSTALLED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, EQUIPMENT + HALLAM (LMR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + MAIN COOLING SYSTEM + VALVES

17-026408

DATED 1968

DICKERMAN CE

HIGHLIGHTS OF THE AIX CONFERENCE ON FAST REACTOR SAFETY

3 PAGES, 35 REFERENCES, REACTOR AND FUEL-PROCESSING TECHNOLOGY, 11(1), PAGES 1-3 (WINTER, 1967-1968)

VERY BRIEF SUMMARY OF TOPICS DISCUSSED. MUCH ATTENTION DEVOTED TO DESCRIBING AND CALCULATING TRANSIENT SODIUM BOILING AND EXPULSION BY VAPORIZATION. SESSION ON OPERATING EXPERIENCE INCLUDED FOLLOWING - (1) DOUNREAY - NAK LEAK WAS IN ONE OF 24 LINKS BETWEEN PRIMARY COOLANT CIRCUIT AND THE REACTOR VESSEL, (2) RAPSODIE - PROBLEMS IN STARTUP INCLUDED THERMAL-STRESS DIFFICULTIES IN ORIGINAL DESIGN, (3) BR-5 - EXTENSIVE OXIDE PLUGGING EXPERIENCED AFTER EXPOSURE OF DRAINED CORE TO AIR, GAS ENTRAPMENT, AND LEAKS IN HEAT EXCHANGER, (4) FERMI - PARTIAL MELTDOWN DUE TO OBJECT IN INLET PLENUM BRIEFLY COVERED.

FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + SODIUM + STRESS + REACTOR, TEST + THERMAL MECHANICAL EFFECT + REACTOR STARTUP EXPERIENCE + DOUNREAY (LMFBR) + FLOW BLOCKAGE + MAIN COOLING SYSTEM + MATHEMATICAL TREATMENT + LEAK + OPERATING EXPERIENCE + BOILING

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-027595 ALSO IN CATEGORIES 5 AND 9 DATED 1968
HARRIGAN PM + MATTEL J + TOTH WL
OPTICAL, LIGHTING, AND PHOTOGRAPHIC EQUIPMENT FOR THE ENRICO FERMI ATOMIC POWER PLANT REPAIR PROGRAM
POWER REACTOR DEVELOPMENT COMPANY + ATOMIC POWER DEVELOPMENT ASSOCIATES
1 PAGE, 1 FIGURE, ANS TRANSACTIONS 11(1), PAGE 358, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND
CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

A MELTDOWN OF TWO CORE SUBASSEMBLIES REQUIRED THE USE OF VERY LONG OPTICAL INSTRUMENTS AND LIGHTING EQUIPMENT FOR THE PURPOSE OF INSPECTION OF THE CORE AND REACTOR COMPONENTS. BECAUSE LITTLE INFORMATION WAS AVAILABLE ON THE DESIGN OF SUITABLE VIEWING AND LIGHTING EQUIPMENT FOR USE IN SODIUM REACTOR ENVIRONMENTS, MUCH DEVELOPMENTAL WORK AND TESTING HAD TO BE DONE TO PROVIDE SUITABLE EQUIPMENT. MUCH DEVELOPMENT WAS REQUIRED TO PROVIDE LIGHTING EQUIPMENT WITH ADEQUATE LAMP LIFE AND MANEUVERABILITY. THE USE OF PHOTOGRAPHY THROUGH THE VIEWING INSTRUMENTS WAS NOT SUCCESSFUL INITIALLY, BUT IMPROVEMENTS BOTH IN TECHNIQUE AND EQUIPMENT LED TO REMARKABLY GOOD RESULTS IN THE LATTER PART OF THE PROGRAM.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + INSTRUMENT, IN CORE + REACTOR, LMCR + MAINTENANCE AND REPAIR + INSTRUMENT, SURVEILLANCE + EQUIPMENT DESIGN + REMOTE MANIPULATING AND VIEWING + MAINTENANCE, REMOTE

17-027596 ALSO IN CATEGORIES 5 AND 9 DATED 1968
DUFFY JG + MATTE J + RYAN JD + WARBERG HS
OBJECT RETRIEVAL TOOLS FOR THE ENRICO FERMI ATOMIC POWER PLANT REPAIR PROGRAM
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN
3 PAGES, 2 FIGURES, ANS TRANSACTIONS 11(1), PAGES 358-360, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

THE METHOD OF RETRIEVING FROM THE BOTTOM OF THE FERMI REACTOR AN OBJECT THAT CAUSED A MELTDOWN INVOLVED PROCEDURES AND TOOLS WHICH ARE UNIQUE AND COULD HAVE APPLICATION IN FUTURE PROBLEMS. TOOLS WERE DEVELOPED TO CUT THE OBJECT INTO PIECES SMALL ENOUGH TO BE REMOVED UPWARD THROUGH THE SUPPORT PLATE. SEVERAL GRIPPERS FOR THIS PURPOSE WERE ALSO DEVELOPED. SIMULTANEOUSLY, SPECIAL TOOLS WERE DEVELOPED TO REMOVE THE OBJECT INTACT VIA THE SODIUM INLET PIPE.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + INSTRUMENT, IN CORE + REACTOR, LMCR + MAINTENANCE AND REPAIR + EQUIPMENT DESIGN + REMOTE MANIPULATING AND VIEWING + MAINTENANCE, REMOTE

17-027774 ALSO IN CATEGORIES 6 AND 9 DATED 1968
ZINIUK MA + WILBER HA
SCRAM TESTS TO DETERMINE SHUTDOWN TRANSIENT CHARACTERISTICS OF THE ENRICO FERMI REACTOR AND ASSOCIATED HEAT TRANSPORT SYSTEM
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-NTS-15 +. 45 PAGES, 8 FIGURES, 7 TABLES, 4 REFERENCES, APRIL 1968

REACTOR OUTLET TEMPERATURE FLUCTUATION AFTER 67 MWH OPERATION BEGAN 15 SEC AFTER SCRAM AND LASTED TIL 220 SEC. THE RATE WAS 44 F/SEC, OVER A CHANGE OF 76 F. TIME CONSTANTS AND TRANSPORT LAGS WERE DETERMINED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + SCRAM, REAL + RESPONSE TIME + INSTRUMENT, TEMPERATURE + REPORT, OPERATIONS ANALYSIS + TEST, INSTRUMENT RESPONSE + MAIN COOLING SYSTEM + REACTOR, LMFBR + THERMAL TRANSIENT

17-027775 ALSO IN CATEGORY 6 DATED 1967
MUELLER RE + WILBER HA
POWER COEFFICIENT MEASUREMENTS FOR THE ENRICO FERMI REACTOR. ENRICO FERMI ATOMIC POWER PLANT NUCLEAR TEST SERIES
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-NTS-9 +. 75 PAGES, 10 FIGURES, 12 TABLES, 22 REFERENCES, NOV. 1967

TESTS WERE MADE OF CONTROL ROD POSITION RESPONSE TO POWER INCREASE/DECREASE, WITH 2 OR 3 LOOP FLOW, FOR POWERS OF 1.3, 2.0, 13.3, 20, 66.7 AND 100 MWH. COEFFICIENTS ARE ABOUT 28% SMALLER THAN PREDICTED. FLOW DEPENDENCE SHOWS VERY LITTLE FEEDBACK INDEPENDENT OF FLOW. THE 20-MWH 3-LOOP COEFFICIENT VALUE IS STILL INCONSISTANT. RELIABLE INLET TEMPERATURE DATA HAVE ELIMINATED THE HYSTERESIS EFFECT BETWEEN POWER INCREASE/DECREASE. CHANGES IN ANALYSIS WERE - INCREASED VALUE OF COMPONENTS DEPENDENT ON CORE SODIUM WORTH, DECREASE IN CORE FUEL PIN AXIAL EXPANSION COEFFICIENT, DECREASE IN RADIAL EXPANSION COEFFICIENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTIVITY EFFECT, EXPANSION + FERMI (LMFBR) + POWER COEFFICIENT + COMPARISON, THEORY AND EXPERIENCE + REPORT, OPERATIONS ANALYSIS + TEST, PHYSICS + FLOW STABILITY + REACTOR, LMFBR

17-027776 DATED 1967
COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT, DECEMBER 1967
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH. POWER REACTOR DEVELOPMENT CO., DETROIT, MICH.
DETROIT EDISON CO., MICH.
APDA-CFE-17 +. 37 PAGES, 11 FIGURES, DEC. 1967

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-027776 *CONTINUED*

DIMENSIONS OF THE UNIDENTIFIED FOREIGN OBJECT WERE ESTIMATED AND SKETCHES REPRODUCED HERE. TOOLS AND VIEWING DEVICES WERE MODIFIED BASED ON USAGE. A SAMPLE OF THE OBJECT TAKEN. AN INDEX TO PAST REPORTS 1 THRU 17 IS MADE, USING OPERATIONALLY SIGNIFICANT TERMS LIKE GALLING.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + REPORT, OPERATIONS + REMOTE MANIPULATING AND VIEWING + REACTOR, LMFBR

17-027844

DATED 1968

HALLAM DECOMMISSIONING EXPERIENCE

CONSUMERS PUBLIC POWER DISTRICT

69 PAGES, FIGURES, TABLES, HNPF MONTHLY RETIREMENT REPORT 21, APRIL 1968, DOCKET 115-3, TYPE--LMR, MFG--A.I., AE--BECHTEL

RESIDUAL SODIUM IN REACTOR VESSEL AND MAIN LOOP WAS REACTED WITH STEAM IN A NITROGEN ATMOSPHERE. INTERMEDIATE HEAT EXCHANGE 3A WAS SHIPPED TO IDAHO, AND THE OTHER STEAM GENERATORS AND PUMPS WERE PREPARED FOR SHIPMENT.

USAEC, PUBLIC DOCUMENT ROOM, 1717 H ST., WASHINGTON, D.C. 20545 (25 CENTS/PAGE)

CHEMICAL REACTION + HALLAM (LMR) + REACTOR, LMR + REPORT, OPERATIONS + SODIUM + DECOMMISSIONING + MAIN COOLING SYSTEM + HEAT EXCHANGERS

17-028562

ALSO IN CATEGORY 6

DATED 1967

WINKLEBLACK RK

RECENT EBR-II OPERATING EXPERIENCE

ARGONNE NATIONAL LABORATORY

15 PAGES, 4 FIGURES, PAPER PRESENTED AT THE 1967 WINTER MEETING OF THE AMERICAN NUCLEAR SOCIETY, CHICAGO, ILLINOIS, NOVEMBER 5-9, 1967

DISCUSSES A FUEL FAILURE AND FISSION PRODUCT RELEASE TO COOLANT AND A SIGNIFICANT DECREASE IN THE POWER COEFFICIENT FOLLOWING A CHANGE IN CORE CONFIGURATION. DESCRIBES ANALYTICAL TECHNIQUE USED TO LOCATE THE FAULTY FUEL. NEED FOR A FAILED FUEL LOCATING METHOD IS DISCUSSED, AND USE OF A TAG MATERIAL IS CONSIDERED. DECREASE IN POWER COEFFICIENT AFTER CORE ENLARGED FROM 73 TO 81 ASSEMBLIES IS ATTRIBUTED TO CHANGE IN CORE AND REFLECTOR SIZE AND COMPOSITION, AND AN INCREASED BOWING OF SUBASSEMBLIES CAUSED BY INSTALLATION OF STAINLESS STEEL SUBASSEMBLIES.

ACTIVITY BUILDUP + EBR 1 AND 2 (RE) + REACTOR, FAST + FUEL ELEMENT BOWING + REACTOR, LMR + POWER COEFFICIENT + REACTOR, RESEARCH + REACTOR, TEST + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + REACTOR, ERDA OWNED + FAILURE, CLADDING + REACTIVITY EFFECT, ANOMALOUS + MAIN COOLING SYSTEM + OPERATING EXPERIENCE

17-028643

DATED 1968

FERMI MONTHLY REPORT MAY 1968

POWER REACTOR DEVELOPMENT COMPANY

EF-57 +. 9 PAGES, MAY 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

THE LATHE TURNINGS RECOVERED IN APRIL ARE 6061 ALUMINUM SCRAPED FROM BURESCOPES AND OTHER TOOLS. THE MIRROR DAMAGED A FUEL ELEMENT ORIFICE PLATE. 4 ZIRCALOY PLATES STILL TO BE RECOVERED. ACCEPTANCE TESTS BEGAN FOR A1 FUEL TRANSPORT MACHINE. *** CORROSION FAILURE OF (1) CARBON STEEL GUARD PIPE CARRYING WASTE GAS PIPES, AND (2) A BITUMASTIC COVERED LIQUID WASTE PIPE AT LOCAL BITUMASTIC FAILURES. WERE REPAIRED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

FERMI (LMFBR) + REPORT, OPERATIONS + FAILURE, PIPE + WASTE TREATMENT, LIQUID + WASTE TREATMENT, GAS + CORROSION + REMOTE MANIPULATING AND VIEWING + DAMAGE + ORIFICE + REACTOR, LMFBR

17-028644

DATED 1968

FERMI MONTHLY OPERATIONS REPORT, APRIL 1968

POWER REACTOR DEVELOPMENT COMPANY

EF-56 +. 7 PAGES, APRIL 1968, DOCKETS 50-16, TYPE-LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

THE ZIRCONIUM PLATE WAS REMOVED LATE IN MARCH. 150 PICTURES TAKEN OF MELTDOWN PAN AND ZIRCONIUM SEGMENTS. FOUR OBJECTS WERE RECOVERED (TWO SMALL LATHE TURNINGS AND 2 OXIDE/CARBON DEPOSITS). ZIRCONIUM PLATE CONTAINED 400 PPM HYDROGEN WAS 76B ROCKWELL HARDNESS. PRIMARY SODIUM ACTIVITY WAS 0.0098 MICROCURIE/GRAM.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

FERMI (LMFBR) + HYDROGEN + REPORT, OPERATIONS + ZIRCONIUM + EMBRITTLEMENT + INCIDENT, RECOVERY FROM + REMOTE MANIPULATING AND VIEWING + REACTOR, LMFBR + OBSTRUCTION

17-028796

ALSO IN CATEGORY 10

DATED 1967

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-028796 *CONTINUED*

CUSTELLO RH

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT, NOVEMBER 1967
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH. POWER REACTOR DEVELOPMENT CO., DETROIT, MICH.
DETROIT EDISON CO., MICH.

APDA-CFE-16 +. 45 PAGES, 25 FIGURES, 2 TABLES, JANUARY 1968

MORE DESCRIPTION OF TOOLS FOR CUTTING UP AND REMOVING UNIDENTIFIED FOREIGN OBJECTS FROM THE REACTOR VESSEL. OVERCROWDED CABLE TRAYS IN THE HIGH AMBIENT TEMPERATURE NEAR SODIUM PIPING CAUSED HEATER FEEDER WIRES INSULATION TO CRACK. CABLES WERE REROUTED AND REPLACED WITH ASBESTOS INSULATED CABLES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, 22151, \$3.00 COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + REPORT, OPERATIONS + REMOTE MANIPULATING AND VIEWING + DAMAGE + ELECTRICAL CONDUCTION + REACTOR, LMFBR

17-028798

DATED 1967

MCCARTHY JF

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT. MONTHLY REPORT NO. 15, OCTOBER 1967

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH. POWER REACTOR DEVELOPMENT CO., DETROIT, MICH.
DETROIT EDISON CO., MICH.

APDA-CFE-15 +. 31 PAGES, FIGURES, TABLES, DECEMBER 1967

SEVERAL PICTURES TAKEN OF UNIDENTIFIED FOREIGN OBJECT FOUND IN REACTOR VESSEL INCLUDED. WAYS TO CUT UP AND REMOVE OBJECT DESCRIBED. PHOTOS OF MELTED CORE ASSEMBLY M-098 INCLUDED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, 22151, \$3.00 COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + FUEL MELTDOWN + REPORT, OPERATIONS + REMOTE MANIPULATING AND VIEWING + REACTOR, LMFBR + OBSTRUCTION

17-028801

ALSO IN CATEGORY 6

DATED 1967

KLICKMAN AE + HORNE RE + WILBER HA

OSCILLATOR TESTS IN THE ENRICO FERMI REACTOR

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

APDA-NTS-11 +. 145 PAGES, 71 FIGURES, 24 TABLES, 10 REFERENCES, AUGUST 1967

ROTATING RODS WHICH WOULD VARY REACTIVITY BY 6 OR BY 2 CENTS BETWEEN 0.001 AND 10 CPS WERE USED, TOGETHER WITH A SYNCHRONOUS AND A NULL BALANCE TRANSFER FUNCTION ANALYZER, AS WELL AS SIMPLE RECORDERS. NEGLIGIBLE DIFFERENCES EXIST FOR SPACE DEPENDENT CALCULATIONS. SYNCHRONOUS ANALYZER SEEMED BEST, BUT TRUE FEEDBACK OBSERVABLE ONLY AT HIGHER POWERS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + OSCILLATOR, REACTIVITY + REACTOR STABILITY + SPACE DEPENDENT DYNAMICS + TRANSFER FUNCTION + TEST, PHYSICS + REACTOR, LMFBR

17-028802

ALSO IN CATEGORY 6

DATED 1967

SEELEY GT + BALL GL + BRANYAN CE + HORNE RE + MUELLER RE + SEGAL BM + WILBER HA

MEASUREMENT OF SOME REACTIVITY EFFECTS IN THE ENRICO FERMI REACTOR

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

APDA-NTS-12 +. 68 PAGES, 18 FIGURES, 13 TABLES, 3 REFERENCES, NOVEMBER 1967

REACTIVITY WORTH OF PERMANENT NEUTRON SOURCE (2.3 CENTS AT 10 WATTS, AND 0.23 CENTS AT 1 KW), OF RETRACTABLE NEUTRON SOURCE (8.3 CENTS), OF COOLANT FLOW RATE (NO GAS ENTRAINMENT AND NEGLIGIBLE STRUCTURAL DEFORMATION) AND PRESSURE COEFFICIENT (NONE AT 15 AND 25 PSIA), AND OF TEMPORARY INSTRUMENT THIMBLE (1.41 CENTS) WERE MADE AND EVALUATED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

PRESSURE, EXTERNAL + FERMI (LMFBR) + INSTRUMENT, IN CORE + REACTIVITY EFFECT + SOURCE, NEUTRON + COOLANT COEFFICIENT + TEST, PHYSICS + FLOW + REACTOR, LMFBR

17-029040

ALSO IN CATEGORY 9

DATED 1968

BATCH ML + KLICKMAN AE

EVALUATION OF NOISE ANALYSIS FOR THE ENRICO FERMI REACTOR

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.

APDA-NTS-13 +. 90 PAGES, FIGURES, TABLES, REFERENCES, JAN. 1968

THE ANALYSIS OF RESULTS COVERED THE FOLLOWING ASPECTS - (1) SEARCH FOR INDICATIONS OF INSTABILITY, (2) EXAMINATION OF POWER SPECTRAL DISTRIBUTION OF VARIOUS SIGNALS, (3) MEASUREMENT OF THE AMPLITUDE OF THE TRANSFER FUNCTION BETWEEN VARIOUS PARAMETERS, (4) MEASUREMENT OF THE TIME LAG BETWEEN THE VARIOUS PARAMETERS, AND (5) MEASUREMENT OF THE RELATIVE AMOUNT OF COHERENT SIGNAL

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-029046 *CONTINUED*

BETWEEN INPUT AND OUTPUT PARAMETERS. THE STUDY INDICATES THAT THROUGH NOISE ANALYSIS MEANINGFUL RESULTS CAN BE OBTAINED IN EACH OF THESE ASPECTS AND POINTS UP CHANGES IN RECORDING TECHNIQUES AND SELECTION OF PARAMETERS WHICH COULD LEAD TO IMPROVED ACCURACY, RELIABILITY, AND UNDERSTANDING OF FUTURE WORK.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + TRANSFER FUNCTION + NOISE ANALYSIS + INSTABILITY + CORRELATION

17-029048

DATED 1968

QUARTERLY TECHNICAL PROGRESS REPORT LIQUID METAL ENGINEERING CENTER (LMEC)
ATOMICS INTERNATIONAL
LMEC-68-20 +. 112 PAGES, FIGURES, TABLES, REFERENCES, APRIL-JUNE 1968

ONE OF A SERIES OF QUARTERLY REPORTS. TOPICS INCLUDE OPERATION OF THE SODIUM COMPONENT TEST INSTALLATION, DESIGN OF COMPONENTS FOR TEST FACILITY, PUMP BEARING TEST PROGRAM, PROGRESS IN DEVELOPMENT OF PRECISION MECHANISMS, MODIFICATION OF AUXILIARY COMPONENT TEST INSTALLATION, INSTRUMENTATION, HIGH TEMPERATURE STRAIN MEASUREMENTS, AND MAINTENANCE OF IDLE EQUIPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

MAINTENANCE AND REPAIR + SODIUM + MEASUREMENT, STRAIN GAGE + COMPONENTS + R AND D PROGRAM + REACTOR, LMFBR + INSTRUMENTS, MISC.

17-029297

ALSO IN CATEGORY 5

DATED 1967

GAJAC G + REYNES L

EXPERIENCE GAINED FROM THE FINAL CONSTRUCTION PHASE AND THE APPROACH OF POWER OF RAPSODIE
CENTRE D ETUDES NUCLEAIRES DE CADARACHE, FRANCE

22 PAGES, 11 FIGURES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPT. 1967

ANALYZES TWO SODIUM LEAKS (OCTOBER 1966). (1) TOO RAPID PREHEATING OF PIPING WHICH HAD TWO COLD SPOTS CAUSED IT TO BURST. AMONG CAUSES ARE FILLING LINE WITH SODIUM TOO SOON (PREHEATING COULD NOT BE ALLOCATED). LEAK INDICATORS DID NOT WORK. DOUBLE WALLS NOW FILLED ALL OVER SYSTEM. (2) UNTIMELY OPENING OF AN ARGON VALVE PRESSURIZED THE SODIUM, DRIVING IT OUT THROUGH CONTROL-ROD-DRIVE PLUGS NOT BOLTED (COMMUNICATION FAILURE). VALVE MADE SLOWER ACTING, SAFETY VALVES MADE LARGER, EQUALIZING OF DRIVES AND COVER GAS PRESSURES DONE. UNCORRECTED, THIS INCIDENT WOULD HAVE BEEN REPEATED YEARLY. ***RAPSODIE INCIDENTS ARE ALL RECORDED AND STUDIED.

AVAILABILITY - G. GAJAC OR L. REYNES, COMMISSARIAT A L'ENERGIE ATOMIQUE - CEN/CADARACHE - DRP

FRANCE + FAILURE, OPERATOR ERROR + REACTOR, LMCR + FAILURE, PIPE + REACTOR, SAFETY RESEARCH + THERMAL MECHANICAL EFFECT + INCIDENT, EQUIPMENT + MODIFICATION + REPORT, OPERATIONS ANALYSIS + FAILURE, DESIGN ERROR + INFORMATION RETRIEVAL + MAIN COOLING SYSTEM + LEAK + PRESSURE TRANSIENT

17-029548

ALSO IN CATEGORY 7

DATED 1968

MCCARTHY WJ + JENS WH

REVIEW OF THE FERMI REACTOR FUEL DAMAGE INCIDENT AND A PRELIMINARY ASSESSMENT OF ITS SIGNIFICANCE TO THE DESIGN AND OPERATION OF SODIUM COOLED FAST REACTORS
POWER REACTOR DEVELOPMENT CO., MCNNDE, MICH.

22 PAGES, BULL. INFORM. SCI. TECH. (PARIS), NO. 124, PAGES 39-50 (MARCH 1968) (IN FRENCH)

THE STATUS OF THE INVESTIGATION OF THE EXTENT AND CAUSE OF THE OCTOBER 5, 1966, FUEL DAMAGE AT THE ENRICO FERMI SODIUM COOLED FAST BREEDER REACTOR IS GIVEN AS OF SEPTEMBER 13, 1967. A FOREIGN OBJECT HAS BEEN FOUND IN THE CORE INLET PLENUM WHICH IS BELIEVED TO HAVE BEEN THE CAUSE OF THE DAMAGE. THE RESULTS OF EXAMINATIONS OF SIX DAMAGED FUEL SUBASSEMBLIES ARE GIVEN AND SOME PRELIMINARY OBSERVATIONS ARE MADE CONCERNING POSSIBLE IMPROVEMENTS WHICH MAY BE MADE IN THE DESIGN OF SODIUM COOLED FAST REACTORS TO AID IN THE PREVENTION, DETECTION AND RECOVERY FROM SIMILAR INCIDENTS.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + REACTOR, LMCR + DAMAGE + COMPARISON, FACILITIES + FUEL ELEMENTS

17-029616

DATED 1968

MATTHEWS RR + HENRY KJ

LOCATION AND REPAIR OF THE DFR LEAK

UKAEA, DOUNREAY WORKS

NUCLEAR ENGINEERING, OCTOBER 1968, PG. 840-844

ON MAY 9, 1967, A 100-LITER/DAY SODIUM LEAK APPEARED, YET DISAPPEARED WHENEVER REACTOR SHUT DOWN EVEN THOUGH TEMPERATURES HELD CONSTANT. LEAK ESTIMATED TO BE 0.015 IN. DIA. HOLE. ANTI-LEAK JACKET 1/8-IN. FROM VESSEL MADE LOCATION DIFFICULT. GOLD ADDED TO NAK SHOWED SLIGHT LEAKAGE EVEN AFTER REACTOR DOWN. LOWERING NAK LEVEL WITH 35 PSIG HE IN VESSEL EVENTUALLY SHOWED LEAK WAS AT BOTTOM OF NAK OUTLET NOZZLES. MILDLY STRAINING EACH NOZZLE IN TURN INDICATED NO. 10 HEAT EXCHANGER, AND ACOUSTIC NOISE ANALYSIS INDICATED A LOCATION ON A PIPE. A FIELD WELD CRACKED FROM FATIGUE WHERE THERE WAS A POOR WELD AND A MISALIGNMENT. INTERNAL WELDING HEAD DEVELOPED FOR THE

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-029616 *CONTINUED*
4-IN. PIPE.

REACTOR, LMCR + FAILURE, PIPE + REACTOR, TEST + WELDING + FAILURE, FATIGUE + DOWNREAY (LMFBR) + REMOTE MANIPULATING AND VIEWING + FAILURE, INSTALLATION ERROR + FAULT + LEAK + REACTOR, LMFBR

17-029714 ALSO IN CATEGORY 11 DATED 1968
FERMI CHANGE 13 - WAIVED CONTAINMENT GROSS LEAKAGE TEST
U.S. ATOMIC ENERGY COMMISSION
1 PAGE, LETTER TO POWEL DEVELOPMENT COMPANY, OCTOBER 21, 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

WAIVES ANNUAL CONTAINMENT GROSS LEAKAGE TEST FOR CURRENT SHUTDOWN TO INVESTIGATE OCT. 66 FUEL DAMAGE INCIDENT. TEST TO BE PERFORMED PRIOR TO RELOADING WHERE THE REACTOR IS LESS THAN \$15 SUBCRITICAL. ***BASIS IS FISSION PRODUCTS IN PRIMARY SODIUM TOTAL LESS THAN 1 CURIE, DECAY HEAT IS LESS THAN ONE WATT PER SUBASSEMBLY.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (108 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + REACTOR, FAST + FERMI (LMFBR) + TEST, LEAK RATE + TECHNICAL SPECIFICATIONS + TEST INTERVAL + REACTOR, LMFBR

17-029732 DATED 1968
DESCRIPTION OF PIN-FUEL SUB ZONE LOADING
ATOMIC INTERNATIONAL
3 PAGES, LETTER TO DIRECTOR, ORL, OCTOBER 23, 1968, DOCKET 50-147

FOR EPITHEMAL CRITICAL EXPERIMENT LABORATORY. SEE AI-4120 FOR COMPLETE DESCRIPTION OF FACILITY. PLATE TYPE FUEL IN A 2 FT DIA CENTRAL TEST REGION WILL BE REPLACED BY PIN TYPE FUEL IN 12X6X6 BOXES FILLED WITH SODIUM AND FUEL PELLETS. ONLY PHYSICAL REARRANGEMENT IS PROPOSED. PROCEDURES AND TESTS WILL BE SIMILAR TO THESE USED IN PREVIOUS 17 CORE LOADINGS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (108 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

CRITICAL ASSEMBLY + REACTOR, LMFBR + FUEL ELEMENTS

17-029737 DATED 1968
HALLAM NUCLEAR POWER FACILITY MONTHLY RETIREMENT REPORT NO. 24 JULY 1968
CONSUMERS PUBLIC POWER DISTRICT
44 PAGES, FIGURES, TABLES, REPORT NO. 24, JULY 1968, DOCKET 115-3, TYPE--LMR, MFG--A.I., AE--BECHTEL

ONE OF A SERIES OF MONTHLY REPORTS. SINCE REACTOR IS BEING DISMANTLED, THE MORE APPROPRIATE INFORMATION RELATES TO MAINTENANCE, CORE-ELEMENT TRANSFER, AND RADIATION SAFETY AND CONTROL. SHORT SUMMARIES OF DAILY WORK GIVEN.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (108 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

17-029754 DATED 1967
BUSSE TH + WOOD DR + BEYER NS
NONDESTRUCTIVE INSPECTION OF EBR-II FUEL-JACKET TUBING USING ELECTROMAGNETIC TECHNIQUES
ARGONNE NATIONAL LABORATORY, SPECIAL MATERIALS AND SERVICES DIV.
ANL-7334 +. 29 PAGES, 12 FIGURES, 4 REFERENCES, MAY 1967

EDDY-CURRENT TECHNIQUES WERE USED TO EXAMINE 136,000 FT OF TUBING. DESCRIBES EQUIPMENT AND TECHNIQUE. OBJECTIVE WAS TO ELIMINATE TUBING WHICH MIGHT PROVE DETRIMENTAL IF USED AS JACKETING FOR FUEL ELEMENTS. 2 AND 1/2 YEARS REQUIRED FOR INSPECTION. FAILURES WERE MINOR. REJECTION RATE CAN AS HIGH AS 28.8%. ALL TUBING HAD BEEN PREVIOUSLY 100% EDDY-CURRENT INSPECTED BY MANUFACTURER. TESTING DEMONSTRATED NEED FOR SECOND INSPECTION AND IMPROVED PRIMARY INSPECTIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (KE) + CLADDING + REACTOR, FAST + REACTOR, LMCR + PROCEDURES AND MANUALS + REACTOR, TEST + REACTOR, ERDA OWNED + TEST, NONDESTRUCTIVE + EXAMINATION + QUALITY ASSURANCE + TUBING + ELECTRICAL CONDUCTION + FUEL ELEMENTS

17-029825 DATED 1968
BASMADIAN JA + HURN GR
IRRADIATION PERFORMANCE OF (U,Pu)N FAST REACTOR FUELS
BATTELLE NORTHWEST
1 PAGE, 1 TABLE, P. 102 OF ANS TRANSACTION, 11(1), 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-029825 *CONTINUED*

CANADIAN NUCLEAR ASSOCIATION, TORONTO, JUNE 10-13, 1968

DISCUSSES TESTS CONDUCTED IN MTR/ETR. THE MIXED FUEL AND 304SS CLAD WERE COMPATIBLE WITH NA BOND AT TEMPS. NEAR 1000 C. ALTHOUGH NA COMPLETELY PENETRATED FUEL PELLETS, NO EVIDENCE WAS FOUND OF FUEL DISSOLUTION OR PELLET CRACKING. LACK OF CENTRAL MELTING AT HIGH POWER DENSITIES IS EVIDENCE OF EXCELLENT THERMAL CONDUCTIVITY. LITTLE DECOMPOSITION AT 2000 C. FUEL HIGHLY PLASTIC AT 1000 TO 2000 C DURING IRRADIATION. UNEXPECTED SINCE GRAIN SIZES WERE 40 MICRONS. PRESSURE GENERATED EXTERNALLY BY PRESENCE OF SOLID NA ABOVE AND BELOW FUEL LEVEL, WHILE LIQUID NA AT 500 C WAS ADJACENT TO FUEL.

ETR (TR) + PRESSURE, EXTERNAL + REACTOR, FAST + FUEL BURNUP + MTR (TR) + PLUTONIUM + SODIUM + REACTOR, TEST + URANIUM + NITRIDE + IRRADIATION TESTING + THERMAL PROPERTY + PLASTICITY + R AND D PROGRAM + REACTOR, LMFBR

17-029831

ALSO IN CATEGORIES 3 AND 5

DATED 1968

RATEICK RC + MCHUGH WE + SALING JH

RESULTS AND CONCLUSIONS FROM THE EXAMINATION OF FUEL SUBASSEMBLIES REMOVED FROM THE ENRICO FERMI REACTOR AFTER THE FUEL DAMAGE INCIDENT

POWER REACTOR DEVELOPMENT COMPANY + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + BATTELLE MEMORIAL INSTITUTE
2 PAGES, 3 REFERENCES, ANS TRANS. 11(1), PAGES 100-101, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

ON OCT. 5, 1966, ZR-CLAD U-MO FUEL PINS PARTIALLY MELTED. OFF-SITE EXAM MADE TO DETERMINE IF FAILURE CAUSED BY FLOW BLOCKAGE. CONCLUSIONS - (1) SHIPMENT OF SUBASSEMBLIES WITHOUT PRIOR CLEANING CAN BE DONE AND EXAM ADEQUATELY PERFORMED IN CONVENTIONAL HOT CELL. (2) MELTING WAS CAUSED BY FLOW BLOCKAGE, (3) EVEN THOUGH SS WRAPPER WALLS LOST INTEGRITY IN MELTED SUBASSEMBLIES, GROSS FUEL LOSSES DID NOT OCCUR. (4) REDISTRIBUTION IN TWO MELTED ASSEMBLIES ACCOUNTED FOR 11.5 CENTS REACTIVITY OF TOTAL LOSS OF 22 CENTS. REMAINING 10.5 CENTS THOUGHT TO BE DUE TO RESULT OF DEFORMATION.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + HOT CELL + REACTOR, LMCR + TRANSPORTATION AND HANDLING + INCIDENT + REACTIVITY EFFECT, SLUMP + EXAMINATION + FLOW BLOCKAGE + REACTOR, LMFBR

17-029834

ALSO IN CATEGORY 7

DATED 1968

SMITH RR + EBERSOLE ER + THALGOTT FW

EXPERIENCE WITH A GAS-TYPE FAILURE IN AN EBR-11 DRIVER FUEL ELEMENT

ARGONNE NATIONAL LABORATORY

2 PAGES, ANS TRANS. 11(1), PAGES 279-80, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

FISSION-GAS MONITOR SHOWED SIGNAL-TO-BACKGROUND RATIO OF 12. (FAILED ELEMENT RADIATION DETECTOR GAVE NO INDICATION). BY RAISING POWER AND STUDYING Xe-133 AND 135 BUILDUP, DETERMINED FAILURE WAS FUEL LOADED IMMEDIATELY PRIOR TO STARTUP. OBSERVED THAT ONCE A GAS BUBBLE WAS RELEASED, DRIVING FORCE FOR ADDITIONAL RELEASE IS NEGLIGIBLE, AND CONCLUDED RUNNING REACTOR WITH GAS-TYPE FAILURE IS POSSIBLE.

ACTIVITY BUILDUP + EBR 1 AND 2 (RE) + REACTOR, FAST + FISSION GAS RELEASE + FISSION PRODUCT, VOLATILE + FAILURE, FUEL ELEMENT + REACTOR, LMCR + MEASUREMENT + NOBLE GAS + REACTOR, TEST + XENON + MONITOR, GAS + COVER GAS + REACTOR, ERDA OWNED + FUEL ELEMENTS

17-029835

DATED 1968

RICE RE + CERUTTI BC + CISSEL DW + DEEGAN GE + KIRN FS

RECENT EBR-11 EXPERIENCE

ARGONNE NATIONAL LABORATORY

1 PAGE, 1 TABLE, ANS TRANSACTIONS 11(1), PAGE 279, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

BRIEFLY SUMMARIZES INTERESTING EVENTS SINCE THOSE REPORTED IN NOVEMBER 1967. REPLACED CONTROL ROD GUIDE THIMBLE DAMAGED DURING FUEL HANDLING. DAMAGE CAUSED INCREASED CLEARANCES, RESULTING IN GAIN OF REACTIVITY. CONTINUING STUDY OF CORE AND REFLECTOR BORING. MAJOR MAINTENANCE INCLUDES REPAIRS TO SECONDARY SODIUM TRACE HEATING FOLLOWING NA FIRE CAUSED BY MELTING A FREEZE PLUG DURING VALVE REPLACEMENT.

EBR 1 AND 2 (RE) + GUIDE + REACTOR, FAST + FIRE + FUEL ELEMENT BOWING + FUEL HANDLING + FAILURE, OPERATOR ERROR + REACTOR, LMCR + MAINTENANCE AND REPAIR + REACTIVITY EFFECT + SODIUM + REACTOR, TEST + OPERATING EXPERIENCE SUMMARY + REACTOR, ERDA OWNED + DAMAGE + CONTROL RODS

17-029851

DATED 1968

SLIPER EG

ENRICO FERMI STEAM GENERATOR EXPERIENCE

DETROIT EDISON COMPANY

1 PAGE, ANS TRANS. 11(1), PAGE 137, 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND CANADIAN NUCLEAR ASSOCIATION, TORONTO, CANADA, JUNE 10-13, 1968

LOW CARBON MO-FERRITIC STEEL USED AS CONSTRUCTION MATERIAL. PROBLEMS AND SOLUTIONS WHICH HAVE OCCURRED ARE (1) STRESS CORROSION BY CAUSTIC EMBRITTLEMENT OF TUBES - RETUBED AND STRESS-RELIEVED BY CIRCULATING HOT ARGON, (2) TUBE VIBRATION - RETUBED AND ADDITIONAL SUPPORTS ADDED, (3) TUBE-TO-TUBE SHEET WELD LEAKAGE - ALL WATER MANIFOLD TUBE JOINTS REWELDED BY INTERNAL BORE-WELD

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-029851 *CONTINUED*

TECHNIQUE, (4) FLOW INSTABILITY - ORIFICES INSTALLED IN WATER MANIFOLD FOR EVEN WATER DISTRIBUTION TO ALL TUBES.

REACTOR, BREEDER + FAILURE, EQUIPMENT + REACTOR, FAST + FERMI (LMFBR) + THERMAL MECHANICAL EFFECT + WELDING + EMBRITTLEMENT + OPERATING EXPERIENCE SUMMARY + MODIFICATION + FAILURE, DESIGN ERROR + FLOW DISTRIBUTION + FLOW STABILITY + STRESS CORROSION + VIBRATION + ORIFICE + STEAM GENERATOR + TUBING + REACTOR, LMFBR

17-029930

DATED 1968

ORLOV VV + PINKHASIK MS + ARISTARKHOV NM + EFIMOV IA + KARPOV AV + NIKULIN MP

SAFETY PROBLEMS CONCERNING THE OPERATION OF THE BR-5 REACTOR

COMITE DETAT POUR L'UTILISATION DE L'ENERGIE NUCLEAIRE, MOSCOW

12 PAGES, BULL. INFORM. SCI. TECH. (PAKIS), NO. 124, PAGES 75-86 (MARCH 1968) (IN FRENCH)

THE SODIUM-COOLED REACTOR BR-5, IN OPERATION SINCE JANUARY 1959, HAS BEEN OPERATING 37,700 HR AS OF JULY 1, 1967 (111,200 HR AT 5 MW). THE OUTLET TEMPERATURE, WHICH HAD BEEN SET AT 450 C, HAS BEEN SET AT 500 C SINCE 1960. THE BURNUP HAS REACHED 7%, WHILE THE DESIGN PROVIDED 2% RATE. OPERATION WAS CONTINUED IN SPITE OF RELEASE OF FISSION PRODUCTS AND CLAD RUPTURE. IN 1965 THE REACTOR WAS FUELED WITH UC. WITH THE USE OF COLD TRAPS, THE OXIDE PERCENTAGE WAS KEPT BELOW 0.003%. THE REACTOR WAS SHUT DOWN SEVERAL TIMES. REASONS THE SHUTDOWNS ARE ANALYZED.

FISSION PRODUCT RELEASE + FRANCE + FUEL BURNUP + FAILURE, FUEL ELEMENT + REACTOR, LMC + SCRAM, REAL + SODIUM + OPERATING EXPERIENCE SUMMARY

17-029931

DATED 1967

WOODRUFF RW + DURAND RE + OWENS JE

SRE AND HNPFC OPERATING AND MODIFICATION EXPERIENCE

17 PAGES, 23 REFERENCES, PP. 2-45-61 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO, CALIFORNIA, APRIL 10-12, 1967, HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY, 1967

THE SODIUM REACTOR EXPERIMENT (SRE) AND THE HALLAM NUCLEAR POWER FACILITY (HNPFC) ARE SODIUM-COOLED GRAPHITE MODERATED REACTORS. THEY WERE DESIGNED AND CONSTRUCTED TO DEMONSTRATE FEASIBILITY OF THE SODIUM-COOLED-GRAPHITE-MODERATED REACTOR CONCEPT, AND IN SO DOING, THEY HELPED SOLVE MANY OF THE PROBLEMS FACED BY NA-COOLED FAST BREEDERS. THE SYSTEMS EXTERNAL TO THE CORE HAVE MANY FEATURES IN COMMON WITH FAST BREEDERS. THE MAIN HEAT TRANSFER SYSTEM, THE PURIFICATION SYSTEMS, AND THE AUXILIARY SYSTEMS ASSOCIATED WITH BOTH REACTORS PERFORMED WELL, THOUGH NOT WITHOUT PROBLEMS.

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMC + SODIUM + OPERATING EXPERIENCE SUMMARY + SRE (RE) + MODIFICATION

17-029932

DATED 1967

MCCARTHY WJ + OLSON WR + ALEXANDERSON EL + BRANYAN CE

RECENT OPERATING EXPERIENCE AT THE ENRICO FERMI ATOMIC POWER PLANT

POWER REACTOR DEVELOPMENT CO., DETROIT

25 PAGES, PP. 178-211 OF FAST BREEDER REACTORS, OXFORD, PERGAMON PRESS, 1967

GIVES RESULTS OF MEASUREMENTS OF TRANSFER FUNCTION, POWER COEFFICIENTS, SYSTEM CHARACTERISTICS DURING TRANSIENTS, AND THERMAL PERFORMANCE. STEAM GENERATOR DESIGN AND OPERATION ARE OUTLINED, AND AN ACCOUNT IS GIVEN OF THE PROBLEMS ENCOUNTERED IN STOPPING SMALL LEAKS FROM THE WATER TO THE NA SIDE. OTHER COMPONENTS AND SYSTEMS FOR WHICH THE BEHAVIOR IS REPORTED INCLUDE FUEL, CONTROL RODS, NEUTRON SOURCE, FUEL HANDLING, REACTOR CONTROL SYSTEM, PRIMARY NA SYSTEM, AND CONTAINMENT.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + TRANSFER FUNCTION + OPERATING EXPERIENCE SUMMARY + MEASUREMENT, REACTIVITY + COMPONENTS + MAIN COOLING SYSTEM + STEAM GENERATOR + REACTOR, LMFBR

17-029933

DATED 1967

ALEXANDERSON EL + BRANYAN CE + FURL JA + NIMS JB + OLSON WR

ENRICO FERMI ATOMIC POWER PLANT OPERATING EXPERIENCE THROUGH 100 MW

POWER REACTOR DEVELOPMENT CO., DETROIT

25 PAGES, PP. 1-21-45 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO, CALIFORNIA, APRIL 10-12, 1967, HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY, 1967

OPERATING EXPERIENCE OF THE FERMI FAST BREEDER THROUGH 200 MW(T) IS OUTLINED. OPERATING STATISTICS ARE TABULATED, RESULTS OF THE HIGH POWER NUCLEAR TEST PROGRAM ARE GIVEN. TRANSFER FUNCTION MEASUREMENTS ARE DISCUSSED, TRANSIENT TESTING RESULTS ARE TABULATED. HEAT BALANCES ARE GIVEN FOR ALL SYSTEMS AT 100 MW(T), CORE AND BLANKET TEMPERATURE MEASUREMENTS ARE DESCRIBED, PRIMARY SODIUM AND COVER GAS IMPURITY ANALYSES ARE TABULATED. PLANT PERFORMANCE AT POWER IS DISCUSSED IN SOME DETAIL.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR TRANSIENT + SODIUM + TRANSFER FUNCTION + COVER GAS + OPERATING EXPERIENCE SUMMARY + THERMAL CONSIDERATION + COOLANT CHEMISTRY + MEASUREMENT, REACTIVITY + TEST, PHYSICS + REACTOR, LMFBR

17-029934

DATED 1967

DUFFY JG + JENS WH + FELDES JG + JOHNSON KP + MCCARTHY WJ

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-029934 *CONTINUED*

INVESTIGATION OF THE FUEL MELTING INCIDENT AT THE ENRICO FERMI ATOMIC POWER PLANT
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT
23 PAGES, PP. 2-15-37 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO, CALIFORNIA, APRIL 10-12, 1967,
HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY, 1967

A DETAILED SUMMARY OF OCTOBER 4-5, 1966 OPERATION OF THE FERMI FAST BREEDER IS GIVEN. THE SUMMARY BEGINS WITH APPROACH TO CRITICAL ON OCT. 4 AND FOLLOWS THROUGH, STEPWISE, TO FINAL MANUAL SCRAM ON OCT. 5, THE FINAL SCRAM DECISION BEING DICTATED BY HIGH RADIATION ALARMS AND SUBSEQUENT ISOLATION OF THE FISSION PRODUCT DETECTOR BUILDING AND CONTAINMENT BUILDING. PLANT INVESTIGATIONS AFTER OCT. 5 ARE OUTLINED. PLANT SYSTEMS ARE ANALYZED FOR OCTOBER, A TEMPERATURE ANALYSIS IS GIVEN, REACTIVITY AND POWER HISTORIES ARE SHOWN AS THEY VARY WITH HOUR OF THE DAY. RESULTS OF PRIMARY NA AND AK ANALYSES ARE TABULATED, 14 CAUSES OF FUEL MELTDOWN ARE POSTULATED. THE DESIGN OF NEW FUEL HANDLING AND INSPECTION EQUIPMENT IS DISCUSSED.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + INCIDENT + REVIEW + OPERATING EXPERIENCE + REACTOR, LMFBR

17-029935

DATED 1967

NOVICK M + BACCA JP + CERUTTI BC + CISSEL DW + KIRN FS + SMITH RR + WHITHAM GK

EBR-II OPERATING EXPERIENCE

ARGONNE NATIONAL LAB., IDAHO FALLS, IDAHO

21 PAGES, PP. 1-20 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO, CALIFORNIA, APRIL 10-12, 1967,
HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY, 1967

EBR-II HAS BEEN OPERATING ROUTINELY AND VERY SATISFACTORILY AS A SODIUM COOLED FAST BREEDER POWER DEMONSTRATION REACTOR WHILE SERVING AS A FAST FLUX IRRADIATION FACILITY IN THE AEC LMFBR PROGRAM. 340 SPECIMENS HAVE UNDERGONE IRRADIATION BY THE END OF 1966. REPROCESSING AND REFABRICATION OF ALL DRIVER FUEL WAS HIGHLY SUCCESSFUL. ALTHOUGH MORE THAN 24,000 FUEL ELEMENTS HAVE UNDERGONE IRRADIATION IN THE REACTOR, NO FAILURE OF ANY ONE HAS BEEN EVIDENT. SOME OF THE FUEL IN THE REACTOR IS NOW IN ITS THIRD CYCLE THROUGH THE FUEL CYCLE FACILITY AND REACTOR. PERFORMANCE OF THE DRIVER FUEL HAS BEEN LESS THAN EXPECTED AND HAS RESULTED IN A RESTRICTION ON BURNUP LIMITS AND THE INITIATION OF AN IMPROVED DRIVER FUEL DEVELOPMENT PROGRAM. NO FAILURE OF ANY MAJOR COMPONENTS.

EBR 1 AND 2 (RE) + REACTOR, FAST + FUEL BURNUP + REACTOR, RESEARCH + OPERATING EXPERIENCE SUMMARY + IRRADIATION TESTING + FUEL REPROCESSING + REACTOR, LMFBR

17-029939

DATED 1968

POWER REACTOR DEVELOPMENT COMPANY ENRICO FERMI ATOMIC POWER PLANT - REPORT FOR JULY 1968 - DOCKET 50-16

POWER REACTOR DEVELOPMENT COMPANY

EF-59, JULY 1968, 7P, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

ONE OF A SERIES OF PROGRESS REPORTS. COVERS OPERATIONS, HEALTH PHYSICS, MAINTENANCE AND MODIFICATION, NONNUCLEAR TEST PROGRAM, AND FACILITY CHANGE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + OPERATING EXPERIENCE SUMMARY + MODIFICATION + OPERATING EXPERIENCE

17-029975

DATED 1968

EGGLESTON RR

RETIREMENT OF THE SODIUM REACTOR EXPERIMENT

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

AE-AEC-12572 +. 114 PAGES, 18 FIGURES, 5 TABLES, 12 REFERENCES, AUG 30, 1968

FINAL SUMMARY OF RETIREMENT EFFORT FROM FEB. 67 TO JULY 1, 68. REVIEWS ALTERNATIVE PLANS, DESCRIBES EVENTUAL, MINIMUM COST PLAN OF INDEFINITE STORAGE UNDER SURVEILLANCE, WITH FUEL AND USABLE ITEMS SHIPPED OUT. ***DESCRIBES STATUS OF VARIOUS SYSTEMS, DEACTIVATION SCHEDULE, AND COSTS (TOTAL \$843,000). SURVEILLANCE COSTS \$2500/MONTH. ***THE MOST SATISFACTORY PROCEDURES WERE THOSE WRITTEN ASSUMING WORKER HAD NO PRIOR PLANT KNOWLEDGE. VALVES HAD TO BE REIDENTIFIED AND RETAGGED. INSTALLATION OF SMALL SUBSYSTEMS ON PALLETS MADE IT EASIER TO SHIP AND BURY THAN CLEANUP. ***BIBLIOGRAPHY OF ALL SRE TOPICAL REPORTS INCLUDED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + PROCEDURES AND MANUALS + SRE (RE) + DECOMMISSIONING

17-029982

DATED 1968

BEAUDRY RJ

OPERATIONAL SHIELD TESTS FOR THE ENRICO FERMI ATOMIC POWER PLANT

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN

APDA-221 +. 205 PAGES, 63 FIGURES, 33 TABLES, 31 REFERENCES, JULY 1968

SHIELD WILL PERFORM ADEQUATELY TO 500 MWTH. PROVIDES MEASUREMENTS AT VARIOUS LOCATIONS IN PUMP

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-029982 *CONTINUED*

CELLS, PIPING, AND VESSEL. DESCRIBED MODIFICATIONS TO CALCULATIONS NECESSARY TO PROVIDE BETTER AGREEMENT. MOCKUP OF ACTUAL CONFIGURATION NEEDED FOR ACCURATE PREDICTION.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MEASUREMENT + COMPARISON, THEORY AND EXPERIENCE + SURVEY, RADIATION + SHIELDING + REACTOR, LMFBR

17-029984

ALSO IN CATEGORY 5

DATED 1968

COSTELLO RH

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT, APRIL 1968

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.

APDA-CFE-21 +. 46 PAGES, 21 FIGURES, JULY 1968

DESCRIBES 4 WAYS OF REMOVING ZIRCONIUM PLATE FASTENER AND PLATES. NINE SMALL OBJECTS WERE REMOVED FROM VESSEL. A MOCKUP OF FUEL ELEMENTS WAS MADE IN A LOOP TO DETERMINE THAT THE OBJECT REMOVED HAD CAUSED THE FLOW BLOCKAGE. A FLOW BLOCKAGE OF 95 AND 96% WAS CALCULATED TO BE REQUIRED FOR THE MELTDOWN. THE OBSERVED BLOCKAGE WAS 92 AND 30% UNDER REALISTIC CONDITIONS (THE OBJECT HAD BEEN DISTORTED IN HANDLING).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + HYDRAULIC EXPERIMENT + REACTOR, LMCR + MOCKUP + REPORT, OPERATIONS ANALYSIS + REMOTE MANIPULATING AND VIEWING + FLOW BLOCKAGE + REACTOR, LMFBR

17-030192

DATED 1968

SELECTED FERMI MAINTENANCE EXPERIENCE

POWER REACTOR DEVELOPMENT COMPANY

PRDC-EF-62 +. 9 PAGES, OCTOBER 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

TWO ZIRCONIUM SEGMENTS WERE REMOVED FROM THE CORE BY ARC MELTING. NO. 1 STEAM GENERATOR WAS DISMANTLED TO FIND LEAKS. TWO TUBES WERE LEAKING AT THE INTERNAL BORE WELD AREA IN THE WATER MANIFOLD TUBE SHEET. THE THIRD TUBE LEAK WAS LOCATED 10 FT BELOW TOP OF THE TUBE SHEET BY PRESSURIZING THE SHELL TO 38 PSIG WITH NITROGEN WHICH PASSED THROUGH A VALVE AT ULTRASONIC VELOCITY. THE FREQUENCY CHANGED AS THE VARYING WATER LEVEL PASSED OVER THE LEAK.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + REPORT, OPERATIONS + WELDING + TEST, LEAK LOCATION + REMOTE MANIPULATING AND VIEWING + FAILURE, TUBING + STEAM GENERATOR + REACTOR, LMFBR

17-030218

DATED 1968

PROPOSED CHANGE 14 - INERT ATMOSPHERE

U.S. ATOMIC ENERGY COMMISSION

2 PAGES, LETTER TO M.C. BECKMAN, PRDC, DEC. 16, 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

WAIVER OF ATMOSPHERIC PURITY (DURING REPLACEMENT OF BOOT SEAL IN PRIMARY SHIELD TANK) IS NOT JUSTIFIED BECAUSE OF GRAPHITE DAMAGE DURING OUTGASSING. *** REFERS TO NATIONAL CARBON CO. CURVE OF BORON LOSS VS TEMPERATURE AND MOISTURE. REFERS TO NUCLEAR METALLURGY (1963 VOL. IX) FOR OTHER INFORMATION. *** SUGGESTS ALTERNATE PLANS BE SUBMITTED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

BORON + REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + GRAPHITE + REACTOR, LMCR + MAINTENANCE AND REPAIR + PRESSURE VESSELS + TECHNICAL SPECIFICATIONS + SHIELDING + REFLECTOR + SEAL + DAMAGE + REACTOR, LMFBR

17-030238

DATED 1967

HALLAM NUCLEAR POWER FACILITY MONTHLY RETIREMENT REPORT NO. 26 SEPT. 1968

CONSUMERS PUBLIC POWER DISTRICT

30 PAGES, FIGURES, TABLES, SEPT. 1968, DOCKET 115-3, TYPE--LMR, MFG--A.I., AE--BECHTEL

ONE OF A SERIES OF MONTHLY REPORTS. SINCE REACTOR IS BEING DISMANTLED, MORE-APPROPRIATE INFORMATION DEALS WITH MAINTENANCE, CORE-ELEMENT TRANSFERS, RADIATION SAFETY AND CONTROL. EACH DAYS WORK BRIEFLY SUMMARIZED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

17-030276

DATED 1968

MADSEN EF

SODIUM SAMPLING IN THE ENRICO FERMI ATOMIC POWER PLANT

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-030278 *CONTINUED*
APDA-306 +. 99 PAGES, REFERENCES, MARCH 1968

THE SAMPLER DESIGNS USED OVER THE YEARS IN THE PRIMARY SODIUM SYSTEM ARE DESCRIBED IN DETAIL. THESE INCLUDE THE EARLIER THIEF TYPE AS WELL AS THE PRESENT PERMANENT FLOW-THROUGH STATION. THE CLEANING PROCEDURES, SAMPLING PROCEDURES, AND ANALYTICAL RESULTS ARE PRESENTED AS THE BASIS FOR EVALUATING THE ADEQUACY OF THE DESIGNS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + SODIUM + SAMPLING + OPERATING EXPERIENCE SUMMARY + MAIN COOLING SYSTEM + COOLANT QUALITY + REACTOR, LMFBR

17-030432 DATED 1968
COSTELLO RH
COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT, MARCH 1968
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH., POWER REACTOR DEVELOPMENT CO., DETROIT, MICH.,
DETROIT EDISON CO., MICH.
APDA-CFE-20 +. 47P., JUNE 1968, 20 FIGURES

MOSTLY DESCRIPTION OF EFFORT TO REMOVE FOLDED ZIRCONIUM FLOW GUIDE FROM LOWER PLENUM, SHOWING OBJECT, TOOLS, AND TECHNIQUE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + REPORT, OPERATIONS ANALYSIS + REMOTE MANIPULATING AND VIEWING + FLOW BLOCKAGE + REACTOR, LMFBR

17-030433 DATED 1968
COSTELLO RH
COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT JUNE 1968
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH., POWER REACTOR DEVELOPMENT CO., DETROIT, MICH.,
DETROIT EDISON CO., MICH.
APDA-CFE-21 +. 34P., 12 FIGURES, OCTOBER 1968

TOOL TO CHISEL OFF SCREW HEADS REMAINING ON CONICAL FLOW GUIDE WAS COMPLETED. ARC MELT TOOL WAS STARTED. *** PRIMARY SYSTEM WAS FILLED AND COLD TRAPPED. PLUGGING INDICATION WAS TRIED WITH A LOW PRESSURE DROP TO MINIMIZE ORIFICE EROSION. *** DESCRIBES FLASHOVER AND SHORT CIRCUIT (CAUSED BY LOOSE BUS LINK ON A 480-V CABINET) THAT DAMAGED THE MOUNTING PLATE OF FOUR CIRCUIT BREAKERS. COOLING WATER PIPE FOR BUILDING VENTILATION SYSTEM WAS REPLACED BECAUSE OF MAJOR LEAKS BELOW GROUND.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + FAILURE, EQUIPMENT + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + ELECTRIC POWER, SHUTDOWN + REPORT, OPERATIONS ANALYSIS + REMOTE MANIPULATING AND VIEWING + INSTRUMENT, COOLANT QUALITY + LEAK + AUXILIARY COOLING SYSTEM + REACTOR, LMFBR

17-030571 ALSO IN CATEGORY 18 DATED 1968
ANDERSON RH + FORD JA + LINDSEY PS
DESIGN, FABRICATION, AND PRELIMINARY OPERATION OF THE STEAM GENERATORS FOR THE ENRICO FERMI ATOMIC POWER PLANT
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-307 +. 301 PAGES, AUGUST 1968

CHRONOLOGICAL HISTORY FROM PURCHASE (1958) THROUGH PRELIMINARY OPERATION (1965). DESCRIPTIVE STUDIES AND EXPERIMENTAL PROGRAM LEADING TO DESIGN CONCEPT (37 PG), DESIGN AND FABRICATION (38 PG), PREOPERATIONAL TESTING AND USE (64 PG). *** COMPACT ONCE-THROUGH DESIGN HAS BEEN GOOD. TUBE FAILURE SHOWED METAL-WATER REACTION COULD BE WITHSTOOD AND DID NOT PROPAGATE. FLOW-INDUCED VIBRATION CAUSED FAILURE DUE TO POOR RESTRAINT AND BAFFLING. TUBE-TO-TUBE-SHEET JOINT-LEAKAGE CAUSE HAS NOT BEEN FOUND. ACCESSIBILITY HAS SAVED MONEY, BUT INTERNAL INSPECTION IS NOT EASY. MANY DEFECTS WERE FOUND IN PREOPERATIONAL TESTING, WHILE FIXING WAS EASY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + METAL WATER REACTION + OPERATING EXPERIENCE SUMMARY + DESIGN STUDY + TEST, PREOPERATIONAL + SYSTEM DESCRIPTION + EXAMINATION + VIBRATION + FAILURE, TUBING + STEAM GENERATOR + REACTOR, LMFBR

17-030675 ALSO IN CATEGORY 11 DATED 1969
ULTRASONIC DETECTOR ON SKATES CHECKS NUCLEAR VESSELS
POWER ENGINEERING 73(1), P.68 (JANUARY 1969)

INCLUDED IN DESIGN OF A1 350-500 MW LMFBR. PERMANENT TRACKS IN THE SPACE BETWEEN REACTOR VESSEL AND ITS JACKET. CAN BE FITTED WITH TV CAMERA, ULTRASONIC INSPECTION UNIT, WELDING, OR CLEANING HEADS (WEIGHT ABOUT 25 LB). DEVELOPED BY NORTH AMERICAN ROCKWELL FOR APPLICATION IN EXPERIMENTAL

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-030675 *CONTINUED*
AIRCRAFT AND APOLLO USAGE.

REACTOR, BWR + PRESSURE VESSELS + REACTOR, PWR + REACTOR, POWER + WELDING + EXAMINATION + REMOTE MANIPULATING
AND VIEWING + R AND D PROGRAM + REACTOR, LMFBR + PIPES AND PIPE FITTINGS

17-030806 DATED 1968
SNEPVANGERS JJM
INCIDENT FOLLOWING FAILURE OF A FUEL IRRADIATION CAPSULE IN THE HFR
REACTOR CENTRUM NEDERLAND, THE HAGUE
RCN-99 +. 52 PAGES, JULY 1968

ON MAY 7, AIRBORNE ACTIVITY DURING SPECIAL NUCLEAR HEATING EXPERIMENTS WITH EXPERIMENTS OUT OF
LUKE LED TO FINDING ELLAS IRRADIATION FACILITY IN A HIGH FLUX REGION (POSITION NOT VERIFIED
DURING STARTUP CHECK) WITHOUT COOLING OR INSTRUMENTATION (TURNED OFF ON SHUTDOWN, SUPPOSEDLY
AFTER WITHDRAWING EXPERIMENT.)

AVAILABILITY - USAEC DEPOSITORY LIBRARIES IN THE U.S. AND OVERSEAS

FISSION PRODUCT, IODINE + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + FAILURE, OPERATOR ERROR + REACTOR, TEST +
NETHERLANDS + REPORT, OPERATIONS ANALYSIS + INCIDENT, HUMAN ERROR + FAILURE, SEQUENTIAL + IRRADIATION
FACILITY + REACTOR, LMFBR

17-030845 ALSO IN CATEGORIES 11 AND 12 DATED 1967
POLOMIK EE + RUSSELL CE + QUINN EP
FLUID FLOW AND VIBRATION TESTS ON 1/2-SCALE MODEL OF THE SEFOR REACTOR VESSEL
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.
GEAP-5440+. APRIL 1967, 100P.

TESTS WERE RUN ON THE FLUID FLOW AND VIBRATION CHARACTERISTICS IN A HALF-SCALE MODEL OF THE SEFOR
(SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR) REACTOR VESSEL. WATER, AT TEMPERATURES UP TO 265 F,
WAS USED TO SIMULATE COOLANT CONDITIONS IN THE SODIUM-COOLED REACTOR. FLOWS WERE VARIED FROM 10
TO 100% RATED REYNOLDS NUMBER IN THE REACTOR. DATA WAS OBTAINED ON RELATIVE CORE FLOW
DISTRIBUTION, FLOW PATTERNS, PRESSURE LOSSES, AND VIBRATION CHARACTERISTICS OF THE CANTILEVER-
MOUNTED CORE SHROUD. FLUID FLOW RESULTS PROVIDED INFORMATION NECESSARY FOR VERIFICATION OF FINAL
DESIGN. VIBRATION RESULTS INDICATED NO SERIOUS EFFECTS OF AMPLITUDE OR STRESS IN THE FULL-SCALE
REACTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA 22151
\$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + MOCKUP + PRESSURE VESSELS + TESTING + WATER + REACTOR, PULSED + SEFOR (RE) +
FLOW DISTRIBUTION + VIBRATION ANALYSIS

17-031271 DATED 1967
HALLAM NUCLEAR POWER FACILITY MONTHLY RETIREMENT REPORT NO. 27
CONSUMERS PUBLIC POWER DISTRICT
29 PAGES, FIGURES, REPORT NO. 27, OCTOBER 1968, DOCKET 115-3, TYPE--LMR, MFG--A.I., AE--BECHTEL

SEALING OF THE CONCRETE HEAT EXCHANGER AND PROCESS CELLS WAS COMPLETED. RADIATION SURVEYS ALSO
COMPLETED. BUILDING DEMOLITION WILL BEGIN, TO BE COMPLETED MARCH 1969.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM
CHARGE \$2.00

HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + BUILDING + DECOMMISSIONING + SEAL

17-031273 DATED 1968
COSTELLO RH
SELECTED FERMI OPERATING EXPERIENCE
POWER REACTOR DEVELOP CO. + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. + DETROIT EDISON CO.
APDA-CFE-19 +. 32 PAGES, 11 FIGURES, MAY 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

DESCRIBES AND PICTURES VARIOUS TOOLS AND GRABBERS FOR RETRIEVING BENT ZIRCONIUM PLATE THROUGH A
HOLE IN A PIPE ELBOW. ***CORE HCLDOWN FINGERS COULD NOT BE INSPECTED BECAUSE OF INSUFFICIENT
LIGHT USING SPHERICAL MIRROR. ***BMI EXAMINATION OF SUBASSEMBLIES SHOWED HYDRIDING OF ZIRCONIUM
LESS THAN 200 PPM GENERALLY, BUT ONE AT 600. FUEL ALLOY TRANSFORMATION TO ALPHA PLUS DELTA PHASE
IS ABOUT 75 TO 90%.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + ZIRCONIUM + HYDRIDE + REPORT, OPERATIONS ANALYSIS + PHASE CHANGE + EXAMINATION + REMOTE
MANIPULATING AND VIEWING + FLOW BLOCKAGE + SUPPORT STRUCTURE + REACTOR, LMFBR + FUEL ELEMENTS

17-031274 DATED 1967

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-031274 *CONTINUED*

POWER REACTOR DEVELOPMENT COMPANY, ENRICO ATOMIC POWER PLANT REPORT FOR FEBRUARY 1968
POWER REACTOR DEVELOPMENT CO., DETROIT, MICH.
PRDC-EF-54 +. 6 PAGES, FEBRUARY 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

OBJECT-REMOVAL TOOLS TRANSFERRED TO CONTAINMENT FOR REMOVAL OF OBJECT THROUGH 14-IN. COOLANT LINE.
PEAK GAMMA LEVEL IN VESSEL IS 1500 R/HR. ***OTHER ITEMS DESCRIBED ARE HEALTH PHYSICS,
MAINTENANCE, AND NONNUCLEAR TESTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM
CHARGE \$2.00

FERMI (LMFBR) + REPORT, OPERATIONS + REACTOR, LMFBR

17-031275

DATED 1968

OBJECT REMOVED FROM FERMI VESSEL

POWER REACTOR DEVELOPMENT CO., DETROIT, MICH.
PRDC-EF-55 +. 7 PAGES, POWER REACTOR DEVELOPMENT COMPANY, ENRICO ATOMIC POWER PLANT REPORT FOR MARCH 1968,
DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

THE OBJECT (ZIRCONIUM PLATE READING 200 MR/HR AT SURFACE, DECONTAMINATED TO 2 MR/HR) WAS REMOVED.
HYDRIDE CONTENT WAS 400 PPM BY VISUAL COMPARISON. PATCH ON PIPE ELBOW WAS REWELDED AFTER 3
WEEKS, INSTEAD OF ONE DAY. OTHER ITEMS DESCRIBED ARE HEALTH PHYSICS, MAINTENANCE, AND NONNUCLEAR
TESTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM
CHARGE \$2.00

FERMI (LMFBR) + MEASUREMENT + REPORT, OPERATIONS + ZIRCONIUM + HYDRIDE + REMOTE MANIPULATING AND VIEWING +
FLOW BLOCKAGE + REACTOR, LMFBR

17-031282

DATED 1968

SELECTED FERMI OPERATING EXPERIENCE

POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-60 +. 8 PAGES, AUGUST 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

ARC-MELT TOOL REMOVED 6 SCREWS, AND TWO PLATES WERE REMOVED. NEW GRIPPER HAD TO BE LIGHTENED FOR
USE. MORE FUEL ELEMENTS MUST BE REMOVED TO MAKE WORKING SPACE IN CORE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM
CHARGE \$2.00

ALLOY + FERMI (LMFBR) + REPORT, OPERATIONS + CHROMIUM + REMOTE MANIPULATING AND VIEWING + REACTOR, LMFBR +
THERMOCOUPLE

17-031283

DATED 1967

SELECTED FERMI OPERATING EXPERIENCE

POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-59 +. 7 PAGES, JULY 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

WORK CONTINUED ON TOOLS TO REMOVE ZIRCONIUM PIECES FROM THE REACTOR VESSEL. OTHER ITEMS DESCRIBED
HEALTH PHYSICS, MAINTENANCE, AND NONNUCLEAR TEST PROGRAM.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM
CHARGE \$2.00

FERMI (LMFBR) + REPORT, OPERATIONS + REACTOR, LMFBR

17-031284

DATED 1968

SELECTED FERMI OPERATING EXPERIENCE

POWER REACTOR DEVELOPMENT COMPANY
PRDC-EF-57 +. 9 PAGES, MAY 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

ADDITIONAL MATERIAL REMOVED FROM REACTOR VESSEL. WHAT LOOKED LIKE LATHE TURNINGS WAS SCRAPINGS
FROM ALUMINUM-HANDLING-TOOL SHAFTS. *** BURIED CARBON STEEL PIPES (ONE COVERED WITH BITUMASTIC,
TWO IN A SECOND CARBON-STEEL GUARD PIPE) FAILED DUE TO LOCAL CORROSION. GAS AND LIQUID LEAKS
WERE WITHIN 10CFR20.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM
CHARGE \$2.00

FERMI (LMFBR) + REPORT, OPERATIONS + FAILURE, PIPE + WASTE DISPOSAL, LIQUID + CORROSION + REACTOR, LMFBR +
PIPES AND PIPE FITTINGS

17-031286

DATED 1967

POWER REACTOR DEVELOPMENT COMPANY ENRICO FERMI ATOMIC POWER PLANT REPORT FOR JUNE 1968
POWER REACTOR DEVELOPMENT COMPANY

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-031286 *CONTINUED*
PRDL-EF-28 +. 8 PAGES, JUNE 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

WHILE REMOVAL TOOLS ARE BEING COMPLETED, REACTOR SODIUM WAS REPLACED, CIRCULATED, AND COLD-TRAPPED. OTHER ITEMS DESCRIBED ARE HEALTH PHYSICS, MAINTENANCE, AND NONNUCLEAR TEST PROGRAMS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

FERMI (LMFBR) + REPORT, OPERATIONS + REACTOR, LMFBR

17-032146 DATED 1963
DONALDSON DM
CHEMICAL EXPERIENCE WITH THE DOUNREAY FAST REACTOR
UKAEA, DOUNREAY WORKS, CAITHNESS, SCOTLAND
CONF-407-1 +. 33 PAGES, 13 FIGURES, 9 TABLES, 19 REFERENCES, SEPTEMBER 1963

IMPURITIES, PARTICULARLY OXYGEN, IN THE LIQUID METAL COOLANT ARE DISCUSSED IN THE LIGHT OF TWO YEARS INFORMATION. PROBABLE SOURCES OF THESE IMPURITIES ARE CONSIDERED AND THE RESULTING EFFECTS OF THEIR PRESENCE IN THE CIRCUIT, SUCH AS CIRCUIT BLOCKAGE, CORROSION OF FUEL AND CLADDING MATERIAL, EMBRITTLEMENT OF CLADDING MATERIAL, ARE DETAILED AND COMPARED WITH DATA OBTAINED IN OUT-OF-PILE RIGS. THE CORRELATION OF THESE DATA ARE DISCUSSED AT LENGTH, PARTICULARLY WITH RESPECT TO THE LEVELS OF OXIDE PRESENT IN THE REACTOR COOLANT.

AVAILABILITY - D.M. DONALDSON, UKAEA, DOUNREAY WORKS, CAITHNESS, SCOTLAND

REACTOR, FAST + REACTOR, LMCR + SODIUM + REACTOR, TEST + UNITED KINGDOM + OPERATING EXPERIENCE SUMMARY + COOLANT CHEMISTRY + DOUNREAY (LMFBR) + COOLANT PURIFICATION SYSTEM + FUEL COOLANT INTERACTION

17-032294 DATED 1969
PROVISIONAL OPERATING LICENSE FOR SEFOR
U.S. ATOMIC ENERGY COMMISSION
40 PAGES, PROVISIONAL OPERATING LICENSE FOR SEFOR, MARCH 4, 1969, DOCKET 50-231

ALLUWS 1-MW OPERATION WITHOUT USING REACTIVITY EXCURSION DEVICE FOR INITIAL TESTING PENDING FULL-POWER EVALUATION. TECHNICAL SPECIFICATION DOES NOT INCLUDE BASES FOR EACH SPECIFICATION.
***WASTE GAS EFFLUENT LIMITS USE CURIE SUMMATION IN GROUPS RATHER THAN SPECIFYING AN ISOTOPE.
DAILY RECORD OF FLOW, TEMPERATURE, POWER, AND REACTIVITY SHALL BE ANALYZED MONTHLY FOR UNEXPLAINED TRENDS. SHORT-TERM CHANGES REQUIRE 50% POWER REDUCTION AND MANAGEMENT NOTIFICATION.
ELABORATE CONTAINMENT-LEAKAGE LIMITS GIVE ALLOWABLE LEAK RATES FOR PENETRATIONS, SPECIFY DEWPOINT AND FREON CONTENT, ATMOSPHERE HEATUP RATE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, FAST + REACTOR, LMCR + WASTE DISPOSAL, GAS + TECHNICAL SPECIFICATIONS + REACTOR, PULSED + SEFOR (RE) + REACTIVITY EFFECT, ANOMALOUS + CONTAINMENT ATMOSPHERE, INERT + LEAK RATE + CONTAINMENT PENETRATION, ELECTRICAL + OPERATOR ACTION

17-032303 ALSO IN CATEGORY 12 DATED 1968
CONTAMINATION FOUND IN WASHED PIPE INTERIORS
CONSUMERS POWER COMPANY
10 PAGES, HALLAM MONTHLY RETIREMENT REPORT NO. 28, NOVEMBER 1968, DOCKET 115-3, TYPE--LMR, MFG--A.E., AE--BECHTEL

DEMOLITION BEGAN. A NOV. 1 COMPLIANCE SURVEY FOUND 1 MRAD/HR AT CONTACT INSIDE WASHED SCRAP PRIMARY PIPING THAT HAD BEEN SURVEYED AT THE ENDS OF THE CUT SEGMENTS. RADIATION LEVELS WERE TO BE MEASURED INSIDE THE PIPE, SO A SURVEY METER WAS MADE TO FIT ON A LONG POLE. PRIMARY 14-IN. PIPING WAS FOUND TO READ 0.1 TO 1.8 MRAD/HR, HIGHEST AT WELDS NEAR TEES OR ELBOWS, ALSO AT SODIUM OXIDE DEPOSITS AT PIPE REDUCTIONS. PRIMARY PIPE WILL BE SEGREGATED AND REMONITORED BEFORE DISPOSAL.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

DECONTAMINATION + HALLAM (LMR) + REACTOR, LMCR + SODIUM + COMPLIANCE + REACTOR, POWER + CONTAMINATION + DECOMMISSIONING + PIPES AND PIPE FITTINGS

17-032304 DATED 1967
HALLAM MONTHLY RETIREMENT REPORT 40
CONSUMERS POWER COMPANY
7 PAGES, OPERATION REPORT NO. 40, FEBRUARY 25, 1969, DOCKET 115-3 TYPE--LMR, MFG--A.E., AE--BECHTEL

BUILDING DEMOLITION CONTINUED WITHOUT ACCIDENT.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-032324

DATED 1969

AMENDED PROPOSED CHANGE 14, PRIMARY SHIELD TANK ATMOSPHERE
POWER REACTOR DEVELOPMENT COMPANY

15 PAGES, 4 FIGURES, LETTER TO AEC FROM M. C. BEEKMAN, MARCH 6, 1969, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

SUBSTITUTES NITROGEN QUALITY DETAILS FOR USE WHEN SODIUM TEMPERATURE EXCEEDS 400 F. REQUIRES SHUTDOWN IF NOT MET. SPECIFIES GRAPHITE PURGE. *** INTENDED TO ALLOW BOOT-SEAL REPLACEMENT, WHEN OXYGEN MUST BE ADMITTED FOR REPAIR OPERATION. HAZARDS ANALYSIS OF BORON LOSS FROM GRAPHITE (DUE TO OXYGEN OR WATER VAPOR NOT REMOVED BY PURGE) COVERED IMPURITY ADDITION TO SECONDARY SODIUM, CHANGED CALIBRATION OF NEUTRON DETECTORS, SECONDARY COOLANT AND STRUCTURE ACTIVATION BY INCREASED NEUTRON LEAKAGE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

BORON + REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + GRAPHITE + REACTOR, LMCR + MAINTENANCE AND REPAIR + PRESSURE VESSELS + TECHNICAL SPECIFICATIONS + SHIELDING + SEAL + CONTAINMENT ATMOSPHERE, INERT + LEAK + REACTOR, LMFBR

17-032469

DATED 1967

HALLAM DECOMMISSIONING CONTINUES

CONSUMERS PUBLIC POWER DISTRICT

8 PAGES, LETTER TO DRL, FEBRUARY 12, 1969, DOCKET 115-3, TYPE--LMR, MFG--A.E., AE--BECHTEL

HALLAM DECOMMISSIONING CONTINUES. NO INCIDENTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

17-032471

DATED 1967

SELECTED FERMI OPERATING EXPERIENCE

POWER REACTOR DEVELOPMENT COMPANY

PRDC-EF-64 +. 7 PAGES OF POWER REACTOR DEVELOPMENT COMPANY, ENRICO FERMI ATOMIC POWER PLANT REPORT FOR DECEMBER 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

ALL FIVE ZIRCALOY SHEETS WERE REMOVED FROM THE CORE. CLEANUP CONTINUES.

AVAILABILITY - U.S. PUBLIC DOCUMENT ROOM, 1717 H. ST., WASHINGTON, D. C. 20545, \$0.25 CENTS/PAGE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR, LMFBR

17-032472

DATED 1968

SELECTED FERMI OPERATING EXPERIENCE

POWER REACTOR DEVELOPMENT COMPANY

PRDC-EF-63 +. 11 PAGES OF POWER REACTOR DEVELOPMENT COMPANY, ENRICO FERMI ATOMIC POWER PLANT REPORT FOR NOVEMBER 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

FAILED WASTE GAS AND LIQUID-WASTE PIPELINES ARE BEING REPLACED. TWO PLATES WERE TORN LOOSE BY FLOW, THE SECOND FOUND WEDGED IN THE BOTTOM OF THE CORE SUPPORT PLATE. REMOVAL OF OTHER PLATES CONTINUED.

AVAILABILITY - U.S. PUBLIC DOCUMENT ROOM, 1717 H. ST., WASHINGTON, D. C. 20545, \$0.25 CENTS/PAGE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + FAILURE, PIPE + CORROSION + REMOTE MANIPULATING AND VIEWING + FLOW BLOCKAGE + REACTOR, LMFBR

17-032587

ALSO IN CATEGORIES 7 AND 18

DATED 1968

STEWART DC + MACIAS ES + BASILE LJ + MILSTED J

BUILDUP OF RADIOACTIVE PRODUCTS IN THERMAL REACTORS. II. RA-226 TARGETS

ARGONNE NATIONAL LAB., 111.

ANL-7478 +. 133 PAGES, FIGURES, TABLES, REFERENCES, JULY 1968

THIS REPORT CONTINUES A SERIES DESCRIBING THE BUILDUP OF ISOTOPIC L, PHYSICS, AND COMPONENT DEVELOPMENT), EBR-2 OPERATION, (2) GENERAL NEUTRON FLUXES AT VARIOUS LEVELS. REFERENCES ARE GIVEN IN ITS DEVELOPMENT, CHEMISTRY), (3) NUCLEAR SAFETY. WEAR TESTS WERE RUN WITH USED ARE DESCRIBED IN DETAIL. AC-227 IS ALSO OF SOME INTEREST AS AT 300 F, SODIUM LUBRICATES. BUT, AT 500 F, IT WETS SURFACE AND A ALPHA EMITTERS. THIS APPLICATION, HOWEVER, WILL PROBABLY ALWAYS LIQUID SODIUM FLASHING, PRESS.-WAVE PROPAGATION, SUPERHEAT, COOLANT ANY QUANTITY. THERE ARE ALSO SEVERAL INTENSE GAMMA EMITTERS AMONG TRANSFER, AND SODIUM-AIR REACTION. PRESENT SOME PROBLEMS IN HANDLING THE MATERIAL FOR HEAT SOURCE APPLICATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 CPY, \$0.65 MICROFICHE

CHEMICAL REACTION + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + MOLYBDENUM + REACTOR, SAFETY RESEARCH +

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-032567 *CONTINUED*
SODIUM + REACTOR, TEST + TESTING + TREAT (PRR) + REACTOR COOLANT + METAL, LIQUID + REACTOR, PULSED +
LUBRICATION + R AND D PROGRAM + REACTOR, LMFBR

17-032607
COSTELLO RM
SELECTED FERMI OPERATING EXPERIENCE
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., MICH.
APDA-CFE-24 +. 36 PAGES, JULY 1968

DATED 1968

CONFIDENCE IN THE ARC-MELT PROCESS FOR REMOVING THE SCREWHEADS HOLDING DOWN THE ZIRCONIUM SEGMENTS
RESULTED IN DE-EMPHASIS OF DEVELOPMENT OF CHISEL TOOL METHOD. DEVELOPMENT WORK IS CONTINUING ON
THE NEW ELECTROHYDRAULIC CONTROL SYSTEM AND ROTATABLE GRIPPER FOR THE SPINE-TYPE MANIPULATOR.
THREE CORE ASSEMBLIES, OPERATED NEXT TO DAMAGED ASSEMBLIES DURING THE MELTDOWN, WERE FOUND TO BE
DISTORTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL ELEMENT BOWING + REACTOR, LMCB + REPORT, OPERATIONS
ANALYSIS + REMOTE MANIPULATING AND VIEWING + REACTOR, LMFBR

17-032619
MCCARTHY WJ + JENS WH

DATED 1967

A REVIEW OF THE FERMI REACTOR FUEL DAMAGE INCIDENT AND A PRELIMINARY ASSESSMENT OF ITS SIGNIFICANCE TO THE
DESIGN AND OPERATION OF SODIUM COOLED FAST REACTORS
POWER REACTOR DEVELOPMENT CO., MICH.
CONF-670916 +. 24 PAGES, PP. VAL.1-23 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST
REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

THE STATUS OF THE INVESTIGATION OF THE EXTENT AND CAUSE OF THE OCTOBER 5, 1966, FUEL DAMAGE IS
GIVEN AS OF SEPTEMBER 13, 1967. A FOREIGN OBJECT WAS FOUND IN THE CORE INLET (BELIEVED TO HAVE
BEEN THE CAUSE OF THE DAMAGE). THE RESULTS OF EXAMINATIONS OF SIX DAMAGED FUEL SUBASSEMBLIES ARE
GIVEN, AND SOME PRELIMINARY OBSERVATIONS ARE MADE CONCERNING POSSIBLE IMPROVEMENTS WHICH MAY BE
MADE IN THE DESIGN OF NA-COOLED FAST REACTORS TO AID IN THE PREVENTION, DETECTION, AND RECOVERY
FROM SIMILAR INCIDENTS.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCB + INCIDENT, RECOVERY FROM +
EXAMINATION + FLOW BLOCKAGE + REACTOR, LMFBR + FUEL ELEMENTS

17-032627 ALSO IN CATEGORIES 5 AND 9 DATED 1968
REPORT ON THE FUEL MELTING INCIDENT IN THE ENRICO FERMI ATOMIC POWER PLANT
POWER REACTOR DEVELOPMENT COMPANY
50 PAGES, FIGURES, TABLES, REFERENCES, DECEMBER 15, 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--
COMMONWEALTH ASSOC.

DESCRIBES THE OCT. 5 HAPPENINGS (FLOW BLOCKAGE DUE TO ZIRCONIUM PLATE CAUSED PARTIAL MELTDOWN,
RELEASING FISSION GAS AND LOSING EXCESS REACTIVITY), RESULTS OF INVESTIGATIONS (ZIRCONIUM FROM
CONICAL FLOW GUIDE, FUEL FAILURE DID NOT PROPAGATE, FLOW REDUCTIONS CAME ONLY FROM ZIRCONIUM
SEGMENT), PLANT MODIFICATION (ON-LINE COMPUTER WILL ALARM ON DEVIATION FROM CALCULATED VALUE OF
REACTIVITY BALANCE, FISSION PRODUCTS, OR SUBASSEMBLY DELTA T AND WILL LATER CAUSE SCRAM).
STARTUP TEST PROGRAM OUTLINED. APPENDICES GIVE DETAILS ON FUEL EXAMINATION. *** OCT. 5
OPERATION WAS TO CHECK EARLIER HIGH DELTA T IN ELEMENTS WHICH MELTED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM
CHARGE \$2.00)

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCB + INCIDENT + INSTRUMENT,
TEMPERATURE + REPORT, OPERATIONS ANALYSIS + INCIDENT, RECOVERY FROM + EXAMINATION + REMOTE MANIPULATING AND
VIEWING + FLOW BLOCKAGE + INSTRUMENT, ABNORMAL INDICATION + INSTRUMENT, OPERATING REACTIVITY + COMPUTER
CONTROL + REACTOR, LMFBR + FUEL ELEMENTS

17-032628 ALSO IN CATEGORY 9
FERMI MELTDOWN INDICATIONS

DATED 1968

POWER REACTOR DEVELOPMENT COMPANY
32 PAGES, FIGURES, TABLES, PP. 5-37 OF REPORT ON FUEL MELTING INCIDENT, DECEMBER 15, 1968, DOCKET 50-16, TYPE--
LMFBR, MFG--APDA, AE--COMMONWEALTH

AUGUST OPERATION SHOWED HIGH OUTLET TEMPERATURES FROM SEVERAL FUEL ELEMENTS. ELEMENTS WERE MOVED
TO SEE IF THE CAUSE WAS POOR FLOW DISTRIBUTION (THE OCT. 5 OPERATION WAS TO TEST THIS). HOT
REACTIVITY AT 1 MW WAS OK. DELAYS DURING THE DAY LEAD TO RECORDING THERMOCOUPLE DATA DURING
POWER USE, RATHER THAN INCREASING POWER IN STEPS AND MONITORING TEMPERATURES. AT 20 MW IN
AUTOMATIC CONTROL THE DN/DT SIGNAL BECAME ERRATIC. AS THIS HAD BEEN EXPERIENCED BEFORE AT THIS
POWER LEVEL AND AS THE NOISE DISAPPEARED IN MANUAL CONTROL, REACTOR POWER AGAIN RAISED IN
AUTOMATIC TIL ALL RODS BEING 9 IN. WITHDRAWN INSTEAD OF 6 IN. CAUSED THE SRO TO HALT POWER
INCREASE, THEN HIGH ELEMENT OUTLET TEMPERATURES (3X NORMAL) AND RADIATION ALARMS RESULTED IN
MANUAL SHUTDOWN. ANALYSIS SHOWED A LOSS OF \$0.005/MW FROM 6 TO 18 MW, THEN LOSS OF \$0.013/MW
ABOVE THIS, WITH A NET LOSS OF \$0.26 (11.5 FUEL LOSS AND 6.9 FUEL BOWING). THE FISSION PRODUCT
DETECTOR (CHARGED WIRE IN REACTOR VESSEL OFFGAS) WAS BEING CALIBRATED DURING POWER INCREASES, AND

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-032628 *CONTINUED*

REQUIRED 3 OR 7 MIN SOAK TIMES. THE LAST MEASUREMENT WAS OFF SCALE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + NOISE + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + REPORT, OPERATIONS ANALYSIS + PERIOD METER + FLOW BLOCKAGE + REACTIVITY EFFECT, ANOMALOUS + INSTRUMENT, ABNORMAL INDICATION + OPERATING EXPERIENCE + REACTOR, LMFBR

17-032629 ALSO IN CATEGORIES 5 AND 11 DATED 1968

EXAMINATION OF MELTED FERMI FUEL

POWER REACTOR DEVELOPMENT COMPANY

55 PAGES, FIGURES, TABLES, APPENDICES B AND C OF REPORT ON THE FUEL MELTING INCIDENT, DECEMBER 15, 1968, DOCKET 50-16, TYPE--LMFBR, MFG--APUA, AE--COMMONWEALTH ASSOC.

AN UNMELTED ELEMENT SHOWED ABOUT 200 PPM HYDROGEN IN CLAD (WELL BELOW 5000 SPALLING LEVEL), HIGH VALUES FROM A PARTICULAR MELT INCOT. OTHERWISE NORMAL. ANOTHER ELEMENT THERMOCOUPLE WENDING HIGHER THAN A MELTED ELEMENT, WAS ONLY BADLY BOWED. *** MOLTEN FUEL ADHERED TO STEEL CLAD BUT DID NOT PENETRATE. HYDROGEN CONTENT REACHED 4000 PPM WITHOUT CRACKING. ONLY XE-KR FOUND IN COVER GAS. CS, SR, I, BA-LA FOUND IN SODIUM COOLANT. WHILE THERE WAS A COMMON HOLE IN THE WRAPPER CANS BETWEEN MELTED ELEMENT, NO EVIDENCE OF INTERCHANGE OF FUEL. FLOW TESTS AND HEAT TRANSFER ANALYSIS SHOWED ABOUT 3% NOMINAL FLOW IN TWO MELTED ELEMENTS, 30 AND 7% FLOW IN DAMAGED ADJACENT ELEMENTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REACTOR, LMCR + SODIUM + EMBRITTLEMENT + HYDRIDE + REPORT, OPERATIONS ANALYSIS + HEAT TRANSFER ANALYSIS + ZIRCALOY + FISSION PRODUCT RETENTION + EXAMINATION + FLOW BLOCKAGE + STRUCTURAL INTEGRITY + REACTOR, LMFBR

17-032632 ALSO IN CATEGORY 18 DATED 1969

REPORT ON SEFOR OPERATING LICENSE APPLICATION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

2 PAGES, LETTER - ACRS TO AEC, MARCH 13, 1969, DOCKET 50-239

ACKS RECOMMENDS OPERATION UP TO 20 MWt BUT INDICATES CONCERN FOR PROVISIONS TO PROTECT CORE INTEGRITY AGAINST PRIMARY LEAKS, INADEQUACY OF SIPHON BREAKER, GUIDELINES FOR ACTIONS IN EVENT OF ANOMALOUS REACTIVITY ON COOLANT TEMPS, COOLANT MONITORING TECHNIQUES AND FREQUENCY, BASES AND ACCEPTABILITY OF CONTINUED OPERATION IN PRESENCE OF KNOWN LOSS OF FUEL CLAD INTEGRITY.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, FAST + REACTOR, LMCR + ACRS + LICENSE APPLICATION + REACTOR, PULSED + SEFOR (RE) + OPERATING LICENSE PROCESS + REACTOR, LMFBR

17-032819 ALSO IN CATEGORIES 5 AND 18 DATED 1967

FLUID FLOW AND VIBRATION TESTS ON 1/2 SCALE VESSEL MODEL

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-4 +. 60 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 4, DEC. 5, 1967, DOCKET 50-231

WATER, AT TEMPERATURES UP TO 265 F, WAS USED TO SIMULATE COOLANT CONDITIONS IN THE SODIUM-COOLED REACTOR. FLOWS WERE VARIED FROM 10 TO 100 PERCENT OF RATED REYNOLDS NUMBER IN THE REACTOR. DATA WAS OBTAINED ON RELATIVE CORE FLOW DISTRIBUTION, FLOW PATTERNS, PRESSURE LOSSES, AND VIBRATION CHARACTERISTICS OF THE CANTILEVER-MOUNTED CORE SHROUD. FLUID FLOW RESULTS PROVIDED INFORMATION NECESSARY FOR VERIFICATION OF FINAL DESIGN. VIBRATION RESULTS INDICATED NO SERIOUS EFFECTS OF AMPLITUDE OR STRESS IN THE FULL-SCALE REACTOR.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, FAST + FLOW THEORY AND EXPERIMENTS + REACTOR, LMCR + PRESSURE VESSELS + TEST, PRESSURE VESSEL + SODIUM + TESTING + REACTOR COOLANT + METAL, LIQUID + REACTOR, PULSED + SEFOR (RE) + PRESSURE DROP + FLOW DISTRIBUTION + VIBRATION + REPORT, PSAR + MODEL TESTING + FLOW + REACTOR, LMFBR

17-032839 ALSO IN CATEGORY 18 DATED 1969

ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT MARCH 1969

ARGONNE NATIONAL LABORATORY

ANL-7561 +. 110 PAGES, FIGURES, TABLES, REFERENCES, MARCH 1969

ONE OF A SERIES. TOPICS ARE - LMFBR DEVELOPMENT OF FUEL (PHYSICS, AND COMPONENTS), FAST BREEDER FUEL DEVELOPMENT, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY. OPERATIONAL ACTIVITIES DISCUSSED FOR EBR-2, ZPR-3, AND TREAT. APPROVAL FOR OPERATION OF ZPPR WAS RECEIVED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-032839 *CONTINUED*

CRITICAL ASSEMBLY + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR, TEST + COMPONENTS + R AND D PROGRAM + ZPR 3 (CAF) + REACTOR, LMFBF + FUEL ELEMENTS

17-032844 ALSO IN CATEGORY 18 DATED 1967

SCOTT FA

SAMPLING FFTF SODIUM COOLANT FOR ANALYTICAL PURPOSES

BATTELLE-NORTHWEST

BNWL-491 +. 33 PAGES, 1 FIGURE, 3 TABLES, 70 REFERENCES, SEPTEMBER 1967

DESCRIBES CONCEPTUAL DESIGN USING A SMALL BYPASS ROOF. ALSO DESCRIBES AND EVALUATES EXPERIENCE WITH 3 ALTERNATIVES (VACUUM THIEF, DIPPING CUP, DRAIN SAMPLING).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + REACTOR, FAST + REACTOR, LMCR + SODIUM + REACTOR, TEST + SAMPLING + FFTF (TR) + OPERATING EXPERIENCE SUMMARY + MAIN COOLING SYSTEM + COOLANT QUALITY + REACTOR, LMFBF

17-032845 ALSO IN CATEGORY 6 DATED 1968

CHAPELET M + EDELINE JC + LHAUBET G

EXPERIMENTAL STUDY OF THE RAPSDIE PROTECTIONS

COMMISSARIAT A L ENERGIE ATOMIQUE, FONTENAY-AUX-ROSES, FRANCE

CEA-R-3626 +. 192 PAGES, 49 FIGURES, 14 TABLES, OCTOBER 1968

GIVES DETAILS OF PROTECTION TESTS MADE BY THE PROTECTION SUB-COMMISSION DURING THE STARTING-UP TRIALS ON THE REACTOR RAPSDIE. THE MAIN TYPES OF TESTS REPORTED CONCERN THE FLUX MEASUREMENTS IN THE CORE, THE REFLECTORS, THE SODIUM AND THE NEAR BIOLOGICAL PROTECTIONS, THE RADIO-PROTECTION MEASUREMENTS IN THE PREMISES, AND THE ACTIVATION OF THE PRINCIPAL FLUIDS. IN THE LAST PART, RESULTS ARE ANALYZED AND COMPARED WITH FORECAST CALCULATIONS.

AVAILABILITY - US DEPOSITORY LIBRARIES FOR THE U. S. AND OVERSEAS

REACTOR, FAST + FLUX DISTRIBUTION + FRANCE + COMPARISON, THEORY AND EXPERIENCE + REACTOR, TEST + REACTOR STARTUP TESTING + REACTOR, LMFBF

17-032933 ALSO IN CATEGORY 18 DATED 1968

ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT

ARGONNE NATIONAL LABORATORY

ANL-7527 +. 140 PAGES, FIGURES, TABLES, REFERENCES, DECEMBER 1968

ONE OF A SERIES. TOPICS ARE - LMFBF DEVELOPMENT PROGRAM, FUEL DEVELOPMENT STUDIES OF OTHER FAST REACTORS, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY. TESTS WITH EBR-2, ZPR-3, AND ZPPR ARE DISCUSSED. CONVERSION OF ZPR-6 AND 9 TO PU OPERATIONS CONTAINER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CRITICAL ASSEMBLY + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + PLUTONIUM + SODIUM + REACTOR, TEST + REACTOR COOLANT + ZPR 9 (CAF) + ZPR 6 (CAF) + MODIFICATION + METAL, LIQUID + ZPPR (CAF) + R AND D PROGRAM + ZPR 3 (CAF) + REACTOR, LMFBF

17-032934 ALSO IN CATEGORY 18 DATED 1969

ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT

ARGONNE NATIONAL LABORATORY

ANL-7548 +. 132 PAGES, FIGURES, TABLES, REFERENCES, JANUARY 1969

ONE OF A SERIES. TOPICS ARE - LMFBF DEVELOPMENT, FUEL DEVELOPMENT FOR FAST REACTORS, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY. EBR-2 OPERATIONS DISCUSSED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + R AND D PROGRAM + REACTOR, LMFBF

17-032939 ALSO IN CATEGORY 18 DATED 1967

ANL - REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JUNE 1967

ARGONNE NATIONAL LABORATORY

ANL-7549 +. 140 PAGES, 24 FIGURES, TABLES, JUNE 1967

SECTION TITLES - PLUTONIUM UTILIZATION, EBWR, LIQUID-METAL FAST-BREEDER REACTORS, GENERAL REACTOR TECHNOLOGY, ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT, AND NUCLEAR SAFETY. SEARCH FOR LOCATION OF FISSION-GAS RELEASE IN EBR-2 CONTINUES. LMFBF TOPICS INCLUDE EBR-2 OPERATION, PHYSICS, COMPONENT AND FUEL DEVELOPMENT, PROGRESS ON FFTF. NUCLEAR SAFETY TOPICS INCLUDE ACCIDENT ANALYSIS, COOLANT DYNAMICS, FUEL MELTDOWN STUDIES WITH TREAT, TREAT OPERATIONS, CONTAINMENT BY ENERGY ABSORPTION, AND PLUTONIUM VOLATILITY STUDIES.

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-032939 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT + REACTOR, FAST + FISSION GAS RELEASE + FUEL MELTDOWN + REACTOR, LMCR + REPORT, OPERATIONS + PLUTONIUM + REACTOR, SAFETY RESEARCH + REACTOR, TEST + TREAT (PRR) + REACTOR COOLANT + ANL + FFTF (TR) + HYDRODYNAMIC ANALYSIS + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBR

17-032940 ALSO IN CATEGORY 18 DATED 1968
ANL - REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, DECEMBER 1967
ARGONNE NATIONAL LABORATORY
ANL-7403 +. 168 PAGES, FIGURES, TABLES, JANUARY 31, 1968

SECTION TITLES - PLUTONIUM UTILIZATION, EBWR, LIQUID-METAL FAST-BREEDER REACTORS, GENERAL REACTOR TECHNOLOGY, ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT, AND NUCLEAR SAFETY. SEARCH FOR LOCATION OF FISSION-GAS RELEASE IN EBR-2 CONTINUES. LMFBR TOPICS INCLUDE EBR-2 OPERATION, PHYSICS, COMPONENT, FUEL DEVELOPMENT, PROGRESS ON FFTF. NUCLEAR SAFETY TOPICS INCLUDE ACCIDENT ANALYSIS, COOLANT DYNAMICS, FUEL MELTDOWN STUDIES WITH TREAT, TREAT OPERATIONS, CONTAINMENT BY ENERGY ABSORPTION, AND PLUTONIUM VOLATILITY STUDIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT + REACTOR, FAST + FISSION GAS RELEASE + FUEL MELTDOWN + REACTOR, LMCR + REPORT, OPERATIONS + PLUTONIUM + REACTOR, SAFETY RESEARCH + REACTOR, TEST + TREAT (PRR) + REACTOR COOLANT + ANL + FFTF (TR) + HYDRODYNAMIC ANALYSIS + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBR

17-032941 ALSO IN CATEGORY 18 DATED 1967
ANL - REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, SEPTEMBER 1967
ARGONNE NATIONAL LABORATORY
ANL-7382 +. 155 PAGES, FIGURES, TABLES, SEPTEMBER 1967

SECTION TITLES - PLUTONIUM UTILIZATION, EBWR, LIQUID-METAL FAST-BREEDER REACTORS, GENERAL REACTOR TECHNOLOGY, ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT, AND NUCLEAR SAFETY. SEARCH FOR LOCATION OF FISSION-GAS RELEASE IN EBR-2 CONTINUES. LMFBR TOPICS INCLUDE EBR-2 OPERATION, PHYSICS, COMPONENT AND FUEL DEVELOPMENT, PROGRESS ON FFTF. NUCLEAR SAFETY TOPICS INCLUDE ACCIDENT ANALYSIS, COOLANT DYNAMICS, FUEL MELTDOWN STUDIES WITH TREAT, TREAT OPERATIONS, CONTAINMENT BY ENERGY ABSORPTION, AND PLUTONIUM VOLATILITY STUDIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

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17-032942 ALSO IN CATEGORY 18 DATED 1967
ANL - REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, AUGUST 1967
ARGONNE NATIONAL LABORATORY
ANL-7371 +. 126 PAGES, 25 FIGURES, TABLES, AUGUST 1967

SECTION TITLES - PLUTONIUM UTILIZATION, EBWR, LIQUID-METAL FAST-BREEDER REACTORS, GENERAL REACTOR TECHNOLOGY, ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT, AND NUCLEAR SAFETY. SEARCH FOR LOCATION OF FISSION-GAS RELEASE IN EBR-2 CONTINUES. LMFBR TOPICS INCLUDE EBR-2 OPERATION, PHYSICS, COMPONENT AND FUEL DEVELOPMENT, PROGRESS ON FFTF. NUCLEAR SAFETY TOPICS INCLUDE ACCIDENT ANALYSIS, COOLANT DYNAMICS, FUEL MELTDOWN STUDIES WITH TREAT, TREAT OPERATIONS, CONTAINMENT BY ENERGY ABSORPTION, AND PLUTONIUM VOLATILITY STUDIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT + REACTOR, FAST + FISSION GAS RELEASE + FUEL MELTDOWN + REACTOR, LMCR + REPORT, OPERATIONS + PLUTONIUM + REACTOR, SAFETY RESEARCH + REACTOR, TEST + TREAT (PRR) + REACTOR COOLANT + ANL + FFTF (TR) + HYDRODYNAMIC ANALYSIS + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBR

17-032943 ALSO IN CATEGORY 18 DATED 1967
ANL - REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JULY 1967
ARGONNE NATIONAL LABORATORY
ANL-7357 +. 148 PAGES, 27 FIGURES, TABLES, JULY 1967

SECTION TITLES - PLUTONIUM UTILIZATION, EBWR, LIQUID-METAL FAST-BREEDER REACTORS, GENERAL REACTOR TECHNOLOGY, ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT, AND NUCLEAR SAFETY. SEARCH FOR LOCATION OF FISSION-GAS RELEASE IN EBR-2 CONTINUES. LMFBR TOPICS INCLUDE EBR-2 OPERATION, PHYSICS, COMPONENT AND FUEL DEVELOPMENT, PROGRESS ON FFTF. NUCLEAR SAFETY TOPICS INCLUDE ACCIDENT ANALYSIS, COOLANT DYNAMICS, FUEL MELTDOWN STUDIES WITH TREAT, TREAT OPERATIONS, CONTAINMENT BY ENERGY ABSORPTION, AND PLUTONIUM VOLATILITY STUDIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-032943 *CONTINUED*

CONTAINMENT + REACTOR, FAST + FISSION GAS RELEASE + FUEL MELTDOWN + REACTOR, LMCR + REPORT, OPERATIONS + PLUTONIUM + REACTOR, SAFETY RESEARCH + REACTOR, TEST + TREAT (PRR) + REACTOR COOLANT + ANL + FFTF (TR) + HYDRODYNAMIC ANALYSIS + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBR

17-032945 ALSO IN CATEGORY 18

DATED 1968

ARGONNE NATIONAL LABORATORY. REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JANUARY 1968

ARGONNE NATIONAL LABORATORY

ANL-7419 +. 158 PAGES, FIGURES, TABLES, JANUARY 1968

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS - LIQUID-METAL FAST-BREEDER REACTORS, FUEL, PHYSICS, AND COMPONENT DEVELOPMENT, EBR-2 OPERATION, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY. NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR KINETICS + REACTOR, RESEARCH + REACTOR, TEST + TREAT (PRR) + ANL + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBR

17-032946 ALSO IN CATEGORY 18

DATED 1968

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, FEBRUARY 1968

ARGONNE NATIONAL LABORATORY

ANL-7427 +. 151 PAGES, FIGURES, TABLES, FEBRUARY 1968

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS - LIQUID-METAL FAST-BREEDER REACTORS, FUEL, PHYSICS, AND COMPONENT DEVELOPMENT, EBR-2 OPERATION, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY. NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR KINETICS + REACTOR, RESEARCH + REACTOR, TEST + TREAT (PRR) + ANL + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBR

17-032947 ALSO IN CATEGORY 18

DATED 1968

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MARCH 1968

ARGONNE NATIONAL LABORATORY

ANL-7438 +. 155 PAGES, FIGURES, TABLES, MARCH 1968

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS - LIQUID-METAL FAST-BREEDER REACTORS, FUEL, PHYSICS, AND COMPONENT DEVELOPMENT, EBR-2 OPERATION, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY. NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR KINETICS + REACTOR, RESEARCH + REACTOR, TEST + TREAT (PRR) + ANL + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBR

17-032949 ALSO IN CATEGORY 18

DATED 1968

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY 1968

ARGONNE NATIONAL LABORATORY

ANL-7457 +. 148 PAGES, FIGURES, TABLES, MAY 1968

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS - LIQUID-METAL FAST-BREEDER REACTORS, FUEL, PHYSICS, AND COMPONENT DEVELOPMENT, EBR-2 OPERATION, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY. NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR KINETICS + REACTOR, RESEARCH + REACTOR, TEST + TREAT (PRR) + ANL + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBR

17-032950 ALSO IN CATEGORY 18

DATED 1968

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JUNE 1968

ARGONNE NATIONAL LABORATORY

ANL-7460 +. 136 PAGES, FIGURES, TABLES, JUNE 1968

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS - LIQUID-METAL FAST-BREEDER REACTORS, FUEL, PHYSICS, AND COMPONENT DEVELOPMENT, EBR-2 OPERATION, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY. NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-032950 *CONTINUED*
REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR KINETICS + REACTOR,
RESEARCH + REACTOR, TEST + TREAT (PRR) + ANL + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBK

17-032951 ALSO IN CATEGORY 18 DATED 1968
REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, SEPTEMBER 1968
ARGONNE NATIONAL LABORATORY
ANL-7500 +. 139 PAGES, FIGURES, TABLES, SEPTEMBER 1968

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS - LIQUID-METAL FAST-BREEDER REACTORS, FUEL, PHYSICS,
AND COMPONENT DEVELOPMENT, EBR-2 OPERATION, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY.
NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR KINETICS + REACTOR,
RESEARCH + REACTOR, TEST + TREAT (PRR) + ANL + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBK

17-032952 ALSO IN CATEGORY 18 DATED 1968
REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, OCTOBER 1968
ARGONNE NATIONAL LABORATORY
ANL-7513 +. 144 PAGES, FIGURES, TABLES, OCTOBER 1968

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS - LIQUID-METAL FAST-BREEDER REACTORS, FUEL, PHYSICS,
AND COMPONENT DEVELOPMENT, EBR-2 OPERATION, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY.
NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR KINETICS + REACTOR,
RESEARCH + REACTOR, TEST + TREAT (PRR) + ANL + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBK

17-032953 ALSO IN CATEGORY 18 DATED 1968
THE REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, NOVEMBER 1, 1968
ARGONNE NATIONAL LABORATORY
OR-ANL-7518 +. 131 PAGES, FIGURES, TABLES, NOVEMBER 1

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS - LIQUID-METAL FAST-BREEDER REACTORS, FUEL, PHYSICS,
AND COMPONENT DEVELOPMENT, EBR-2 OPERATION, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY.
NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS.

968AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151
\$3.00 COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR
KINETICS + REACTOR, RESEARCH + REACTOR, TEST + TREAT (PRR) + ANL + REACTOR, PULSED + R AND D PROGRAM +
REACTOR, LMFBK

17-032990 ALSO IN CATEGORY 18 DATED 1968
ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT
ARGONNE NATIONAL LABORATORY
ANL-7487 +. 112 PAGES, FIGURES, TABLES, REFERENCES, AUGUST 1968

ONE OF A SERIES. TOPICS ARE - LMFBK DEVELOPMENT PROGRAM, STUDIES OF GENERAL REACTOR TECHNOLOGY,
AND NUCLEAR SAFETY. EBR-2 AND TREAT OPERATIONS DISCUSSED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + SODIUM + REACTOR, TEST + TREAT (PRR) + METAL, LIQUID + R
AND D PROGRAM + REACTOR, LMFBK

17-033016 ALSO IN CATEGORY 11 DATED 1969
CHANGE 14 - PRIMARY SHIELD TANK ATMOSPHERE
U.S. ATOMIC ENERGY COMMISSION
5 PAGES, LETTER TO M.C. BEEKMAN, PRDC, MARCH 26, 1969, DOCKET 50-16 TYPE--LMFBK, MFG--APDA, AE--COMMONWEALTH
ASSOC.

ALLOWS MAINTENANCE OPERATION WITH AIR IN PRIMARY SHIELD TANK FOR 10,000 HR OR LESS TO REPLACE BOOT
SEAL. PROVIDES FOR GRAPHITE PURGING. *** DRL SAFETY EVALUATION CONSIDERS BORON LOSS FROM
GRAPHITE SHIELDING, SHIELD TANK COLLAPSE, AND LEAKAGE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM
CHARGE \$2.00

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-033016 *CONTINUED*

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + PRESSURE VESSELS + TECHNICAL SPECIFICATIONS + SEAL + CONTAINMENT ATMOSPHERE, INERT + SAFETY EVALUATION + REACTOR, LMFBR

17-033429 ALSO IN CATEGORIES 1 AND 18 DATED 1968
LMFBR PROGRAM PLAN. VOLUME I - OVERALL PLAN
ARGONNE NATIONAL LAB.
WASH-1101 +. 350 PAGES, FIGURES, TABLES, AUGUST 1968

CONTENTS - THE OBJECTIVES, SCOPE, BACKGROUND INFORMATION, REQUIREMENTS AND GOALS, PLAN ORGANIZATION, NEEDS, PLAN OF ACTION, SCHEDULING AND PRIORITIES, AND FACILITIES. APPENDICES CONTAIN COLLECTION OF TASKS AND CONTRACT WORK, FACILITIES TABLE, GLOSSARY, AND PRIVATELY SUPPORTED WORK. NINE OTHER ELEMENTS OF THE PLAN, WHICH ARE PRESENTED IN SEPARATE VOLUMES, ARE PLANT DESIGN, COMPONENTS, INSTRUMENTATION AND CONTROL, SODIUM TECHNOLOGY, CORE DESIGN, FUELS AND MATERIALS, FUEL RECYCLE, PHYSICS, AND SAFETY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, BREEDER + REACTOR, FAST + SODIUM + REACTOR COOLANT + DESIGN STUDY + REVIEW + R AND D PROGRAM + REACTOR, LMFBR + JACOBS

17-033489 DATED 1968
MATTHEWS RR + HENRY KJ
LOCATION AND REPAIR OF THE DFR LEAK
UNITED KINGDOM ATOMIC ENERGY AUTHORITY, SCOTLAND
5 PAGES, NUCL. ENG., 13, PAGES 840-44 (OCTOBER 1968)

THE NAK LEAK WAS INDICATED BY A SPARK-PLUG-TYPE DETECTOR IN THE LEAK JACKET CONTAINING THE REACTOR VESSEL. LEAK LOCATION INCLUDED MONITORING THE LEAK JACKET FOR COOLANT AND FOR PRESSURIZED BLANKET GAS, THE USE OF TRACERS, AND BUBBLE-NOISE MEASUREMENT ON PRESSURIZING THE LEAK JACKET. THE ERRATIC BEHAVIOR OF THE LEAK, WHICH CAUSED CONSIDERABLE DIFFICULTY IN ITS LOCATION, WAS LATER ATTRIBUTED TO THERMAL DISTORTION OF THE PIPEWORK WHILE THE REACTOR WAS ON POWER. A CRACK WAS FOUND IN THE AREA OF A WELD ON ONE OF THE 4-IN.-DIA OUTLET PIPES. THE CUTTING, WELDING, AND INSPECTION TECHNIQUES INVOLVED IN THE REPAIR OF THE SECTION ARE DESCRIBED. DETAILS ARE GIVEN OF THE ORGANIZATION OF THE 11-MONTH OPERATION.

REACTOR, FAST + MAINTENANCE AND REPAIR + REACTOR, TEST + UNITED KINGDOM + WELDING + REPORT, OPERATIONS ANALYSIS + TEST, LEAK LOCATION + DOUNREAY (LMFBR) + MAIN COOLING SYSTEM + FAULT + LEAK + REACTOR, LMFBR

17-033982 DATED 1967
BICKEL W + ENGELMANN P + HAFELE W + WITTEK G
SAFETY CONSIDERATIONS FOR THE KARLSRUHE FAST CRITICAL FACILITY SNEAK
KERNFORSCHUNGSZENTRUM KARLSRUHE, INST. FÜR ANGEWANDTE REAKTORPHYSIK
33 PAGES, 9 FIGURES, 1 TABLE, 7 REFERENCES, PAPER PRESENTED AT THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

SNEAK IS A FLEXIBLE FAST ZERO POWER REACTOR. IT IS THE FIRST FAST REACTOR AND THE FIRST WITH LARGE AMOUNTS OF PLUTONIUM FUEL IN GERMANY. IT HAS BEEN BUILT IN THE DENSELY POPULATED AREA OF KARLSRUHE. DUE TO THESE FACTS THE SAFETY ASPECTS OF SNEAK HAD TO BE EMPHASIZED. THE SAFETY FEATURES INCORPORATED IN THE DESIGN OF SNEAK AND THE ADMINISTRATIVE MEASURES TO GUARANTEE SAFE OPERATION OF THE REACTOR ARE DESCRIBED IN DETAIL. A SHORT REVIEW IS GIVEN OF THE OPERATION EXPERIENCE AND OF THE LICENSING PROCEDURE OF SNEAK.

AVAILABILITY - W. BICKEL/P. ENGELMANN/W. HAFELE/G. WITTEK, INST. FÜR ANGEWANDTE REAKTORPHYSIK, KERNFORSCHUNGSZENTRUM, KARLSRUHE, GERMANY

ADMINISTRATIVE CONTROL + DESIGN CRITERIA + SITING, REACTOR + OPERATING EXPERIENCE SUMMARY + GERMANY + SAFETY PRINCIPLES AND PHILOSOPHY + R AND D PROGRAM + REACTOR, LMFBR

17-034636 ALSO IN CATEGORY 18 DATED 1968
SUMMARY OF TREAT SODIUM LOOP EXPERIMENTS ON BEHAVIOR OF SINGLE EBR-II MARK I PINS UNDER ACCIDENT CONDITIONS
12 PAGES, 10 FIGURES, 1 TABLE, 31 REFERENCES, NUCLEAR ENGINEERING AND DESIGN, VOL. 7, PAGES 442-454 (MAY 1968)

PRESENTED IS A STUDY, EXPERIMENTALLY AND BY COMPUTATION, OF MECHANISMS OF FUEL FAILURE AND ASSOCIATED MOVEMENT OF FUEL AND COOLANT DURING CORE MELTDOWN ACCIDENT. THESE TWO PHENOMENA COULD LEAD TO EXTENSION OF ANY INITIAL FUEL FAILURE AND WOULD PROBABLY BE FACTORS CAUSING SERIOUS MELTDOWN. *** FUEL FAILURE EXPERIMENTS IN INSTRUMENTED SODIUM LOOPS WERE CONDUCTED AT VARIOUS CONDITIONS OF TEMP. AND FLOW OF SODIUM COOLANT. POWER BURSTS OF THE TREAT REACTOR, INTO WHICH LOOPS WERE INSERTED, WAS ADJUSTED TO SPECIFIED INTENSITIES. RESULTS OF TESTS AND ANALYSIS ARE PRESENTED.

COMPUTER PROGRAM + EBR 1 AND 2 (RE) + REACTOR, FAST + FAILURE, FUEL ELEMENT + IN PILE EXPERIMENT + REACTOR, LMCR + SODIUM + COMPARISON, THEORY AND EXPERIENCE + REACTOR, TEST + TREAT (PRR) + ANALYTICAL MODEL + CORE MELTDOWN + IN PILE EXPERIMENT + REACTOR, PULSED + MATHEMATICAL TREATMENT

17-034706 ALSO IN CATEGORIES 3 AND 12 AND 13 DATED 1969

IX

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-034706 *CONTINUED*

JOHNSON KP + MATTE J

OPERATING EXPERIENCE WITH THE FERMI FUEL HANDLING CASK CAR

POWER REACTOR DEVELOPMENT COMPANY, MONROE, MICHIGAN + ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN

15 PAGES, 9 FIGURES, PP. 135-149 OF PROCEEDINGS OF 16TH CONFERENCE ON REMOTE SYSTEMS TECHNOLOGY, 1969

THE ENRICO FERMI ATOMIC POWER PLANT WAS PROVIDED WITH A CASK CAR FOR TRANSPORTING SPENT FUEL BETWEEN THE REACTOR BUILDING AND THE FUEL AND REPAIR BUILDING. THIS CAR, A MULTI-STATION, SELF-CONTAINED UNIT, EXHIBITED A VARIETY OF PROBLEMS AND DEFECTS IN EARLY OPERATION. ANALYSIS OF THESE PROBLEMS AND THE AVAILABLE SOLUTIONS REQUIRED TO ACHIEVE SATISFACTORY PERFORMANCE UNDER DESIGNED OPERATING CONDITIONS LED TO CONSIDERATION OF REPLACING THE CASK CAR WITH A NEW FACILITY.

AVAILABILITY - K.P. JOHNSON, POWER REACTOR DEVELOPMENT COMPANY, P. O. BOX 725, MONROE, MICHIGAN 48161

REACTOR, BREEDER + FAILURE, EQUIPMENT + REACTOR, FAST + FERMI (LMFBR) + FUEL HANDLING + REACTOR, LMCR + SHIPPING CONTAINER + TRANSPORTATION AND HANDLING + FUEL HANDLING MACHINE + OPERATING EXPERIENCE + REACTOR, LMFBR

17-035763

DATED 1969

RETIREMENT PROGRESS AT HALLAM

CONSUMERS POWER COMPANY

13 PAGES, FIGURES, TABLES, HALLAM MONTHLY RETIREMENT REPORT NO. J2, MARCH 1969, DOCKET 115-3, TYPE--LMR, MFG--A.E., AE--BECHTEL

LOSS OF 4 DAYS DUE TO INCLEMENT WEATHER AND OTHER PERIODS OF COLD TEMPERATURE AND STRONG WIND ALONG WITH ERRONEOUS ESTIMATE OF IRONWORKER PROFICIENCY HAS SET COMPLETION DATE BACK TO MID-MAY FOR REACTOR BUILDING DEMOLITION. COMPLETION OF ISOLATION-STRUCTURE WEATHERPROOFING HAS BEEN SET BACK TO JULY 1969. REACTOR-BUILDING SIDING REMOVAL IS 70 PERCENT COMPLETE. AUXILIARY-BAY STRUCTURAL STEEL AND GENERATOR-ROOM REMOVAL ARE 60 PERCENT COMPLETE. NOTED NO DEVIATION FROM SAFE PRACTICES.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + DECOMMISSIONING

17-035802

DATED 1969

PROPOSED CHANGE 15 - MODIFICATION TO FUEL HANDLING CASK

POWER REACTOR DEVELOPMENT CORPORATION

75 PAGES, FIGURES, TABLES, LETTER TO P.A. MORRIS, AEC-DRL, MARCH 26, 1969, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

CASK CAR CONTAINING 17,500 LB OF DEPLETED U-238 WAS DISMANTLED AND REPLACED WITH FUEL-TRANSPORT FACILITY, THUS REQUIRING CHANGE REQUEST. CHANGE ITEM D (B) ON PAGE 2 TO READ - (B) 9,000 LB DEPLETED U-238 CONSTITUTING SHIELDING FOR THE FUEL TRANSPORT FACILITY. *** INCLUDED IN TECH.SPEC. 15 CHANGES ARE - DIESEL GENERATOR SHALL BE OPERATING AND CONNECTED TO BUS DURING CLEANING CHAMBER OPERATIONS INVOLVING FUEL WITH MORE THAN 100 WATTS OF DECAY HEAT. I-131 ACTIVITY SHALL BE LESS THAN 10,000 CI IN TRANSPORT FACILITY AND LESS THAN 125 CI IN CLEANING CHAMBER OR TRANSFER TANK. MAXIMUM PERMISSIBLE HEAT GENERATION RATE SHALL BE 4 KW. INERT ATM. SHALL BE MAINTAINED WITHIN FACILITY CAVITY. ONE-AT-A-TIME LIMITATIONS PLACED ON SOME COMPONENTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL HANDLING + REACTOR, LMCR + MODIFICATION + TECHNICAL SPECIFICATIONS + FUEL HANDLING MACHINE + REACTOR, LMFBR + GENERATOR, DIESEL

17-036053

DATED 1969

ARTERBURN JO

SEFOR STARTUP AND INITIAL OPERATION

GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIFORNIA

14 PAGES, 2 FIGURES, 1 REFERENCE, PAPER PRESENTED AT THE AMERICAN POWER CONFERENCE, MARCH 1969

ALTHOUGH SEFOR PERSONNEL HAD A NUMBER OF YEARS OF EXPERIENCE IN OPERATIONS AND MAINTENANCE OF WATER REACTORS, ONLY ONE HAD EXPERIENCE WITH SODIUM. FILLING AND PURIFICATION (OXIDE COLD TRAPS) OF SODIUM SYSTEM WAS DONE EXPEDITIOUSLY AND SAFELY. NITROGEN COOLANT AND FREON SYSTEMS REQUIRED GREATEST EFFORT TO MAKE OPERATIONAL. LOWERING THE SODIUM LEVEL JUST BELOW TOPS OF HEXAGONAL FUEL CHANNELS OFFERED A GOOD LIGHT-REFLECTING SURFACE FOR COKE LOCATION-IDENTIFICATION DURING REMOTE REFUELING. LOW-POWER TEST WILL REQUIRE 4 TO 6 MOS. IN TRANSIENT TESTS, REACTIVITY WILL BE INSERTED FROM LOW VALUES IN SUBPROMPT TO SUPERPROMPT RANGE AS HIGH AS 1.3%.

AVAILABILITY - J.O. ARTERBURN, GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIFORNIA

REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + SODIUM + TRAINING + REACTOR STARTUP EXPERIENCE + REACTOR, PULSED + SEFOR (RE) + REMOTE MANIPULATING AND VIEWING + COOLANT PURIFICATION SYSTEM

17-036078

ALSO IN CATEGORIES 6 AND 9

DATED 1969

SMITH RR + BUMP TR + CUSHMAN RA + HYNDMAN RW + KIRN FS + LOEWENSTEIN WB + LONG KJ + MADELL JT + PERSIANI PJ +

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-036078 *CONTINUED*

WALLIN WK
THE EFFECTS OF AN OVER-COOLED STAINLESS STEEL REFLECTOR ON THE EBR II POWER COEFFICIENT
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS
ANL-7544 +. 115 PAGES, 60 FIGURES, 20 TABLES, 42 REFERENCES, MAY, 1969

DURING A ROUTINE STARTUP FOR EBR-II RUN 25 ON APRIL 21, 1967, SIGNIFICANT CHANGES WERE NOTED IN THE BASIC BEHAVIOR OF THE POWER COEFFICIENT. THE MAGNITUDE OF THE INTEGRAL POWER COEFFICIENT DECREASED BY A FACTOR OF APPROXIMATELY TWO WHILE THE MAGNITUDE OF THE DIFFERENTIAL POWER COEFFICIENT IN THE MIDPOWER REGION DECREASED BY A FACTOR OF APPROXIMATELY FOUR. CONSIDERABLE ANALYTICAL EFFORT SHOWED THE CHANGE IN THE POWER COEFFICIENT TO BE IDENTIFIED WITH THERMOMECHANICAL DEFORMATION EFFECTS IN A STAINLESS STEEL REFLECTOR THAT WAS INSTALLED BEFORE THE STARTUP.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

POWER COEFFICIENT + REACTOR STABILITY + STEEL, STAINLESS + DEFORMATION + MEASUREMENT, REACTIVITY + REFLECTOR + SYSTEM DESCRIPTION + REACTIVITY EFFECT, ANOMALOUS + REACTOR PHYSICS + OPERATING EXPERIENCE + REACTOR, LMFBR

17-036110

DATED 1969

HALLAM MONTHLY RETIREMENT REPORT NO. 34, MAY, 1969
CONSUMERS PUBLIC POWER DISTRICT
20 PAGES, HALLAM MONTHLY RETIREMENT REPORT, NO. 34, MAY 1969, DOCKET 115-3, TYPE--LMR, MFG--A.E., AE--BECHTEL

ISOLATION STRUCTURE RETAINING WALLS AND WEATHER PROOFING WAS STARTED. DISASSEMBLY OF REACTOR BUILDING STRUCTURE CONTINUED. A WHOLE-BODY COUNTER WAS RECEIVED FOR CHECKING 20 CPPD EMPLOYEES PARTICIPATING IN PROGRAM. NO DEVIATION FROM SAFE-PRACTICE PROCEDURES NOTED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + MONITOR, PERSONNEL + REPORT, OPERATIONS + BUILDING + DECOMMISSIONING

17-036111

DATED 1969

HALLAM MONTHLY RETIREMENT REPORT NO. 31, FEBRUARY 1969
CONSUMERS PUBLIC POWER DISTRICT
10 PAGES, HALLAM MONTHLY RETIREMENT REPORT NO. 31, FEBRUARY 1969, DOCKET 155-3, TYPE--LMR, MFG--A.E., AE--BECHTEL

THE AUXILIARY BAY AND REACTOR BAY CRANES WERE REMOVED AND SHIPPED, AND THE BAY ROOFS WERE REMOVED. REACTOR BAY AND AUXILIARY BAY WALLS ARE BEING REMOVED. *** FUTURE ACTIVITIES WILL BE TO COMPLETE REMOVAL OF WALL BETWEEN EAST-WEST COLUMNS 13-15 AND NORTH-SOUTH COLUMNS A AND E. A RETAINING WALL WILL BE INSTALLED AND CONVENTIONAL PLANT BUILDING RESEALED. *** NO ACTIVITIES VIOLATED SAFE-PRACTICE PROCEDURES.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + REPORT, OPERATIONS + BUILDING + DECOMMISSIONING + EQUIPMENT

17-036625

ALSO IN CATEGORY 18

DATED 1969

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT JUNE 1969
ARGONNE NATIONAL LABORATORY
ANL-7581 +. 136 PAGES, FIGURES, TABLES, JUNE 1969

ONE OF A SERIES. TOPICS - LMFBR DEVELOPMENT, EBR-2 OPERATIONS, FAST-REACTOR FUEL DEVELOPMENT, FAST-REACTOR ANALYSIS METHODS AND DATA, CHEMICAL SEPARATIONS, REACTOR KINETICS, FUEL MELTDOWN STUDIES, TREAT OPERATIONS, AND CHEMICAL REACTIONS AND ENERGY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CHEMICAL REACTION + REACTOR, FAST + FUEL MELTDOWN + REACTOR, LMCR + METAL WATER REACTION + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR, RESEARCH + REACTOR, TEST + TREAT (PRR) + REACTOR, ERDA OWNED + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBR

17-036627

ALSO IN CATEGORY 11

DATED 1969

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT APRIL-MAY 1969
ARGONNE NATIONAL LABORATORY
ANL-7577 +. 206 PAGES, FIGURES, TABLES, JULY 7, 1969

EBR-2 OPERATIONS - A SMALL SODIUM FIRE OCCURRED WHILE A SAMPLE WAS BEING TAKEN FROM THE PRIMARY SYSTEM. A STEADY DRIPPING OF SODIUM FROM THE SAMPLING PUMP WAS THE CAUSE. MARK-1A FUEL PINS SHORTENED DURING IRRADIATION - CAUSE IS BEING INVESTIGATED. ***LMFBR DEVELOPMENTS ARE SUMMARIZED. *** NUCLEAR SAFETY - DISCUSSES REACTOR KINETICS STUDIES, TREAT OPERATIONS, CHEMICAL REACTIONS AND ENERGY, AND ENGINEERED SAFETY FEATURES.

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-036627 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CHEMICAL REACTION + REACTOR, FAST + FIRE + REACTOR, LMCR + METAL WATER REACTION + REPORT, OPERATIONS +
REACTOR, SAFETY RESEARCH + REACTOR KINETICS + REACTOR, RESEARCH + SODIUM + REACTOR, TEST + TREAT (PRR) +
IRRADIATION TESTING + REACTOR, PULSED + R AND D PROGRAM + ENGINEERED SAFETY FEATURE + REACTOR, LMFBR + FUEL
ELEMENTS

17-036630 ALSO IN CATEGORY 18 DATED 1969
NOTICE OF ISSUANCE OF OPERATING LICENSE FOR 20-MW(T) OPERATION OF SEFOR
USAEC DIVISION OF REACTOR LICENSING
20 PAGES, LETTER - DRL TO GENERAL ELECTRIC COMPANY - AUGUST 21, 1969, DOCKET 50-231

AMENDMENT TO PROVISIONAL OPERATING LICENSE WOULD PERMIT (1) OPERATION AT STEADY STATE UP TO 20
MW(T) AND IN THE PULSED MODE, AND (2) THE RECEIPT POSSESSION AND USE OF UP TO 10 MILLICURIES OF
KR-85, I-131, XE-133, AND CS-137. RELATED SAFETY EVALUATION AND ACRS REPORT ARE ENCLOSED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM
CHARGE \$2.00

REACTOR, FAST + REACTOR, LMCR + ACRS + REACTOR, PULSED + SEFOR (RE) + SAFETY EVALUATION + OPERATING LICENSE
PROCESS

17-036779 ALSO IN CATEGORY 11 DATED 1969
MATTHEWS KR + HENRY KF
LOCATION AND REPAIR OF A LEAK IN THE DOUNREAY FAST REACTOR PRIMARY CIRCUIT
10 PAGES, 13 FIGURES, JOURNAL OF THE BRITISH NUCLEAR ENERGY SOCIETY, 8(3), PAGES 168-177 (JULY 1969)

DESCRIBES METHODS USED TO DETECT, LOCATE, AND REPAIR THE LEAK. THE NAK COOLANT WAS DISCHARGED
COMPLETELY, AND A HELIUM-GAS PRESSURE TEST STARTED. LEAK WAS SMALL, SO, TO INCREASE LEAK RATE,
EACH OF THE 24 HEAT EXCHANGERS WAS MOVED 4 IN. TO OPEN THE LEAK AND OBTAIN A BIGGER LEAK RATE TO
FACILITATE LOCATION. THIS ASSOCIATED LEAK WITH CIRCUIT 10A OR 10B. A BUBBLE-NOISE TECHNIQUE
WITH MICROPHONES CLAMPED ON PIPES WAS USED TO LOCATE LEAK MORE PRECISELY. A CRACK IN A FAULTY
WELD WAS FOUND. DETAILED DESCRIPTION OF REPAIR IS GIVEN.

REACTOR, FAST + MAINTENANCE AND REPAIR + FAILURE, PIPE + REACTOR, TEST + UNITED KINGDOM + WELDS + NAK +
FAILURE, FABRICATION ERROR + TEST, LEAK LOCATION + METAL, LIQUID + DOUNREAY (LMFBR) + MAIN COOLING SYSTEM +
LEAK + OPERATING EXPERIENCE + REACTOR, LMFBR

17-037406 ALSO IN CATEGORY 9 DATED 1969
VOLK ER + GREEN DM + BATCH ML
DESIGN AND CAPABILITIES OF THE MALFUNCTION DETECTION ANALYZER INSTALLED ON THE ENRICO FERMI FAST BREEDER
REACTOR
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-L3 +. 76 PAGES, FIGURES, TABLES, 19 REFERENCES, JULY 1969

AS A RESULT OF THE FUEL MELTING INCIDENT (FERMI, OCTOBER 5, 1966), A STUDY WAS INITIATED TO
DETERMINE A PROMPT, RELIABLE, AND ECONOMIC MEANS TO DETECT MALFUNCTIONS THAT COULD LEAD TO A
SIMILAR INCIDENT. CONDITIONS THAT COULD RESULT IN A SECOND INCIDENT WERE REVIEWED. CALCULATIONS
CONSIDERED CRITICAL FOR DETECTION WERE SELECTED FOR DETAILED ANALYSIS. THE SELECTION AND
DESCRIPTION OF THE REAL-TIME COMPUTER SYSTEM INSTALLED ARE INCLUDED.

REACTOR, BREEDER + COMPUTER PROGRAM + COMPUTER, DIGITAL + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN +
INSTRUMENT, PROTECTIVE + REACTOR, LMCR + REACTOR SAFETY SYSTEM + INCIDENT + INSTRUMENT, DETECTION FAILED FUEL
ELEMENT + MONITOR + SYSTEM DESCRIPTION + REACTIVITY EFFECT, ANOMALOUS + COMPUTER CONTROL + REACTOR, LMFBR +
FAILURE, INCIPIENT

17-037469 DATED 1969
MUELLER RE
COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT, JANUARY 1969
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN
APDA-CFE-29 +. 42 PAGES, FIGURES, TABLES, JUNE 1969

PRINCIPAL ACTIVITY INVOLVED COMPLETING THE LOWER PLENUM RESTORATION OPERATIONS UNDERTAKEN AFTER
REMOVAL OF THE ZIRCONIUM SEGMENTS. WELDING OF SCREW SHANKS TO FLOW CONE WAS COMPLETED. A VACUUM
CLEANER WAS USED IN REMOVING DEBRIS IN LOWER PLENUM. MODIFICATIONS TO NEUTRON DETECTOR TUBES IN
GRAPHITE SHIELD WERE MADE. A TUBE SEAL DESIGN WAS FOUND INADEQUATE ALLOWING COOLING GAS TO
BYPASS DETECTION. GASKET REPLACEMENT FOR STEAM GENERATOR WATER MANIFOLD COVER WAS BEGUN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + INSTRUMENT, NUCLEAR + REACTOR, LMCR +
MAINTENANCE AND REPAIR + REPORT, OPERATIONS + MODIFICATION + INCIDENT, RECOVERY FROM + SEAL + STEAM GENERATOR
+ OPERATING EXPERIENCE + COOLING + REACTOR, LMFBR

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-037487 ALSO IN CATEGORY 9 DATED 1969
ENRICO FERMI ATOMIC POWER PLANT REPORT FOR MARCH 1969
POWER REACTOR DEVELOPMENT COMPANY
EF-67 +. 11 PAGES, MARCH 1969, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

RESTORATION OF PRIMARY PIPING WAS COMPLETED AND THE VESSEL FILLED WITH SODIUM AND HEATED TO 450 F FOR SODIUM CLEANUP. 53 HR OF STARTUP TRAINING WERE LOGGED BY THE OPERATORS ON THE SIMULATOR. A DEFECTIVE RELAY COIL IN THE BUS THROWOVER CIRCUIT CAUSED THE CONTROL POWER FUSE TO BLOW. ELECTRICAL CONTAINMENT PENETRATIONS WERE LEAK-RATE TESTED. AN IBM SERIES-1800 COMPUTER WAS RECEIVED. THE COMPUTER WILL PERFORM COMPUTATIONS EVERY 5 SEC UNDER NORMAL CONDITIONS FOR ANOMALOUS REACTIVITY, SUBASSEMBLY OUTLET TEMP., AND FISSION-PRODUCT BEHAVIOR VS POWER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + COMPUTER, DIGITAL + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + MAINTENANCE AND REPAIR + REPORT, OPERATIONS + TEST, LEAK RATE + TRAINING + FAILURE, INSTRUMENT + INSTRUMENT, SURVEILLANCE + SIMULATION + CONTAINMENT PENETRATION, ELECTRICAL + REACTOR, LMFBR + RELAYS

17-037838 ALSO IN CATEGORIES 5 AND 12 DATED 1968
HIGH-TEMPERATURE LIQUID-METAL TECHNOLOGY REVIEW
BROOKHAVEN NATIONAL LABORATORY
BNL-50115(PK-32) +. 88 PAGES, 8 FIGURES, REFERENCES, APRIL 1968

THE PURPOSE OF THE REVIEW IS TO PROVIDE UP-TO-DATE INFORMATION ON THE VARIOUS RESEARCH AND DEVELOPMENT PROGRAMS IN THE UNITED STATES IN THE FIELD OF HIGH-TEMPERATURE LIQUID-METAL TECHNOLOGY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + REACTOR, BREEDER + REACTOR, FAST + HEAT TRANSFER + REACTOR, LMCR + THERMODYNAMICS + REACTOR COOLANT + CORROSION + METAL, LIQUID + MASS TRANSFER + PROPERTY, PHYSICAL + REVIEW + R AND D PROGRAM

17-038113 DATED 1969

PROGRESS REPORT ON GAS-COOLED FAST BREEDER REACTOR
GULF GENERAL ATOMIC, INC.
GA-9359 +. 53 PAGES, FIGURES, TABLES, 3 REFERENCES, JUNE 12, 1969

PROGRAM UTILIZES WORK ON THE LMFBR THAT WOULD BE APPLICABLE TO GCFR. VANADIUM BASED ALLOYS MAY NOT BE APPLICABLE ABOVE 800 C. MANIFOLDED FUEL-ROD SELECTED AS REFERENCE CONCEPT. IN-PILE AND OUT-OF-PILE TESTS ARE BEING PERFORMED ON FUEL-CLADDING INTERACTING STUDIES. PLANS UNDERWAY FOR FAST FLUX TESTING IN EBR-II. CRITICAL EXPERIMENTS TO BE PERFORMED COOPERATIVELY WITH BNWL USING PLUTONIUM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + BATTELLE NORTHWEST + VANADIUM + REACTOR, FAST, GAS COOLED + REACTOR, LMFBR

17-040390 DATED 1969
ADDITIONAL INFORMATION ON THE FERMI FUEL MELTING INCIDENT, PLANS FOR STARTUP AND POWER ASCENSION AND PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS
POWER REACTOR DEVELOPMENT COMPANY
75 PAGES, PROPOSED CHANGES TO TECH SPECS, AUGUST 20, 1969, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

ANSWERS 6 AEC QUESTIONS ON THE FUEL-MELTING INCIDENT; 4 ON LMFBR EXPERIENCE AND R AND D PROGRAMS AND 3 ON THE READINESS FOR RESUMPTION OF OPERATIONS. REVIEWS CREDIBLE AND HYPOTHETICAL ACCIDENTS, CORROSION OF THE PRIMARY SYSTEM, AND PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT ANALYSIS + NRC QUESTION + REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + CORROSION + TECHNICAL SPECIFICATIONS + INCIDENT, RECOVERY FROM + MAIN COOLING SYSTEM + R AND D PROGRAM + REACTOR, LMFBR

17-040402 ALSO IN CATEGORIES 5 AND 6 DATED 1969
HORI M + HOSLER A
A STUDY OF GAS ENTRAINMENT AND ABSORPTION-DESORPTION IN THE ENRICO FERMI REACTOR AND THEIR EFFECT ON CORE PERFORMANCE
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-LA-6 +. 50 PAGES, FIGURES, TABLES, 19 REFERENCES, JULY 1969, DOCKET 50-16, TYPE--LMFBR, MFG--APDA, AE--COMMONWEALTH ASSOC.

CATEGORY 17
OPERATIONAL SAFETY AND EXPERIENCE

17-040402 *CONTINUED*

THE POSSIBLE PRESENCE OF GAS IN THE PRIMARY COOLANT AND ITS EFFECT ON CORE HEAT TRANSFER ARE DISCUSSED. GAS COULD BE SLOWLY INTRODUCED INTO THE PRIMARY SYSTEM BY GAS BUBBLE ENTRAINMENT AND/OR DIFFERENTIAL TEMPERATURE SOLUBILITY BETWEEN THE HOT AND COLD PORTIONS OF THE PRIMARY SYSTEM. A FEW POTENTIAL SITES FOR THE ACCUMULATION OF THIS GAS HAVE BEEN FOUND AND THE MAXIMUM ACCUMULATED VOLUMES AND RATES OF GAS RELEASE FROM THESE SITES DETERMINED. THE EFFECT OF A SINGLE LARGE BUBBLE AND ALSO THE CONTINUOUS PRESENCE OF GAS ON THE CORE HEAT TRANSFER HAVE BEEN STUDIED.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + HEAT TRANSFER + REACTOR, LMCR + REACTOR STABILITY + COOLANT CHEMISTRY + INCIDENT + BUBBLE + REACTIVITY EFFECT, VOID + GAS + REACTOR, LMFBR

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-002241 DATED 1957
KOCH L + MONSON H + JIKRENT D + LEVENSON M + SIMMONS W + HUMPHREYS J + HAUGSNES J + JANKUS V + LOEWENSTEIN W
HAZARD SUMMARY REPORT EXPERIMENTAL BREEDER REACTOR II (EBR-II)
ARGONNE NATIONAL LABORATORY, LEMONT, ILLINOIS
ANL-5719 +. 383 PAGES, 29 TABLES, 148 FIGURES, 11 REFERENCES, MAY 1957

62.5-MW THERMAL, FAST, SODIUM-COOLED REACTOR INTENDED TO GIVE REACTOR, FUEL, AND REPROCESSING
FACILITY EXPERIENCE. POWER DENSITY, 1.3 KW/LITER.

EBR 1 AND 2 (RE) + REACTOR, LMCR + REPORT, PSAR

18-003622 DATED 1964
LIQUID METAL FAST BREEDER REACTOR DESIGN STUDY. VOLUME I - INTRODUCTION--SUMMARY AND EVALUATION--DESCRIPTION
OF THE PLANT
COMBUSTION ENGINEERING, INC.
78 PAGES, 5 TABLES, 20 FIGURES- JANUARY 1964, CEND-200 VOL. I OF II

THIS IS VOLUME I OF A CONCEPTUAL DESIGN OF A LARGE SODIUM-COOLED FAST BREEDER. THIS VOLUME DOES
NOT CONTAIN MATERIAL OF DIRECT INTEREST TO NSIC BUT MIGHT SERVE AS A BASIS FOR VOLUME II WHICH IS
EXPECTED TO CONTAIN SUCH INFORMATION.

REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION

18-005832 DATED 1963
1000-MWE SGR AND PROTOTYPE EVALUATION STUDY, VOLUME 2 - 200-MWE SGR PROTOTYPE PRELIMINARY DESIGN
ATOMICS INTERNATIONAL
339 PAGES, 17 TABLES, 16 FIGURES- DECEMBER 1963, NAA-SR-9213, VOL. 2- CFSTI

THE ACCIDENT ANALYSES CONSIDERED THE MOST SIGNIFICANT RADIOLOGICAL ACCIDENTS WHICH COULD OCCUR
WITH AN SGR AT THE AEC REFERENCE SITE. ALL OTHER ACCIDENTS CONSIDERED RESULTED IN LOWER
RADIATION DOSES TO THE PUBLIC. IN EACH ACCIDENT STUDIED, A NUMBER OF IMPROBABLE FAILURES WERE
ASSUMED TO OCCUR SIMULTANEOUSLY OR SEQUENTIALLY IN ORDER TO CREATE A POTENTIALLY HAZARDOUS
CONDITION. THE EFFECTS OF EACH ACCIDENT ON THE REQUIRED SITING DISTANCES ARE SHOWN IN TABLE 17.
THE MAXIMUM CREDIBLE ACCIDENT (MCA) FOR THIS STUDY IS DEFINED AS THAT ACCIDENT WHOSE OFF-SITE
RADIOLOGICAL CONSEQUENCES ARE NOT EXCEEDED BY ANY CREDIBLE ACCIDENT.

ACCIDENT ANALYSIS + ADMINISTRATIVE CONTROL + DESIGN CRITERIA + FIRE + FUEL HANDLING + REACTOR, GRAPHITE
MODERATED + HEAT TRANSFER + INSTRUMENT, NUCLEAR + REACTOR, LMCR + MAINTENANCE AND REPAIR + ACCIDENT, MAXIMUM
CREDIBLE (MCA) + REPORT, OPERATIONS + REACTOR DESCRIPTION + SODIUM + VENTILATION SYSTEM + ELECTRIC POWER

18-005877 DATED 1963
1000-MWE SGR AND PROTOTYPE EVALUATION STUDY, VOLUME I-LARGE SGR DESIGN BASES, ECONOMICS, AND DEVELOPMENT
PROGRAM
ATOMICS INTERNATIONAL
294 PAGES- DECEMBER 1963, NAA-SR-9213, VOL. I- CFSTI

TECHNOLOGICAL DEVELOPMENTS IN THE ELECTRIC POWER INDUSTRY POINT TO THE INSTALLATION OF LARGER
UNITS (400 MWE AND ABOVE) IN UTILITY SYSTEMS. THE SODIUM COOLED, GRAPHITE MODERATED REACTOR
(SGR) IS MOST ATTRACTIVE AT THESE RATINGS. THIS DOCUMENT DISCUSSES THE SODIUM GRAPHITE REACTOR
CONCEPT AS APPLIED TO LARGE CENTRAL POWER STATIONS. VOLUME I PRESENTS THE DESIGN BASES,
ECONOMICS, CONCEPTUAL DESIGN, AND DEVELOPMENT PROGRAM ASSOCIATED WITH A 1000-MWE SGR POWER PLANT.
AS AN INITIAL STEP TOWARDS THE LARGE SGR PLANT, A PROTOTYPE SGR OF 200-MWE CAPACITY IS
ENVISIONED WITH INTENT TO START IN 1964. THE DESIGN BASES AND DEVELOPMENT PROGRAM FOR THE
PROTOTYPE PLANT ARE ALSO COVERED IN VOLUME I. IN VOLUME II ARE CONTAINED THE PRELIMINARY DESIGN,
DESCRIPTION, AND DRAWINGS OF THE 200-MWE PROTOTYPE PLANT. THE SGR FEATURES THE HIGHEST DEGREE OF
INHERENT SAFETY OF ANY NUCLEAR REACTOR CONCEPT AND PROMISES UNMATCHED OPERATIONAL RELIABILITY AND
ECONOMICS.

DESIGN CRITERIA + ECONOMICS + REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + REACTOR DESCRIPTION + SODIUM

18-006305 ALSO IN CATEGORY 8 DATED 1962
DIETZ HB
HAZARD ANALYSIS OF A SODIUM-WATER REACTION IN AN SRE PIPE GALLERY
ATOMICS INTERNATIONAL
25 PAGES- SEPTEMBER 28, 1962, NAA-SR-MEMO-7743- CFSTI

THE EFFECTS OF A POSSIBLE SODIUM-WATER REACTION IN THE SRE (SODIUM REACTOR EXPERIMENT) PIPE
GALLERIES WERE ANALYZED. POSSIBILITIES FOR THIS REACTION WERE CONSIDERED VERY REMOTE.

REACTOR, LMCR + METAL WATER REACTION + SRE (RE)

18-006306 DATED 1959
LIQUID METAL FUEL REACTOR EXPERIMENT - ANNUAL TECHNICAL REPORT
THE BABCOCK AND WILCOX COMPANY
174 PAGES, 64 FIGURES- MARCH 25, 1959, BAW-1136- CFSTI

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-006306 *CONTINUED*

THE MAXIMUM CREDIBLE INCIDENT FOR THE LMFRE-I HAS BEEN REEVALUATED IN LIGHT OF DESIGN DECISIONS AND IS TENTATIVELY CONSIDERED AS TWO SEPARATE INCIDENTS- MAXIMUM CREDIBLE REACTIVITY ADDITION, AND MAXIMUM CREDIBLE RELEASE OF RADIOACTIVITY.

ACCIDENT ANALYSIS + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA)

18-006307

DATED 1960

THOMAS CE + ALCORN FM + BINGHAM BE + BUNTS JL + BURKART CA + CARLTON JD + DENISE RP + LOWE AL + SHAW H + SCHOMER RT + WEISSENFLUH JA
THE LIQUID METAL THORIUM BREEDER REACTOR
THE BABCOCK AND WILCOX COMPANY
204 PAGES, 53 FIGURES, 23 TABLES, 78 REFERENCES- FEBRUARY, 1960, BAW-1171- CFSTI

A CONCEPTUAL DESIGN STUDY HAS BEEN CONDUCTED TO EVALUATE THE TECHNICAL AND ECONOMIC FEASIBILITY OF A LARGE LIQUID METAL FUEL REACTOR (LMFR) WITH A BREEDING RATIO GREATER THAN 1.00 AND A DOUBLING TIME LESS THAN 25 YR.

REACTOR, LMCR + REACTOR DESCRIPTION

18-007188

ALSO IN CATEGORY 5

DATED 1965

SAFETY ASPECTS OF THE EXPERIMENTAL POWER STATION KNK
3 PAGES, 2 FIGURES- NUCLEAR ENGINEERING 10(112)- 335-337 (SEPTEMBER 1965)

THE PROJECT FOR A 20 MEGAWATT ELECTRICAL EXPERIMENTAL POWER STATION WITH A SODIUM-COOLED, ZIRCONIUM HYDRIDE-MODERATED REACTOR (TYPE KNK), IS THE RESULT OF EXTENSIVE DEVELOPMENT WORK UNDERTAKEN BY INTERATOM SINCE THE SPRING OF 1960. WORK ON THE PLANNING OF THE PROJECT IS PRACTICALLY COMPLETE. START ON CONSTRUCTION AT THE KARLSRUHE RESEARCH CENTRE IS ESTIMATED FOR THE END OF THIS YEAR. THIS REPORT DEALS WITH THE LAYOUT OF THE PLANT. SPECIAL ATTENTION IS GIVEN TO THE SAFETY ASPECTS.

ACCIDENT ANALYSIS + CONTROL SYSTEM + REACTOR, LMCR + REACTIVITY EFFECT + SAFETY REVIEW + REACTOR SAFETY SYSTEM + SODIUM + GERMANY + TECHNICAL SPECIFICATIONS + ENGINEERED SAFETY FEATURE

18-007253

DATED 1965

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR PRELIMINARY SAFEGUARDS SUMMARY REPORT VOL. I AND II
GENERAL ELECTRIC, ATOMIC POWER EQUIPMENT DIVISION
710 PAGES, FIGURES, TABLES, REFERENCES- 1965, NP-15077 DOCKET NO. 50-231- CFSTI

SEFOR WILL BE A 20 MWT FAST-SPECTRUM, SODIUM-COOLED REACTOR DESIGNED TO BE REPRESENTATIVE OF LARGE FAST BREEDER POWER REACTORS FUELED WITH MIXED PLUTONIUM DIOXIDE - URANIUM DIOXIDE. SEFOR WILL BE USED TO OBTAIN PHYSICS AND ENGINEERING DATA AT VARIOUS FUEL COMPOSITIONS, TEMPERATURES, AND CRYSTALLINE STATES, PARTICULARLY FOR THE SYSTEMATIC DETERMINATION OF THE DOPPLER COEFFICIENT OF REACTIVITY UP TO AND BEYOND FUEL CONDITIONS CORRESPONDING TO NORMAL OPERATION OF A POWER REACTOR. THE REACTOR WILL BE FUELED WITH 1-INCH DIAMETER, STAINLESS-STEEL-CLAD PELLETS. THE CORE WILL BE 34 INCHES LONG AND 34 INCHES IN DIAMETER. THE FUEL WILL BE 17 PERCENT PLUTONIUM DIOXIDE AND 83 PERCENT NATURAL URANIUM DIOXIDE. BERYLLIUM OXIDE WILL SOFTEN THE NEUTRON SPECTRUM. MOVABLE RADIAL REFLECTORS LOCATED OUTSIDE OF THE REACTOR VESSEL CONTROL ABOUT 14% IN REACTIVITY. A SPECIAL REACTIVITY INSERTION DEVICE WILL INITIATE EXCURSIONS WITH A ROD OSCILLATOR FOR OTHER TESTS.

REACTOR, BREEDER + CERAMICS + DOPPLER COEFFICIENT + DOPPLER EFFECT + REACTOR, FAST + REACTOR, LMCR + REPORT, SAR + SEFOR (RE)

18-007317

DATED 1965

ANDERSON DR + FOSTER KW
200-MWE PROTOTYPE LARGE SGR - REACTOR STRUCTURE DESIGN AND EVALUATION
ATOMIC INTERNATIONAL
296 PAGES, 144 FIGURES, 14 TABLES- JULY 25, 1965, NAA-SR-11314- CFSTI

BASIC CONCEPT DESCRIBED IN NAA-SR-9213 (DECEMBER 1963), THIS REPRESENTS ADDITIONAL STUDIES ON THERMO-MECHANICAL DESIGN OF THE REACTOR STRUCTURE. DESCRIBES EXPECTED ENVIRONMENT FOR 30 YEAR LIFE, EFFECTS ON STRUCTURE, BASIS OF MATERIAL SELECTION, MECHANICAL-STRUCTURAL ANALYSIS, CALANDRIA LEAK DETECTION, REMOTE MAINTENANCE TECHNIQUES, REACTOR STEADY STATE AND TRANSIENT STUDIES.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + MAINTENANCE AND REPAIR + REACTOR KINETICS + REACTOR STABILITY + THERMAL MECHANICAL EFFECT + POWER DISTRIBUTION + STRESS ANALYSIS + DESIGN STUDY + MATERIAL

18-008423

DATED 1956

ENRICO FERMI FAST BREEDER REACTOR PLANT
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
APDA-115 +. 107 PAGES, 54 FIGURES, 9 TABLES- NOVEMBER 1, 1956- CFSTI, \$9.10 CY, \$3.41 MN

THIS IS AN EARLY DESCRIPTION OF THE REACTOR PORTION OF THE ENRICO FERMI PLANT. NUMEROUS MODIFICATIONS HAVE BEEN MADE SINCE THIS REPORT WAS WRITTEN.

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-008423 *CONTINUED*

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + REACTOR
DESCRIPTION

18-008704

DATED 1965

FACT BOOK ON U. S. NUCLEAR POWER PROJECTS

88 PAGES, REFERENCES- AUGUST 1965, BOZELL AND JACOBS, INC., PUBLISHER

ONE PAGE DESCRIPTION OF U. S. POWER REACTORS, MOSTLY MANAGEMENT LIFE LINE COST, FINANCING, CHIEF
EXECUTIVE OFFICERS.

REACTOR, BWR + REACTOR, GCR + REACTOR, LMCR + REACTOR, ORGANIC COOLED + REACTOR, PMR + REACTOR DESCRIPTION +
REACTOR, POWER

18-009456

ALSO IN CATEGORY 5

DATED 1959

SPIEGLER P

SODIUM-24 SPECIFIC ACTIVITY IN HNPf U-MO AND U-C REACTOR SYSTEMS

ATOMICS INTERNATIONAL

NAA-SR-MEMO-3830 +. 17 PAGES, MAY 21, 1959, CFSTI, \$1.00 CY

THE SPECIFIC ACTIVITY OF THE PRIMARY SODIUM IN THE HNPf WAS DETERMINED FOR SIX POSSIBLE CORE
CONFIGURATIONS (3 U-MO AND 3 U-C). THE ANALYSIS WAS PERFORMED USING A NEUTRON ECONOMY METHOD.
SINCE THIS METHOD OF CALCULATING THE SPECIFIC ACTIVITY IS DIFFERENT FROM THAT PREVIOUSLY USED, IT
WAS FIRST APPLIED TO THE SRE, AND THE RESULTANT SODIUM ACTIVITY WAS COMPARED WITH AVAILABLE
EXPERIMENTAL DATA. AGREEMENT WITH EXPERIMENT WAS GOOD, AND THE METHOD WAS THEREFORE APPLIED TO
THE HNPf CORES. THE RESULTS OF THE CALCULATIONS ARE SUMMARIZED IN TABLE I.

ACTIVATION + ACTIVITY BUILDUP + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + SODIUM + SRE (RE)

18-009746

ALSO IN CATEGORY 11

DATED 1965

SIMMONS WR + PERSIANI PJ + BUMP TR

FAST REACTOR TEST FACILITY (FARET) AND INITIAL EXPERIMENTAL PROGRAM

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ANS-100(SUPPL.) + CONF. 650417(SUPPL.) +. 19 PAGES, 3 FIGURES, PAGES 37-55 OF FAST REACTOR OF TECHNOLOGY, AND
NATIONAL TOPICAL MEETING, DETROIT, MICHIGAN, APRIL 26-28, 1965, SOURCE

THE FAST REACTOR TEST FACILITY SITE, BUILDING, CONTAINMENT STRUCTURE, FUEL-HANDLING SYSTEMS,
SHIELDING, COOLANT SYSTEMS, REACTOR PROPER, CONTROL SYSTEMS, EXPERIMENTAL CAPABILITIES, DESIGN
SPECIFICATIONS, LOADING, AND FUEL-TESTING PROGRAM AND OBJECTIVES ARE DESCRIBED.

REACTOR, BREEDER + CONTAINMENT, HIGH PRESSURE + REACTOR, FAST + REACTOR, LMCR + PRESSURE VESSELS + REACTOR
DESCRIPTION + FTF (TR) + CONTAINMENT STRUCTURE

18-009964

DATED 1963

HARULDSSEN RO + MCGINNIS FD + NOVICK M + SMITH RR + THALGOTT FW

SAFETY ANALYSIS REPORT EBR-I, MARK VI

ARGONNE NATIONAL LABORATORY

ANL-6411 +. 152 PAGES, 48 FIGURES, 24 TABLES, 41 REFERENCES, FEBRUARY 1963, CFSTI, \$2.75 CY

MARK IV, FUELED WITH METALLIC PLUTONIUM, IS INTENDED PRIMARILY FOR PHYSICS INFORMATION SINCE THE
MATERIAL AND THE LOW SPECIFIC POWER ARE INCOMPATIBLE WITH SECOND-GENERATION BREEDERS.
INSTABILITIES AND CURES OF PREVIOUS CURES ANALYZED AND SHOWN NOT TO APPLY. MELTING COULD LEAD TO
FUEL SLUMP AND EXCURSION. FIFTY PERCENT RELEASE ANALYZED.

EBR 1 AND 2 (RE) + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + PLUTONIUM + REPORT, SAR + ACCIDENT, FUEL
SLUMP + EXCURSION, LARGE + INSTABILITY + FAILURE, CLADDING

18-010107

DATED 1966

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, FEBRUARY 1966

ARGONNE NATIONAL LABORATORY

ANL-7176 +. 95 PAGES, 28 FIGURES, 20 TABLES, REFERENCES, MARCH 23, 1966, CFSTI, \$4.00 CY, \$0.75 MN

THIS MONTHLY PROGRESS REPORT (FEBRUARY 1966) CONTAINS SECTIONS ON LIQUID-METAL-COOLED REACTORS,
GENERAL REACTOR TECHNOLOGY, ADVANCED SYSTEMS R AND D, NUCLEAR SAFETY, AND PUBLICATIONS. THE
SAFETY SECTION DEALS WITH REACTOR KINETICS, TREAT, CHEMICAL ENERGY PROBLEMS IN REACTOR SAFETY,
AND PLUTONIUM VOLATILITY SAFETY.

CHEMICAL REACTION + REACTOR, LMCR + PLUTONIUM + REACTOR KINETICS + ANL

18-010118

DATED 1966

NEWCOMB L

CALANDRIA REMOTE MAINTENANCE TOOL DEVELOPMENT

ATOMICS INTERNATIONAL, A DIVISION OF NORTH AMERICAN AVIATION, INC.

NAA-SR-11202 +. 86 PAGES, 38 FIGURES- APRIL 15, 1966- CFSTI, \$3.00

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-010118 *CONTINUED*
EQUIPMENT DESIGNED AND TESTED FOR INSTALLATION AND MAINTENANCE OF PROCESS CHANNELS AND GRAPHITE IN THE CALANDRIA OF A LARGE SGR CORE.

REACTOR, GRAPHITE MODERATED + REACTOR, LMCR + MAINTENANCE AND REPAIR + DESIGN STUDY

18-010318
HAEFELE W + SMIDT D + WIRTZ K
THE KARLSRUHE REFERENCE DESIGN OF A 1000 MWE SODIUM COOLED FAST BREEDER REACTOR
KERNFORSCHUNGSZENTRUM, KARLSRUHE, WEST GERMANY
KFK-369 +. 51 PAGES, OCTOBER 1965, DEP.MN

DATED 1965

THE KARLSRUHE FAST BREEDER PROJECT EXTENDS TO 1967 AND IS AIMED TOWARDS THE INVESTIGATION OF THE FEASIBILITY OF 1000 MWE FAST BREEDER REACTORS. THE PRINCIPAL COOLANT BEING CONSIDERED IS SODIUM. THE PURPOSE OF THE PRESENT KARLSRUHE STUDY CALLED NA-1, OF A 1000 MWE SODIUM COOLED BREEDER REACTOR WAS TO SERVE AS A REFERENCE DESIGN. ON THE BASIS OF THIS REFERENCE DESIGN THE FREQUENTLY DISCUSSED COMPETITION BETWEEN BREEDING, SAFETY AND ECONOMY IS BEING EXPLORED IN A FASHION WHICH IS AS QUANTITATIVE AS POSSIBLE. IT IS IN PARTICULAR THE IMPACT OF SAFETY ON BREEDING AND ECONOMY WHICH IS BEING STUDIED. IT IS THE PURPOSE OF THIS PAPER TO DESCRIBE THE PRESENT NA-1 DESIGN AND ITS SAFETY FEATURES WITH RESPECT TO INCIDENTS, THE MAXIMUM HYPOTHETICAL ACCIDENT (MHA) AND THE DESIGN BASIS ACCIDENT (DBA), IN VIEW OF THE FORTHCOMING NA-2 STUDY.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + DESIGN STUDY + GERMANY

18-010356 ALSO IN CATEGORIES 11 AND 12 DATED 1965
SAUVAGE M + GUMUCHIAN M
1000-MWE REACTOR DESIGN STUDY OF A HANDLING CELL
COMMISSARIAT A L ENERGIE ATOMIQUE, CADARACHE, FRANCE, CENTRE D ETUDES NUCLEAIRES
ORNL-TR-988 + PNR/SEPR/65,008 +. 41 PAGES, FEBRUARY 18, 1965, JCL, \$4.60 FS, \$1.43 MF

THIS REPORT REPRESENTS A SUPPLEMENTARY CHAPTER TO THE 1000 MW REACTOR DESIGN AS PRESENTED IN GENEVA (CEA REPORT 2554). IT DESCRIBES A DESIGN MAKING USE OF A CELL INSTEAD OF ROTATING PLUGS TO CARRY OUT HANDLING. THE PURPOSE OF A STUDY OF THIS VARIANT IS TO PROPOSE A REACTOR DESIGN TO SERVE AS A BASIS FOR A MORE-DETAILED STUDY WITHIN THE SCOPE OF A DESIGN PROJECT. THUS, THE PROBLEMS POSED BY THE REACTOR BLOCK, STORAGE, HANDLING AND THE HANDLING CELL, HAVE BEEN EXAMINED, WITH THE STRESS PLACED ON THE LAST TWO POINTS, WHICH PRESENT THE GREATEST DIFFICULTIES. MUCH REMAINS UNKNOWN ABOUT THE CONTAMINATION OF THIS CELL BY SODIUM AT THE TIME OF HANDLING. METHODS ARE PROPOSED TO LIMIT THIS CONTAMINATION. THE PRINCIPAL ADVANTAGES ARE THE FOLLOWING - (A) HANDLING TAKES PLACE WITHIN SIGHT. THIS VISIBILITY IN CERTAIN CASES IS DIRECT THROUGH PORT HOLES PROVIDED IN THE CELL, OR INDIRECT BY MEANS OF PERISCOPES OR TELEVISION CAMERAS. (B) SINCE THIS CELL IS LEAK-PROOF AND FILLED WITH ARGON, THE PROBLEMS OF TIGHTNESS BETWEEN REACTOR AND CELL ARE CONSIDERABLY SIMPLIFIED. (C) THIS CELL ALSO PERMITS HANDLING OF CONTAMINATED EQUIPMENT WITHOUT REQUIRING THE USE OF AUXILIARY, HIGHLY SHIELDED PROTECTIVE EQUIPMENT. (D) FINALLY, THE CELL PLAYS THE ROLE OF A TIGHT CHAMBER WHICH CAN RETAIN ACTIVE PRODUCTS, ITS STRUCTURE PERMITS IT TO OFFER CONTAINMENT OF A REACTOR EXPLOSION.

CONTAINMENT + DESIGN CRITERIA + REACTOR, FAST + HOT CELL + REACTOR, LMCR + MAINTENANCE AND REPAIR

18-010748 DATED 1966
GREEN L + LEVINE MM
THE RANDOMLY PACKED SETTLED BED FAST REACTOR CONCEPT - 1000-MW(E) REACTOR DESIGN
BROOKHAVEN NATIONAL LABORATORY, UPTON, NEW YORK
BNL-887 +. 56 PAGES, 36 FIGURES, 20 TABLES, 32 REFERENCES, JUNE 1, 1965, CFSTI, \$3.00 CY, \$0.50 MN

CONCEPTUAL DESIGN FOR SODIUM-COOLED U-PU CARBIDE SPHERES. MAJOR LIMITATION IS INTEGRITY OF UNCLAD 0.12-IN. FUEL PARTICLES. BREEDING RATIO IS 1.7 TO 1.8. REPORT CONSISTS OF BED PHYSICS AND ENGINEERING (13 PAGES), THERMAL-HYDRAULIC DESIGN (7 PAGES), PLANT DESIGN (12 PAGES) AND ECONOMICS (14 PAGES).

REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + DESIGN STUDY + FLUIDIZED BED

18-010862 DATED 1966
CASE EG
AMENDMENT 3 TO LICENSE DPR-9, POWER REACTOR DEVELOPMENT COMPANY
DIVISION OF REACTOR LICENSING, USAEC
8 PAGES, JULY 27, 1966, DOCKET 50-16, PDR

AMENDMENT 3 ALLOWS AN INCREASE OF SB-124 (NEUTRON SOURCE) ACTIVITY TO 15 KILOCURIES FROM 3.6, AND LIMITS NUMBER OF SOURCES TO FOUR. SOURCE MAY BE CONTAINED IN TANTALUM OR STAINLESS STEEL.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + SOURCE, NEUTRON + TECHNICAL SPECIFICATIONS

18-011021 DATED 1966
QUARTERLY STATUS REPORT ON ADVANCED REACTOR TECHNOLOGY (ART) FOR PERIOD ENDING APRIL 30, 1966
LOS ALAMOS SCIENTIFIC LABORATORY
LA-3524 +. 69 PAGES, 5 FIGURES, 26 TABLES, MAY, 1966, CFSTI \$3.00 CY, \$0.75 MN

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-011021 *CONTINUED*

THIS REPORT IS ONE OF A SERIES OF SUCH REPORTS ON THE SUBJECTS ENUMERATED BELOW. PROGRAMS REPORTED ON MULTEN PLUTONIUM SUPPORTING RESEARCH, UHTREX, PLASMA THERMOCOUPLE, AND HEAT PIPES.

REACTOR, BREEDER + REACTOR, LMCR + PLUTONIUM

18-011152

DATED 1966

AEC UNCLASSIFIED PROGRAMS. QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY - MARCH 1966
ATOMICS INTERNATIONAL

NAA-SR-11900 +. 170 PAGES, FIGURES, TABLES, JUNE 1966, CFSTI \$5.00 CY, \$1.00 MN

A PRELIMINARY STUDY OF TRITIUM IN A 2117 MW(TH) FAST BREEDER INDICATE THAT 9200 CURIES PER YEAR MAY BE PRODUCED AND MAY LEAK TO COOLANT. THIS WOULD PROBABLY BE MORE OF A NUISANCE THAN A HAZARD. HYDRAULIC TESTS ON THE FUEL ELEMENTS (UO₂ CLAD IN SINTERED ALUMINUM POWDER) FOR THE PIQUA CORE II WERE FOUND IN ERROR BECAUSE THE FLOWMETER READ ABOUT 40 PERCENT LOW. VARIOUS TEMPERATURE AND FLOW CALCULATIONS ARE MADE.

REACTOR, FAST + HYDRAULIC EXPERIMENT + REACTOR, LMCR + REACTOR, ORGANIC COOLED + PIQUA (OCR) + TRITIUM + URANIUM DIOXIDE

18-011247

DATED 1965

LEYPUNSKII AI

THE BN-350 ATOMIC POWER STATION

JPRS-31,193 + TT65-31690 +. PAGES 1-30, TRANSLATED FROM ATOMNAYA STANTSIIYA, APRIL 1965, CFSTI, \$2.00 CY, \$0.50 MN

REVIEW OF DESIGN OF 1000 MW(TH), FAST, SODIUM-COOLED REACTOR FOR POWER AND DESALTING. BR-5 TESTS SHOW THAT SWELLING OF THE PLUTONIUM DIOXIDE CORE CRACKED THE CLAD WHERE THE CLAD-CORE GAP WAS MINIMAL. MIXED (PU,U)O₂ CAN GET A 10 PERCENT BURNUP. TWO RADIAL ZONES ARE USED TO EVEN THE POWER DISTRIBUTION.

REACTOR, BREEDER + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, DESALINATION + USSR + URANIUM DIOXIDE + PLUTONIUM DIOXIDE

18-011295

DATED 1966

FERMI TECHNICAL SPECIFICATIONS POWER COEFFICIENT REQUIREMENT LOWERED

POWER REACTOR DEVELOPMENT COMPANY, DETROIT

7 PAGES, AUGUST 1966, DOCKET NO. 50-16, PDR

EXTREMELY CONSERVATIVE ASSUMPTIONS AND USE OF ACTUAL REACTOR PARAMETERS ALLOW ADEQUATE LEeway, SO THE ACTUAL COEFFICIENT OF 0.18 PLUS OR MINUS 0.013 (RATHER THAN 0.2) LEADS TO THE SAME SAFE RESULTS IN VARIOUS ACCIDENT ANALYSES.

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + POWER COEFFICIENT + TECHNICAL SPECIFICATIONS

18-011333

DATED 1965

ARONSTEIN RE + CAMPISE AV + SPEIGLER P + VITTI J + PEARSON E + CASEY D + HOLZ HB + KEIM M
HNPF ALTERNATE CORE MODIFICATION STUDIES

ATOMICS INTERNATIONAL

NAA-SR-MEMO-11046 +. 23 PAGES, FEBRUARY 18, 1965, CFSTI, DOCKET NO. 115-3

DESCRIBES RESULTS OF A STUDY OF THE HNPF TO ACCOMMODATE TWO TYPES OF CORES - AN ADVANCED THERMAL REACTOR AND A PROTOTYPE OF A FAST REACTOR. THE REQUIRED MODIFICATIONS ARE LISTED.

REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + MODIFICATION

18-011336

DATED 1966

TAUBE M + POTURAJ-GUTNIAK S + KOWALEW A + MIELCARSKI M

MERCURY-BOILING FUSED-SALT FAST REACTOR WARS

INSTITUTE OF NUCLEAR RESEARCH, WARSAW, POLAND

INR-706/L +. 7 PAGES, APRIL 1966, DEP.

BRIEFLY DESCRIBES THE CONCEPT OF A 1000-MW(TH) REACTOR IN WHICH LIQUID MERCURY IS THE COOLANT, FUSED PLUTONIUM CHLORIDE THE FUEL, AND URANIUM-238 TRICHLORIDE AND NaCl OR KCl THE DILUENT.

REACTOR, HOMOGENEOUS + REACTOR, LMCR + REACTOR DESCRIPTION + POLAND

18-011425

DATED 1965

HURST KM

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 6

GENERAL ELECTRIC COMPANY, SAN JOSE, CALIFORNIA

GEAP-4994 +. 129 PAGES, 45 FIGURES, 17 TABLES, OCTOBER 1965, CFSTI, \$4.00 CY, \$1.00 MN

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT. THIS REPORT IS ONE OF A SERIES OF SUCH REPORTS ON THE SUBJECTS ENUMERATED BELOW. THESE INCLUDE - EXPERIMENTAL

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-011429 *CONTINUED*

PROGRAM IN ZPR III, FUEL, SPECIAL INSTRUMENTATION, CORE ASSEMBLY STEADY-STATE AND TRANSIENT (HEAT TRANSFER) CHARACTERISTICS, REACTIVITY CONTROLS FOR EXPERIMENTATION, FUEL HANDLING, CORE DESIGN.

REACTOR, BREEDER + CRITICAL ASSEMBLY + REACTOR, FAST + FUEL HANDLING + HEAT TRANSFER + REACTOR, LMCR + TEST, PHYSICS + SEFOR (RE) + RESEARCH CONTRACT + INSTRUMENTS, MISC.

18-011665
BEUSMAN CC

DATED 1959

PRELIMINARY SAFETY ANALYSIS OF THE SODIUM-DEUTERIUM REACTOR (SDR)
NUCLEAR DEVELOPMENT CORPORATION OF AMERICA

NDA-84-14 +. 127 PAGES, 15 TABLES, 37 FIGURES, 22 REFERENCES, MARCH 6, 1959, CFSTI \$2.50 CY

PRELIMINARY SAFETY EVALUATION MADE OF A PROPOSED SODIUM-DEUTERIUM REACTOR. A SODIUM COOLED, HEAVY WATER-MODERATED REACTOR UTILIZING A FUEL TUBE AND CALANDRIA DESIGN CONCEPT. A STUDY OF MECHANICAL EQUIPMENT MALFUNCTION, FLUID SYSTEM FAILURES, AND OPERATIONAL ACCIDENTS SHOWS THAT THE REACTOR HAS SUFFICIENT INHERENT AND DESIGN SAFETY FEATURES TO RENDER IT SAFE FROM MAJOR ENERGY RELEASE HAZARDS. THE SUCCESSFUL OPERATION OF AN INTEGRATED SODIUM-WATER SYSTEM IN A MOCKUP OF THE REACTOR HAS VERIFIED THE FEASIBILITY OF THE SEPARATION OF SODIUM AND WATER UNDER SIMULATED REACTOR OPERATING CONDITIONS. THIS REPORT COVERS AN EXAMINATION OF THE REACTOR AND ITS SUPPORTING SYSTEMS. IT DOES NOT COVER THE CONSEQUENCES OF REACTOR INCIDENTS TO THE PROPOSED SITE AND ITS SURROUNDINGS.

REACTOR, HWR + REACTOR, LMCR + METAL WATER REACTION + REACTOR, PRESSURE TUBE + REPORT, PSAR

18-011672
HART RS

DATED 1960

RADIOLOGICAL IMPLICATIONS CONCERNING THE USE OF ARGON AS THE CORE COVER GAS IN SODIUM GRAPHITE REACTORS
ATOMIC INTERNATIONAL

NAA-SR-MEMO-5817 +. 10 PAGES, OCTOBER 24, 1960, CFSTI \$1.80 FS, \$1.80 MF

THE SUBSTITUTION OF ARGON FOR HELIUM AS THE INERT CORE-COVER GAS WOULD RESULT IN AN ECONOMIC ADVANTAGE. THE MAJOR OBJECTION IS THE INDUCED ARGON RADIOACTIVITY THAT WOULD OCCUR IN THE COVER GAS. THIS REPORT CALCULATES THE ACTUAL AMOUNT OF EXTRA RADIOACTIVITY WHICH WOULD OCCUR, AND EVALUATES THE RADIOLOGICAL SIGNIFICANCE OF THIS ACTIVITY ON EXISTING DESIGN AND OPERATING CHARACTERISTICS OF SODIUM-GRAPHITE REACTORS. THERE DOES NOT APPEAR TO BE ANY RADIOLOGICAL REASON WHY ARGON COULD NOT BE SUBSTITUTED FOR HELIUM AS THE COVER GAS IN PRESENTLY CONCEIVED SODIUM-GRAPHITE REACTORS. MOREOVER, AN INCREASE OF THE THERMAL NEUTRON FLUX IN THE COVER-GAS REGION BY AT LEAST A FACTOR OF TEN COULD BE TOLERATED WITHOUT MATERIALLY CHANGING THE SITUATION. SOME CONFIRMATION OF THESE CONCLUSIONS MAY BE INFERRED FROM THE SELECTION OF ARGON FOR THE INERT GAS SYSTEM FOR BOTH EBR-II AND FERMI.

EBR 1 AND 2 (RE) + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + ARGON + COOLANT CHEMISTRY + HELIUM + SHIELDING

18-012007

DATED 1965

ASTLEY ER + HALAS DR + FINCH LM + HOFMANN PL + LITTLE WW
SAFETY AND NUCLEAR CONSIDERATIONS IN THE DESIGN OF THE FTR
BATTTELLE-NORTHWEST, RICHLAND

ANL-7120 +. 12 PAGES, 13 FIGURES, 4 TABLES, 4 REFERENCES, PAGES 141-152 FROM PROCEEDINGS OF THE CONFERENCE ON SAFETY FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS, OCTOBER 11-14, 1965, CFSTI \$9.25 CY, \$3.15 MN.

NUCLEAR AND SAFETY FEATURES OF THE CONCEPTUAL DESIGN OF THE FAST TEST REACTOR (FTR) ARE DISCUSSED. IT WILL BE SODIUM COOLED AND PLUTONIUM-OXIDE--STAINLESS-STEEL FUELED. OPERATION AT 300 MW(T) SHOULD PRODUCE A PEAK FLUX OF 10 TO THE 16 AT THE CORE CENTER. IN-CORE CONTROL WITH POISON RODS HAS BEEN SELECTED.

REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, TEST + FFTF (TR) + DESIGN STUDY

18-012008

DATED 1965

LINK LE + AYER JE + HUB KA + MECHAM WJ + TRICE WG + WEILLS JT
A LARGE METAL-FUELED FAST REACTOR STUDY. PART II. CORE DESIGN AND FUEL CYCLE
ARGONNE NATIONAL LABORATORY, ARGONNE

ANL-7120 +. 10 PAGES, 8 FIGURES, 4 TABLES, 12 REFERENCES, PAGES 94-103 FROM PROCEEDINGS OF THE CONFERENCE ON SAFETY FUELS, AND CORE DESIGN IN LARGE TEST POWER REACTORS, OCTOBER 11-14, 1965, CFSTI \$9.25 CY, \$3.75 MN.

THE CORE DESIGN FOR A 1000-MW(E) FAST-BREEDER REACTOR IS DISCUSSED (SEE ALSO ANL-7120, PG 85). THE PERIOD OF THE REFUELING CYCLE FOR THIS REACTOR IS 206 DAYS, WITH A PLANT LOAD FACTOR OF 80 PERCENT AND 6 PERCENT AVERAGE BURNUP. RECENT IRRADIATION DATA INDICATES 7 PERCENT AVERAGE BURNUP IS POSSIBLE. DESCRIBES THE FUEL CYCLE FOR FISSION-PRODUCT REMOVAL AND RESTORATION OF THE IRRADIATED FUEL TO A FORM SUITABLE FOR RETURN TO THE REACTOR.

REACTOR, FAST + FUEL BURNUP + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, POWER + DESIGN STUDY + RADIOCHEMICAL PROCESSING

18-012009

DATED 1965

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-012009 *CONTINUED*

BUTLER DK + LOEWENSTEIN WB + MENEGHEITI D + PHILLIPS KE
A LARGE METAL-FUELED FAST REACTOR STUDY. PART I. NEUTRONICS CALCULATIONS FOR A 1,000-MWE PLUTONIUM-URANIUM,
METAL-FUELED, FAST BREEDER REACTOR
ARGONNE NATIONAL LABORATORY, ARGONNE
ANL-7120 +. 9 PAGES, 3 FIGURES, 11 TABLES, 5 REFERENCES, PAGES 85-93 FROM PROCEEDINGS OF THE CONFERENCE ON
SAFETY, FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS, OCTOBER 11-14, 1965, CFSTI \$9.25 CY, \$3.75 MN.

CALCULATED RESULTS OF THE NUCLEAR AND REACTIVITY PROPERTIES OF THE CONCEPTUAL REACTOR ARE GIVEN.
THE GENERAL DESIGN IS THAT OF A RING OF SIX CYLINDRICAL CORE MODULES SEPARATED BY AND SURROUNDED
BY A FERTILE BLANKET. A BERYLLIUM REGION WAS INSERTED AT THE CORE-BLANKET INTERFACES OF THE
MODULES. TWO-ZONE ENRICHMENT IS USED. THE EFFECTS OF THE BE REGIONS AND FERTILE-BLANKET
THICKNESS ARE DISCUSSED. REACTIVITY COUPLING BETWEEN MODULES IS SMALL. RESULTS OF THERMAL
EXPANSION ON REACTIVITY COEFFICIENTS ARE SUMMARIZED. THE DYNAMIC BEHAVIOR OF THE MODULES IS
DISCUSSED FOR BOTH KAMP INSERTIONS OF REACTIVITY AND LOSS OF COOLANT FLOW. RESULTS INDICATE
FEEDBACK FROM STAINLESS-STEEL EXPANSION IS GREATER THAN THE DOPPLER EFFECT. ALL LOSS-OF-COOLANT-
FLOW STUDIES INDICATED INITIAL BOILING NEAR THE AXIAL CORE BOUNDARY, RESULTING IN NEGATIVE VOID
REACTIVITY (SEE ALSO ANL-7120, PAGE 94).

DOPPLER EFFECT + REACTIVITY EFFECT, EXPANSION + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT +
VOID COEFFICIENT + REACTOR, POWER + COUPLED CORES + DESIGN STUDY

18-012010

DATED 1965

FARMER WS + STRAWSON DG
A RE-EXAMINATION OF DESIGN FACTORS FOR A 1000-MWE FAST BREEDER REACTOR
ALLIS-CHALMERS MFG. CO., BETHESDA
ANL-7120 +. 6 PAGES, 1 FIGURE, 2 TABLES, 9 REFERENCES, PAGES 124-29 FROM PROCEEDINGS OF THE CONFERENCE ON
SAFETY FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS, OCTOBER 11-14, 1965, CFSTI \$9.25 CY, \$3.75 MN.

FOUR DIFFERENT DESIGNS OF A LARGE FAST-BREEDER REACTOR ARE REVIEWED. THREE CORE GEOMETRIES WERE
CONSIDERED - AN ANNULAR, TWO PANCAGED, AND ONE MODULAR. NO ONE GEOMETRY APPEARED TO BE BEST.
EACH DESIGN RESULTED IN A BREEDING RATIO GREATER THAN 1.2. FUEL-CYCLE COSTS RANGED FROM A
MINIMUM OF 0.43 MILL PER KWH TO A MAXIMUM OF 0.85. THE MAJOR SAFETY CONSIDERATION IN THESE
DESIGNS WAS THE REACTIVITY EFFECT OF VOIDS IN THE SODIUM COOLANT. IT IS CONCLUDED THAT THE HIGH
PLUTONIUM INVENTORY AND LARGE OUTSIDE DIAMETER OF THE ANNULAR CORE IS NOT A DISADVANTAGE.

ECONOMICS + REACTOR, FAST + REACTOR, LMCR + PLUTONIUM + REACTOR DESCRIPTION + REACTOR, POWER + DESIGN STUDY

18-012011

DATED 1965

HAFFLE W + SMIDT D + WIRTZ K
THE KARLSRUHE REFERENCE DESIGN OF A 1000 MWE SODIUM-COOLED FAST BREEDER REACTOR
KERNFORSCHUNGSZENTRUM KARLSRUHE, GERMANY
ANL-7120 + 23 PAGES, 8 FIGURES, 5 TABLES, PAGES 162-84 FROM PROCEEDINGS OF THE CONFERENCE ON SAFETY FUELS AND
CORE DESIGN IN LARGE FAST POWER REACTORS, OCTOBER 11-14, 1966, CFSTI \$9.25 CY, \$3.75 MN.

DESCRIBES THE DESIGN AND SAFETY FEATURES OF A REACTOR CONCEPT. THE FUEL IS UO2-PUO2, CLAD WITH
INCOLLOY 800. THE EXPECTED FUEL-CYCLE COST IS 0.7 MILL/KWHR. THE CAPITAL COSTS ARE 100
DOLLARS/KWHR. THE BREEDING RATIO 1.4, AND THE DOUBLING TIME 7 YEARS. AN ANALYSIS OF POSSIBLE
ACCIDENTS IS GIVEN IN WHICH THE FOLLOWING ARE CONSIDERED - (1) LOSS OF COOLANT FLOW, (2)
REACTIVITY KAMP INSERTIONS, (3) MAJOR BREAKS IN PIPES CONTAINING THE COOLANT AND (4) PLUGGING OF
THE CENTRAL SUBASSEMBLY.

ACCIDENT ANALYSIS + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + REACTOR DESCRIPTION + REACTOR,
POWER + DESIGN STUDY + GERMANY

18-012012

DATED 1965

COHEN KP + C NEIL GL
SAFETY AND ECONOMIC CHARACTERISTICS OF A 1000-MWE FAST SODIUM-COOLED REACTOR DESIGN
GENERAL ELECTRIC, SAN JOSE
ANL-7120 +. 20 PAGES, 12 FIGURES, 19 TABLES, 6 REFERENCES, PAGES 185-204 FROM PROCEEDINGS OF THE CONFERENCE
ON SAFETY FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS, OCTOBER 11-14, 1965, CFSTI \$9.25 CY, \$0.75 MN.

A DESIGN FOR A 1000-MWE FAST BREEDER IS PRESENTED. EMPHASIS HAS BEEN PLACED ON SAFETY AND
ASSURANCE OF TECHNICAL FEASIBILITY. ACCIDENT ANALYSES HAVE BEEN PERFORMED IN SUFFICIENT DETAIL
TO DETERMINE THE CHARACTERISTIC BEHAVIOR OF THE PLANT IN COMMON ACCIDENT SITUATIONS AND TO
DEMONSTRATE THAT THE DESIGN SATISFIES THE SAFETY CRITERIA. FUEL COSTS HAVE BEEN CALCULATED.
CALCULATIONS OF CAPITAL AND OPERATING COSTS HAVE NOT BEEN PERFORMED, EXCEPT INsofar AS PREVIOUS
ESTIMATES FOR SIMILAR DESIGNS MAY BE APPLICABLE. SOME CONSIDERATION HAS BEEN GIVEN TO THE
EFFECTS OF UNCERTAINTIES IN THE PHYSICS DATA AND FUEL PERFORMANCE ON ECONOMICS.

ACCIDENT ANALYSIS + ECONOMICS + REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, POWER + DESIGN
STUDY

18-012013

DATED 1965

WRIGHT JH + GUNSON WE + HECK FM + DALEAS RS
CONCEPTUAL DESIGN AND PRELIMINARY ACCIDENT ANALYSIS OF A SODIUM-COOLED, CARBIDE-FUELED, LARGE MODULAR FAST
REACTOR
WESTINGHOUSE ELECTRIC CORP., PITTSBURGH

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-012013 *CONTINUED*

ANL-7120 +. 25 PAGES, 14 FIGURES, TABLES, PAGES 205-29 FROM PROCEEDINGS OF THE CONFERENCE ON SAFETY FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS, OCTOBER 11-14, 1965, CFSTI \$9.25 CY, \$3.75 MN.

PRESENTS THE CONCEPTUAL DESIGN AND DISCUSSES THE ADVANTAGES OF THE MODULAR CONCEPT, POTENTIAL ACCIDENTS ARE REVIEWED (PIPE RUPTURE, LOSS OF PUMP POWER, FLOW BLOCKAGE, MELTDOWN, EXPULSION OF A CONTROL ROD, AND FUEL SLUMPING). GIVES A BRIEF SUMMARY OF THE FUEL-CYCLE-COST ANALYSES.

ACCIDENT ANALYSIS + REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + DESIGN STUDY

18-012138
GEIER JD

DATED 1966

FAST REACTOR TEST FACILITY (FARET), VOL. I-DESCRIPTION AND PROGRAM
ARGONNE NATIONAL LABORATORY, ARGONNE, ILL.

ANL-7168 VOL. I +. 343 PAGES, 123 FIGURES, 21 TABLES, 6 REFERENCES, MARCH 1966, CFSTI \$7.00 CY

IT IS NOT A DESIGN REPORT, BUT IT GIVES DESCRIPTION AND DRAWINGS AS OF THE TIME THAT THE PROJECT WAS TERMINATED BY AEC (DEC. 1965). THE 50-MW(TH) FARET WOULD USE A VARIETY OF FUELS AS STANDARD ELEMENTS (THUS GIVING A LONG-TERM PROOF TEST) WHILE HAVING IRRADIATION LOOPS (TO TEST ADVANCED FUEL). IN-CORE INSTRUMENTS WOULD GIVE BETTER DATA ON IRRADIATION CONDITIONS OF STANDARD FUEL. FARET MARKEDLY RESEMBLES EBR I AND II.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, TEST + FFTF (TR) + REACTOR, ERDA OWNED

18-012145
HARDE R + STOHR KW
THE SODIUM-COOLED EXPERIMENTAL ATOMIC POWER STATION NKK

DATED 1966

7 PAGES, 9 FIGURES, 1 TABLE, ATOMWIRTSCHAFT 11(7), PAGES 354-360 (JULY 1966)

BRIEF DESCRIPTION OF 20-MW(E) SODIUM-COOLED, ZIRCONIUM-HYDRIDE-MODERATED REACTOR DUE TO BEGIN OPERATION IN 1969. IT IS A PROTOTYPE FOR A 300-MW(E) STATION.

REACTOR, LMCR + REACTOR DESCRIPTION + ZIRCONIUM + GERMANY + HYDRIDE

18-012165
DAVIES DK
NUCLEAR STEAM GENERATORS

DATED 1965

THE BABCOCK AND WILCOX COMPANY, BARBERTON, OHIO
11 PAGES, 6 FIGURES, 1 REFERENCE, PAGES 302-312 OF PROCEEDINGS OF THE AMERICAN POWER CONFERENCE, VOL. 27, AMERICAN POWER CONFERENCE, TWENTY-SEVENTH ANNUAL MEETING CHICAGO, ILLINOIS, APRIL 27-29, 1965, PUBLISHED BY AMERICAN POWER CONFERENCE, ILLINOIS INSTITUTE OF TECHNOLOGY, CHICAGO, ILLINOIS, \$10.00

REVIEWS VARIOUS KINDS OF BOILERS. DESCRIBES RECENT TESTS WHICH SHOW THAT, IN HIGH-VELOCITY (GREATER THAN 60 FT/SEC) ONCE-THROUGH BOILERS, NUCLEATE BOILING CAN BE MAINTAINED UP TO 85 PERCENT QUALITY, WITH LOW MASS FLOWS AND HEAT FLUXES LESS THAN 150,000 BTU/HR-FT SQ. DESCRIBES SODIUM-HEATED-BOILER EFFORTS.

REACTOR, LMCR + REACTOR, PWR + REACTOR DESCRIPTION + NUCLEATE BOILING + HEAT EXCHANGERS

18-014646
VAUTREY L

DATED 1963

LIQUID METALS AND FAST REACTORS. THE THERMAL LOOPS OF THE RAPSODIE REACTOR
CENTRE D'ETUDES NUCLEAIRES CADARACHE, FRANCE
EURFNR-95F + EUR-1852.F +. 17 PAGES, 1963, IN FRENCH

THE RAPSODIE COOLING LOOPS ARE BRIEFLY DESCRIBED, AND THE MOST IMPORTANT EXPERIMENTS ON THE TEST LOOPS ARE MENTIONED.

AVAILABILITY - JOHN CREAR LIBRARY, 35 WEST 33RD STREET, CHICAGO, ILLINOIS 60616 \$1.60 COPY, \$0.80 MICRONEGATIVE

REACTOR, FAST + FRANCE + REACTOR, LMCR + SYSTEM DESCRIPTION + MAIN COOLING SYSTEM

18-014651
MUSTNER R

DATED 1963

RAPSODIE, AN EXPERIMENTAL FAST BREEDER REACTOR
CENTRE D'ETUDES NUCLEAIRES CADARACHE, FRANCE
EURFNR-93F + EUR-1699.F +. 30 PAGES, 1963

RAPSODIE, UNDER CONSTRUCTION AT CADARACHE SINCE 1961, WILL BE THE FIRST REACTOR OF THIS TYPE IN CONTINENTAL EUROPE. AS A LOGICAL CONSEQUENCE OF FRENCH PLUTONIUM PRODUCTION AND FRENCH EXPERIENCE IN LIQUID METAL TECHNOLOGY, IT WILL BE A PLUTONIUM FUELED, SODIUM-COOLED REACTOR. MASURCA AND HARMONIE, RESPECTIVELY A CRITICAL ASSEMBLY AND A FAST-SOURCE REACTOR, WILL BE THE MAIN TOOLS FOR NEUTRON STUDIES OF THE LARGE FAST POWER STATIONS OF THE FUTURE. THIS ARTICLE GIVES A BRIEF DESCRIPTION OF THE RAPSODIE REACTOR, SOME DETAILS OF THE HISTORY AND PRESENT STATUS OF THE PROJECT, AND OUTLINES THE FUTURE DEVELOPMENT OF A FAST-REACTOR PROGRAM.

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-014651 *CONTINUED*

AVAILABILITY - JOHN CREKAR LIBRARY, 35 WEST 33RD ST., CHICAGO, ILLINOIS 60616, \$2.60 COPY

REACTOR, FAST + FRANCE + REACTOR, LMCR + PLUTONIUM + REACTOR DESCRIPTION

18-014675

DATED 1966

QUARTERLY STATUS REPORT ON ADVANCED REACTOR TECHNOLOGY (ART) FOR PERIOD ENDING OCTOBER 31, 1966

LOS ALAMOS SCIENTIFIC LABORATORY

LA-3625-MS +. 28 PAGES, NOVEMBER 1966

THIS REPORT IS ONE OF A SERIES OF SUCH REPORTS ON THE FOLLOWING SUBJECTS - UHTREX, GENERAL STUDIES RELEVANT TO SODIUM-COOLED REACTORS, III PLASMA THERMOCOUPLE, IV HEAT PIPES, AND MOLTEN PLUTONIUM WORK (BEING PHASED OUT). CERAMIC PLUTONIUM FUEL STUDIES WILL BE PART OF THE LASL ADVANCED PLUTONIUM FUELS PROGRAM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA 22151, \$3.00 COPY, \$0.65 MICRONEGATIVE

REACTOR, GCR + REACTOR, LMCR + PLUTONIUM + UHTREX (GCR) + MOLTEN FUEL + LASL + DIRECT ENERGY CONVERSION DEVICES

18-014800

ALSO IN CATEGORY 6

DATED 1965

PFLASTERER GR + CALDIROLA L

SEFOR EXPERIMENTAL PROGRAM PLANNING. VOLUME II. DESCRIPTIONS OF PLANNED TESTS

GENERAL ELECTRIC, SAN JOSE, ADVANCED PRODUCTS OPERATION

GEAP-5092 (VOL. 2) +. 116 PAGES, AUGUST 1965

VOL. 1 CONTAINS FUNCTIONAL REQUIREMENTS (FOR MAJOR EQUIPMENT ITEMS), BASED ON INFORMATION IN VOL. 2. VOL. 2 DESCRIBES THE TESTS, ANALYSES, AND REQUIRED MEASUREMENTS. TESTS INCLUDE (1) CRITICAL, (2) STATIC, (3) FREQUENCY RESPONSE, (4) REACTIVITY OSCILLATOR, (5) SUPERCRITICAL TRANSIENTS, AND (6) SUPER-PROMPT-CRITICAL TRANSIENTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + DESIGN CRITERIA + REACTOR, LMCR + OSCILLATOR, REACTIVITY + REACTOR KINETICS + REACTOR TRANSIENT + NOISE ANALYSIS + TEST, PLANT RESPONSE + SEFOR (RE)

18-014804

ALSO IN CATEGORY 17

DATED 1966

ENRICO FERMI REACTOR- USE FOR IRRADIATION TESTING AND BACKGROUND INFORMATION

JOINT COMMITTEE ON ATOMIC ENERGY

270 PAGES, FIGURES, TABLES, APRIL 5, 1966, U.S. GOVERNMENT PRINTING OFFICE, 1966

37 PAGES OF TWO HOURS OF ORAL TESTIMONY. THE REST CONSISTS OF LETTERS, SUMMARIES OF FINANCES AND COSTS, ACMS, ORAL, LEGAL OPINIONS, ETC. CONTENT IS A COMPLETE SUMMARY OF FERMI-REACTOR ADMINISTRATIVE HISTORY TO THE MELTDOWN OF OCT. 10, 1966. THE POINT WAS MADE THAT FAST-NEUTRON FUEL-CLADDING DAMAGE CAN BE OBTAINED QUICKER AND IN THE ACTUAL OPERATING CONDITIONS THROUGH FERMI USAGE.

AVAILABILITY - GOVERNMENT PRINTING OFFICE, WASHINGTON, D. C. 20402

REACTOR, BREEDER + CONGRESSIONAL ACTIVITY + REACTOR, FAST + FERMI (LMFBR) + REACTOR, LMCR + IRRADIATION TESTING + FUEL ELEMENTS

18-016379

DATED 1966

ARMSTRONG RH + CARTER JC

EXPERIMENTAL BOILING-LIQUID-METAL FAST REACTOR.

ARGONNE NATIONAL LABORATORY, ILL.

ANL-7212 +. 83 PAGES, 16 REFERENCES, 29 FIGURES, 10 TABLES, JUNE 1966

PROPOSES THE USE OF THE EBR-1 TO STUDY CHARACTERISTICS OF BOILING LIQUID METALS IN FAST-REACTOR CORES. DESCRIBES THE 1-MW, SEALED, TEST UNIT WHICH IS TO BE PLACED IN THE CORE-CAVITY LINER AFTER REMOVING STANDARD EBR-1 FUEL. THE OBJECTIVES ARE TO DETERMINE THE REACTIVITY EFFECT OF VAPOR FORMATION AND THE FEASIBILITY OF USING THE VAPOR AS THE WORKING FLUID IN A POWER CYCLE. DISCUSSES THE TECHNICAL ASPECTS, SAFETY, THE ANALYTICAL AND TEST PROGRAM, AND COSTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE.

HEAT TRANSFER, BOILING + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + REACTIVITY EFFECT + REACTOR DESCRIPTION + VOID COEFFICIENT + TEST, PHYSICS + METAL, LIQUID + R AND D PROGRAM

18-016530

DATED 1966

HUB KA + CHARAK I + LUTZ DE + THOMPSON DH + GAST PF + MENELEY DA + KEYFITZ IM

FEASIBILITY STUDY OF NUCLEAR STEAM SUPPLY SYSTEM USING 10,000-MW, SODIUM-COOLED BREEDER REACTOR

WESTINGHOUSE ELECTRIC CORP., PITTSBURGH, PA. ATOMIC POWER DIV. + ARGONNE NATIONAL LABORATORY, ILL.

ANL-7183 +. 210 PAGES, 54 FIGURES, 45 TABLES, 57 REFERENCES, SEPTEMBER 1966

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-016530 *CONTINUED*

DESIGN WAS REGARDED AS FEASIBLE BY ANL-WESTINGHOUSE EFFORT FOR 10,000-MW(T) PLANT WITH ANNULAR CARBIDE FUEL CORE, 1050 F REACTOR-OUTLET TEMPERATURE, AND 7-YEAR DOUBLING TIME. NO MCA WAS ESTABLISHED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + DESIGN STUDY

18-016534 ALSO IN CATEGORY 17 DATED 1967

ADAMS RM + GLASSNER A

EBR II POWER COEFFICIENT, AARR LOSS OF FLOW STUDIES

ARGONNE NATIONAL LAB., ARGONNE, ILLINOIS

ANL-7329 +. 103 PAGES, 28 FIGURES, 29 TABLES, 23 REFERENCES, FROM REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, APRIL 1967.

(PAGE 2-4) EBR-II STARTUP WITH STAINLESS-STEEL ASSEMBLIES REPLACING SOME DEPLETED URANIUM SHOWED A DECREASING POWER COEFFICIENT, BUT TESTS SHOWED THAT IT INCREASED ABOVE HALF POWER. (PAGE 75-82) AARR PRIMARY COOLANT SYSTEM BURSTING PRESSURE/STRAIN-ENERGY ARE CALCULATED AS 5000 PSI/524 MW SEC, AND THE VESSEL IS STRONGER. A 163-MW SEC NUCLEAR EXCURSION FOLLOWED BY A 1300-MW SEC METAL-WATER REACTION, WOULD GIVE A 4270-PSI VESSEL PRESSURE THAT EQUALIZES AT 3300 PSI. FLOW-COASTDOWN ANALOG-COMPUTER STUDY SHOWS PCNY MOTOR FLOW IS NECESSARY TO PREVENT HIGH FUEL-ELEMENT TEMPERATURES AT END OF LIFE. HOLKED-END BEAM-HOLE SEAL RINGS WERE ACCEPTABLY LEAK-RATE TESTED AT 55 PSI DELTA P.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + EBR I AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + METAL WATER REACTION + POWER COEFFICIENT + PRESSURE VESSELS + TEST, LEAK RATE + AARR (RR) + REACTOR, ERDA OWNED + BEAM HOLE + SEAL + REACTIVITY EFFECT, ANOMALOUS + REACTOR, FLUX TRAP + PERFORMANCE LIMIT

18-016576 ALSO IN CATEGORY 11 DATED 1965

SUPPLEMENT TO SRE SAFETY ANALYSIS REPORT. A PLAN FOR USE OF A FAST-REACTOR FUEL TEST SECTION IN THE SRE ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-11234 +. 78 PAGES, FIGURES, MAY 3, 1965

SRE MODIFICATIONS INCLUDE REPLACEMENT OF THE CENTRAL SEVEN PEP MODERATOR ELEMENTS BY A FAST-FUEL FLUX-TRAP TEST SECTION, ADDITION OF A 20-MW(T) HEAT SINK TO THE HEAT-TRANSFER SYSTEM TO ALLOW 50-MW(T) OPERATION, IMPROVEMENT OF ENVIRONMENTAL CONTROL AROUND FAST FUEL ELEMENTS DURING FUEL HANDLING, AND CHANGES IN SERVICE FACILITIES. CONCLUSIONS - (1) TECHNICALLY, NO ADDITIONAL BUILDING CONTAINMENT NEEDED BECAUSE OF ADDITION OF FAST TEST SECTION. (2) FOR UNMODIFIED REACTOR, ENERGY RELEASE OF UP TO 50 LB OF TNT WILL NOT LIFT THE LOADING FACE SHIELD OR DAMAGE CORE TANKS. (3) ADDITIONAL CONSTRAINT OF LOADING-FACE SHIELD NEEDED FOR LARGER RELEASES, BUT THOSE UP TO 200 LB OF TNT CAN BE CONTAINED. (4) PRESENCE OF PU DOES NOT MATERIALLY AFFECT SITE-BOUNDARY DOSE RATE WITH NONCREDIBLE DETONATION ACCIDENT OF UP TO 50 LB OF TNT. (5) ADDITION OF PUOZ FUEL DOES NOT SUBSTANTIALLY ALTER RADIOLOGICAL CONSIDERATIONS AT SITE BOUNDARY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + REACTOR, LMCR + REPORT, SAR + SRE (RE) + IN PILE EXPERIMENT + IRRADIATION TESTING + REACTOR, FLUX TRAP

18-016652 DATED 1967

FOSTER J

PRELIMINARY CONCEPT STUDY - CENTRAL TEST FACILITY FOR SPACE NUCLEAR ELECTRIC POWER SYSTEM

OAK RIDGE NATIONAL LABORATORY, OAK RIDGE, TENN.

ORNL-TM-1756 +. 81 PAGES, FIGURES, 3 TABLES, JUNE 1967

DESIGN CONCEPT IS STRONGLY INFLUENCED BY CHEMICAL-ENERGY RELEASE IN LIQUID-METAL-COOLED REACTORS TO 40 MW(T). DESCRIBES ENERGY RELEASES, CONTAINMENT, AND CALCULATION OF OFF-SITE DOSE.

AVAILABILITY - J. FOSTER, OAK RIDGE NATIONAL LABORATORY, OAK RIDGE, TENN.

AIRBORNE RELEASE + CONTAINMENT + DOSE + REACTOR, LMCR + REACTOR, SPACE + DESIGN STUDY + REACTOR TEST FACILITY

18-016925 DATED 1964

HOOKS KR

A FEASIBILITY STUDY OF A HIGH TEMPERATURE LIQUID ALUMINUM COOLED FLUIDIZED BED NUCLEAR POWER REACTOR

AIR FORCE INSTITUTE OF TECHNOLOGY

AO-603 610 + GNE/PHYS/64-10 +. 46 PAGES, FIGURES, AUGUST 1964

PRESENTS A FEASIBILITY STUDY OF A HIGH-TEMPERATURE LIQUID-ALUMINUM-COOLED POWER REACTOR. THE CRITERIA OF A COOLANT OUTLET TEMPERATURE OF 2000 C AND A LOW-PRESSURE CORE LED TO THE CHOICE OF LIQUID ALUMINUM AS THE COOLANT. A COMPATIBILITY STUDY DETERMINED THAT ONLY ALUMINUM MONONITRIDE IS INERT TO ALUMINUM AT 2000 C, AND MUST BE USED FOR CONSTRUCTING THE CORE. FUEL ELEMENTS OF URANIUM MONONITRIDE WERE CHOSEN ON THE BASIS OF PROBABLE COMPATIBILITY WITH THE ALUMINUM NITRIDE

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-016925 *CONTINUED*

CLADDING. CONSIDERATION OF THE HEAT-TRANSFER EFFECTS LED TO THE CHOICE OF MICRON-SIZE COATED FUEL PARTICLES IN A CYLINDRICAL FLUIDIZED-BED CORE. RESULTS INDICATE A FAST CORE WITH A CRITICAL MASS ABOVE 10 KG AND THAT THE REACTOR MAY BE INHERENTLY STABLE BY CORRECT DESIGN OF THE CORE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

ALUMINUM + COATED PARTICLE + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, POWER + DESIGN STUDY + METAL, LIQUID

18-017092

DATED 1965

COHEN KP + ONEILL GL

SAFETY AND ECONOMIC CHARACTERISTICS OF A 1000 MWE FAST SODIUM-COOLED REACTOR DESIGN

GENERAL ELECTRIC CO., SUNNYVALE, CALIF., ADVANCED PRODUCTS OPERATION

GEAP-4939 +. 170 PAGES, REFERENCES, DECEMBER 1965

DESIGN FOR A MIXED-OXIDE-FUELED REACTOR BASED ON UPDATED SET OF SAFETY AND DESIGN CRITERIA. CRITERIA DEVELOPED HAVE IMPLICATIONS ON REACTIVITY COEFFICIENTS, CORE AND FUEL DESIGN, AND PRIMARY SYSTEM DESIGN. 1965 CALCULATIONS REPORTED HERE DO NOT TAKE INTO ACCOUNT CHANGES SUCH AS DISCOVERY THAT PU-239 DOPPLER CONTRIBUTION IS ALMOST ZERO, THUS CHANGING DOPPLER COEFFICIENT VALUES SOMEWHAT. PRIMARY EMPHASIS IS ON SAFETY AND ASSURANCE OF TECHNICAL FEASIBILITY, AND OPTIMIZATION IS LIMITED. ACCIDENT ANALYSES PERFORMED IN SUFFICIENT DETAIL TO DETERMINE CHARACTERISTIC BEHAVIOR OF PLANT IN COMMON ACCIDENT SITUATIONS AND TO DEMONSTRATE THAT SAFETY CRITERIA SATISFIED. EFFECTS OF UNCERTAINTIES IN NUCLEAR DATA AND FUEL PERFORMANCE CONSIDERED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + DESIGN STUDY + SAFETY ANALYSIS

18-017095

DATED 1966

FEASIBILITY STUDY OF A 1000-MWE SODIUM-COOLED FAST REACTOR

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-11378 (VOL 2) +. 68 PAGES, FIGURES, TABLES, JULY 1966

DESCRIBES PROGRAM IN FAST-REACTOR TECHNOLOGY TO BE UNDERTAKEN BY AI, WHICH WILL BE COMBINED WITH WORK OF OTHER CONTRACTORS AND NATIONAL LABORATORIES TO YIELD COMPLETE DEVELOPMENT PROGRAM IN THIS AREA. DEALS WITH SOLID-FUELED, SODIUM-COOLED REACTOR CONCEPTS AND THE AI PLANT TO INITIATE FAST-REACTOR PROTOTYPE PROJECT BY 1970. CONCEPTUAL DESIGN STUDIES TO BEGIN IN FY 66 (TO EXPLORE NUCLEAR AND SAFETY UNCERTAINTIES, FUEL AND MATERIALS TECHNOLOGY AND MECHANICAL AND PLANT DESIGN ASPECTS OF FAST REACTORS). KEY FACTORS CONSIDERED IN STUDY INCLUDE (1) COUPLED FAST-THERMAL, MIXED CYCLE (PU-TH), AND U-235-PU ZONED CORES, (2) CONTROL OF ALL POSTULATED ACCIDENTS BY DOPPLER IS POSSIBLE OR CONTROL-ROD ACTION TO PREVENT FUEL VAPORIZATION, AND CONSIDERATION OF ACCIDENTS SUCH AS CONTROL ROD WITHDRAWAL, FUEL SUMPING, LOSS OF PUMP, CHANNEL PLUGGING, MAXIMUM SODIUM VOIDING ACCIDENTS, AND OTHERS (3) OTHER MIXED OXIDES (FAST SECTION) WITH MIXED CARBIDES, (4) BE, BEO, AND ZRHZ AS MODERATOR-REFLECTORS, (5) OPTIMIZATION OF TEMPERATURES IN SODIUM FOR LOW FUEL-CYCLE COSTS AND OPTIMUM STEAM CONDITIONS, (6) ELIMINATION OF STEEL LININGS AND N2 COOLING SYSTEMS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + R AND D PROGRAM

18-017113

DATED 1966

YEVICK JG

FAST REACTOR TECHNOLOGY. PLANT DESIGN

MASSACHUSETTS INSTITUTE OF TECHNOLOGY, CAMBRIDGE, MASSACHUSETTS AND LONDON, ENGLAND

754 PAGES, FIGURES, TABLES, REFERENCES, 1966

ATTEMPTS TO CONSOLIDATE 15 YEARS OF AVAILABLE INFORMATION FOR USE IN PRESENT FAST-REACTOR DESIGN. CHAPTERS, EACH WRITTEN BY AN EXPERT, INCLUDE (1) COOLANT PROPERTIES, HEAT TRANSFER, AND FLUID FLOW OF LIQUID METALS, (2) STRUCTURAL ANALYSIS, (3) HEAT-TRANSPORT SYSTEMS, (4) FUEL HANDLING, (5) SHIELDING, AND OTHERS. MATERIALS AND PHYSICS ASPECTS OF FUEL AND CORE DESIGN ARE NOT DISCUSSED FOR LACK OF SPACE, BUT DESIGN ASPECTS OF EBR-I AND II, USSR BR-5, LAMPRE-I, DOUNREAY, FERMI, AND RAPSODIE ARE.

AVAILABILITY - THE M.I.T. PRESS, CAMBRIDGE, MASS., \$35.00 COPY

REACTOR, FAST + REACTOR, LMCR + DESIGN STUDY + SYSTEM DESCRIPTION

18-017341

DATED 1963

SIMON NK

PRELIMINARY RADIOLOGICAL HAZARD ANALYSIS FOR THE FARET REACTOR

ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA

NAA-SR-MEMO-8364 +. 36 PAGES, APRIL 2, 1963

THE RADIOLOGICAL HAZARD TO THE ENVIRONMENT IS FROM TWO REACTOR ACCIDENTS - (1) CORE MELTDOWN FORMS

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-017341 *CONTINUED*

A SUPERCRITICAL MASS WHICH RELEASES ENERGY EQUIVALENT TO THE DETONATION OF 300 LB TNT, AND (2) A FUEL ELEMENT FALLS OUT OF THE FUEL-HANDLING MACHINE AND BURNS. A COMPUTER CODE DEVELOPED RECENTLY AT ATOMICS INTERNATIONAL CALCULATES RELEASE QUANTITY, CLOUD CONCENTRATION, DIRECT BUILDING DOSE, EXTERNAL CLOUD DOSE, AND INHALED CLOUD DOSE VERSUS DISTANCE FOR EACH. THE EXCLUSION RADIUS WAS DETERMINED TO BE 0.06 MILE, AND THE LOW-POPULATION-ZONE RADIUS 1.6 MILES FOR DESIGN CONDITIONS UNDER WHICH THE LARGEST RADIOACTIVE RELEASE OCCURRED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REFUELING + ACCIDENT, FUEL SLUMP + FFTF (TR) + HAZARDS ANALYSIS

18-017355
MUENCHOW HU

DATED 1964

STATUS SUMMARY OF APDA SMALL LEAK SODIUM - WATER REACTION TEST PROGRAM
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT
TID-20936 +. 11 PAGES, JUNE 30, 1964

ONE OF A SERIES OF PROGRESS REPORTS ON A PROGRAM WHOSE SCOPE IS (1) DESIGN AND OPERATION OF A SMALL-LEAK SODIUM-WATER REACTION TEST LOOP, (2) CONDUCTION OF A SERIES OF 12 TESTS TO DETERMINE EFFECT OF SMALL LEAKS ON TUBE WASTAGE, (3) ANALYSIS OF ALL TEST DATA. SPECIFIC OBJECTIVES ARE (1) FIND MEANS OF ESTIMATING THE EXTENT OF TUBE WASTAGE, (2) DETERMINE DOMINANT PARAMETERS FOR TUBE WASTAGE FOR DESIGN STUDIES. DESCRIPTION OF TEST-STAND SYSTEMS IS GIVEN, AND ENGINEERING AND DESIGN STATUS OF THE SYSTEMS IS DISCUSSED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + METAL WATER REACTION + FAILURE, TUBING + R AND D PROGRAM + STEAM GENERATOR

18-017396

DATED 1965

SMIUT D + FRISCH W + GIORIANO P + HEUSENER G + KESSLER G + KREWER KH + MERK W + MALMBERG T + SCHONFELD E
SAFETY AND COST ANALYSIS OF A 1000-MWE SODIUM COOLED FAST POWER REACTOR
KERNFORSCHUNGSZENTRUM, KARLSRUHE (WEST GERMANY). INSTITUT FUER NEUTRONENPHYSIK UND REAKTORTECHNIK
EURFNR-77F + KFK-398 +. 40 PAGES, OCTOBER, 1965

OUTLINES STUDY OF DYNAMIC BEHAVIOR USING ANALOG PROGRAMS OF THE CORE AND TOTAL PRIMARY CIRCUIT, AND A REVISED VERSION OF DIGITAL CODE FORE. MAXIMUM HYPOTHETICAL ACCIDENT IS FAST EXCURSION WITH EFFECTS OF A POSITIVE VOID COEFFICIENT, WITH (1) DISASSEMBLY OF THE CORE BY FASTEST POSSIBLE REACTIVITY INPUT RATE DUE TO VOIDING OF CORE CENTER OR MELTDOWN, AND (2) GATHERING OF MOLTEN CORE MATERIAL IN CONTAINMENT AFTER CORE DISASSEMBLY. IT IS CONCLUDED THAT (1) INFINITE RAMPS OF 10 DOLLARS/SEC CAN BE BALANCED BY A SPRING-DRIVEN SAFETY SYSTEM, (2) COOLANT BOILING IN A SINGLE SUBASSEMBLY MAY BE DANGEROUS WITH SODIUM SUPERHEAT, (3) POSITIVE VOID EFFECT IS IMPORTANT ONLY FOR MHA, (4) ADVANTAGES IN COST AND SAFETY FAVOR LOWER COOLANT FRACTIONS AND A NOT TOO COMPACT CARBIDE CORE. ADDITION OF MO TO FUEL OXIDE IS NOT SATISFACTORY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + EURATOM + REACTOR, FAST + REACTOR, LMCR

18-017500

ALSO IN CATEGORY 17

DATED 1967

ANOMALOUS SWELLING OF EBR-II MARK 1A FUEL
ARGONNE NATIONAL LAB., ARGONNE, ILL.

ANL-7317 +. 5 PAGES, 1 TABLE, PAGES 8-12 FROM REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MARCH 1967

WHILE EARLIER 48% ENRICHED FUEL SWELLED 6-7% AT 1.2 A/O FUEL BURNUP, THE 1A 52% ENRICHED FUEL IS UNEXPECTEDLY SWELLING TWICE THAT AMOUNT, UNLESS THE FUEL CONTAINS REMELTED REJECT PINS AND PIN ENDS. VARIOUS MEASUREMENTS TAKEN, AND IRRADIATION STUDIES BEGUN IN NOV. 1966 ARE SUMMARIZED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + EBR 1 AND 2 (RE) + FUEL BURNUP + REACTOR, LMCR + REACTOR, ERDA OWNED + EXPANSION + FUEL ELEMENTS

18-017699

DATED 1965

GENIE ATOMIQUE. TOME TROIS (NUCLEAR ENGINEERING. VOLUME III)
INSTITUT NATIONAL DES SCIENCES ET TECHNIQUES NUCLEAIRES, SACLAY

992 PAGES, FIGURES, TABLES, 1965, INSTITUT NATIONAL DES SCIENCES ET TECHNIQUES NUCLEAIRES, SACLAY AND PRESSES UNIVERSITAIRES DE FRANCE, PARIS

A SERIES OF MONOGRAPHS ON NUCLEAR REACTORS, WITH EMPHASIS ON THOSE IN FRANCE. THE PRINCIPLES, THE EXPERIMENTS, AND THE APPLICATIONS ARE GIVEN FOR EACH REACTOR. LOW-FLUX REACTORS AND CRITICAL ASSEMBLIES ARE DESCRIBED. GENERAL CONSIDERATIONS ARE PRESENTED ON THE RESEARCH REACTORS WITH HIGH FLUX. THE REACTORS DESCRIBED IN DETAIL ARE THE EL-3, MELUSINE, TRITON, PEGASE, SILOE, AND ULYSSE. GRAPHITE REACTORS WITH GASEOUS COOLANTS--BOTH WITH NATURAL AND ENRICHED URANIUM--ARE

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-017699 *CONTINUED*

DISCUSSED. THE CONSTRUCTION TECHNIQUES FOR SOLID MODERATORS ARE DESCRIBED. THE THERMAL PROBLEMS CONNECTED WITH THE COOLING OF GASEOUS REACTORS ARE EXAMINED. HEAVY-WATER, LIGHT-WATER, AND ORGANIC REACTORS ARE DESCRIBED. PROBLEMS OF FAST REACTORS ARE CONSIDERED. THE USE OF LIQUID METALS FOR COOLING IS DISCUSSED, ALONG WITH THE PRESENT DESIGN TENDENCIES FOR POWER REACTORS. DESCRIPTIONS OF REACTORS FOR NAVAL PROPULSION ARE INCLUDED.

AVAILABILITY - PRESSES UNIVERSITAIRES DE FRANCE, 108 BOULEVARD ST-GERMAINE, PARIS, FRANCE

FRANCE + REACTOR, GCR + REACTOR, HWR + REACTOR, LMCR + REACTOR, ORGANIC COOLED + REACTOR DESCRIPTION + REACTOR, RESEARCH + REACTOR, POWER

18-018745

DATED 1966

GREIZY CE

SODIUM-COOLED REACTORS. FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 16, AUGUST - OCTOBER 1965

GEAP-4982 +. 69 PAGES, JANUARY 1966

ONE OF A SERIES OF PROGRESS REPORTS WHOSE CONTENTS PERTAIN TO (TASK B) VENTED FUEL DEVELOPMENT, (TASK C) TRANSIENT TESTING OF FUEL, (TASK E) FUEL-PERFORMANCE EVALUATION, (TASK F) FAST-FLUX IRRADIATION OF FUEL, (TASK G) REACTOR PHYSICS AND CORE ANALYSIS, (TASK K) REACTOR DESIGN AND SAFETY CHARACTERISTICS, (TASK L) FOR CLADDING EVALUATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, NATIONAL BUREAU OF STANDARDS, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + REACTOR, LMCR + IRRADIATION TESTING + R AND D PROGRAM + FUEL ELEMENTS

18-018942

DATED 1958

HOLZ H6

DESIGN BASIS FOR THE HNPf PRIMARY NA PIPING ARRANGEMENT BETWEEN THE REACTOR AND THE EQUIPMENT CELLS, AND RECOMMENDATIONS FOR THE SELECTION OF A CONCEPTUAL ARRANGEMENT FOR THE HNPf

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-MEMO-2663 +. 28 PAGES, REFERENCES, APRIL 21, 1958

GIVES THE GROUND RULES THAT AFFECT THE PIPING-ARRANGEMENT CONCEPTS BETWEEN THE REACTOR AND EQUIPMENT CELLS, AND PRESENTS RECOMMENDATIONS FOR SELECTING A CONCEPTUAL ARRANGEMENT.

DESIGN CRITERIA + REACTOR, GRAPHITE MODERATED + HALLAM (LMR) + REACTOR, LMCR + SODIUM + DESIGN STUDY + MAIN COOLING SYSTEM + PIPES AND PIPE FITTINGS

18-019142

DATED 1967

COMPACT SODIUM COOLED REACTOR PLANT KARLSRUHE (KNK-KARLSRUHE) SAFETY REPORT, VOL. I, II, AND III

INTERATOM GMBH, GERMANY

INTAT-70 +. 450 PAGES, FIGURES, TABLES, APRIL 10, 1967. IN GERMAN

VOL. I - DESCRIPTIONS OF THE NUCLEAR PLANT (INTRODUCTION, SITE, DESCRIPTION, ENERGY SOURCES, INSTRUMENTATION AND CONTROL, DESIGN DATA, RADIATION FIELD, NORMAL OPERATION). VOL. II - ABNORMAL CONDITIONS AND COUNTER MEASURES (REACTIVITY DISTURBANCES, DISTURBANCES IN THE COOLING SYSTEM, ABNORMAL HANDLING, RADIOLOGICAL CONSEQUENCES, MAXIMUM CREDIBLE ACCIDENT, PLANNING FOR CATASTROPHE) PROTECTION AND SAFETY MEASURES (RADIATION PROTECTION, RADIATION PROTECTION ZONES, PERSONNEL RADIATION EXPOSURE IN NORMAL OPERATION, FUEL HANDLING, PROTECTION OF THE SURROUNDINGS. APPENDIX COLLECTION OF OPERATION AND DESIGN DATA. VOL. III - DRAWINGS.

AVAILABILITY - INTERATOM, INTERNATIONALE ATOMREAKTORBAU GMBH, 506 BENSBERG/KOLN, POSTFACH, GERMANY

ENERGY SOURCE + FUEL HANDLING + HYDRAULIC ANALYSIS + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR TRANSIENT + REPORT, SAR + CONTROL + GERMANY + RADIATION SAFETY AND CONTROL + KARLSRUHE + INSTRUMENTS, MISC.

18-019560

DATED 1967

LEONARD BH + PLANCHAN JD

CONSTRUCTION AND DESIGN PROBLEMS OF SODIUM COOLED FAST REACTORS

BECHTEL CORPORATION, SCIENTIFIC DEVELOPMENT DEPARTMENT, SAN FRANCISCO, CALIFORNIA

8 PAGES, 5 FIGURES, APRIL 1967, PAPER PRESENTED AT THE NATIONAL TOPICAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, APRIL 10-12, 1967, FAST REACTORS-OPERATING, UNDER CONSTRUCTION, PLANNED FOR CONSTRUCTION

DISCUSSES THE THERMAL, CHEMICAL, AND RADIOACTIVE PROPERTIES OF SODIUM AND REACTOR DESIGN REQUIREMENTS DICTATED BY THESE PROPERTIES. THERMAL CONSIDERATIONS MUST INCLUDE MATERIAL SELECTION, STRESSES, PIPE FLEXIBILITY, HEATING CAPABILITY, AND SYSTEM PLUGGING DUE TO VAPOR CONDENSATION. CHEMICAL CONSIDERATION INCLUDE ELABORATELY SEALED VALVES AND AN INERT, PURE, BLANKET GAS. RADIOACTIVITY REQUIRES CONSIDERATION OF SHIELDING.

DESIGN CRITERIA + FRANCE + REACTOR, LMCR + SODIUM + THERMAL PROPERTY + PROPERTY, PHYSICAL + MAIN COOLING SYSTEM

18-019562

DATED 1967

CARLE R + FAURE J + LEDUC J + RUBIN M + RASTOIN J + ROZENHOLC M + SEBILLEAU F + VAUTREY L + VENDRYES G +

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-019562 *CONTINUED*
VILLENEUVE J + ZALESKI CP

THE PHENIX REACTOR

COMMISSARIAT A L-ENERGIE ATOMIQUE, FRANCE

25 PAGES, 8 FIGURES, 3 REFERENCES, APRIL 1967, PAPER PRESENTED AT THE NATIONAL TOPICAL MEETING OF THE AMERICAN NUCLEAR SOCIETY, APRIL 10-12, 1967, FAST REACTORS-OPERATING, UNDER CONSTRUCTION, PLANNED FOR CONSTRUCTION

DESCRIBES THE 600-MWT/250-MWE REACTOR. THE MAIN CHARACTERISTICS ARE UO₂-PUO₂ FUEL WITH SS CLAD, SODIUM COOLING (SIMILAR TO RAPSODIE EXCEPT THAT PHENIX IS A POWER-REACTOR PROTOTYPE). CONSTRUCTION IS TO BEGIN IN 1969.

REACTOR, BREEDER + REACTOR, FAST + FRANCE + REACTOR, LMCR + REACTOR DESCRIPTION + DESIGN STUDY

18-019891

DATED 1966

HOLZ HB + SCHNEIDER G

1000-MWE FBR PARAMETRIC STUDY OF SECONDARY SODIUM-STEAM SYSTEM

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-12060 +. 151 PAGES, 14 FIGURES, 4 TABLES, 9 REFERENCES, AUG. 31, 1966

INITIAL SCOPING PHASE TO PROVIDE COST TRENDS FOR 1000-MWE PLANT. ELEVEN COMBINATIONS OF SODIUM/STEAM-GENERATORS/TURBINE WERE USED, USING 1050 F SECONDARY SODIUM HOT-LEG TEMP., AND A 2400 PSI/1000 F-1000 F STEAM SYSTEM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

ECONOMICS + REACTOR, LMCR + MAIN COOLING SYSTEM + R AND D PROGRAM + STEAM GENERATOR + TURBINE

18-020046

DATED 1966

CASEY DF

DESIGN, PERFORMANCE, AND ECONOMICS EVALUATION OF SODIUM-HEATED, MODULAR, SUPERCRITICAL, STEAM GENERATORS

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-8508 +. 71 PAGES, 16 FIGURES, TABLES, OCTOBER 31, 1966

PRESENTS DESIGN DETAILS FOR 37-TUBE MODULES FOR A 450-MWE PLANT AND 61-TUBE MODULES FOR A 350-MWE PLANT WITH SINGLE REHEAT. DISCUSSES STEADY-STATE AND TRANSIENT DESIGN CRITERIA, RESULTS OF COMPUTER CALCULATIONS OF THERMAL PERFORMANCE, COMPLEMENTARY EQUIPMENT FOR THE MODULES, COST ESTIMATES, AND MAINTENANCE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + ECONOMICS + REACTOR, LMCR + MAINTENANCE AND REPAIR + SODIUM + THERMAL ANALYSIS + DESIGN STUDY + STEAM GENERATOR

18-020482

DATED 1966

PFLASTERER GR

SEFOR EXPERIMENTAL PROGRAM PLANNING. VOLUME 1. SPECIFICATIONS FOR THE PLANT AND EXPERIMENTAL EQUIPMENT.

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

GEAP-5092(VOL. 1) +. 99 PAGES, REFERENCES, AUGUST 1966

PROGRAM IS TO OBTAIN PHYSICS AND ENGINEERING DATA ON SOFT-SPECTRUM FAST-BREEDER REACTORS. VOL. 1 CONTAINS EQUIPMENT FUNCTIONAL REQUIREMENTS ON THE ROD-OSCILLATOR MECHANISM, FAST-REACTIVITY-EXCURSION DEVICE, TRANSIENT-POWER DETECTORS, DATA-ACQUISITION SYSTEM, INSTRUMENTED FUEL ASSEMBLIES, REACTOR-FUEL-TEMPERATURE CHANGES, REACTOR-POWER-DISTRIBUTION MEASUREMENTS, PLANT CONTROL, PLANT INSTRUMENTATION, AND PLANT-TEMPERATURE CHANGES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + DESIGN CRITERIA + REACTOR, LMCR + OSCILLATOR, REACTIVITY + REACTOR TRANSIENT + DATA PROCESSING + INSTRUMENT, CONTROL + INSTRUMENT, TEMPERATURE + SEFOR (REI) + EQUIPMENT + R AND D PROGRAM + INSTRUMENTS, MISC.

18-021080

DATED 1959

MASON DG

SODIUM SYSTEMS HAZARDS EVALUATION

ATOMICS INTERNATIONAL, CANOGA PARK, CALIF.

NAA-SR-MEMO-3937 +. 12 PAGES, REFERENCES, JUNE 1, 1959

RECOMMENDS AREAS TO BE INVESTIGATED TO IMPROVE SODIUM-SYSTEMS HAZARDS EVALUATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + SODIUM + HAZARDS ANALYSIS + REVIEW

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-021086 DATED 1964
PROBERT P
SODIUM-HEATED STEAM GENERATOR DEVELOPMENT. PROGRESS REPORT NO. 9
BABCOCK AND WILCOX CO., BARBERTON, OHIO
TID-23133 +. 60 PAGES, JANUARY 1964

THE PRELIMINARY DESIGN OF THE FUEL-SIZE ONCE-THROUGH STEAM GENERATOR IS 52% COMPLETE. DEVELOPMENT PROGRESS IS REPORTED ON HEAT TRANSFER AND CORROSION STUDIES, PROCEDURES FOR WELDING, RADIOGRAPHIC INSPECTION, AND CHEMICAL SIMULATION OF THE SODIUM ENVIRONMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., \$3.00 COPY, \$0.65 MICROFICHE

HEAT TRANSFER + REACTOR, LMCR + SODIUM + WELDING + CORROSION + RADIOGRAPHY + R AND D PROGRAM + STEAM GENERATOR

18-021111 DATED 1964
HORST KM
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 2, JULY-SEPTEMBER 1964
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
GEAP-4742 +. 73 PAGES, REFERENCES, OCTOBER 1964

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS INCLUDE STUDIES OF REACTIVITY EFFECTS OF FUEL, CLAD, AND COOLANT, MOCK UP TESTING, FUEL DEVELOPMENT, INSTRUMENTATION FOR FUEL-TEMPERATURE MEASUREMENT, STEADY-STATE AND TRANSIENT CHARACTERISTICS OF THE CORE ASSEMBLY, REACTIVITY CONTRL, FUEL HANDLING, AND CORE DESIGN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00 COPY, \$0.65 MICROFICHE

FUEL HANDLING + INSTRUMENT, IN CORE + REACTOR, LMCR + MOCKUP + REACTIVITY EFFECT + REACTOR STABILITY + REACTOR CONTROL + DESIGN STUDY + SEFOR (NE) + R AND D PROGRAM + FUEL ELEMENTS

18-021112 DATED 1965
HORST KM
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 4, FEBRUARY-APRIL 1965
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
GEAP-4864 +. 142 PAGES, MAY 1965

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS INCLUDE STUDIES OF REACTIVITY EFFECTS OF FUEL, CLAD, AND COOLANT, MOCKUP TESTING, FUEL DEVELOPMENT, INSTRUMENTATION FOR FUEL-TEMPERATURE MEASUREMENT, STEADY-STATE AND TRANSIENT CHARACTERISTICS OF THE CORE ASSEMBLY, REACTIVITY CONTROL, FUEL HANDLING, AND CORE DESIGN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00 COPY, \$0.65 MICROFICHE

FUEL HANDLING + INSTRUMENT, IN CORE + REACTOR, LMCR + MOCKUP + REACTIVITY EFFECT + REACTOR STABILITY + REACTOR CONTROL + DESIGN STUDY + SEFOR (RE) + R AND D PROGRAM + FUEL ELEMENTS

18-021113 DATED 1965
HORST KM
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. FIFTH QUARTERLY REPORT, MAY-JULY 1965
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
GEAP-4934 +. 122 PAGES, AUGUST 1965

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS INCLUDE STUDIES OF REACTIVITY EFFECTS OF FUEL, CLAD, AND COOLANT, MOCKUP TESTING, FUEL DEVELOPMENT, INSTRUMENTATION FOR FUEL-TEMPERATURE MEASUREMENT, STEADY-STATE AND TRANSIENT CHARACTERISTICS OF THE CORE ASSEMBLY, REACTIVITY CONTROL, FUEL HANDLING, AND CORE DESIGN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00 COPY, \$0.65 MICROFICHE

FUEL HANDLING + INSTRUMENT, IN CORE + REACTOR, LMCR + MOCKUP + REACTIVITY EFFECT + REACTOR STABILITY + REACTOR CONTROL + DESIGN STUDY + SEFOR (RE) + R AND D PROGRAM + FUEL ELEMENTS

18-021114 DATED 1966
HORST KM
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 8, FEBRUARY-APRIL 1966
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
GEAP-5160 +. 124 PAGES, MAY 1966

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS INCLUDE STUDIES OF REACTIVITY EFFECTS OF FUEL, CLAD, AND COOLANT, MOCKUP TESTING, FUEL DEVELOPMENT, INSTRUMENTATION FOR FUEL-TEMPERATURE MEASUREMENT, STEADY-STATE AND TRANSIENT CHARACTERISTICS OF THE CORE ASSEMBLY, REACTIVITY CONTROL, FUEL HANDLING, AND CORE DESIGN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-021114 *CONTINUED*
COPY, \$0.65 MICROFICHE

FUEL HANDLING + INSTRUMENT, IN CORE + REACTOR, LMCR + MOCKUP + REACTIVITY EFFECT + REACTOR STABILITY + REACTOR CONTROL + DESIGN STUDY + SEFOR (RE) + R AND D PROGRAM + FUEL ELEMENTS

18-021115 DATED 1967
HORST KM
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 11, NOVEMBER 1966-JANUARY 1967
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.
GEAP-5442 +. 86 PAGES, REFERENCES, FEBRUARY 1967

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS INCLUDE IRRADIATION TESTING OF FUEL ELEMENTS AND THERMOCOUPLES, STEADY-STATE AND TRANSIENT CHARACTERISTICS OF THE CORE, CONTROL-ROD DESIGN, FUEL HANDLING, AND CORE DESIGN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL HANDLING + INSTRUMENT, IN CORE + REACTOR, LMCR + DESIGN STUDY + IRRADIATION TESTING + SEFOR (RE) + R AND D PROGRAM + FUEL ELEMENTS

18-021116 DATED 1966
HORST KM
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 10, AUGUST-OCTOBER 1966
GENERAL ELECTRIC CO., SUNNYVALE, CALIF., ADVANCED PRODUCTS OPERATION
GEAP-5301 +. 108 PAGES, REFERENCES, NOVEMBER 1966

ONE OF A SERIES OF PROGRESS REPORTS. TOPICS INCLUDE IRRADIATION TESTING OF FUEL PELLETS, TESTING OF IN-PILE THERMOCOUPLES, HYDRAULIC FLOW TESTS, FUEL HANDLING, CORE DESIGN, AND FUEL FABRICATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00
COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, FAST + FUEL HANDLING + HYDRAULIC ANALYSIS + INSTRUMENT, IN CORE + REACTOR, LMCR + DESIGN STUDY + IRRADIATION TESTING + SEFOR (RE) + R AND D PROGRAM + FUEL ELEMENTS

18-021211 DATED 1967
SCHLECHTENDAHL EG + CRAMER M + GAST K
SAFETY FEATURES OF A 300MWE SODIUM COOLED FAST BREEDER REACTOR (NA2)
KERNFORSCHUNGSZENTRUM, KARLSRUHE, GERMANY
KFK-611 + EURFNR-368 +. 25 PAGES, 10 FIGURES, 2 TABLES, 10 REFERENCES, JUNE 1967

PRESENTS A BRIEF DESCRIPTION OF THE DESIGN AND SUMMARIZES RESULTS OF THE ACCIDENT ANALYSIS FOR REACTIVITY PERTURBATIONS, COOLANT-SYSTEM FAILURES, AND SAFETY-SYSTEM FAILURE. THE CONTAINMENT DESIGN IS BASED ON THE REQUIREMENTS OF A NUCLEAR ACCIDENT RESULTING FROM A SEVERE COMPONENT FAILURE COINCIDING WITH FAILURE OF THE PREVENTIVE SAFEGUARDS. THE DESIGN-BASIS ACCIDENT WOULD RESULT IN 2-HR DOSES AT 500 METERS OF 0.98 REM TO THE BONES AND 0.17 REM TO THE THYROID.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + ACCIDENT, REACTIVITY + DESIGN STUDY + GERMANY

18-022105 DATED 1966
HARDE R + STROEHR KW
THE COMPACT SODIUM-COOLED NUCLEAR REACTOR FACILITY IN KARLSRUHE (KNK)
ANS-TRANS-519 +. 18 PAGES, FIGURES, TRANSLATED FROM ATOMWIRT. ATOMTECH., 11, PAGE 354-360, (1966)

THE 20-MWE EXPERIMENTAL POWER PLANT AT THE NUCLEAR RESEARCH CENTER HAS STEEL-JACKETED UO₂ FUEL AND ZIRCONIUM HYDRIDE MODERATOR. THE THERMAL REACTOR WILL BE USED TO TEST FAST-BREEDER FUELS. A FAST CORE IS A POSSIBILITY UNDER STUDY.

REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, TEST + ZIRCONIUM + GERMANY + HYDRIDE + IRRADIATION TESTING + FUEL ELEMENTS

18-022583 DATED 1966
EPEL LG + LEVINE MM + NUGENT G + PARSICK RJ
THE ORDERED-BED FAST REACTOR CONCEPT (1000-MW(E) REACTOR DESIGN)
BROOKHAVEN NATIONAL LABORATORY, INC.
BNL-50027 +. 37 PAGES, 15 FIGURES, 19 TABLES, 20 REFERENCES, JUNE 1966

1000-MWE PLANT WILL HAVE A DOUBLING TIME OF 11 YEARS, TOTAL ENERGY COST 3.28 MILLS/KWE-HR. USES URANIUM CARBIDE SPHERES COOLED BY LIQUID SODIUM. *** BRIEF REVIEW OF REACTOR PHYSICS AND ENGINEERING ASPECTS, ECONOMIC ANALYSIS.

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-022583 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + DESIGN STUDY + REACTOR, ORDERED BED, FAST

18-023870 ALSO IN CATEGORIES 6 AND 8 DATED 1966

DEFELICE J + HYLAND M

FAST TEST REACTOR HYPOTHETICAL MELTDOWN ACCIDENT. PROGRESS REPORT. OCTOBER 1965 - DECEMBER 1965

BATTELLE NORTHWEST, RICHLAND, WASH.

CUO-1430-3 +. 65 PAGES, 3 FIGURES, 17 TABLES, 10 REFERENCES, JAN. 21, 1966

A STUDY OF THE ENERGY YIELD FROM A MELTDOWN IS IN PROGRESS. FUEL-ZONING EFFECTS WERE EXPLORED IN DETAIL WITH AX-1 AND WEAK-EXPLOSION COMPUTER CODES. AX-1 INDICATES A SMALLER ENERGY RELEASE DUE TO FUEL ZONING THAN EXPECTED, AND INSENSITIVENESS TO DEGREE OF ZONING. WEAK-EXPLOSION RESULTS IN A LARGE ENERGY RELEASE INCREASE WITH ZONING, HOWEVER ITS LACK OF TREATMENT OF SURFACE EFFECTS UNDOUBTEDLY OVERESTIMATES ZONING EFFECTS. WORK CONTINUES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00
COPY, \$0.65 MICROFICHE

COMPUTER PROGRAM + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REACTIVITY + REACTOR, TEST + FFTF (TR) + CORE MELTDOWN + ACCIDENT, LOSS OF COOLANT

18-023947 ALSO IN CATEGORY 9 DATED 1966

QUARTERLY TECHNICAL PROGRESS REPORT ON AEC-SPONSORED ACTIVITIES. MARCH 15-JUNE 15, 1966

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.

APDA-193 +. 107 PAGES, 26 FIGURES, 9 TABLES, REFERENCES, MARCH 15, 1966

INDIVIDUAL PROJECTS ARE - (PROJECT 8) PASTE CONCEPT FOR BLANKETS (29 PG) - TEST LOOP OPERATED, IN-PILE LOOP DESIGNED, 1000-MWE REF.-DESIGN MADE. (10) SMALL-LEAK SODIUM-WATER REACTION (10 PG) - SOME TESTS ON TUBE WASTAGE MADE. (11) SODIUM LOOP OPERATION FOR IMPURITY DETECTION AND REMOVAL (43 PG). (14) UNDER-SODIUM SCANNER (8 PG) - DISPLAY OF ULTRASONIC SIGNAL ON TV TUBE BEING ASSEMBLED. (15) COMPILATION OF FERMI EXPERIENCE (3 PG) - LISTS TOPICAL REPORTS IN PROGRESS. (16) FAST-FLUX TEST FACILITY (3 PG) - WORK REPORTED IN APDA 186, 187, 188.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + SODIUM + REACTOR, TEST + OUT OF PILE EXPERIMENT + FFTF (TR) + CORROSION + REMOTE MANIPULATING AND VIEWING + COOLANT PURIFICATION SYSTEM + FAILURE, TUBING + INSTRUMENT, COOLANT QUALITY + R AND D PROGRAM

18-023948 DATED 1967

QUARTERLY TECHNICAL PROGRESS REPORT ON AEC-SPONSORED ACTIVITIES, DECEMBER 15, 1966-MARCH 15, 1967

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.

APDA-204 +. 63 PAGES, 6 FIGURES, 5 TABLES, REFERENCES, MARCH 15, 1967

INDIVIDUAL PROJECTS ARE - (PROJECT 8) SODIUM-WATER REACTION STUDIES (24 PG) - TEST AND EVALUATION CONTINUED. (11) SODIUM LOOP FOR COOLANT-IMPURITY DETECTION AND CLEANUP (18 PG). (14) UNDER-SODIUM SCANNER (5 PG) - NARROW-BEAM TRANSDUCER BEING TRIED IN A TANK OF SODIUM. (15) CURRENT FERMI EXPERIENCE (3 PG) - LISTS REPORTS IN PROGRESS. (16) FAST-FLUX TEST FACILITY (3 PG). WORK BEGUN ON ENERGY RELEASE FROM GROSS MELTDOWN OF METAL CORE AND CARBIDE CORE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + SODIUM + REACTOR, TEST + OUT OF PILE EXPERIMENT + FFTF (TR) + CORROSION + REMOTE MANIPULATING AND VIEWING + COOLANT PURIFICATION SYSTEM + FAILURE, TUBING + INSTRUMENT, COOLANT QUALITY + R AND D PROGRAM

18-024002 DATED 1968

KELLER DL

PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING OCTOBER THROUGH DECEMBER 1967

BATTELLE MEMORIAL INSTITUTE, COLUMBUS, OHIO

BMI-1826 +. FIGURES, TABLES, JANUARY 1, 1968

PROGRESS REPORTED ON STUDIES TO EVALUATE POTENTIAL OF MIXED U AND PU NITRIDE FUELS FOR LIQUID-METAL FAST-BREEDER REACTORS INCLUDES MATERIAL PROPERTIES, COMPATIBILITY, FABRICATION, AND IRRADIATION TESTING. SECOND TOPIC IS STUDY OF RADIATION EFFECTS ON VARIOUS CLADDING AND STRUCTURAL MATERIALS, WITH EMPHASIS ON MECHANICAL-PROPERTY CHANGES, PARTICULARLY FOR 347 STAINLESS STEEL. THE THIRD TOPIC - CHARACTERISTICS OF COATED-PARTICLE FUEL MATERIALS. FABRICATION, COMPATIBILITY, THERMAL SHOCK RESISTANCE, AND IRRADIATION TESTING ARE DISCUSSED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + COATED PARTICLE + FABRICATION + CLADDING + REACTOR, LMCR + PLUTONIUM + THERMAL MECHANICAL EFFECT + URANIUM + STEEL, STAINLESS + MATERIAL + NITRIDE + IRRADIATION TESTING + PROPERTY, PHYSICAL + REVIEW

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-024002 *CONTINUED*
+ R AND D PROGRAM + FUEL ELEMENTS

18-024156 ALSO IN CATEGORY 11 DATED 1967
BERGSTROM RN + CHITTENDEN WA + MCCLINTOCK DC
SODIUM-COOLED FAST REACTORS
SARGENT + LUNDY
3 PAGES, 1 FIGURE, POWER ENGINEERING, 71(10), PAGE 56-58, (OCT. 1967)

QUITE GENERAL DISCUSSION OF STRUCTURAL PROBLEMS OF THE BUILDING, WHICH, WHILE NOT BASICALLY DIFFERENT FROM WATER REACTORS, ARE COMPLEX AND HARDER TO ANALYZE.

REACTOR, LMCR + STRESS ANALYSIS + BUILDING

18-024515 DATED 1967
SHAVER RE + WITTENBROCK NG
REVIEW OF REACTOR SAFETY ANALYSES OF FAST AND LIQUID METAL COOLED REACTORS
BATTELLE-NORTHWEST, RICHLAND, WASH.
BNWL-477 +. 58 PAGES, TABLES, NOVEMBER, 1967

TO GUIDE DESIGN AND SAFETY ANALYSIS OF THE FAST-FLUX TEST FACILITY, THIS COMPILATION WAS MADE FROM SAFETY-ANALYSIS REPORTS ON U.S. FAST AND LIQUID-METAL-COOLED REACTORS. FACILITIES ARE - FERMI, HALLAM, SEFOR, FARET, EBR 1 AND 2, ZPR III. TABULATED INFORMATION ON CONTROL AND SAFETY SYSTEMS, REACTOR PROTECTION SYSTEMS, BACKUP SYSTEMS, CONTAINMENT, REACTIVITY EFFECTS AND PARAMETERS, FUEL AND FUEL HANDLING, ACCIDENTS AND CHEMICAL PROBLEMS, SITE, VENTILATION SYSTEM, AND WASTE EFFLUENTS. NO CONCLUSIONS GIVEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT + CONTAINMENT + CONTROL SYSTEM + REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + REACTIVITY EFFECT + REACTOR SAFETY SYSTEM + SITING + VENTILATION SYSTEM + FFTF (TR) + PLANT PROTECTIVE SYSTEM + WASTE MANAGEMENT + COOLANT CHEMISTRY + REVIEW + COMPARISON, REACTOR CHARACTERISTICS + FUEL COOLANT INTERACTION

18-025002 DATED 1968
KELLER DL
PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING OCTOBER THROUGH DECEMBER 1967
BATTELLE MEMORIAL INSTITUTE, COLUMBUS, OHIO
BMI-1826 +. FIGURES, TABLES, JANUARY 1, 1968

PROGRESS REPORTED ON STUDIES TO EVALUATE POTENTIAL OF MIXED U AND PU NITRIDE FUELS FOR LIQUID-METAL FAST-BREEDER REACTORS INCLUDES MATERIAL PROPERTIES, COMPATIBILITY, FABRICATION, AND IRRADIATION TESTING. SECOND TOPIC IS STUDY OF RADIATION EFFECTS ON VARIOUS CLADDING AND STRUCTURAL MATERIALS, WITH EMPHASIS ON MECHANICAL-PROPERTY CHANGES, PARTICULARLY FOR 347 STAINLESS STEEL. THE THIRD TOPIC - CHARACTERISTICS OF COATED-PARTICLE FUEL MATERIALS. FABRICATION, COMPATIBILITY, THERMAL SHOCK RESISTANCE, AND IRRADIATION TESTING ARE DISCUSSED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VIRGINIA \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + COATED PARTICLE + FABRICATION + CLADDING + REACTOR, LMCR + PLUTONIUM + THERMAL MECHANICAL EFFECT + URANIUM + STEEL, STAINLESS + MATERIAL + NITRIDE + IRRADIATION TESTING + PROPERTY, PHYSICAL + REVIEW + R AND D PROGRAM + FUEL ELEMENTS

18-025157 DATED 1967
BOWLES LF + TAYLOR D
PROTOTYPE FAST REACTOR SODIUM PUMPS
UKAEA, FAST REACTOR DESIGN OFFICE, RISLEY + ENGLISH ELECTRIC CO., LTD., WHETSTONE, GREAT BRITAIN
TRG-1538(R) +. 9 PAGES, 4 FIGURES, 4 REFERENCES, MARCH 1967, NUCLEAR ENGINEERING 12(132), PAGE 36-366

DESCRIBES 11,600- PRIMARY AND 17,000-GPM SECONDARY SODIUM PUMPS TO BE BUILT FOR THE PROTOTYPE FAST REACTOR (PFR) AT DOUNREAY. MAIN PROBLEMS SUCH AS NEED FOR LONG SHAFTS, SEALING, AND SODIUM-LUBRICATED BEARINGS ARE DISCUSSED. SPECIAL NOISE AND CAVITATION PROBLEMS OF PRIMARY PUMPS ALSO OUTLINED. DESIGN BASED ON DEVELOPMENT FROM 6000-GPM PROTOTYPE PUMP AT UKAEA REACTOR ENGINEERING LABORATORY, RISLEY. FIRST PUMP WILL UNDERGO FULL-SCALE TESTS IN WATER IN MID-1968, AND ALL PUMPS WILL BE INSTALLED BY END OF 1969.

REACTOR, FAST + SODIUM + UNITED KINGDOM + EQUIPMENT DESIGN + REACTOR, LMCR + PUMPS

18-025454 DATED 1968
DIECKAMP H + RODDIS LH
THE GENERAL PUBLIC UTILITIES/ATOMICS INTERNATIONAL FAST BREEDER REACTOR DEMONSTRATION PLANT
ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA + PENNSYLVANIA ELECTRIC COMPANY
40 PAGES, 16 FIGURES, 3 TABLES, PRESENTED AT AMERICAN POWER CONFERENCE, CHICAGO, ILLINOIS, APRIL 23-25, 1968

DESCRIBES A PROPOSED CONCEPT TO BE EVALUATED ON BASIS OF DESIGN, TEST RESULTS, AND FINANCIAL ASPECTS, WITH OPERATION BY 1975 ANTICIPATED. POWER SET AT 500 MWE. POT CONCEPT FOR CORE, PUMPS,

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-025454 *CONTINUED*

HEAT EXCHANGERS ETC. CLADDING SELECTION DIFFICULT BECAUSE FUEL LIFETIME LIMITED BY RADIATION DAMAGE TO CLAD. AEC DESIGN-CRITERIA BASIS FOR SAFETY GUIDELINES. PLANT HAS CAPABILITY OF LOAD FOLLOWING. MAINTENANCE AND INSPECTION SYSTEMS DISCUSSED BRIEFLY. FUEL BURNUP OF 75 MWD/KG (OF HEAVY ISOTOPE, PU PLUS U) DURING 1980S EXPECTED, WITH INCREASE TO 100. LINEAR POWER IS 15KW/FT OF ROD. FUEL-CYCLE COSTS FIRST 5 YEARS 1 MILL/KWHR.

AVAILABILITY - L. H. RUDDIS, JR., DIRECTOR, NUCLEAR ACTIVITIES, GENERAL PUBLIC UTILITIES CORPORATION

REACTOR, BREEDER + ECONOMICS + REACTOR, LMCR + REACTOR DESCRIPTION

18-025462

DATED 1968

RAMEY, JT

UTILITY PARTICIPATION IN THE AEC FAST BREEDER REACTOR PROGRAM

U.S. ATOMIC ENERGY COMMISSION

15 PAGES, FROM THE AMERICAN POWER CONFERENCE, APRIL 23, 1968, CHICAGO, ILL.

AEC PLANS NOW PROVIDE FOR FINANCIAL ASSISTANCE FOR AT LEAST THREE DEMONSTRATION PLANTS. RECOMMENDATIONS FOR INCREASED INVOLVEMENT BY THE UTILITIES INCLUDE (1) PLANNING AND ENSURING THAT LIGHT-WATER NUCLEAR-EQUIPMENT SUPPLIERS ESTABLISH AND USE CODES, STANDARDS, AND EFFECTIVE QUALITY-ASSURANCE PROGRAMS, (2) EXPEDITE AND AUGMENT PROGRAMS FOR EARLY DEMONSTRATION OF PLUTONIUM RECYCLE IN COMMERCIAL LIGHT-WATER REACTORS, AND (3) ENHANCE THE FAST-BREEDER-REACTOR TECHNICAL BACKGROUNDS OF PERSONNEL BY ASSIGNING TO LMFR AND OTHER BREEDER PROGRAMS.

AVAILABILITY - J.T. RAMEY, COMMISSIONER, U.S. ATOMIC ENERGY COMMISSION

ADMINISTRATIVE CONTROL + REACTOR, BREEDER + REACTOR, LMCR + TRAINING + REVIEW + R AND D PROGRAM

18-025730

ALSO IN CATEGORY 17

DATED 1968

INFORMATION REQUESTED ON SEFOR EMERGENCY PLANS

USAEC, DIVISION OF REACTOR LICENSING

4 PAGES, MAY 27, 1968, LETTER TO K. P. COHEN FROM P. A. MORRIS, DOCKET 50-231

DRL HAS BEEN REVIEWING STATUS OF MEDICAL PLANS, INCLUDING FACILITIES AND PERSONNEL FOR TREATMENT OF CONTAMINATED OR IRRADIATED PERSONNEL, AND REQUESTS INFORMATION ON (1) ON-SITE DECONTAMINATION FACILITIES, (2) QUALIFICATIONS AND TRAINING OF LOCAL PHYSICIANS, (3) ARRANGEMENTS FOR TRANSPORTING THE INJURED, (4) DISTANCE TO NEAREST HOSPITAL AGREEABLE TO ACCEPTING INJURED PEOPLE, (5) QUALIFICATIONS AND TRAINING OF HOSPITAL PERSONNEL, (6) LIMITATIONS ON HOSPITAL AVAILABILITY, (7) WRITTEN PLAN OF ACTION AT LOCAL HOSPITAL. ***SEPARATE LETTER INDICATES TRAINING SEMINAR THAT LOCAL MEDICAL PERSONNEL CAN ATTEND.

AVAILABILITY - USAEC PUBLIC DOCUMENT ROOM, WASHINGTON, D. C.

NRC QUESTION + REACTOR, FAST + REACTOR, LMCR + RADIATION INJURY TREATMENT + EMERGENCY PROCEDURE + REACTOR, PULSED + SEFOR (RE) + RADIOLOGICAL ASSISTANCE + OPERATING LICENSE PROCESS

18-025880

DATED 1968

MORE INFORMATION NEEDED ON SEFOR

USAEC, DIVISION OF REACTOR LICENSING

16 PAGES, LETTER TO K.P. COHEN FROM P.A. MORRIS, MAY 1, 1968, DOCKET NO. 50-231

REVIEW OF SEFOR FDSAR AND SUPPLEMENTS 1 THROUGH 9 INDICATES THAT A NUMBER OF AREAS WERE INADEQUATELY DESCRIBED AND EVALUATED AND THAT SUPPLEMENTS 10 THROUGH 15 HAVE NOT BEEN REVIEWED. WE UNDERSTAND THAT FURTHER SUPPLEMENTS ARE DUE. ENCLOSED IS A 15-PAGE LIST OF QUESTIONS ON ADDITIONAL INFORMATION REQUIRED.

AVAILABILITY - USAEC, PUBLIC DOCUMENT ROOM

NRC QUESTION + REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + REPORT, SAR + SEFOR (RE)

18-025935

ALSO IN CATEGORY 17

DATED 1968

MACPHERSON RE

FAST REACTOR SAFETY CONFERENCE

OAK RIDGE NATIONAL LABORATORY, OAK RIDGE, TENNESSEE

8 PAGES, TABLE, REFERENCE, NUCLEAR SAFETY, 9(3), PAGES 202-209 (MAY-JUNE, 1968)

BRIEFLY SUMMARIZES THE SESSIONS COVERING (1) CORE MELTDOWN ACCIDENTS, (2) SODIUM BOILING AND SUPERHEATING, (3) INCIDENTS IN REACTORS, AND (4) SAFETY. MALFUNCTIONS IN THE BRS WERE (1) FUEL-ELEMENT LEAKS CAUSED BY HEATER SHORTING, PLUS AIR IN LEAKAGE CAUSING MASSIVE SODIUM OXIDE PLUGGING, AND (2) REACTIVITY PERTURBATIONS DUE TO ENTRAINMENT OF ARGON BY THE SODIUM STREAM. ***PLUGGING OF SUBASSEMBLIES BY OXIDE HAS BEEN DETECTED PRIMARILY BY THE EFFECT ON PRIMARY FLOW. A PANEL DISCUSSION SHOWED THAT ATTENTION IS SWITCHING FROM THE LARGE, LESS PROBABLE ACCIDENT TO THE SMALLER, MORE FREQUENT ACCIDENTS WHICH AFFECT PLANT USE.

ACCIDENT ANALYSIS + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + SODIUM + INCIDENT + SAFETY PRINCIPLES AND PHILOSOPHY + REVIEW + FLOW BLOCKAGE + COOLANT QUALITY

APPARATUS FOR PROVIDING AIR ATMOSPHERE DIRECTLY OVER A REACTOR INSIDE LARGER ARGON-FILLED CHAMBER COMPRISED OF AXIALLY COLLAPSIBLE BELLOW WITH ONE END SEALED TO REACTOR AND OTHER END TO OPENING IN TOP OF CHAMBER. BELLOW PROTECTS CONTROL-ROD ELECTRICAL CONNECTIONS AND PERMITS DIRECT-CONTACT-MAINTENANCE CONTROL-ROD DRIVES. REACTOR COVER CAN BE RAISED, COLLAPSING BELLOW, FOR REPLACEMENT OF FUEL ELEMENTS.

RESPONDS TO DRL QUESTIONS OF MAY 3, 1968, REGARDING FACILITY DESIGN, CONTAINMENT, PHYSICS, REACTOR VESSEL, INSTRUMENTATION, REFLECTOR-CONTROL SYSTEM, REFUELING CELL, SAFETY ANALYSES, AND ENVIRONS AND RADIOLOGICAL EFFECTS.

REQUESTS PERMISSION PRIOR TO OPERATING LICENSE TO INSERT NO MORE THAN 6 FUEL RODS (LESS THAN 1/3 A CRITICAL MASS) IN THE AIR-FILLED REACTOR VESSEL TO CHECK CORE-CHANNEL ALIGNMENT. PROCEDURE INCLUDED.

PROGRESS REPORTED ON ACQUISITION OF INSTRUMENTS, FUEL FABRICATION, PLANT CONSTRUCTION, AND TRAINING.

REVISÉD PAGES (IN GERMAN) TO KNK SODIUM-COOLED REACTOR SAR.

ACRS BELIEVES REACTOR CAN BE OPERATED AT 1 MW AS REQUESTED, UNTIL REACTOR CHARACTERISTICS ARE WELL ENOUGH UNDERSTOOD.

REACTOR, FAST + REACTOR, IMCR + REACTOR, SAFETY RESEARCH + ACRS + REACTOR, PULSED + SEFOR (RE)

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029301 ALSO IN CATEGORY 6 DATED 1967
SAFETY PROBLEMS RELATED TO THE ASSEMBLING OF MOCK-UPS IN MASURCA
22 PAGES, 16 FIGURES, FROM INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPT. 1967

PUNCTUAL ADIABATIC MODEL WITH SIX GROUPS OF DELAYED NEUTRONS, PROGRAMMED FOR 7094 COMPUTER WITH
BRITTAN LINEARIZATION METHOD TO ALLOW SAME PRECISION WITH PROMPT AND DELAYED CRITICALITY.
STARTUP ACCIDENT WITH DELAYED SCRAM GIVES MAXIMUM ALLOWABLE \$0.2/SEC, AND \$5/SEC WITH INHERENT
SAFETY. STUDY OF LOADING ERRORS LEADS TO GRAPPLE SPEED LIMIT OF 1 M/MIN. DOSE RATES IN CELL FOR
500 W OPERATION CALCULATED.

CRITICAL ASSEMBLY + ACCIDENT, CONTROL ROD WITHDRAWAL + SAFETY ANALYSIS + REACTOR, LMFBR

18-029374 ALSO IN CATEGORY 5 DATED 1955
BRITTAN RU + CERUTTI B + LICHTENBERGER HV + RICE R + THALGOTT FW
HAZARD EVALUATION REPORT ON THE FAST REACTOR ZERO POWER EXPERIMENT (ZPR-III)
ARGONNE NATIONAL LABORATORY, LEMONT, ILLINOIS
ANL-FWT-105 +. 75 PAGES, 19 FIGURES, TABLES, APRIL 1955

THE ASSEMBLY IS SMALL PIECES IN DRAWERS PLACED IN ARRAY OF SQUARE TUBES ON TWO CARRIAGES SO THAT
TWO HALF-CRITICAL SECTIONS MAY BE BROUGHT TOGETHER AND CONTROLLED. PART OF PROGRAM TO ESTABLISH
FEASIBILITY OF SODIUM COOLED BREEDER. PROGRAM COVERS USE OF U-235. SUMMARIZES SAFETY
MECHANISMS, REACTIVITY COEFFICIENTS, AND KINETIC CHARACTERISTICS. THE MOST SEVERE ACCIDENT
OCCURS IF A LARGE POSITIVE TEMP COEFFICIENT EXISTS. SAFETY RODS DO NOT OPERATE AND A HIGH RATE
OF REACTIVITY INSERTION OCCURS. DETAILED CALCULATIONS GIVEN IN APPENDIX. DIRECT DOSE FROM CLOUD
ASSUMING 100% FISSION PRODUCTS ARE IN CLOUD IS GIVEN FOR CLOUD HEIGHTS OF 30 TO 200 METERS AND
WIND SPEEDS OF 1/2 AND 5 METERS/SEC.

ACCIDENT, CONSEQUENCES + CRITICAL ASSEMBLY + DOSE + REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION +
REPORT, SAR + TESTING + CRITICALITY EXPERIMENT + HAZARDS ANALYSIS + REACTOR PHYSICS + ZPR 3 (CAF) + OFF SITE

18-029375 ALSO IN CATEGORY 5 DATED 1959
GERSHUN IL
PRELIMINARY SAFEGUARDS REPORT BASED ON URANIUM-MOLYBDENUM FUEL FOR THE HALLAM NUCLEAR POWER FACILITY
ATOMIC INTERNATIONAL
NAA-SR-3379 +. 100 PAGES, FIGURES, TABLES, FEBRUARY 10, 1959

THE HNPF IS A 240 MWT SODIUM-COOLED GRAPHITE-MODERATED REACTOR. FUEL ELEMENTS ARE 19-ROD CLUSTERS
OF U10% MO ALLOY WITH STAINLESS STEEL CLADDING. CONTROL RODS ARE BORON-NICKEL RINGS ASSEMBLED
ON AN SS PULL TUBE. CONTAINMENT IS SUBATMOSPHERIC. RADIOACTIVE COMPONENTS ARE LOCATED IN
CONCRETE SHIELDED CELLS. AN INERT GAS ATMOSPHERE IS USED IN ALL AREAS CONTAINING PRIMARY-SYSTEM
PIPING AND COMPONENTS. DOSE FOR MAJOR STACK RELEASE AT EXCLUSION BOUNDARY UNDER NEUTRAL
CONDITIONS GIVEN (BUT TIME FOR THIS DOSE IS NOT GIVEN.) THE MCA IS ONE IN WHICH CONTROL ROD
THIMBLE FAILS IN REGION OF HELIUM ATMOSPHERE AND GAS LEAKS OUT. DOSE RATE AT SITE BOUNDARY IS
LESS THAN 0.1 MR/HR.

ACCIDENT ANALYSIS + ACCIDENT, CONSEQUENCES + CONTAINMENT DESIGN + DOSE + FISSION PRODUCT RELEASE + HALLAM
(LMR) + REACTOR, LMCR + ACCIDENT, MAXIMUM CREDIBLE (MCA) + REACTOR DESCRIPTION + REACTOR TRANSIENT +
INSTRUMENT, CONTROL + HAZARDS ANALYSIS + SYSTEM DESCRIPTION + EQUIPMENT + REPORT, PSAR + ENGINEERED SAFETY
FEATURE + AUXILIARY COOLING SYSTEM + OFF SITE

18-029433 ALSO IN CATEGORIES 2 AND 11 DATED 1968
RESPONSE TO QUESTION A-1. DEFINITIONS AND FUNCTION OF CLASS I AND II ITEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIFORNIA
11 PAGES, 3 FIGURES, 2 TABLES, PAGE A-1 THRU A-11 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE
APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS THE EXTENT TO WHICH THE DEFINITIONS AND FUNCTIONS OF CLASS I AND CLASS II ITEMS HAVE BEEN
MODIFIED IN THE COURSE OF FINAL DESIGN AND CONSTRUCTION. (ANSWER ON PGS A-1 THRU -11.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION +
EARTHQUAKE ENGINEERING

18-029434 ALSO IN CATEGORIES 2 AND 11 DATED 1968
RESPONSE TO QUESTION A-2 - EARTHQUAKE EXPOSURE
GENERAL ELECTRIC CO., SAN JOSE, CALIFORNIA
4 PAGES, 1 TABLE, PAGE A-12 THRU A-15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE
TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

INDICATE THE EARTHQUAKE EXPOSURE, DESCRIBED BY RESPONSE SPECTRA, USED IN THE DESIGN FOR BOTH THE
DESIGN AND MAXIMUM EARTHQUAKES. (ANSWER ON PGS A-12 THRU -15.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029434 *CONTINUED*
ENGINEERING + RESPONSE SPECTRUM

18-029435 ALSO IN CATEGORY 2 DATED 1968
RESPONSE TO QUESTION A-3 - EARTHQUAKE DESIGN CRITERIA
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE A-15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE THE CRITERIA USED IN DESIGN OF THE CLASS I AND CLASS II ITEMS FOR BOTH THE DESIGN AND MAXIMUM EARTHQUAKES, INCLUDING CRITERIA FOR COMBINED LOADING CONDITIONS FOR ACCEPTANCE LIMITS IN TERMS OF STRESS, DEFORMATION, OR OTHER SELECTED CONDITION FOR THESE LOADING COMBINATIONS, AND FOR THE DAMPING COEFFICIENTS AND THE RESPONSE SPECTRA PLOTS OR ACCELERATION COEFFICIENTS TO BE USED FOR THE DYNAMIC COMPONENTS OF LOADING. (ANSWER) SEE APPENDIX A OF PSAR, SUPPLEMENT 6 OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + RESPONSE SPECTRUM

18-029436 ALSO IN CATEGORIES 2 AND 11 DATED 1968
RESPONSE TO QUESTION A-4 - SEISMIC ANALYSIS OF CLASS I ITEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
66 PAGES, 21 FIGURES, 2 TABLES, PAGE A-16 THRU A-21 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE IN GENERAL THE METHODS USED FOR SEISMIC ANALYSIS OF CLASS I ITEMS. PROVIDE MINIMUM INFORMATION FOR THE CONTAINMENT AND A FEW OF THE MOST ESSENTIAL CLASS I SYSTEMS. (A) ANALYTICAL MODEL USED, E.G., LOCATION OF LUMPED MASSES, VALUES OF PARAMETERS USED, AND SUPPORT CONDITIONS. (ANSWER ON PAGES A-15 THRU -21 PLUS 60 PAGES OF TABLES AND FIGURES.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + REACTOR, LMCR + REPORT, SAR + ANALYTICAL TECHNIQUE + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING

18-029437 ALSO IN CATEGORY 2 DATED 1968
RESPONSE TO QUESTION A-5 - ACCOMMODATION OF EARTHQUAKE-INDUCED EXCITATION BY PUMP HANGERS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE A-28 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE HOW VERTICAL FORCES AND MOTIONS, DUE TO EARTHQUAKE INDUCED EXCITATION, ARE ACCOMMODATED BY THE CONSTANT TENSION HANGERS SUPPORTING THE E-M PUMPS. (ANSWER) SPRING CONSTANT CLOSE TO ZERO AND ACTS AS UPWARD LOADS ON PUMPS, COUNTERBALANCING GRAVITY. PERMIT BUILDING MOTION WITHOUT TRANSMITTING TO PUMP.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + SUPPORT STRUCTURE + DAMPING + EARTHQUAKE ENGINEERING + PUMPS

18-029438 ALSO IN CATEGORY 2 DATED 1968
RESPONSE TO QUESTION A-7 - SEISMIC ANALYSIS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE A-29 AND A-30 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EXPAND TABLES 1 AND 2 OF SUPPLEMENT 6 BY CORRECTING THE BUILDING ACCELERATION, AND BY ADDING A COLUMN EXPLAINING, IN A SHORT NOTE FOR EACH LISTED ITEM, THE PRINCIPAL CAUSE OF THE SEISMIC AMPLIFICATION. INDICATE IN A SEPARATE COLUMN THE PERCENTAGE OF SEISMIC STRESSES WITH RESPECT TO THE TOTAL SUM OF COMBINED STRESSES. FOR EACH LISTED CLASS I ITEM, INDICATE THE NATURAL FREQUENCY OR THE PERIOD OF THE ITEM, INCLUDING ITS IMMEDIATE SUPPORTS. FOR PIPING LISTED IN TABLE 1, INDICATE WHETHER THE SYSTEM FOR WHICH THE PERIOD IS GIVEN, INCLUDES THE SUPPORT. (ANSWER ON PGS A-29 AND -30.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + SUPPORT STRUCTURE + ACCELERATION + EARTHQUAKE ENGINEERING + INTERACTION, FOUNDATION AND STRUCTURE + PIPES AND PIPE FITTINGS

18-029439 ALSO IN CATEGORIES 2 AND 10 DATED 1968

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029439 *CONTINUED*

RESPONSE TO QUESTION A-8 - BATTERIES AND RACKS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE A-30 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ARE THE BATTERIES AND BATTERY RACKS LOCATED UNDER THE CONTROL ROOM CLASS I ITEMS. DISCUSS THE CRITERIA AND METHODS OF ANALYSIS USED IN THE DESIGN OF THE BATTERIES, AND THE RACKS HOLDING THE BATTERIES, TO ENSURE THAT THEY WILL FUNCTION PROPERLY UNDER THE MAXIMUM EARTHQUAKE LOADING.
(ANSWER) BATTERIES AND RACKS ARE CLASS I ITEMS, CERTIFIED EARTHQUAKE RESISTANT BY MANUFACTURER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + EMERGENCY POWER, ELECTRIC + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + BATTERIES & CHARGERS

18-029440 ALSO IN CATEGORIES 2 AND 9 DATED 1968

RESPONSE TO QUESTION A-9 - SEISMIC EFFECTS ON CRITICAL INSTRUMENTATION

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, PAGE A-30 THRU A-32 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CERTAIN CONTROLS AND INSTRUMENTATION IN THE CONTROL ROOM AND OTHER PARTS OF THE FACILITY ARE CRITICAL TO SAFE SHUTDOWN OF THE FACILITY. PROVIDE A DISCUSSION OF THE CRITERIA AND METHODS OF ANALYSIS EMPLOYED IN THE DESIGN OF THESE CRITICAL CONTROLS AND INSTRUMENTATION FOR SEISMIC EFFECTS INCLUDING SEISMIC LOADINGS, POSSIBLE TILT (I.E., MERCURY SWITCHES AND RELAYS), ETC.
(ANSWER ON PGS A-30 THRU -32.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + INSTRUMENT, CONTROL + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + CONTROL PANEL/ROOM

18-029441 ALSO IN CATEGORY 2 DATED 1968

RESPONSE TO QUESTION A-10 - PIPING

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

13 PAGES, 8 FIGURES, PAGE A-32 THRU -44 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ELABORATE ON PIPING STRESS ANALYSIS PARTICULARLY COMBINING HORIZONTAL AND VERTICAL COMPONENT OF EARTHQUAKE MOTION. SUBSTANTIATE STATEMENT THAT DIFFERENTIAL MOTION OF POINTS OF SUPPORT OF PIPING IS ZERO. IS RELATIVE OR OBSOLETE ACCELERATION OF EQUIPMENT FROM BUILDING MOTION ASSUMED. WHAT DESIGN PROVISIONS ENSURE NO DAMAGE TO SERVICE LINES WHICH PASS CLOSE TO WALLS. DISCUSS DISLODGEEMENT POTENTIAL OF FLOOR GRATINGS UNDER EARTHQUAKE LOADING AND PROTECTION PROVIDED PIPING TO PREVENT PUNCTURE BY GRATING RODS AT OPENINGS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + STRESS ANALYSIS + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + EARTHQUAKE ENGINEERING + PIPES AND PIPE FITTINGS

18-029442 ALSO IN CATEGORY 2 DATED 1968

RESPONSE TO QUESTION A-11 - PROTECTION OF CLASS I EQUIPMENT FROM TORNADOES

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE A-45 AND A-46 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS THE DESIGN PROVISIONS MADE FOR PROTECTION OF CLASS I EQUIPMENT FROM TORNADO WINDS AND TORNADO GENERATED MISSILES. DISCUSS THE TORNADO GENERATED MISSILES IN DETAIL. (ANSWER) GRATING SURROUNDS HEAT EXCHANGERS AND DIESEL GENERATOR TO PROTECT FROM FLYING DEBRIS. SEE ALSO SUPPLEMENT 1 OF PSAR PG 3-62 AND APPENDIX E.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + MISSILE GENERATION AND PROTECTION + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + DESTRUCTIVE WIND + GENERATOR, DIESEL + HEAT EXCHANGERS

18-029443 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION B-1 - STEADY STATE THERMAL GRADIENTS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

12 PAGES, 7 FIGURES, PAGE 6-1 THRU -12 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE A RATIONAL NUMERICAL ESTIMATE OF THE ERRORS INTRODUCED INTO THE CALCULATIONS FOR THE

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029443 *CONTINUED*

STRUCTURE BY THE USE OF STEADY STATE THERMAL GRADIENTS INSTEAD OF TRANSIENT GRADIENTS. COVER ALL THREE CASES - (1) OPERATING CONDITION, (2) ACCIDENT CONDITION, (3) SPILLING OF SODIUM. CONSIDER ONE OR TWO EXAMPLES ONLY, BUT PROVIDE NUMERICAL INFORMATION AND SKETCHES AS REQUIRED FOR EVALUATION. (ANSWER ON PG 8-1 THRU -12.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONCRETE + REACTOR, LMCR + REPORT, SAR + THERMAL MECHANICAL EFFECT + STRESS ANALYSIS + TEMPERATURE GRADIENT + CONTAINMENT STRUCTURE + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029444 ALSO IN CATEGORY 2 DATED 1968

RESPONSE TO QUESTION A-6 - DYNAMIC LOADING OF CLASS I ITEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE A-28 AND -29 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS METHODS USED TO TAKE INTO ACCOUNT THE INFLUENCE OF ROCKING, HORIZONTAL TRANSLATION AND ELASTICITY OF THE SUPPORTING STRUCTURES ON THE DYNAMIC LOADING OF CLASS I ITEMS. DISCUSS THE RESULTS OBTAINED FOR A REPRESENTATIVE CLASS I STRUCTURE AND CLASS I SYSTEM. (ANSWER) SEE QUESTION B-2. PIPING IS DISCUSSED IN QUESTIONS A-7 AND A-10B AND C.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + SUPPORT STRUCTURE + EARTHQUAKE ENGINEERING + INTERACTION, SOIL AND STRUCTURE + PIPES AND PIPE FITTINGS

18-029445 ALSO IN CATEGORIES 2 AND 11 DATED 1968

RESPONSE TO QUESTION B-2 - MATHEMATICAL MODEL FOR SEISMIC DESIGN
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, PAGE B-13 THRU -15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THE MATHEMATICAL MODEL USED FOR THE SEISMIC DESIGN OF THE CONTAINMENT STRUCTURE ASSUMES A VERTICAL FLEXIBLE CANTILEVER. ROCKING AND DIFFERENTIAL HORIZONTAL TRANSLATION OF THE BASE AND THE INTERACTION BETWEEN THE PRIMARY AND SECONDARY CONTAINMENTS APPEAR TO HAVE BEEN NEGLECTED. DISCUSS THE ERROR INTRODUCED INTO THE DESIGN BY THESE ASSUMPTIONS. (ANSWER ON PGS B-13 THRU B-15.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + REACTOR, LMCR + REPORT, SAR + CONTAINMENT STRUCTURE + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING + MATHEMATICAL TREATMENT + INTERACTION, SOIL AND STRUCTURE

18-029446 ALSO IN CATEGORIES 5 AND 11 DATED 1968

RESPONSE TO QUESTION B-3 - PRESSURE LOADING
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE B-16 AND -17 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

IT IS STATED IN THE APPLICATION THAT PRESSURE LOADING CAN BE APPLIED AS UNIFORM LOAD SIMULTANEOUSLY IN ALL CELLS AND THAT THE COMMUNICATIONS PROVIDED BETWEEN CELLS ARE SUCH THAT THIS ASSUMPTION IS VALID. DISCUSS A TRANSIENT PRESSURE DISTRIBUTION AND ITS INFLUENCE, OR LACK OF INFLUENCE, ON THE DESIGN. INCLUDE A DISCUSSION ON COMMUNICATIONS BETWEEN CELLS. (ANSWER) SEE SECTION XVI OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + PRESSURE, INTERNAL + RESPONSE TO NRC QUESTION

18-029447 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION B-4 - CONCRETE REINFORCEMENT
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE B-18 AND -19 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DOES CONCRETE HAVE ENOUGH REINFORCING TO COVER ALL TENSION IN TENSION BLOCK OF HOMOGENEOUS UNCRACKED CONCRETE SECTION. HAVE METHODS OF ACI CODE FOR CRACKED SECTION BEEN USED. ON WHAT BASIS HAS CONCRETE COMPRESSION BEEN ESTABLISHED. HAVE PROVISIONS BEEN MADE TO PREVENT BUCKLING OF REINFORCING BARS IN COMPRESSION ZONE, ESPECIALLY WHERE PLASTIC DEFORMATIONS ARE PERMITTED TO OCCUR. HAVE LOCAL PLASTIC STRAINS BEEN EVALUATED. PROVIDE ASSUMED ELASTICITY AND SHEAR MODULUS OF CONCRETE, CONCRETE POISSON'S RATIO, ASSUMED SHRINKAGE STRAIN FOR CONCRETE UNDER LONG-TERM TEMPERATURE LOADING. (ANSWER ON PAGES B-18 AND -19.)

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029447 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONCRETE + CONTAINMENT DESIGN + CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + CONTAINMENT
STRUCTURE + SEFOR (RE) + RESPONSE TO NRC QUESTION + STRUCTURAL INTEGRITY

18-029448 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION B-5 - LINER DESIGN

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

34 PAGES, 3 FIGURES, 2 TABLES, PAGE B-20 THRU -31 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE
APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CONCERNING LINER, DISCUSS VARIATIONS IN THICKNESS AND YIELD POINTS AND ERECTION INACCURACIES
FACTORED INTO DESIGN, UNBALANCED LOADS AT ANCHORS AND BUCKLING, DEFORMATION FROM BUCKLING,
STRAINS WHERE BUCKLING IS NOT PERMITTED, TESTING, WELDING ELECTRODE AND QUALITY CONTROL OF
PROCESS, DESIGN CRITERIA, LEAD OF MAXIMUM LOCAL STRESSES AT STRESS CONCENTRATION POINTS, USE OF
2% OF COLUMN LOAD FOR LATERAL RESTRAINT OF ANCHOR BARS, SODIUM TEMPERATURE FOR ACCIDENTAL SPILL,
DESIGN CRITERIA AND MARGINS FOR BELLOW, PENETRATION SLEEVES, AND OTHER FLEXIBLE JOINTS. (ANSWER
ON PG B-20 THRU -31 PLUS ATTACHMENT.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + STRESS + WELDING + SEFOR (RE) + RESPONSE
TO NRC QUESTION + QUALITY ASSURANCE + CONTAINMENT LINER + PENETRATIONS, PRIMARY CONTAINMENT

18-029449 ALSO IN CATEGORIES 2 AND 11 DATED 1968

RESPONSE TO QUESTION C-1 - INTERACTION OF INNER AND OUTER CONTAINMENTS DURING EARTHQUAKE

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE C-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE THE LOCAL LOADING GENERATED DURING AN EARTHQUAKE BY THE INTERACTION OF THE INNER AND
OUTER CONTAINMENTS. INCLUDE THE ROCKING AND THE DIFFERENTIAL (WITH RESPECT TO THE GROUND)
HORIZONTAL TRANSLATION OF THE BUILDING. PROVIDE THE CORRESPONDING MAXIMUM STRESSES AND STRAINS.
(ANSWER) SEE ANSWER TO QUESTION B-2.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + EARTHQUAKE
ENGINEERING + INTERACTION, SOIL AND STRUCTURE

18-029450 ALSO IN CATEGORIES 2 AND 11 DATED 1968

RESPONSE TO QUESTION C-2 - LOCAL STRESSES DUE TO EARTHQUAKE

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE C-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE THE LOCAL STRESSES, DUE TO AN EARTHQUAKE, AT THE JUNCTION OF THE STEEL SHELL AND HEAVY
CONCENTRATED MASSES, SUCH AS LOCKS, HATCHES, AND LARGE PENETRATIONS. FOR EXAMPLE, AN EMERGENCY
ESCAPE LOCK PROTRUDES FROM THE CONTAINMENT SHELL AT GROUND LEVEL AND APPEARS TO BE SUPPORTED BY
THE SHELL. THIS PROTRUSION IS MASSIVE AND LONG AND MAY REQUIRE ADDITIONAL SUPPORT TO PREVENT IT
FROM OVERLOADING THE CONTAINMENT SHELL AT THE POINT OF FASTENING. ADDITIONAL INFORMATION IS
REQUESTED AS TO THE DESIGN OF THIS ESCAPE LOCK AND THE PROVISIONS THAT HAVE BEEN OR WILL BE TAKEN
TO INSURE THAT IT CANNOT IN ANY WAY JEOPARDIZE THE CONTAINMENT OF THE STRUCTURE UNDER ACCIDENT
AND EARTHQUAKE LOADING CONDITIONS. (ANSWER) SEE SECT. 7.3 AND SUPPL. 7 TO FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + REACTOR, LMCR + REPORT, SAR + STRESS + CONTAINMENT AIR LOCK + SEFOR (RE) +
RESPONSE TO NRC QUESTION + EARTHQUAKE ENGINEERING

18-029451 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION D-1 - VACUUM BREAKER SYSTEM

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, 2 FIGURES, PAGE D-1 THRU D-3 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE
TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS THE FOLLOWING REGARDING THE VACUUM BREAKER SYSTEM - TYPE, SIZE MOTIVE FORCE, AND SETTING
OF VACUUM BREAKER VALUES, FREQUENCY AT WHICH THEIR OPERABILITY WILL BE CHECKED, AND METHOD OF
TESTING. (ANSWER) 4 VALVES INSTALLED IN PARALLEL (2 INSIDE AND 2 OUTSIDE CONTAINMENT). VALVES
ARE 6 INCH, 150 LB SWINGING DISC TYPE. OPERABILITY TESTS ANNUALLY. LEAKAGE TESTS BY PLACING
BLIND FLANGES ON OUTLET PIPING AND PRESSURIZING TO 30 PSIG.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
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CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029451 *CONTINUED*
REACTOR, LMCR + REPORT, SAR + TESTING + CONTAINMENT VACUUM BREAKER + SEFOR (RE) + SYSTEM DESCRIPTION +
RESPONSE TO NRC QUESTION + VALVES

18-029452 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-2 - PROTECTION OF CLOSED LOOPS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE D-4 AND -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THERE ARE, IN ADDITION TO THE SODIUM LINES, EIGHT REFRIGERANT LINES, ONE RADIOACTIVE GASEOUS WASTE
LINE, ONE ARGON SUPPLY LINE, ONE BREATHING AIR LINE, AND ONE NITROGEN SUPPLY LINE THAT PENETRATES
THE OUTER CONTAINMENT BARRIER, ALL OF WHICH ARE CLASSIFIED AS TYPE 3 PENETRATIONS, I. E., THEY
ARE CLOSED LOOPS AND, THEREFORE, DO NOT REQUIRE ISOLATION VALVES. DISCUSS EXTENT TO WHICH THESE
LOOPS HAVE BEEN PROTECTED FROM (A) FALLING OBJECTS, (B) MISSILES, (C) SEISMIC DISTURBANCES, (D)
WIND LOADS, (E) FIRE, (F) OUTER CREDIBLE MISHAPS. (ANSWER) SEE SECTION VII OF FDSAR AND ANSWER
TO QUESTION D-15.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + FIRE + REACTOR, LMCR + MISSILE GENERATION AND PROTECTION + REPORT, SAR + SEFOR
(RE) + EARTHQUAKE + RESPONSE TO NRC QUESTION + DESTRUCTIVE WIND + CONTAINMENT PENETRATION, PIPE

18-029453 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-3 - CLOSURE DEVICES OF SODIUM LINES FOR CONTAINMENT ISOLATION
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE D-5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THE TWO 10-INCH MAIN SECONDARY SODIUM LINES AND THE TWO 3-INCH AUXILIARY SECONDARY SODIUM LINES
HAVE NO CONTAINMENT ISOLATION CLOSURE DEVICES AT THE OUTER CONTAINMENT BARRIER NOR AT ANY OTHER
POINT THROUGHOUT THEIR CIRCUITS. DISCUSS THE OPERATIONAL CONSIDERATIONS, THE CONTAINMENT
ISOLATION CAPABILITIES UNDER ALL ACCIDENT CONDITIONS AND THE ASSOCIATED RADIOLOGICAL HAZARDS THAT
LED TO THE ABOVE DESIGN DECISION. (ANSWER) THEY QUALIFY AS TYPE 3 PENETRATIONS. COVER GAS
PRESSURE WILL BE KEPT HIGHER THAN THAT IN PRIMARY LOOPS, SO LEAKAGE WILL BE INTO PRIMARY SYSTEM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING
SYSTEM + CONTAINMENT PENETRATION, PIPE + VALVES + PIPES AND PIPE FITTINGS

18-029454 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-4 - CONTAINMENT PENETRATION DESIGN
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE D-6 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FROM FIGURE VII-3 IN FDSAR, DEPICTING CONTAINMENT PENETRATION DESIGN, IT IS NOTED THAT 13
TABULATED LINES EACH HAVE ONE COMMON PENETRATION SLEEVE WHICH PASSES THROUGH BOTH INNER AND OUTER
CONTAINMENT BARRIERS. IT IS NOTED ON PAGE 13 OF SUPPLEMENT 7 THAT THE SILICONE RUBBER DIAPHRAGM
SEALS BETWEEN THE PENETRATION SLEEVES AND THE OUTER CONTAINMENT HAVE BEEN REPLACED WITH METALLIC
SEALS. SUPPLY A DISCUSSION OF THE NEW PENETRATION DETAILS FOR THE 13 ABOVE MENTIONED LINES,
INCLUDING A TYPICAL DRAWING, AND THE METHOD THAT WILL BE EMPLOYED TO VERIFY THE LEAK-TIGHT
INTEGRITY OF THE JUNCTION BETWEEN THE PENETRATION SLEEVE AND (A) THE INNER CONTAINMENT AND (B)
THE OUTER CONTAINMENT. (ANSWER) FIGURE GIVEN OF TYPICAL SEAL. INTEGRITY DETERMINED AS PART OF
ENTIRE INNER CONTAINMENT TEST.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + INTEGRITY + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION +
INDEPENDENCE + PENETRATIONS, PRIMARY CONTAINMENT

18-029455 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-5 - BELLOWS TEMPERATURES FROM SODIUM LINE CONTAINMENT PENETRATIONS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, 1 FIGURE, PAGE D-6 AND -7 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS EFFECTS OF LOCALIZED PEAK TEMPERATURES EXPERIENCED BY METAL BELLOWS AS A RESULT OF CALROD
HEATERS IN THE SODIUM LINE CONTAINMENT PENETRATIONS. TAKE ANTICIPATED PIPE DISPLACEMENT AND COLD
STARTUP INTO ACCOUNT WHEN ESTABLISHING PEAK TEMPERATURES. (ANSWER) SEE SUPPL. 7, SECT. 2,3, TO
FDSAR. PEAK TEMP. OF 850 F FOR BELLOWS. DESIGN TEMP. IS 1000 F.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029455 *CONTINUED*
REACTOR, LMCR + REPORT, SAR + THERMAL CONSIDERATION + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION +
CONTAINMENT PENETRATION, PIPE

18-029456 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-6 - ROLLER GUIDE SUPPORT
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE D-8 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

SUBMIT DRAWING OF TYPICAL ROLLER GUIDE SUPPORT, SUCH AS WOULD BE USED ON PENETRATION MK-2.
(ANSWER) FIGURE GIVEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

GUIDE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + SUPPORT
STRUCTURE + CONTAINMENT PENETRATION, PIPE

18-029457 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-7 - REFUELING CELL INTEGRITY AND CHEMICAL REACTION THEREIN AFTER ACCIDENT
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, 1 FIGURE, PAGE D-8 AND -9 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS MOST ADVERSE CHEMICAL REACTION CONSIDERED DURING REFUELING CELL DESIGN, AND ABILITY OF
REFUELING CELL TO CONTAIN RADIOACTIVE AND NONRADIOACTIVE MATERIALS WITHIN REFUELING CELL SHOULD
SUCH AN ACCIDENT OCCUR. (ANSWER) CELL ENVIRONMENT TO BE MAINTAINED AT NOT MORE THAN 10 PPM O₂ OR
H₂O. NO FIRES OR CHEMICAL REACTIONS CONSIDERED IN EVALUATING PRESSURE REQUIREMENTS. TO AVOID
PRESSURIZING, A 20 INCH DIAMETER RUPTURE DISK IS PROVIDED TO RELIEVE AT DELTA P OF 4 PSI. SEE
SECT. 7.2 OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CHEMICAL REACTION + FIRE + REACTOR, LMCR + ACCIDENT, REFUELING + REPORT, SAR + CONTAINMENT INTEGRITY +
PRESSURE RELIEF + SEFOR (RE) + PRESSURE, INTERNAL + RESPONSE TO NRC QUESTION

18-029458 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-8 - CONSEQUENCES OF LINE BREAK ON CLOSED-LOOP SYSTEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
4 PAGES, PAGE D-9 THRU -12 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FROM TABLE VII-1 OF FDSAR, IT IS NOTED THAT THERE ARE IN EXCESS OF 60 INNER CONTAINMENT
PENETRATIONS RANGING IN SIZE FROM 30-INCHES IN DIAMETER TO 1/4-INCH IN DIAMETER THAT DO NOT HAVE
ISOLATION VALVES IN THE LINES BECAUSE THEY FORM CLOSED LOOPS AND ARE THUS CONSIDERED AN INTERNAL
OR EXTERNAL EXTENSION OF THE CONTAINMENT. DISCUSS CONSEQUENCES SHOULD A LINE BREAK AND EXTENT TO
WHICH THESE INTERNAL AND EXTERNAL CLOSED SYSTEMS HAVE BEEN PROTECTED FROM (A) FALLING OBJECTS,
(B) MISSILES, (C) SEISMIC DISTURBANCES, (D) WIND LOADS AND TORNADOES, (E) FIRES, (F) OTHER
CREDIBLE MISHAPS. (ANSWER ON PGS D-9 THRU -12.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + FIRE + REACTOR, LMCR + MISSILE GENERATION AND PROTECTION + CONTAINMENT ISOLATION
+ REPORT, SAR + CONTAINMENT INTEGRITY + SEFOR (RE) + EARTHQUAKE + RESPONSE TO NRC QUESTION + DESTRUCTIVE WIND
+ VALVES

18-029459 ALSO IN CATEGORY 5 DATED 1968
RESPONSE TO QUESTION D-9 - ARGON GAS COOLANT LOOP LINES
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE D-12 AND -13 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ARGON GAS COOLANT LOOP PASSES THROUGH AIR ZONE IN ORDER TO REACH FAN AND COOLER LOCATED IN
NITROGEN ZONE. SHOULD EITHER LINE DEVELOP LARGE LEAK, (A) HOW LONG WOULD IT TAKE ARGON FAN TO
EMPTY ALL GAS INTO OUTER CONTAINMENT AIR ZONE AND REPLACE IT WITH AIR FROM OUTER CONTAINMENT, (B)
WHAT WOULD BE COURSE OF SUCH ACCIDENT, AND POTENTIAL DOSES SHOULD IT OCCUR AT TIME WHEN REACTOR
VESSEL HEAD AND IRRADIATED FUEL STORAGE TANK SHIELDING COVER WERE OFF, (C) WHAT MEANS TERMINATE
REACTION BETWEEN EXPOSED SODIUM AND CONTAINMENT AIR, (D) COULD IT BE TERMINATED BEFORE ALL OXYGEN
IN OUTER CONTAINMENT WAS CONSUMED. (ANSWER ON PGS D-12 AND -13).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + REACTOR, LMCR + FAILURE, PIPE + REPORT, SAR + ACCIDENT, HYPOTHETICAL + SEFOR (RE) +
RESPONSE TO NRC QUESTION + AUXILIARY COOLING SYSTEM

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029460 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-10 - BRAZED JOINTS OF COPPER REFRIGERANT LINES
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE D-14 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

REFRIGERANT LINES ARE COPPER TUBING BRAZED TO BOTH END CAPS OF PENETRATION SLEEVE PASSING THROUGH THE OUTER CONTAINMENT BARRIER. DISCUSS ADEQUACY OF BRAZED JOINTS TO WITHSTAND CONCURRENT PRESSURE, TEMPERATURE, AND PIPE REACTION LOADS EXPECTED UNDER ACCIDENT CONDITIONS. (ANSWER) PIPING IS BRAZED TO OUTER CLOSURE PLATE OF PENETRATION SLEEVES AND TO A CLOSURE PLATE ATTACHED THROUGH A BELLOWS JOINT TO INNER END OF PENETRATION SLEEVES. PIPE REACTIONS RELIEVED BY BELLOWS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION + AUXILIARY COOLING SYSTEM + PIPES AND PIPE FITTINGS

18-029461 ALSO IN CATEGORIES 9 AND 11 DATED 1968
RESPONSE TO QUESTION D-11 - INSTRUMENT LEAD PENETRATION
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE D-14 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

SLIDING FRICTION IS THE SOLE MEANS OF HOLDING SHIELDING PLUGS IN PLACE WHEN INSTRUMENT LEADS PASS THROUGH THE PENETRATION. DISCUSS RELATIONSHIP OF THE REFUELING CELL PRESSURE AND FORCE REQUIRED TO SLIDE PLUGS OUT. (ANSWER) WHEN IN USE, UTILITY WALL SLEEVE PLUG WOULD BE REPLACED, WITH CONFIGURATION HELD IN PLACE BY BATTING. NO DESIGN AT PRESENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION + ELECTRICAL CONDUCTION + CONTAINMENT PENETRATION, ELECTRICAL + INSTRUMENTS, MISC.

18-029462 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-12 - MULTIPLE QUAD-RING SEALS OF MANIPULATOR
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE D-15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FRICTION LOAD OF MULTIPLE QUAD-RING SEALS IS SOLE MEANS OF HOLDING MANIPULATOR IN PLACE UNTIL THE SLAVE ARM CONTACTS THE CELL WALL. DISCUSS (A) RELATIONSHIP OF REFUELING CELL PRESSURE AND THE FORCE REQUIRED TO MOVE THE MANIPULATOR OUTWARD AND (B) MEANS OF ATTACHMENT OF THE SLAVE ARM AND ITS LOAD CARRYING CAPABILITY WHEN IT CONTACTS THE REFUELING CELL WALL. (ANSWER) SEAL HAS BEEN PROOF TESTED AT 2000 LB AXIAL LOAD WITHOUT SLIPPAGE (EQUIVALENT TO 25 PSIG DELTA P ACROSS CELL BARRIER).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

HOT CELL + REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + SEAL + REMOTE MANIPULATING AND VIEWING + RESPONSE TO NRC QUESTION

18-029463 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-13 - CONNECT-DISCONNECT FITTINGS OF BREATHING AIR SYSTEM
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, 1 FIGURE, PAGE D-15 AND -16 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

SUPPLY DRAWINGS OF CONNECT-DISCONNECT FITTINGS UTILIZED IN BREATHING AIR SYSTEM TO PERMIT AN EVALUATION TO BE MADE OF POSSIBILITY OF AIR BEING RELEASED IN INNER CONTAINMENT. (ANSWER) VENDOR DATA SHEETS SHOWN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + PERSONNEL PROTECTIVE DEVICE + REPORT, SAR + AIR + SEFOR (RE) + SYSTEM DESCRIPTION + SEAL + RESPONSE TO NRC QUESTION + VALVES

18-029464 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION D-14 - CONTAINMENT LEAKAGE TESTS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
4 PAGES, 1 TABLE, PAGE D-17 THRU -20 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE PLANS FOR INNER AND OUTER CONTAINMENT LEAKAGE TESTS. (ANSWER ON PGS D-17 THRU -20.)

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029464 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + TEST, LEAK RATE + CONTAINMENT INTEGRITY + TEST, LEAK LOCATION + SEFOR (RE) +
RESPONSE TO NRC QUESTION

18-029465 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION D-15 - ISOLATION SYSTEMS FOR CONTAINMENT PENETRATIONS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE D-20 THRU -46 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

TABULATE ISOLATION SYSTEMS FOR CONTAINMENT PENETRATIONS, INCLUDING FUNCTIONAL LOCATION, PIPE AND
VALVE SIZES, DESIGN CONDITIONS, AND REQUIRED FUNCTIONAL PERFORMANCE. (ANSWER) 25 TABULATION
SHEETS GIVEN.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029466 ALSO IN CATEGORY 6 DATED 1968

RESPONSE TO QUESTION E-1 - REMOVAL OF BEO FROM CORE

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE E-1 AND -2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS MORE FULLY THE ANTICIPATED EFFECTS THAT REMOVAL OF BEO WILL HAVE ON REACTIVITY EFFECTS AND
MAGNITUDE OF THE DOPPLER COEFFICIENT. IS EXPERIMENTAL INFORMATION EQUIVALENT TO THAT USED IN
PREDICTING PHYSICS PARAMETERS FOR THE INITIAL CORE LOADING ALSO AVAILABLE FOR PREDICTING THESE
PARAMETERS WITH BEO ABSENT. (ANSWER) REPLACING BEO WITH NICKEL RESULTS IN REDUCING DOPPLER
COEFF. 11 PER CENT. SEE SECT. 3.3.5 OF FDSAR SUPPLEMENT 10.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

BERYLLIUM + DOPPLER COEFFICIENT + REACTOR, LMCR + OXIDE + REACTIVITY EFFECT + REPORT, SAR + NICKEL + SEFOR
(RE) + RESPONSE TO NRC QUESTION + REACTOR PHYSICS

18-029467 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION F-1 - REACTOR VESSEL SPECIFICATIONS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

6 PAGES, PAGE F-1 THRU -6 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

INDICATE REVISIONS OR EXPANSIONS OF REACTOR VESSEL SPECIFICATIONS. WERE COMPLETE REACTOR VESSEL
CALCULATIONS PERFORMED FOR COMPLETE OPERATIONAL MODE OF VESSEL, INCLUDING ANY NEW OR REVISED
TRANSIENTS. WAS COMPLIANCE WITH ASME CODE - SECTION III OBTAINED. (ANSWER) SPECS. WERE REVISED
3 TIMES. FINAL IS GIVEN IN APPENDIX A, SUPPLEMENT 16 OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + REACTOR, LMCR + PRESSURE VESSELS + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029468 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION F-2 - O-RING FLEXING MARGINS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE F-6 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

HAS YOUR EVALUATION VERIFIED THAT U-RING FLEXING MARGINS ARE SUFFICIENT TO ENSURE A GOOD SEAT AT
THE MAXIMUM CLOSURE FLANGE DEFLECTIONS. (ANSWER) VESSEL HEAD AND FLANGES DESIGNED FOR 1500 PSIG.
MAX. FLANGE DEFLECTIONS ARE LESS THAN 1 MIL. SPRINGBACK OF SEAL IS 5 AND 6 MILS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + PRESSURE VESSELS + REPORT, SAR + SEFOR (RE) + FLANGE + SEAL + RESPONSE TO NRC QUESTION

18-029469 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION F-3 - BOLTS IN REACTOR VESSEL AFTER ACCIDENT

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, 1 TABLE, PAGE F-6 THRU -8 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029469 *CONTINUED*

FOR BOLTS HOLDING REACTOR VESSEL OUTER HEAD, CENTER HEAD, AND BOTTOM SHIELD PLUG WHICH ARE TO ABSORB ENERGY OF DESIGN BASIS ACCIDENT BLAST BY PLASTIC DEFORMATION (STRETCHING), EVALUATE CRITICAL VELOCITY AND COMPARE IT WITH DEFORMATION VELOCITY OF BOLTS. (ANSWER ON PAGES F-6 THRU - 8.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + DESIGN CRITERIA + REACTOR, LMCR + PRESSURE VESSELS + REPORT, SAR + FASTENER + SEFOR (RE) + RESPONSE TO NRC QUESTION + STRUCTURAL INTEGRITY

18-029470 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION F-4 - PRESSURE VESSEL DESIGN

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE F-9 AND -10 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS DESIGN BASIS FOR CYLINDRICAL SECTION BETWEEN REACTOR VESSEL SUPPORT FLANGE AND HEAD FLANGE AND CYLINDRICAL SECTION OF REACTOR VESSEL BELOW HEAD FLANGE. INCLUDE MAXIMUM OPERATING AND ACCIDENT LOADING CONDITIONS, COMPONENT AND NET STRESSES, AND DESIGN MARGINS FOR THOSE REGIONS. DISCUSS EFFECT OF IMPACT FORCES ON DESIGN MARGINS FOR ABOVE REGIONS FOR REACTOR VESSEL HEAD DROP ACCIDENT. CONSIDER CASES FOR ASYMMETRIC IMPACT OF (A) CENTER HEAD, (B) OUTER HEAD, AND (C) COMBINED CENTER HEAD AND OUTER HEAD. CONSIDER EFFECT ON DESIGN MARGINS OF CORE SUPPORT STRUCTURE FOR CASE WHERE CENTER IMPACTS ON VESSEL SHIELD SECTION ATTACHED TO CENTER HEAD DETACHES AND FALLS AT ANGLE ONTO TOP OF CORE. (ANSWER ON PAGES F-9 AND -10.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT VESSEL LOADING + DESIGN CRITERIA + REACTOR, LMCR + PRESSURE VESSELS + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + SUPPORT STRUCTURE

18-029471 ALSO IN CATEGORY 9 DATED 1968

RESPONSE TO QUESTION G-1 - PROTECTION INSTRUMENTS DURING TRANSIENT TEST

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, 2 REFERENCES, PAGE G-1 AND -2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

IDENTIFY AND EVALUATE ADEQUACY OF PROTECTION INSTRUMENT SYSTEM TO BE USED DURING PLANNED TRANSIENT TEST PROGRAM. (ANSWER ON PGS G-1 AND -2.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

INSTRUMENT, PROTECTIVE + REACTOR, LMCR + REACTOR TRANSIENT + REPORT, SAR + TESTING + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029472 ALSO IN CATEGORY 9 DATED 1968

RESPONSE TO QUESTION G-2 - INSTRUMENTATION DURING FUEL HANDLING

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, PAGE G-3 THRU -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE INSTRUMENTATION USED DURING FUEL HANDLING OPERATIONS. JUSTIFY ADEQUACY. (ANSWER ON PGS G-3 THRU -5.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FUEL HANDLING + REACTOR, LMCR + REPORT, SAR + MONITOR + SEFOR (RE) + RESPONSE TO NRC QUESTION + RADIATION MONITORS

18-029473 ALSO IN CATEGORY 9 DATED 1968

RESPONSE TO QUESTION G-3 - RELIABILITY OF ION CHAMBER NITROGEN COOLING SYSTEM

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

7 PAGES, 2 FIGURES, PAGE G-5 THRU -11 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS RELIABILITY OF ION CHAMBER NITROGEN COOLING SYSTEM AND ASSOCIATED TEMPERATURE SENSORS. CAN TEMPERATURE MONITORING SYSTEM BE DISABLED BY SINGLE FAILURE. (ANSWER ON PGS G-5 THRU -11.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CHAMBER, ION + REACTOR, LMCR + REPORT, SAR + NITROGEN + INSTRUMENT, TEMPERATURE + SEFOR (RE) + RESPONSE TO NRC QUESTION + INSTRUMENT, POWER RANGE + AUXILIARY COOLING SYSTEM

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029474 ALSO IN CATEGORY 9 DATED 1968
RESPONSE TO QUESTION G-4 (PT.1) - BYPASS MODE OF PROTECTIVE INSTRUMENTS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE G-12 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE SCHEMATIC DIAGRAM OF THOSE PORTIONS OF THE PROTECTION CIRCUITRY INSTRUMENT WHICH ACTUATE
AND REMOVE THE OPERATIONAL BYPASSES. (ANSWER) SEE SUPPLEMENT 2 OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

INSTRUMENT, PROTECTIVE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029475 ALSO IN CATEGORY 9 DATED 1968
RESPONSE TO QUESTION G-4 (PT.2) - PROTECTION SYSTEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
8 PAGES, 4 FIGURES, PAGE G-12 THRU -15 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE
TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE ABILITY TO MEET PROPOSED IEEE STANDARD, NUCLEAR POWER PROTECTION SYSTEMS. (ANSWER ON PGS
G-12 THRU -15.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CODES AND STANDARDS + INSTRUMENT, PROTECTIVE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC
QUESTION

18-029476 DATED 1968
RESPONSE TO QUESTION G-5 - SAFE SHUTDOWN AFTER ABANDONING CONTROL ROOM
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE G-16 AND -17 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS PROCEDURES TO EFFECT SAFE SHUTDOWN IN EVENT CONTROL ROOM MUST BE ABANDONED DURING
OPERATION. (ANSWER ON PGS G-16 AND -17.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + EMERGENCY PROCEDURE + SEFOR (RE) + RESPONSE TO NRC QUESTION + CONTROL PANEL/ROOM
+ OPERATOR ACTION

18-029477 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION G-6 - MAINTAINANCE OF CONTAINMENT GAS QUALITY
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, PAGE G-17 THRU -19 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE AND EVALUATE THE INSTRUMENTATION AND PROCEDURES WHICH ENSURE THAT THE DESIRED INERT GASES
ARE THE ONLY ONES INTRODUCED INTO CONTAINMENT. (ANSWER ON PGS G-17 THRU -19.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + CONTAINMENT AIR COOLING + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC
QUESTION + CONTAINMENT ATMOSPHERE, INERT

18-029478 ALSO IN CATEGORY 10 DATED 1968
RESPONSE TO QUESTION G-7 - EFFECT OF SODIUM VAPOR ON ELECTRICAL SYSTEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE G-19 AND -20 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231 MAY 24, 1968

DISCUSS ADVERSE EFFECTS, IF ANY, OF SODIUM VAPOR ON ELECTRICAL EQUIPMENT SUCH AS CABLES,
INSULATORS, CONNECTORS, AND TERMINAL BOARDS. (ANSWER) SEE SECT. 3.4 OF SUPPLEMENT 12 TO FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SODIUM + VAPOR PRESSURE + ENVIRONMENT + SEFOR (RE) + RESPONSE TO NRC QUESTION +
DAMAGE + ELECTRICAL CONDUCTION

18-029479 DATED 1968
RESPONSE TO QUESTION G-8 - INSTRUMENT MONITORING ACCIDENTAL RELEASE FROM BATCH TANKS

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029479 *CONTINUED*
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE G-20 AND -21 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT IS THE BASIS FOR THE DESIGN OF THE RADIATION INSTRUMENTATION WHICH MONITORS ACCIDENTAL RELEASES FROM THE BATCH TANKS. CAN A CREDIBLE RELEASE SATURATE THE INSTRUMENTATION. (ANSWER) ONE SYSTEM MONITORS BEFORE DILUTION WITH STACK DISCHARGE, OTHER MONITORS AFTER DILUTION. THE GAS EFFLUENT SHUT-OFF VALVE CLOSES IF INSTRUMENT FAILS OR BECOMES SATURATED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + MONITOR, STACK + REPORT, SAR + SOURCE, CONTINUOUS + STACK + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + RADIATION MONITORS

18-029480 DATED 1968
RESPONSE TO QUESTION G-9 - MONITORING SYSTEM FOR CORE GAS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, 1 FIGURE, PAGE G-21 AND -22 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE RADIATION MONITORING SYSTEM FOR REACTOR CORE GAS. JUSTIFY ITS ADEQUACY. (ANSWER) FIGURE GIVEN. COMBINATION OF PERIODIC FUEL EXAMINATION AND GAS SAMPLING WILL PROVIDE ADEQUATE INFORMATION ON FUEL OPERATION AND CONDITION OF FUEL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + MONITOR, GAS + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + SEFOR (RE) + OFF GAS + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION

18-029481 ALSO IN CATEGORY 9 DATED 1968
RESPONSE TO QUESTION G-10 - IN-CORE INSTRUMENT TO SENSE PRESSURE PULSE DURING TRANSIENT TEST
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE G-23 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ARE THERE IN-CORE INSTRUMENTS CAPABLE OF SENSING PRESSURE PULSES WHICH MAY OCCUR DURING TRANSIENT TEST PROGRAM. (ANSWER) ANALYSIS OF EXCURSIONS DID NOT INDICATE A SIGNIFICANT PRESSURE PULSE. NO INSTRUMENTS PROVIDED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

INSTRUMENT, IN CORE + REACTOR, LMCR + REACTOR TRANSIENT + REPORT, SAR + TEST, PHYSICS + SEFOR (RE) + RESPONSE TO NRC QUESTION + PRESSURE PULSE + INSTRUMENT, PRESSURE

18-029482 ALSO IN CATEGORY 9 DATED 1968
RESPONSE TO QUESTION H-1 - SINGLE CONTROL SCHEME
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, PAGE H-1 THRU -3 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS BASIS FOR YOUR CONCLUSION THAT ONE CONTROL SCHEME IS SUFFICIENT FOR SEFOR. INCLUDE CONSIDERATIONS GIVEN TO FAILURES THAT COULD DISABLE THE REACTIVITY CONTROL SYSTEM (I.E., SHIELD PLUG MOVEMENT). (ANSWER ON PGS H-1 THRU -3.) NOT SURE OF INTENT OF QUESTION. DRL PREVIOUSLY CONCLUDED THAT CONCEPT IS SOUND.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + REACTOR SAFETY SYSTEM + INSTRUMENT, CONTROL + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029483 ALSO IN CATEGORY 2 DATED 1968
RESPONSE TO QUESTION H-2 - FRICTION FORCES BETWEEN REFLECTORS AND EXTENSION SHAFTS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE H-3 AND -4 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE AND DISCUSS POSSIBILITIES OF HIGH FRICTIONAL FORCES BEING CREATED BETWEEN REFLECTORS AND/OR EXTENSION SHAFTS AND FIXED STRUCTURES DUE TO VARIATIONS IN COEFFICIENT OF FRICTION BETWEEN REFLECTOR PADS AND GUIDE RAILS AND MINIMUM RUNNING CLEARANCES AS A RESULT OF FABRICATION TOLERANCES, INSTALLATION TOLERANCES, AND DISPLACEMENTS DUE TO SEISMIC LOADINGS. (ANSWER) SEE SECT. 7 OF SUPPLEMENT 11 TO FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029483 *CONTINUED*

GUIDE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + EARTHQUAKE + REFLECTOR + RESPONSE TO NRC QUESTION + CORE COMPONENTS + STRUCTURAL INTEGRITY + ABLATION + DISPLACEMENT DESIGN

18-029484 ALSO IN CATEGORY 9 DATED 1968

RESPONSE TO QUESTION H-3 - MEASURING FRICTIONAL LOADS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE H-4 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THE FRICTIONAL FORCES RESISTING MOTION OF THE REFLECTORS CAN BE ESTABLISHED ONLY INDIRECTLY BY MEANS OF A CHANGE IN THE SCRAM TIME. DISCUSS SENSITIVITY OF THIS METHOD AND FEASIBILITY OF PROVIDING A MORE DIRECT MEANS OF MEASURING FRICTIONAL LOADS THROUGHOUT THE STROKE OF UNITS. (ANSWER) TEST PROGRAM INDICATED NO CHANGE FOR LIFETIME. MEASUREMENT CAN BE MADE BY UNBOLTING DRIVE UNIT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

GUIDE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + REFLECTOR + RESPONSE TO NRC QUESTION + CORE COMPONENTS + ABLATION + DISPLACEMENT DESIGN + CONTROL ROD DRIVE MECHANISMS

18-029485 ALSO IN CATEGORY 9 DATED 1968

RESPONSE TO QUESTION H-4 - ACCUMULATOR DRIVING FORCE DURING SCRAM

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE H-5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

ACCUMULATORS GIVE A RELATIVELY SMALL DOWNWARD FORCE TO OVERCOME POSSIBLE LARGE FRICTION FORCES OR TO ACCELERATE THE REFLECTOR. WHAT ARE THE CONSIDERATIONS THAT LED TO THIS LIMITED DRIVING FORCE ASSISTANCE AVAILABLE DURING A SCRAM. (ANSWER) THE 200 PSI ACCUMULATOR PRESSURE RESULTS IN 600 LB OF ADDED FORCE FOR INITIALLY DRIVING THE REFLECTOR CONTROL ELEMENT DOWN OR A SCRAM. ACCUMULATOR ACTS AS BACKUP TO LOW PRESSURE SYSTEM WHICH PROVIDES ADDITIONAL FORCE TO START ROD MOTION. FRICTION OF RODS OF 300 LB MEASURED, SO ACCUMULATOR COURSES INCREASE IN DOWNWARD FORCE FROM 1100 LB TO 1700 LB.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + REACTOR SAFETY SYSTEM + SCRAM, REAL + CONTROL ROD SCRAM MECHANISM + SEFOR (RE) + REFLECTOR + RESPONSE TO NRC QUESTION + ACCELERATION + ABLATION + CONTROL ROD DRIVE MECHANISMS + ACCUMULATORS

18-029486 ALSO IN CATEGORY 9 DATED 1968

RESPONSE TO QUESTION H-5 - OPERATING PRESSURES AND RANGES FOR ACCUMULATORS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, 1 FIGURE, PAGE H-6 AND -7 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT WAS THE BASIS FOR ESTABLISHING THE OPERATING PRESSURES AND THEIR RANGES FOR THE ACCUMULATORS. IT APPEARS THAT THE FUNCTION OF THE ACCUMULATORS COULD BE NULLIFIED IF THE OPERATING PRESSURES APPROACHED THE EXTREMES OF THE PRESENTLY ALLOWABLE RANGES. (ANSWER) PRESSURE NOW SET OF 200 PSI. CHANGE WILL BE MADE DURING FINAL CHECKOUT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + OPERATION + REPORT, SAR + CONTROL ROD SCRAM MECHANISM + SEFOR (RE) + REFLECTOR + RESPONSE TO NRC QUESTION + ACCUMULATORS

18-029487 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION I-1 - SEAL RING

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE I-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

SINCE ENTIRE WEIGHT OF SHROUD AND ITS INTERNALS IS TRANSFERRED TO REACTOR VESSEL WALL THROUGH SEAL RING, DISCUSS CAPABILITY OF TWELVE 3/4 UNC BOLTS TO PREVENT SHROUD AND CORE FROM DROPPING IN EVENT SEAL RING SHOULD FAIL. SURVEILLANCE AND QUALITY CONTROL PROGRAM THAT HAS BEEN OR WILL BE FOLLOWED TO ENSURE THAT RING WILL NOT FAIL AND MEANS WHEREBY IT WILL BE PROMPTLY DETECTED SHOULD IT FAIL. HOW RESIDUAL WELDING STRESSES IN SEAL RING WERE ESTABLISHED IN ORDER TO DETERMINE ITS MAXIMUM WORKING STRESS. (ANSWER) STRESS ON BOLT IS 6100 PSI. CODE ALLOWS 8500. SEAL FAILURE ALLOWS FLOW BYPASS CORE, WHICH IS EASILY DETECTED. RING WAS FORGED - NO WELDS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, COMPONENT + FABRICATION + REACTOR, LMCR + REPORT, SAR + STRESS + THERMAL MECHANICAL EFFECT + WELDING + FASTENER + SEFOR (RE) + SEAL + RESPONSE TO NRC QUESTION + CORE COMPONENTS + STRUCTURAL INTEGRITY + QUALITY ASSURANCE + SURVEILLANCE PROGRAM + SYSTEM CAPABILITY

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029488 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION 1-2 - OUTLET SLEEVE INTEGRITY
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 1 PAGE, PAGE 1-2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT DESIGN PROVISIONS HAVE BEEN MADE TO ENSURE THAT MOUNT FAILURE, AND SUBSEQUENT COOLANT BYPASS DUE TO OUTLET SLEEVE DISPLACEMENT, IS A REMOTE POSSIBILITY. DISCUSS CONSEQUENCES OF FAILURE OF SLEEVE MOUNTING. HOW WOULD SUCH A FAILURE BE DETECTED. (ANSWER) FAILURE REQUIRES SHEARING OF 3/4-IN.-OD PIN. NO LOADS ON PIN. FAILURE DETECTED BY SUDDEN INCREASE IN COOLANT FLOWS. CONSEQUENCES ARE FLOW THROUGH CORE REDUCED. MAX. TEMP. STILL SAFELY BELOW BOILING POINT OF SODIUM. SCRAM OCCURS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, COMPONENT + ACCIDENT, CONSEQUENCES + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + CORE COMPONENTS

18-029489 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION 1-3 - FASTENERS FOR VESSEL INTERNALS. BOLT LOADS
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 1 PAGE, PAGE 1-3 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

IT HAS BEEN NOTED THAT MANY REACTOR VESSEL INTERNALS ARE TO BE ASSEMBLED BY MEANS OF THREADED FASTENERS, MADE FROM SIMILAR MATERIALS AND WITHOUT LUBRICANTS. IT IS PRESUMED THAT STIPULATED TORQUE VALUES FOR VARIOUS FASTENERS IMPLIED A REQUIRED BOLT LOAD. DISCUSS ANTICIPATED VARIATION IN BOLT LOADS FOR GIVEN TORQUE WHEN THREADED MEMBERS ARE BOTH CLEAN, UNLUBRICATED STAINLESS STEEL TYPE 304. CITE EXPERIENCE UPON WHICH ABOVE INFORMATION IS BASED. (ANSWER) VARIATION IN BOLT LOAD WITHIN $\pm 25\%$ OF NOMINAL. LOAD VARIATION BASED ON GE TESTS. ALL FASTENERS LOCKED BY PINS, LOCKWIRE, AND WELDING, AND DO NOT DEPEND ON LOAD TO PREVENT UNFASTENING.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + STRESS + STEEL, STAINLESS + FASTENER + SEFOR (RE) + RESPONSE TO NRC QUESTION + CORE COMPONENTS + STRUCTURAL INTEGRITY + LUBRICATION

18-029490 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION 1-4 - CONSEQUENCES OF LOOSE PIECES IN PRIMARY COOLANT SYSTEM, PIPING INTEGRITY, ETC
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 3 PAGES, PAGE 1-3 THRU -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CONSIDERING LARGE NUMBER OF SMALL PIECES SPOT, PLUG, OR TACK WELDED IN PLACE WITHIN CORE ASSEMBLY AND LARGE NUMBER OF BOLTS, NUTS AND PINS, DISCUSS QUALITY ASSURANCE PROGRAM TO MINIMIZE LOSING SUCH PARTS. DISCUSS CONSEQUENCES ASSOCIATED WITH PRESENCE OF LOOSE PIECES IN PRIMARY COOLANT SYSTEM. IF LOOSE PIECES CAN SIGNIFICANTLY AFFECT CORE HEAT TRANSFER, PIPING OR VESSEL INTEGRITY, OR OPERABILITY OR DEVICES WHICH HAVE SAFETY-RELATED FUNCTION (E.G., CHECK VALVES), DISCUSS POSSIBLE NEED FOR DESIGN REMEDIAL MEASURES AND/OR DEVELOPMENT OR SURVEILLANCE MEASURES. (ANSWER) ALL FASTENERS SECURED BY LOCKWIRES OR WELDING. ALSO DISCUSSES PROBABILITY AND CONSEQUENCES OF LOOSE PIECES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FAILURE, COMPONENT + ACCIDENT, CONSEQUENCES + REACTOR, LMCR + REPORT, SAR + FASTENER + SEFOR (RE) + RESPONSE TO NRC QUESTION + CORE COMPONENTS + OBSTRUCTION + VALVES

18-029491 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION 1-5 - INCONEL INTEGRITY
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 4 PAGES, 2 FIGURES, PAGE 1-5 THRU -8 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

AVAILABLE TEST DATA ON PERFORMANCE OF INCONEL 718 IN ENVIRONMENT TYPICAL OF THAT EXPECTED IN SEFOR DO NOT APPEAR COMPLETE ENOUGH TO VERIFY THAT PARTS CONSTRUCTED OF THIS MATERIAL (AND ASSOCIATED WELDS) WILL MAINTAIN THEIR INTEGRITY FOR DESIGN LIFE OF CORE. DISCUSS MEASURES TO BE TAKEN TO ASCERTAIN CONTINUED SERVICEABILITY OF PARTS MADE OF INCONEL 718 DURING COURSE OF SEFOR TEST PROGRAM. (ANSWER) SEAL RING FAILURE WILL BE DETECTED BY INCREASED COOLANT FLOW. TORQUE RODS FOR CORE CLAMP TIGHTENED ONLY TO 1/4 OF 0.2% YIELD STRENGTH. MAX. STRESS FROM SEVERE TRANSIENT IS ONLY 1/3 OF YIELD STRENGTH.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

INTEGRITY + REACTOR, LMCR + REPORT, SAR + TENSILE PROPERTY + WELDING + EMBRITTLEMENT + FASTENER + SEFOR (RE) + STRESS STRAIN DATA + RESPONSE TO NRC QUESTION + INCONEL + CORE COMPONENTS + RADIATION EFFECT

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029442 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION I-6 - FAILURES OF AUXILIARY SODIUM INLET LINE CHECK VALVE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, PAGE I-9 THRU -11 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FOR THE AUXILIARY SODIUM INLET LINE CHECK VALVE, DISCUSS TYPE OF FAILURES CONSIDERED IN ITS
DESIGN, MEANS BY WHICH THESE FAILURES WILL BE DETECTED, AND CONSEQUENCES FOLLOWING EACH FAILURE.
BY WHAT MEANS AND HOW FREQUENTLY WILL OPERABILITY OF CHECK VALVE BE VERIFIED. (ANSWER ON PGS I-9
THRU -11.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + FASTENER + TEST, SYSTEM OPERABILITY + SEFOR (RE) + RESPONSE TO NRC QUESTION +
AUXILIARY COOLING SYSTEM + FAILURE MODE ANALYSIS + VALVES

18-029443 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION I-7 - MARMON COUPLINGS, INTEGRITY
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE I-11 AND -12 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

INLET AND OUTLET AUXILIARY SODIUM LINES, AND COOLANT OVERFLOW LINE INSIDE REACTOR VESSEL, UTILIZE
MARMON TYPE COUPLINGS. DISCUSS CONSIDERATIONS GIVEN TO POSSIBILITY OF COUPLINGS BECOMING LOOSE,
AND DESCRIBE THE CONSEQUENCES OF SUCH A FAILURE. BY WHAT MEANS CAN COUPLING FAILURE BE DETECTED.
(ANSWER ON PGS I-11 AND -12.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + FAILURE + REACTOR, LMCR + REPORT, SAR + FASTENER + SEFOR (RE) + FLANGE + SEAL +
RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM

18-029444 ALSO IN CATEGORY 12 DATED 1968
RESPONSE TO QUESTION J-1 - DIRECT RELEASE TO ATMOSPHERE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE J-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

THERE ARE SITUATIONS CITED IN THE FD-SAR THAT NORMALLY INERTED REGIONS (N AND/OR AR) MAY BE VENTED
DIRECTLY TO THE ATMOSPHERE, RATHER THAN BEING DIRECTED TO THE VOLUME LIMITED GAS STORAGE TANKS.
DISCUSS BASIS FOR THIS DIRECT RELEASE TO THE ATMOSPHERE, INCLUDING ACTIVITY LIMITS PLACED ON THIS
RELEASE AND METHOD BY WHICH ACTIVITY CONTENT IS DETERMINED. (ANSWER) SEE QUESTION M-11 OF THIS
SUPPLEMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

AIRBORNE RELEASE + REACTOR, LMCR + REPORT, SAR + SOURCE, CONTINUOUS + ARGON + NITROGEN + STACK + COVER GAS +
SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029495 ALSO IN CATEGORY 12 DATED 1968
RESPONSE TO QUESTION J-2 - COVER GAS ISOLATION VALVE SYSTEM
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE J-1 AND -2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS
IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS BASIS FOR SELECTING ISOLATION VALVE SYSTEM ASSOCIATED WITH COVER GAS IN SODIUM EXPANSION
TANK FOR SECONDARY COOLANT SYSTEM. (ANSWER) SECONDARY COOLANT SYSTEMS ARE CLOSED LOOPS, BUT
SINCE COVER GAS IS PROVIDED ON FEED AND BLEED BASIS IT IS NECESSARY TO ISOLATE GAS LINES TO CLOSE
OUTSIDE OF CONTAINMENT PART OF LOOP.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + CONTAINMENT ISOLATION + REPORT, SAR + COVER GAS + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN
COOLING SYSTEM + VALVES

18-029497 DATED 1968
RESPONSE TO QUESTION J-4 - OUTLINE OF FILL PROCEDURE FOR COOLANT SYSTEM
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE J-2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE AN OUTLINE OF YOUR FILL PROCEDURE WHICH DISCLOSES THE SEQUENCE, HEAT RATES, TEMPERATURES,
AND RELATED PARAMETERS FOR INITIAL FILLING OF PRIMARY AND SECONDARY SODIUM SYSTEMS. (ANSWER) NOT
YET PREPARED.

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029497 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + PROCEDURES AND MANUALS + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING
SYSTEM

18-029498

DATED 1968

RESPONSE TO QUESTION K-1 - OPERATION OF CRANES AND FUEL HANDLING GRAPPLE

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, 4 FIGURES, 7 TABLES, PAGE K-1 THRU -3 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION,
RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

SUPPLY ADDITIONAL INFORMATION ON OPERATION, INCLUDING DRAWINGS, OF MULTI-TON GRAPPLE FOR 10-TON
HOIST, FUEL ROD GRAPPLE, AND ONE-HALF-TON HOIST GRAPPLE. ALSO DISCUSS ABNORMAL OPERATING
CONDITIONS AND FAILURE MODES CONSIDERED IN DESIGN OF THESE SYSTEMS. (ANSWER ON PGS K-1 THRU -3
PLUS 7 TABLES, AND 3 FIGURES.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

FAILURE, EQUIPMENT + REACTOR, LMCR + OPERATION + REPORT, SAR + FUEL HANDLING MACHINE + SEFOR (RE) + SYSTEM
DESCRIPTION + RESPONSE TO NRC QUESTION

18-029499

DATED 1968

RESPONSE TO QUESTION K-2 - REFUELING OPERATIONS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

5 PAGES, 1 FIGURE, PAGE K-3 THRU -7 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

SUPPLY ADDITIONAL INFORMATION ON REFUELING OPERATION WHEN VERTICAL FUEL TRANSFER PORT, SPENT FUEL
SHIPPING CASK, AND REFUELING CELL EQUIPMENT ARE EMPLOYED. THE ABOVE SHOULD INCLUDE A SEQUENTIAL
LISTING OF ALL OPERATIONS THAT ARE TO BE CARRIED OUT AND DRAWINGS OF SUFFICIENT DETAIL OF PIECES
OF EQUIPMENT SUCH THAT IS POSSIBLE TO MORE FULLY EVALUATE POSSIBILITY OF AN INADVERTENT RELEASE
OF RADIOACTIVITY DURING REFUELING. ALSO, DISCUSS ABNORMAL OPERATING CONDITIONS AND FAILURE MODES
CONSIDERED IN THE DESIGN OF THESE SYSTEMS. (ANSWER ON PGS K-3 THRU -7.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + OPERATION + PROCEDURES AND MANUALS + REPORT, SAR + SHIPPING CONTAINER + REFUELING + SEFOR (RE)
+ RESPONSE TO NRC QUESTION

18-029500

DATED 1968

RESPONSE TO QUESTION K-3 - KEEPING SODIUM LEVEL ABOVE IRRADIATED FUEL STORAGE TANK

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

4 PAGES, 1 FIGURE, PAGE K-7 THRU -9 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS MEANS FOR PREVENTING THE SODIUM LEVEL FROM BEING INADVERTENTLY LOWERED IN THE IRRADIATED
FUEL STORAGE TANK, ABILITY TO DETECT LOW SODIUM LEVELS, CONSEQUENCES OF LOW SODIUM LEVELS, AND
MEANS WHEREBY LEVEL CAN BE RETURNED TO NORMAL. (ANSWER) CHANGES MADE IN DRAINING TANK SINCE
FDSAR WRITTEN. DIP LINE WILL NOT BE INSTALLED, INSTEAD DRAIN LINE IN NOZZLE K-4 WHICH EXTENDS
DOWN INTO TANK TO ELEVATION 2 1/4 INCHES ABOVE TOP OF STORED FUEL RODS. NO OTHER NOZZLES OR DIP
LINES ON TANKS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SODIUM + FUEL STORAGE + STORAGE CONTAINER + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029501

ALSO IN CATEGORY 6

DATED 1968

RESPONSE TO QUESTION L-1 - REACTIVITY OF INSTRUMENTED FUEL ASSEMBLY

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

1 PAGE, PAGE L-1 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT IS REACTIVITY OF AN INSTRUMENTED FUEL ASSEMBLY (IFA) AS COMPARED TO A SINGLE FUEL ELEMENT.
DISCUSS NORMAL AND SPECIAL CONDITIONS UNDER WHICH AN IFA WOULD BE LOADED INTO OR REMOVED FROM
CORE AT SEFOR. CONSIDER IN THIS DISCUSSION DESIGN CRITERION WHICH LIMITS REACTIVITY WORTH OF ANY
ITEM BEING ROUTINELY HANDLED NEAR CORE TO LESS THAN 1%. (ANSWER) SEE SECT. 16.2.4.3 OF FDSAR.
NO PLANS TO HANDLE IFA ROUTINELY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + FUEL HANDLING + REACTOR, LMCR + REACTIVITY EFFECT + REPORT, SAR + INSTRUMENT, COMPONENT +
SEFOR (RE) + RESPONSE TO NRC QUESTION + FUEL ELEMENTS

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029502 DATED 1968
RESPONSE TO QUESTION L-2 - NATURAL CIRCULATION TEST PROGRAM
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE L-1 AND -2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DISCUSS IN DETAIL THE NATURAL CIRCULATION TEST PROGRAM IN LIGHT OF SYSTEM DEPENDENCE ON NATURAL CIRCULATION CAPABILITY OF REACTOR COOLANT SYSTEM FOR ABNORMAL OPERATION CONDITIONS. (ANSWER) CAPABILITIES WILL BE VERIFIED BUT REQUIRES NUCLEAR HEATING OF CORE. WILL PERFORM AFTER 5-MW OPERATION ESTABLISHED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FLOW THEORY AND EXPERIMENTS + REACTOR, LMCR + REPORT, SAR + COOLING, SHUTDOWN + TEST, SYSTEM OPERABILITY + SEFOR (RE) + HEAT TRANSFER, CONVECTION + RESPONSE TO NRC QUESTION + EMERGENCY COOLING + SYSTEM CAPABILITY + FLOW, CONVECTION

18-029503 ALSO IN CATEGORY 5 DATED 1968
RESPONSE TO QUESTION L-3 - COOLANT FLOW TIME CONSTANTS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE L-2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

INDICATE ON A SIMPLIFIED FLOW DIAGRAM OF THE NA COOLANT SYSTEM THE TIME DELAYS THROUGH THE PRIMARY AND SECONDARY SYSTEMS AS WELL AS SIGNIFICANT TIME CONSTANTS (E.G., FUEL CLAD, GAP, PELLETS, IHX TUBES, ETC.) FOR FORCED AND NATURAL CONVECTION FLOW RATES. (ANSWER) SEE APPENDIX A OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + COOLING, SHUTDOWN + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + FLOW + RATE + FLOW, CONVECTION

18-029504 ALSO IN CATEGORY 5 DATED 1968
RESPONSE TO QUESTION L-4 - AVOIDING FUEL MELTING
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE L-2 AND -3 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CLARIFY STATEMENT ON TOP OF PAGE (16-5) REGARDING AVOIDING FUEL MELTING. DEFINE TEMPERATURE IN TRANSIENT RESPONSE FIGURES AT END OF CHAPTER XVI. WHAT IS SIGNIFICANCE OF THERMAL TEMPERATURE WAVE INDICATED IN FIGURES XVI-5 AND XVI-7, TO REST OF THE SYSTEM (ALSO, REFERENCE P.16-6). (ANSWER) NO MELTING CRITERIA GIVEN IN SUPPL. 3 OF FDSAR. THERMAL TRANSIENTS DISCUSSED IN SECT. 16 OF FDSAR, ACCIDENT TRANSIENTS IN APPENDIX A OF SUPPLEMENT 16 TO FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FUEL MELTDOWN + REACTOR, LMCR + REACTOR TRANSIENT + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + SAFETY ANALYSIS + THERMAL TRANSIENT

18-029505 ALSO IN CATEGORY 5 DATED 1968
RESPONSE TO QUESTION L-5 - LOSS-OF-PUMPING-POWER ACCIDENT
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, PAGE L-3 AND -4 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

TABLE XVI-13 (PAGE 16-14 PRESENTS ELEVEN LOSS OF PUMPING POWER ACCIDENTS. CASES 2, 3, 4, AND 6 RESULT IN MAXIMUM COOLANT TEMPERATURES IN EXCESS OF (SYSTEM DESIGN TEMPERATURE) 1000 F. STATEMENTS BELOW TABLE IMPLY CVER-DESIGN COOLANT TEMPERATURES PRODUCE NO DETRIMENTAL EFFECTS ON SYSTEM. DETAIL RATIONALE BEHIND THESE FINDINGS, INCLUDING CONSIDERATIONS OF FUEL TEMPERATURE AND THERMAL STRESSES IN CRITICAL LOCATIONS IN PRIMARY SYSTEM AND SECONDARY SODIUM COOLANT SYSTEMS. ALSO PROVIDE TIME-TEMPERATURE GRAPH OF THESE FOUR CASES - CARRYING EACH IN TIME UNTIL NORMAL CONDITIONS ARE RE-ESTABLISHED. DEFINE THE MEANING OF MAXIMUM COOLANT TEMPERATURE AND INDICATE CORRESPONDING MAXIMUM HOT CHANNEL-HOT SPOT TEMPERATURES. *** (ANSWER ON PAGES L-3 AND -4.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + REPORT, SAR + STRESS + THERMAL MECHANICAL EFFECT + HOT SPOT + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + HOT CHANNEL + THERMAL TRANSIENT

18-029506 ALSO IN CATEGORIES 5 AND 6 DATED 1968
RESPONSE TO QUESTION L-6 - ENERGY DENSITIES OF FUEL IN TRANSIENT TESTS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, 1 TABLE, PAGE S-4 AND -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029506 *CONTINUED*

WHAT ARE EXPECTED ENERGY DENSITIES IN CAL/GM OF FUEL IN PLANNED SUB-PROMPT AND PROMPT CRITICAL TRANSIENT TESTS. (ANSWER) TABLE OF ENERGY DENSITIES GIVEN BASED ON TRANSIENT ANALYSIS DESCRIBED IN SECTION SII OF FDSAR AND SUPPLEMENT 10.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REACTOR TRANSIENT + REPORT, SAR + THERMAL CONSIDERATION + TEST, PHYSICS + SEFOR (RE) + HEAT TRANSFER EXPERIMENT + RESPONSE TO NRC QUESTION + FUEL ELEMENTS

18-029507 ALSO IN CATEGORY 5 DATED 1968

RESPONSE TO QUESTION L-7 - CLAD FAILURE

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE L-5 AND -6 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

WHAT IS CLADDING FAILURE THRESHOLD AND UNDER WHAT CONDITIONS WILL CLADDING FAIL. (ANSWER) SEE SUPPLS. 3 AND 10 OF FDSAR. ONLY OBSERVED EFFECT OF CYCLIC OPERATION THROUGH FUEL MELTING POINT WAS SOME LOCAL CLAD EXPANSION. ENERGY DENSITY OF 400 CAL/GRAM NEEDED FOR CLAD TO FAIL FROM INTERNAL PRESSURE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CLADDING + INTEGRITY + REACTOR, LMCR + REPORT, SAR + FAILURE, CLADDING + SEFOR (RE) + PRESSURE, INTERNAL + RESPONSE TO NRC QUESTION

18-029508 ALSO IN CATEGORY 5 DATED 1968

RESPONSE TO QUESTION L-8 - ACCIDENT ANALYSES FOR LOSS OF EXTERNAL HEAT SINKS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

3 PAGES, PAGE L-6 THRU -8 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

DESCRIBE BASES, TIME HISTORY, AND RESULTS OF ACCIDENT ANALYSES FOR PARTIAL AND FINALLY COMPLETE LOSS OF EXTERNAL HEAT SINKS AS CLASS I COMPONENTS AND DESIGN MARGINS ARE AFFECTED. INCLUDE - (A) LOSS OF THREE INDEPENDENT FAN-COOLERS WHICH PROVIDE HEAT SINKS FOR INNER CONTAINMENT ATMOSPHERE. (B) LOSS OF SECONDARY SODIUM COOLANT FROM MAIN COOLANT LOOP. (C) COMBINATION OF (A) AND (B.3). (D) LOSS OF DIESEL GENERATOR AND OFF-SITE POWER. (E) COMBINATION OF (A) AND (D). (F) COMBINATION OF (B.3) AND (D). (G) COMBINATION OF (A), (B.3) AND (D). (ANSWER ON PGS L-6 THRU - 8.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT ANALYSIS + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + REPORT, SAR + HEAT SINK + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + AUXILIARY COOLING SYSTEM

18-029509 ALSO IN CATEGORY 5 DATED 1968

RESPONSE TO QUESTION L-9 - CORE BOUNDARY PROBLEMS IN BETHE-TAIT ANALYSIS OF MHA

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE L-8 AND -9 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

HOW ARE THE CORE BOUNDARY PROBLEMS TREATED IN THE BETHE-TAIT ANALYSIS OF THE MHA. ALSO, WHEN THE FRED IS PLACED IN THE CORE, HOW IS ITS BOUNDARY CONSIDERED IN THE FUEL MELTDOWN AND BETHE-TAIT ANALYSES. (ANSWER ON PGS L-8 AND -9.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

BETHE TAIT + FUEL MELTDOWN + REACTOR, LMCR + REPORT, SAR + ACCIDENT, HYPOTHETICAL + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029510 DATED 1968

RESPONSE TO QUESTION M-1 - FISSION PRODUCTS QUANTITY IN RADWASTE

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

2 PAGES, PAGE M-1 AND -2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE THE MAXIMUM QUANTITIES OF THE VARIOUS RADIOACTIVE FISSION PRODUCTS THAT COULD BE PRESENT IN THE LIQUID AND GASEOUS RADIOACTIVE SYSTEM. ANALYZE THE CONSEQUENCES OF THE RELEASE OF THE MAXIMUM CONTENTS OF THE GASEOUS DECAY TANK TO THE ATMOSPHERE. JUSTIFY ALL ASSUMPTIONS. (ANSWER ON PGS M-1 AND -2.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

ACTIVITY BUILDUP + FISSION PRODUCT RELEASE + REACTOR, LMCR + REPORT, SAR + WASTE SOURCE AND TYPE + WASTE

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029510 *CONTINUED*
MANAGEMENT + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029511 ALSO IN CATEGORIES 2 AND 16 DATED 1968
RESPONSE TO QUESTION M-2 - ON-SITE METEOROLOGICAL DATA
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE M-2 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC
REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE ON-SITE METEOROLOGICAL DATA DESCRIBED IN TABLE III-5 OF THE FDSAR. (ANSWER) REPORT IN
FINAL STAGES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + SITE CLIMATOLOGY + METEOROLOGY + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029512 ALSO IN CATEGORY 2 DATED 1968
RESPONSE TO QUESTION M-3 - LIQUID WASTE STORAGE FIELD
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
4 PAGES, 1 FIGURE, 1 TABLE, PAGE M-2 THRU -5 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION,
RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE ELEVATION OF THE FIELD INTO WHICH LIQUID RADIOACTIVE WASTES WILL BE DISCHARGED. DURING A
PEAK PRECIPITATION PERIOD ASSOCIATED WITH PEAK RUNOFF CONDITIONS (MAXIMUM PROBABLE FLOODING
CONDITIONS) PROVIDE LEVEL TO WHICH FALL CREEK WOULD RISE AND DISCUSS POTENTIAL FOR FLOODING OF
THE FIELD CARRYING ITS ACCUMULATED RADIOACTIVE FISSION PRODUCTS. (ANSWER ON PGS M-2 THRU -5.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + WASTE STORAGE + WASTE DISPOSAL, LIQUID + SEFOR (RE) + RESPONSE TO NRC QUESTION +
FLOOD

18-029513 DATED 1968
RESPONSE TO QUESTION M-4 - PLUTONIUM ISOTOPES IN REACTOR CORE VS OPERATING TIME
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, 1 TABLE, PAGE M-5 AND -6 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE THE QUANTITIES OF THE PLUTONIUM ISOTOPES PRESENT IN THE REACTOR CORE VS REACTOR OPERATING
TIME (INCLUDING 238, 239, 240, 241 AND 242). (REFERENCE PAGE 16-32, 16-4.3.3.1). (ANSWER) TABLE
GIVEN. ALSO SEE ATTACHMENT 3 OF SUPPLEMENTARY INFORMATION TO THE ATOMIC SAFETY AND LICENSING
BOARD, JULY 17, 1965.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

ACTIVITY BUILDUP + REACTOR, LMCR + PLUTONIUM + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029514 ALSO IN CATEGORY 12 DATED 1968
RESPONSE TO QUESTION M-5 - COMBINATION FILTER FOR GASEOUS WASTE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
15 PAGES, 1 FIGURE, PAGE M-6 THRU -9 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO
QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE A DESCRIPTION OF THE COMBINATION FILTER THROUGH WHICH GASEOUS WASTES ARE DISCHARGED TO THE
ATMOSPHERE. ALSO, PROVIDE THE DESIGN SPECIFICATIONS FOR THESE FILTERS, THE FILTER TESTING
PROCEDURES, AND THE PROPOSED FREQUENCY OF TESTING. (ANSWER ON PGS M-6 THRU -9 PLUS 1 FIGURE AND
6 PAGES OF FILTER SPECIFICATIONS.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

AIRBORNE RELEASE + DESIGN CRITERIA + REACTOR, LMCR + REPORT, SAR + TEST, FILTER + WASTE DISPOSAL, GAS + FILTER
CHARACTERISTICS + FILTER, COMBINATION + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION

18-029515 DATED 1968
RESPONSE TO QUESTION M-6 - LOCATION OF MONITORING STATIONS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
4 PAGES, 1 FIGURE, 1 TABLE, PAGE M-9 THRU -12 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION,
RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE A MAP WITH THE LOCATIONS OF RADIATION ENVIRONMENTAL MONITORING STATIONS. (ANSWER ON PGS M-
9 THRU -12.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029515 *CONTINUED*
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + MONITOR, ENVIRONMENTAL + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029516 ALSO IN CATEGORY 7 DATED 1968
RESPONSE TO QUESTION M-7 - FISSION PRODUCT RELEASE ASSUMPTIONS FOR DESIGN OF CONTROL ROOM SHIELDING
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
9 PAGES, 1 FIGURE, 1 TABLE, PAGE M-12 THRU -20 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE FISSION PRODUCT RELEASE ASSUMPTIONS USED IN THE DESIGN OF THE SHIELDING OF THE CONTROL ROOM. PROVIDE DETAILS OF THE VENTILATION SYSTEM FOR THE CONTROL ROOM, INCLUDING ITS MODE OF OPERATION DURING ACCIDENT CONDITIONS. PROVIDE DOSE THAT WOULD BE RECEIVED BY OPERATORS DURING COURSE OF DBA. (ANSWER ON PAGES M-12 THRU -20.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DESIGN CRITERIA + DOSE + FISSION PRODUCT RELEASE + REACTOR, LMCR + REPORT, SAR + VENTILATION SYSTEM + ACCIDENT MODEL + PERSONNEL EXPOSURE, RADIATION + SEFOR (RE) + SHIELDING + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + SYSTEM OPERABILITY IN ACCIDENT + CONTROL PANEL/ROOM

18-029517 ALSO IN CATEGORY 1 DATED 1968
RESPONSE TO QUESTION M-8 - OFF-SITE EMERGENCY PLANS, DETAILS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
3 PAGES, PAGE M-21 THRU -23 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

PROVIDE DETAILS OF ANTICIPATED OFF-SITE EMERGENCY PLANS. (ANSWER IS A VERY BRIEF DISCUSSION OF ON-SITE ACTIONS.)

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + RADIATION PROTECTION, ORGANIZATION + EMERGENCY PROCEDURE + RADIATION SAFETY AND CONTROL + SEFOR (RE) + RESPONSE TO NRC QUESTION + OFF SITE

18-029518 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION M-9 - LEAK RATE VS TIME FOR CONTAINMENT BARRIERS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, 1 FIGURE, PAGE M-23 AND -24 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

FOR THE CALCULATIONAL BASIS FOR RADIOLOGICAL EFFECTS, PROVIDE THE LEAK RATE VERSUS TIME FOR EACH OF THE CONTAINMENT BARRIERS THAT WERE USED FOR THE DOSE CALCULATIONS PRESENTED IN TABLE D-8. (ANSWER ON PGS M-23 AND -24).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CONTAINMENT ANALYSIS + DOSE + FISSION PRODUCT RELEASE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + LEAK RATE + OFF SITE

18-029519 DATED 1968
RESPONSE TO QUESTION M-10 - DOSE AT NEAREST OFF-SITE AND LOW POPULATION LOCATIONS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
2 PAGES, 3 REFERENCES, PAGE M-25 AND -26 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

EVALUATE DOSES AT NEAREST OFF-SITE LOCATION AND LOW POPULATION ZONE. TABULATE AVERAGE HOURLY DOSE OVER ACCIDENT PERIOD. MAXIMUM TWO-HOUR DOSE AND INFINITE DOSE BASED ON STATIONARY POINT ON SITE AND AT LOW POPULATION BOUNDARIES. (ANSWER) SEE TABLE D-8, APPENDIX D, OF FDSAR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + DOSE + REACTOR, LMCR + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + OFF SITE

18-029520 DATED 1968
RESPONSE TO QUESTION M-11 - DIRECT VENTING TO STACK OF INNER CONTAINMENT NITROGEN AND ARGON
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
1 PAGE, PAGE M-27 OF SUPPLEMENT 17 (AMENDMENT 16) TO SEFOR LICENSE APPLICATION, RESPONSE TO QUESTIONS IN AEC REQUEST OF MAY 3, 1968, DOCKET 50-231, MAY 24, 1968

UNDER WHAT CIRCUMSTANCES WILL THE INNER CONTAINMENT NITROGEN AND ARGON ATMOSPHERES BE VENTED DIRECTLY TO THE STACK. HOW WILL THE RADIOACTIVITY LEVELS IN THOSE ATMOSPHERES BE DETERMINED

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029520 *CONTINUED*

PRIOR TO VENTING, AND WHAT ARE THE CRITERIA FOR MAXIMUM ACTIVITY LEVELS ABOVE WHICH DIRECT VENTING TO THE ENVIRONS WILL NOT BE ALLOWED. (ANSWER) FREQUENCY FOR DIRECT VENTING WILL BE LOW. PURGE OF REFUELING CELLS WILL BE MONITORED AT STACK.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, LMCR + REPORT, SAR + STACK + COVER GAS + WASTE DISPOSAL, GAS + SEFOR (RE) + RESPONSE TO NRC QUESTION

18-029606

ALSO IN CATEGORY 5

DATED 1968

PETERS NT

DEVELOPMENT OF THE PRIMARY SYSTEM FOR THE ENRICO FERMI ATOMIC POWER PLANT
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

APDA-308 +. 173 PAGES, JUNE 1968

DOCUMENTS THE COURSE OF DEVELOPMENT OF THE PRIMARY HEAT TRANSPORT SYSTEM, STARTING WITH THE EARLY HISTORY OF THE OBJECTIVES AND CONTINUING THROUGH THE EVOLUTION OF THE SYSTEM DESIGN CONCEPTS TO THE FINAL DESIGN. PRESENTS CONSTRUCTION, TESTING, AND OPERATION OF SYSTEM WITH REACTOR POWER UP TO 100 MWT WITH EVALUATION OF THE GENERAL PERFORMANCE AND IMPORTANT FEATURES, INCLUDING RECOMMENDATIONS FOR USE IN DESIGN OF NEW LMFBRS. PRESENTS INTERACTION WITH DESIGN OF OTHER FEATURES OF PLANT (SUCH AS REACTOR CONTAINMENT BUILDING). SYSTEM COMPONENTS AND AUXILIARY FEATURES ARE DESCRIBED ONLY AS RELEVANT TO THE DISCUSSION OF PRIMARY SYSTEM AS AN INTEGRATED, FUNCTIONAL UNIT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + FERMI (LMFBR) + SYSTEM DESCRIPTION + MAIN COOLING SYSTEM + R AND D PROGRAM + REACTOR, LMFBR

18-029722

ALSO IN CATEGORIES 1 AND 5

DATED 1967

BAKER L + EGGEN DT + IVINS RD + SZAWLEWICZ SA

US LMFBR SAFETY PROGRAM

ARGONNE NATIONAL LABORATORY + US ATOMIC ENERGY COMMISSION

25 PAGES, 2 FIGURES, 2 TABLES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPTEMBER 1967

DESCRIBES OBJECTIVES AND STATUS OF PLANNING EFFORT FOR SAFETY ANALYSES AND EVALUATIONS IN THREE TECHNICAL AREAS - (1) ANALYSIS OF ACCIDENTS AND ENERGY RELEASE, (2) PREVENTION AND MITIGATION OF THE CONSEQUENCES OF ACCIDENTS, AND (3) ESTABLISHMENT OF SAFETY-RELATED CRITERIA, CODES AND STANDARDS. BASED ON A TABULATION OF POSSIBLE KNOWN OR POTENTIAL OFF-NORMAL OPERATIONS, INSTABILITIES, AND MALFUNCTIONS, DISCIPLINE-ORIENTED REQUIREMENTS WERE DEVELOPED TO SPECIFY THE RESEARCH AND DEVELOPMENT EFFORT. A BRIEF INDICATION OF THE OBJECTIVES AND CURRENT EFFORTS PROCEEDING UNDER EACH TASK ARE GIVEN.

AVAILABILITY - L. BAKER, JR., ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ACCIDENT ANALYSIS + CODES AND STANDARDS + SAFETY PROGRAM + R AND D PROGRAM + ENGINEERED SAFETY FEATURE + REACTOR, LMFBR

18-029812

DATED 1968

JOHNSON HG + GUNBY AL

FFTF PRIMARY PUMP LOCATION

BATTELLE NORTHWEST, RICHLAND, WASHINGTON

BNWL-729 +. 46 PAGES, 13 FIGURES, 7 TABLES, 8 REFERENCES, MAY 1968

A STUDY WAS COMPLETED OF THE EFFECTS OF LOCATING FFTF PRIMARY PUMPS IN EITHER THE HOT OR COLD LEG. MAIN CONSIDERATIONS WERE (1) SYSTEM COSTS, (2) SYSTEM THERMAL TRANSIENTS, (3) PUMP RELIABILITY, AND (4) PLANT SAFETY. IT WAS CONCLUDED THAT (1) HOT AND COLD LEG PUMP LOCATIONS ARE BOTH PRACTICAL, (2) PLANT CAPITAL COSTS ARE ABOUT \$1 MILLION LESS FOR COLD LEG, ASSUMING EQUAL EMERGENCY COOLING COSTS, (3) EXPERIENCE AT OTHER PLANTS INDICATES A HIGH PROBABILITY OF SUCCESS FOR PUMP OPERATION AT FFTF COLD LEG TEMPERATURES, AND (4) SAFETY CONSIDERATIONS AND THERMAL TRANSIENTS DO NOT LEAD TO A PREFERENCE OF PUMP LOCATION AT THIS TIME. STUDY SUPPORTS THE COLD LEG LOCATION AT THIS TIME, BUT A FIRM DECISION MUST REST ON WHETHER THE EMERGENCY COOLING CONCEPT BECOMES A FACTOR.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + MAIN COOLING SYSTEM + PUMPS

18-029328

DATED 1968

HEBEISEN JC + PRICE DE + MOAK DP

NITRIDE FUEL-CLADDING COMPATIBILITY

BATTELLE MEMORIAL INSTITUTE

1 PAGE, P. 104 OF ANS TRANSACTIONS, 11(1), 1968 ANNUAL MEETING OF THE AMERICAN NUCLEAR SOCIETY AND THE CANADIAN NUCLEAR ASSOCIATION, TORONTO, JUNE 10-13, 1968

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-029828 *CONTINUED*

DISCUSSES STUDIES OF COMPATIBILITY OF NITRIDE FUELS IN CONTACT WITH AND SODIUM-BONDED TO POTENTIAL CLAD MATERIALS. 304 SS HAS SHOWN BEST COMPATIBILITY OF MATERIALS STUDIED. CHEMICAL PURITY AND STOICHIOMETRY OF FUELS HAS MARKED EFFECT ON COMPATIBILITY. HYPERSTOICHIOMETRIC (U,Pu)N HAS EXCELLENT COMPATIBILITY, WHILE REACTIONS OCCUR WITH HYPOSTOICHIOMETRIC (U,Pu)N.

CHEMICAL REACTION + CLADDING + FUEL INTEGRITY + PLUTONIUM + SODIUM + URANIUM + NITRIDE + STOICHIOMETRY + REACTOR, LMFBR + FUEL ELEMENTS

18-029865 ALSO IN CATEGORIES 1 AND 12 DATED 1968
LMFBR PROGRAM PLAN. VOL. 3 - COMPONENTS
ARGONNE NATIONAL LAB., ILL.
WASH-1103 +. 437 PAGES, FIGURES, TABLES, AUGUST 1968

TWELVE TASK AREAS INCLUDE (1) COMPONENT REQUIREMENTS, (2) DEVELOPMENT OF QUALITY AND RELIABILITY ASSURANCE PROGRAMS, (3) COMPONENT MATERIALS REQUIREMENTS, (4) DEVELOPMENT OF FABRICATION AND NDT PROCESSES, (5) FACILITIES FOR COMPONENT DEVELOPMENT, SPECIFIC COMPONENT DEVELOPMENT PROGRAMS ARE (6) STEAM GENERATOR, (7) CORE HANDLING EQUIPMENT (8) COOLANT TRANSPORT AND STORAGE SYSTEMS, (9) REACTOR VESSEL AND RELATED COMPONENTS, (10) REACTIVITY CONTROL DRIVES, (11) LIQUID METAL PUMPS, (12) HEAT EXCHANGERS. ***EACH SECTION BRIEFLY STATES THE GOALS, INTERACTION WITH OTHER PROJECTS, CURRENT STUDIES, AND GIVES A SCHEDULE OR FLOW SHEET FOR THE EFFORT.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

FABRICATION + PRESSURE VESSELS + RELIABILITY, COMPONENT + TEST, NONDESTRUCTIVE + FUEL HANDLING MACHINE + PROPERTY, PHYSICAL + QUALITY ASSURANCE + MAIN COOLING SYSTEM + R AND D PROGRAM + STEAM GENERATOR + REACTOR, LMFBR + PUMPS + PIPES AND PIPE FITTINGS + JACOBS + HEAT EXCHANGERS + CONTROL ROD DRIVE MECHANISMS

18-030209 DATED 1968
SEFOR AMENDMENT 25
GENERAL ELECTRIC COMPANY
36 PAGES, FIGURES, 19 REFERENCES, AMENDMENT 25 TO SEFOR PSAR, DEC. 10, 1968, DOCKET 50-231

CONSISTS OF SUPPLEMENT 21 (RESPONSE TO INFORMAL DRL QUESTIONS OF OCT. 18, 1968) AND ERRATA 4 (REVISED PAGES FOR PSAR AND SUPPLEMENTS).

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

NRC QUESTION + REACTOR, FAST + REPORT, SAR + REACTOR, PULSED + SEFOR (RE) + REACTOR, LMFBR

18-030210 DATED 1968
RESPONSE TO QUESTION 1, FUEL PERFORMANCE
GENERAL ELECTRIC COMPANY
11 PAGES, 1 FIGURE, PAGES 1-10 OF SUPPLEMENT 21 TO SEFOR PSAR, DEC. 10, 1968, DOCKET 50-231

FUEL PERFORMANCE UNDER STEADY STATE AND TRANSIENT OPERATION DISCUSSED. ***20 MWTH POWER DENSITY IS 1/10TH THAT SUCCESSFULLY TESTED. LOCAL SWELLING OF FUEL (OF 10 MILS) WOULD NOT AFFECT COOLING, AND 20 MILS WILL NOT CAUSE FAILURE. REACTION OF SODIUM WITH FUEL WAS TESTED DURING TREAT TRANSIENT, WITH NO CLAD DAMAGE OBSERVED. IF UPPER 1/3 OF A FUEL BUNDLE SLUMPS INTO CENTER 1/3 DURING A COOLANT FLOW BLOCKAGE AND MELTDOWN, REACTIVITY ADDED WOULD ONLY BE 0.65.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

CHEMICAL REACTION + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR TRANSIENT + REPORT, SAR + SODIUM + REACTIVITY EFFECT, SLUMP + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + FLOW BLOCKAGE + FAILURE, FUEL ELEMENT WATERLOGGING + REACTOR, LMFBR

18-030252 ALSO IN CATEGORY 6 DATED 1968
RESPONSE TO QUESTION 3 - CRITICALITY ACCIDENTS
NORTHERN STATES POWER COMPANY
2 PAGES, PP. 18-20 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

PROTECTION AGAINST CRITICALITY ACCIDENTS DURING STARTUP OR PLANT SHUTDOWN DESCRIBED. *** (A) PERIOD TRIPS BACKED UP BY POWER TRIPS, ARE PROVIDED FOR POWER LEVELS ABOVE 100 WATTS. SHORT PERIODS CANNOT OCCUR AT POWER BECAUSE OF THE DOPPLER EFFECT. (B) WHENEVER CONTAINMENT OR REACTOR COOLING IS INOPERABLE, CONTROL ROD DRIVE POWER SUPPLY BREAKERS WILL BE OPENED. 5 OF THE 10 REFLECTOR SEGMENTS WOULD HAVE TO BE RAISED FOR CRITICALITY, DUE TO THE SHUTDOWN MARGIN.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

ACCIDENT, CONTROL ROD WITHDRAWAL + REPORT, SAR + SHUTDOWN MARGIN + ACCIDENT, CRITICALITY + SEFOR (RE) + RESPONSE TO NRC QUESTION + INSTRUMENT, INTERMEDIATE RANGE + REACTOR, LMFBR

18-030253 ALSO IN CATEGORIES 5 AND 7 DATED 1968

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-030253 *CONTINUED*

RESPONSE TO QUESTION 5 - SOURCE TERMS IN RELEASE CALCULATION

NORTHERN STATES POWER COMPANY

11 PAGES, PP. 23-33 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

FISSION PRODUCT SOURCE TERMS USED IN ACCIDENT CALCULATIONS AND OPERATIONAL CONSIDERATIONS IS DISCUSSED. ***THREE ACCIDENTS (CORE DISASSEMBLY IN MELTDOWN, SECONDARY LOOP RUPTURE, GAS RADWASTE TANK RUPTURE) WERE DISCUSSED IN SAR. ESSENTIAL FEATURES OF THESE ANALYSES ARE REPEATED HERE. (A) 2/3 OF FUEL MELTS BUT 99% OF IODINE REACTS WITH SODIUM, (B) ALTHOUGH SOME PRIMARY SODIUM MIGHT LEAK THROUGH TO THE SECONDARY SYSTEM, IT WOULD SPILL IN LOW OXYGEN AREAS. ONLY 0.0187 CU FT OF PRIMARY SODIUM WAS ESTIMATED RELEASED. (C) ASSUMED ONE TANK RELEASES 15% OF GASEOUS FP FROM CENTER FUEL ELEMENTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

FISSION GAS RELEASE + FAILURE, FUEL ELEMENT + FUEL MELTDOWN + REPORT, SAR + WASTE STORAGE + ACCIDENT, HYPOTHETICAL + WASTE TREATMENT, GAS + SEFOR (RE) + RESPONSE TO NRC QUESTION + REACTOR, LMFBR

18-030254 ALSO IN CATEGORY 9

DATED 1968

RESPONSE TO QUESTION 2 - FUEL FAILURE DETECTION

NORTHERN STATES POWER COMPANY

7 PAGES, PP. 11-17 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10 1968, DOCKET 50-231

FUEL FAILURE DETECTION DISCUSSED. ***FOLLOWING EACH TRANSIENT TWO FUEL RODS WILL BE EXAMINED AND COVER GAS SAMPLES ANALYZED BEFORE/AFTER. THREE TYPES OF CONTINUOUS MONITORS DISCUSSED (DELAYED NEUTRON, CHARGED WIRE PRECIPITATOR, GROSS GAMMA OF COVER GAS). THE THIRD WAS CHOSEN AND WILL BE INSTALLED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REPORT, SAR + INSTRUMENT, DETECTION FAILED FUEL ELEMENT + SEFOR (RE) + RESPONSE TO NRC QUESTION + REACTOR, LMFBR

18-030255 ALSO IN CATEGORIES 5 AND 10 AND 12 DATED 1968

RESPONSE TO QUESTION 4 - EMERGENCY COOLING

NORTHERN STATES POWER COMPANY

2 PAGES, PP. 21-22 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

EMERGENCY COOLING IS THAT FOLLOWING LOSS OF SECONDARY HEAT SINKS AND TOTAL LOSS OF SITE POWER. ***REFERS TO SUPPLEMENT 19, QUESTION 2, WHERE SAME QUESTION ANSWERED. A 50-KW AUXILIARY POWER UNIT HAS BEEN PROVIDED FOR AUX. PC PUMP, AND ALSO POWERS A GAS BLOWER TO COOL THIS PUMPS WINDINGS. LATTER GAS BLOWER SYSTEM IS DESCRIBED HERE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

EMERGENCY POWER, ELECTRIC + ACCIDENT, LOSS OF POWER + REPORT, SAR + SEFOR (RE) + RESPONSE TO NRC QUESTION + MAIN COOLING SYSTEM + AUXILIARY COOLING SYSTEM + REACTOR, LMFBR + PUMPS

18-030256 ALSO IN CATEGORY 3

DATED 1968

RESPONSE TO QUESTION 6 - SPENT FUEL STORAGE

NORTHERN STATES POWER COMPANY

2 PAGES, PP. 34-35 OF SUPPLEMENT 21 TO SEFOR LICENSE APPLICATION, DEC. 10, 1968, DOCKET 50-231

INITIAL OPERATION OF THE IRRADIATED FUEL STORAGE TANK DESCRIBED. ***REFERS TO SAR SUPPLEMENT 13. ORIGINAL DESIGN WAS FOR STORAGE OF 1.5 CORES, BUT PRESENT PLAN IS FOR NO REFUELING SO INITIAL OPERATION WILL BE FOR 30 RODS MAX. STORED IN GAS RATHER THAN SODIUM. CALCULATIONS (PRESENTED) SHOW PEAK TEMPERATURE OF CLAD IS 820 F. EVEN IF 6 RODS WERE PLACED IN ONE CHANNEL, TEMPERATURE WOULD BE ONLY 1225 F.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REPORT, SAR + FUEL STORAGE + HIGH TEMPERATURE + SEFOR (RE) + RESPONSE TO NRC QUESTION + REACTOR, LMFBR + FUEL ELEMENTS

18-030399

DATED 1965

LUCHBAUM R + COVERT R + BAYSDEN J + HYATT M

MCR SYSTEM PARAMETRIC STUDIES

GENERAL MOTORS CORP., INDIANAPOLIS, IND.

EDR-4186, VOLUMES I-IV + GAMD-3078 (VOLUMES I-4), MAY 3, 1965

A COMPREHENSIVE PARAMETRIC ANALYSIS OF MCR-TYPE MOBILE NUCLEAR POWERPLANTS IN THE RANGE OF 1.5 TO 15 MW NET ELECTRICAL POWER OUTPUT IS PRESENTED IN THE FOUR VOLUMES OF THIS REPORT. THE SEVERAL PHASES OF THE STUDY ARE REPORTED AS FOLLOWS - VOLUME I - PARAMETRIC STUDIES OF THE BASIC CONCEPT SYSTEM. VOLUME II - PARAMETRIC PACKAGING STUDIES OF THE BASIC CONCEPT SYSTEM. VOLUME III - PARAMETRIC STUDIES OF SYSTEMS WITH ALTERNATE COMPONENT CONCEPTS. VOLUME IV - SUPPORT STUDIES OF

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-030395 *CONTINUED*

OTHER ALTERNATE COMPONENT CONCEPTS. THE ONLY SAFETY ITEM DISCUSSED AT LENGTH IS SHIELDING, IN VOLUMES III AND IV.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, POWER + DESIGN STUDY + SHIELDING + COMPONENTS

18-030498
ROZENHOLC M
PHENIX

DATED 1968

COMMISSARIAT A L ENERGIE ATOMIQUE, PARIS
11 PAGES, ENERG. NUCL. (PARIS) 10, PAGES 65-75 (MAR.-APR. 1968) (IN FRENCH)

DESCRIBES PRINCIPAL FEATURES OF THE FAST BREEDER SODIUM-COOLED PHENIX REACTOR. THE OVERALL PLANT LAYOUT IS ILLUSTRATED, AND DESIGN CHARACTERISTICS OF THE FUEL ELEMENTS, CORE, CONTROL ELEMENTS, FUEL LOADING MACHINE, STEAM GENERATOR, POWER PLANT EQUIPMENT, AND CONTROL SYSTEM ARE DISCUSSED.

FRANCE + REACTOR DESCRIPTION + REACTOR, POWER + INSTRUMENT, CONTROL + FUEL HANDLING MACHINE + SYSTEM DESCRIPTION + EQUIPMENT + CORE COMPONENTS + STEAM GENERATOR + REACTOR, LMFBR + CONTROL RODS + FUEL ELEMENTS

18-030571 ALSO IN CATEGORY 17

DATED 1968

ANDERSON RH + FORD JA + LINDSEY PS
DESIGN, FABRICATION, AND PRELIMINARY OPERATION OF THE STEAM GENERATORS FOR THE ENRICO FERMI ATOMIC POWER PLANT

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICH.
APDA-307 +. 301 PAGES, AUGUST 1968

CHRONOLOGICAL HISTORY FROM PURCHASE (1958) THROUGH PRELIMINARY OPERATION (1965). DESCRIPTIVE STUDIES AND EXPERIMENTAL PROGRAM LEADING TO DESIGN CONCEPT (37 PG), DESIGN AND FABRICATION (38 PG), PREOPERATIONAL TESTING AND USE (64 PG). *** COMPACT ONCE-THROUGH DESIGN HAS BEEN GOOD. TUBE FAILURE SHOWED METAL-WATER REACTION COULD BE WITHSTOOD AND DID NOT PROPAGATE. FLOW-INDUCED VIBRATION CAUSED FAILURE DUE TO POOR RESTRAINT AND BAFFLING. TUBE-TO-TUBE-SHEET JOINT-LEAKAGE CAUSE HAS NOT BEEN FOUND. ACCESSIBILITY HAS SAVED MONEY, BUT INTERNAL INSPECTION IS NOT EASY. MANY DEFECTS WERE FOUND IN PREOPERATIONAL TESTING, WHILE FIXING WAS EASY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FERMI (LMFBR) + METAL WATER REACTION + OPERATING EXPERIENCE SUMMARY + DESIGN STUDY + TEST, PREOPERATIONAL + SYSTEM DESCRIPTION + EXAMINATION + VIBRATION + FAILURE, TUBING + STEAM GENERATOR + REACTOR, LMFBR

18-030574
HAEFELE W

DATED 1968

FEDERAL GERMAN FAST BREEDER PROGRAMME
KERNFORSCHUNGSZENTRUM, KARLSRUHE, GERMANY
EURFNR-403 + STI/PUB-165 (VOL. 1) + CONF-671043 +. 1968

STARTED IN 1960 AT KARLSRUHE, CONCENTRATED ON PHYSICS, SAFETY, AND PRINCIPAL ENGINEERING STUDIES. FOUR MAJOR EXPERIMENTAL STEPS WERE SUAK, STARK, SNEAK, AND SEFOR. ABOUT 1970 TWO PROTOTYPES OF 300 MW(E), (ONE LIQUID SODIUM AND ONE STEAM COOLANT) WILL BE BUILT TO DEMONSTRATE THE BREEDING AND ELECTRICITY POTENTIAL ON A SIGNIFICANT SCALE. THEREAFTER, TWO DEMONSTRATION PLANTS OF 1000 MW(E) WILL MEET THE ULTIMATE OBJECTIVE - 1000 MW(E) FAST OXIDE BREEDER REACTORS AVAILABLE ON A COMMERCIAL AND COMPETITIVE BASIS BY THE END OF THE 1970S.

REACTOR, BREEDER + STEAM + GERMANY + R AND D PROGRAM + REACTOR, LMFBR

18-030576

DATED 1968

ISHIKAWA H + AN S + MISHIMA Y
FAST REACTOR RESEARCH AND DEVELOPMENT PROGRAMME IN JAPAN
JAPAN ATOMIC ENERGY RESEARCH INSTITUTE, TOKYO
STI/PUB-165 (VOL. 1) + CONF-671043- (VOL. 1) +. 12 PAGES, PAGES 15-26 OF FAST REACTOR PHYSICS. VOL. I., VIENNA, INTERNATIONAL ATOMIC ENERGY AGENCY, 1968, FROM SYMPOSIUM ON FAST REACTOR PHYSICS AND RELATED SAFETY PROBLEMS, KARLSRUHE, GERMANY

ARTICLE STRESSES FAST REACTOR PHYSICS AND RELATED SAFETY PROBLEMS. THE 100-MW(TH) EXPERIMENTAL REACTOR AND PROTOTYPE REACTOR ARE TO BE SODIUM-COOLED. THE FORMER IS TO BE CRITICAL IN 1972, SOME R AND D ACTIVITIES HAVE ALREADY COMMENCED. THE PROTOTYPE IS TO BE DEVELOPED BY THE EARLY 1980S. A PROTOTYPE ADVANCED CONVERTER PROJECT IS NOW UNDER WAY IN JAPAN, AND A NEW ORGANIZATION WAS ESTABLISHED IN OCTOBER 1967 TO DEVELOP BOTH THE FAST REACTOR AND THE ADVANCED CONVERTER. THE JAERI FAST CRITICAL ASSEMBLY IS TO OBTAIN INFORMATION ON REACTIVITY EFFECTS, TO CHECK DESIGNS OF THE EXPERIMENTAL AND THE PROTOTYPE REACTOR. IN MANY ESTABLISHMENTS THEORETICAL RESEARCH IS INCREASING ON THE FOLLOWING TOPICS - NUCLEAR DATA NEUTRON ENERGY SPECTRUM, REACTIVITY COEFFICIENTS, DESIGN STUDY, ANALYSIS OF KINETICS AND ACCIDENTS. THE STUDY OF FAST REACTOR SAFETY MAINLY CONFINED TO THE THEORETICAL APPROACH. THE EXPERIMENTAL RESEARCH PROGRAM IS ALSO BEING PREPARED ON SUCH SUBJECTS AS CONTAINMENT SYSTEM DESIGN STUDIES, SODIUM TECHNOLOGY AND FUEL DEVELOPMENT.

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-030576 *CONTINUED*
JAPAN + R AND D PROGRAM + REACTOR, LMFBR

18-030609 ALSO IN CATEGORIES 11 AND 12 DATED 1967
SEFOR FSAR VOL. I - REACTOR DESCRIPTION
GENERAL ELECTRIC CO.
200 PAGES, FIGURES, TABLES, SEFOR FACILITY DESCRIPTION AND SAFETY ANALYSIS REPORT VOL. 1, JULY 21, 1967,
DOCKET 50-231

CONTAINS SECTIONS ON SUMMARY DESCRIPTION, SITE, REACTOR, COOLANT SYSTEM, SHIELDING, CONTAINMENT, SAFEGUARDS WASTE CONTROL, AND INSTRUMENTATION. A 20-MWT SODIUM-COOLED FAST-BREEDER PU02-U23802 FUELED EXPERIMENTAL REACTOR TO GAIN EXPERIENCE WITH SUCH REACTORS AND PARTICULARLY THE DOPPLER COEFF. AND ITS EFFECT ON REACTOR RESPONSE. CONTROLLED BY 10 MOVABLE NICKEL REFLECTORS AND 18 FIXED B4C RODS. CONTAINMENT BY STEEL-LINED REINFORCED-CONCRETE BARRIER SURROUNDED BY WELDED A-212B STEEL VESSEL. SODIUM INLET TEMP., 700 F, OUTLET TEMP., 820 F AT 5000 GPM. DELTA P IN MAIN LOOP, 31-PSI. PUMP DISCHARGE, 55 PSIG. SAFEGUARDS INCLUDE CONTAINMENT INERTING SAFETY VESSEL OF STEEL AROUND REACTOR VESSEL, 4 COOLANT SYSTEMS, AND CORE DISPERSION ASSEMBLY TO PREVENT CRITICAL MASS IF CORE AND VESSEL DISASSEMBLE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

CONTAINMENT DESIGN + CONTAINMENT, MULTIPLE + DOPPLER COEFFICIENT + DOPPLER EFFECT + REACTOR, LMC + PRESSURE VESSELS + REACTOR DESCRIPTION + REACTOR STABILITY + REPORT, SAR + SITING, REACTOR + SODIUM + ARGON + NITROGEN + INSTRUMENT, CONTROL + WASTE MANAGEMENT + SEFOR (RE) + SHIELDING + SYSTEM DESCRIPTION + EXPERIMENT + CONTAINMENT ATMOSPHERE, INERT + CORE COMPONENTS + PLUTONIUM DIOXIDE + MAIN COOLING SYSTEM + R AND D PROGRAM + ENGINEERED SAFETY FEATURE + REACTOR, LMFBR + FUEL ELEMENTS

18-030633 ALSO IN CATEGORY 5 DATED 1967
SEFOR FSAR VOL. II - SAFETY ANALYSIS
GENERAL ELECTRIC COMPANY
150 PAGES, FIGURES, TABLES, SEFOR FACILITY DESCRIPTION AND SAFETY ANALYSIS REPORT, VOL. 2, JULY 21, 1967,
DOCKET 50-231

CONTAINS SECTIONS ON AUXILIARY SYSTEMS, EXPERIMENTAL PROGRAM AND EQUIPMENT, CONDUCT OF OPERATIONS, ELECTRIC POWER SUPPLY, AND THE SAFETY ANALYSIS. APPENDICES INCLUDE AN ANALOG MODEL OF SEFOR, RESULTS OF MOCKUP TESTS IN ZPR-3, BLAST SHIELDING AND DESIGN, AND RADIOLOGICAL EFFECTS CALCULATIONS. SAFETY ANALYSIS CONSIDERS THE SAFETY FEATURES FOR OPERATION, POSITIVE REACTIVITY INSERTIONS, AND LOSS OF COOLANT FLOW. THE MHA IS A SUDDEN INCREASE IN POWER DENSITY FOLLOWED BY LOSS OF COOLANT FLOW LEADING TO RAPID FUEL MELTING AND COMPACTION OF CORE INTO A SUPERCRITICAL MASS, WHEREAS THE CONTAINMENT DESIGN CRITERIA IS SUCH THAT IT MEETS A SET OF CONDITIONS MORE SEVERE THAN THOSE PREDICTED FOR THE MHA. THE CALCULATIONAL BASIS FOR RADIOLOGICAL EFFECTS ASSUMES CONDITIONS MORE THAN THOSE FOR THE MHA AND CONTAINMENT CAPABILITY LESS EFFECTIVE THAN THOSE OF THE DESIGN CRITERIA.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

COMPUTER, ANALOG + CONTAINMENT DESIGN + DOPPLER COEFFICIENT + DOPPLER EFFECT + DOSE + REACTOR, LMC + ACCIDENT, LOSS OF FLOW + ACCIDENT, MAXIMUM CREDIBLE (MCA) + MOCKUP + REACTOR DESCRIPTION + REPORT, SAR + ANALYTICAL MODEL + ACCIDENT, HYPOTHETICAL + HAZARDS ANALYSIS + ANALOG SIMULATION + ELECTRIC POWER, NORMAL + TEST, PHYSICS + SEFOR (RE) + SHIELDING + SYSTEM DESCRIPTION + EXPERIMENT + SIMULATION + SAFETY ANALYSIS + R AND D PROGRAM + ENGINEERED SAFETY FEATURE + SYSTEM CAPABILITY + OFF SITE + REACTOR, LMFBR

18-030761 ALSO IN CATEGORY 1 DATED 1967
DIECKAMP HM + BEELEY RJ + STOKER DJ + ASH EB
ATOMICS INTERNATIONAL DEMONSTRATION FAST BREEDER REACTOR
ATOMIC INTERNATIONAL, CANOGA PARK, CALIF.
CONF-670413-(AND SUPPL) +. 22 PAGES, PAGES 3.1-22 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO, CALIF., APRIL 10-12, 1967. HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY, 1967

DESCRIBES A 500-MWE NA-COOLED FAST-BREEDER CONCEPTUAL DESIGN TO ESTABLISH THE BASIS FOR EARLY DEMONSTRATION PLANT OPERATION. THE CONCEPTUAL DESIGN BEING OBTAINED BY SYSTEMATICALLY EXAMINING ALTERNATE SYSTEM AND SUBSYSTEM CHOICES IS GIVEN. COST-EFFECTIVENESS TECHNIQUES (CAPITAL, FUEL AND DEVELOPMENT COSTS, PRIOR EXPERIENCE, MAINTAINABILITY AND RELIABILITY) ARE EVALUATED. MAJOR SYSTEM EVALUATIONS HAVE BEEN OR ARE BEING CONDUCTED RELATIVE TO BASIC PLANT OPERATING PARAMETERS, CORE GEOMETRY, PRIMARY SYSTEM CONFIGURATION, FUEL HANDLING AND STEAM GENERATORS, AS WELL AS OVERALL PLANT DESIGN AND LAYOUT.

ECONOMICS + AI + DESIGN STUDY + REACTOR, LMFBR

18-030762 ALSO IN CATEGORY 1 DATED 1967
GAST K + SCHLECHTENDAUHL E + SMIDT D + GUTHMANN E + HELD C + KOEHLER M + MORGENSTERN F
NA-2 DESIGN - 300 MWE GERMAN FCR
KERNFORSCHUNGSZENTRUM KARLSRUHE, GERMANY
CONF-670413-(AND SUPPL) +. 15 PAGES, PAGES 3.23-38 OF FAST REACTORS NATIONAL TOPICAL MEETING, SAN FRANCISCO, CALIF., APRIL 10-12, 1967, HINSDALE, ILL., AMERICAN NUCLEAR SOCIETY

PRESENTS PRELIMINARY RESULTS OF THE CURRENT CONCEPTUAL DESIGN STUDY FOR A 300-MWE NA-COOLED FAST

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-030762 *CONTINUED*

BREEDER. DUE CONSERVATISM IS APPLIED IN THE DESIGN AND IN THE SAFETY ASSESSMENT TO ACCOUNT FOR THE FIRST-OF-A-KIND-CHARACTERISTIC OF THE REACTOR. IT INCLUDES MOST OF THE BASIC FEATURES PREVIOUSLY ESTABLISHED FOR LARGE COMMERCIAL BREEDER REACTORS. THOSE DESIGN FEATURES WHICH ARE OF VITAL IMPORTANCE TO THE SAFETY AND OPERATING RELIABILITY OF THE REACTOR ARE DISCUSSED. EMPHASIS IS LAID UPON THE CRITICAL DESIGN ASPECTS OF THE CORE AND OF THE PRIMARY COOLANT SYSTEM, AND CONSEQUENCE-LIMITING SAFEGUARDS ARE DESCRIBED.

DESIGN STUDY + GERMANY + SAFETY ANALYSIS + REACTOR, LMFBR

18-031032

DATED 1968

PFLASTERER GR
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM
GENERAL ELECTRIC CO.
GEAP-5673 +. 31 PAGES, FIGURES, TABLES, AUGUST 1968

327 FUEL RODS WOULD NOT MEET PELLET GAS-CONTENT SPECIFICATIONS, SO AN OUTGASSING FURNACE WAS BUILT TO REWORK THE RODS. ***OTHER THAN THIS DESCRIPTIVE MATERIAL, THE REPORT LISTS STATUS OF VARIOUS PROJECTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, FAST + REACTOR, LMCR + FUEL, PELLET TYPE + FAILURE, FABRICATION ERROR + REACTOR, PULSED + SEFOR (RE) + OFF GAS + R AND D PROGRAM + FUEL ELEMENTS

18-031448

ALSO IN CATEGORY 5

DATED 1968

KIRSCHLER LH + HILTZ RH + RODGERS SJ
THE EFFECT OF HIGH TEMPERATURE SODIUM ON THE MECHANICAL PROPERTIES OF CANDIDATE ALLOYS FOR THE LMFBR PROGRAM
MSA RESEARCH CORPORATION, PA.
MSAR-68-135 +. 100 PAGES, 37 FIGURES, 13 TABLES, SEPTEMBER 1968

THREE LOOPS (ONE OPERATING, ONE EVALUATING A CARBON METER), FOUR RETORTS (ALL USED IN SCREENING TESTS). 500-HR HIGH-TEMPERATURE OPERATION OF 304 AND 316 SS COMPLETED, CREEP RUPTURE TESTING BEGUN. PRELIMINARY DATA INDICATE 50% REDUCTION IN 1400 F RUPTURE LIFE, COMPARED WITH 1200 F. A CHANGE IN OXIDATION MECHANISM SUGGESTED.

ALLOY + REACTOR, LMCR + METAL + SODIUM + OUT OF PILE EXPERIMENT + STEEL, STAINLESS + CREEP BEHAVIOR + COOLANT CHEMISTRY + PROPERTY, PHYSICAL + INSTRUMENT, COOLANT QUALITY + R AND D PROGRAM + REACTOR, LMFBR + PIPES AND PIPE FITTINGS

18-031574

ALSO IN CATEGORY 5

DATED 1963

KLICKMAN AE + NIMS JB + MCCARTHY WJ + NICHOLSON RB
SAFETY CONSIDERATIONS IN THE ENRICO FERMI REACTOR DESIGN
COMITATO NAZIONALE ENERGIA NUCLEARE
CONF-193(VOL.2) +. 38 PAGES, PP. 593-630 OF REATTORI VELOCI, FROM THE 8TH NUCLEAR CONGRESS, ROME, JUNE 1963

THE GENERAL APPROACH TAKEN TO GUARD AGAINST SEVERE REACTOR ACCIDENTS IS DESCRIBED. A HYPOTHETICAL MELTDOWN ACCIDENT IS POSTULATED AND ANALYZED FOR ENERGY RELEASE AND CONTAINMENT REQUIREMENTS. PRECAUTIONS TAKEN IN DESIGN OF THE FERMI FAST BREEDER REACTOR TO PREVENT AUTOCATALYTIC EFFECTS AND RESONANCE INSTABILITY ARE DISCUSSED.

REACTOR, BREEDER + DESIGN CRITERIA + REACTOR, FAST + FERMI (LMFBR) + FUEL MELTDOWN + REACTOR, LMCR + ACCIDENT, HYPOTHETICAL + ITALY + SAFETY PRINCIPLES AND PHILOSOPHY + ENGINEERED SAFETY FEATURE + REACTOR, LMFBR

18-032265

ALSO IN CATEGORY 7

DATED 1968

HANSON JE
EXPERIMENT DESCRIPTION AND HAZARDS EVALUATION FOR THE PACIFIC NORTHWEST LABORATORY MIXED OXIDE (UO₂-PUO₂) IRRADIATIONS IN EBR-11, TASK A, SUBTAS, 1 IRRADIATIONS
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-650 +. 190 PAGES, FIGURES, TABLES, REFERENCES, JULY 1968

RECENT DEVELOPMENTS IN MIXED OXIDE FUEL IRRADIATION EXPERIENCE PERMITTED CONSIDERATION OF SOME REDUCTION IN NUMBER OF FUEL-DESIGN PARAMETERS IN THE FTR DRIVER FUEL DEVELOPMENT PROGRAM. GENERALLY, IN ADDITION TO PROVIDING FAST-FLUX IRRADIATED FUEL SPECIMENS FOR TRANSIENT IRRADIATION, THE 19-CAPSULE SUBASSEMBLIES WILL ALSO INCLUDE SPECIMENS FOR EVALUATING EFFECTS OF CLADDING THICKNESS REDUCTION AND ALTERNATE FUEL FORMS ON FUEL PERFORMANCE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + FUEL INTEGRITY + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + HAZARDS ANALYSIS + IN PILE EXPERIMENT + URANIUM DIOXIDE + PLUTONIUM DIOXIDE + REACTOR, LMFBR + FUEL ELEMENTS

18-032542

ALSO IN CATEGORY 5

DATED 1967

WOODFIELD FW
FAST FLUX TEST FACILITY DEVELOPMENT MONTHLY PROGRESS REPORT, OCTOBER 1967

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032542 *CONTINUED*
BATTTELLE NORTHWEST
BNWL-615 +. 38 PAGES, 3 REFERENCES, NOVEMBER 10, 1967

FUEL FAILURE ACTIVITY IN THE EX-REACTOR PORTION WILL EXCEED CRUD ACTIVITY BY FACTOR OF 10 (DETAILS IN BNWL-585). LIQUID NA SAMPLING TOPICAL REPORT IS BNWL-491. ****A RESISTANCE HEATER WAS OPERATED AT 1016TH BTU/HR-FT² IN 300 F WATER. ***CONCEPTUAL STUDIES ON NOZZLE SEAL CLEARANCE PROGRESS. ***FUEL CLAD IRRADIATION SAMPLES HAVE BEEN SENT TO EBR-*. ***CONSTRUCTION COMPLETED ON POSTIRRADIATION CREEP-RUPTURE TEST DEVICE. ***UNIRRADIATED SERPENTINE DOES NOT LOSE WATER OR HYDROGEN ON CONTACT WITH 930 F SODIUM, ALTHOUGH IT COULD NOT BE KEPT IN CONTACT FOR LONG.

AVAILABILITY - F.W. WOODFIELD, BATTELLE-NORTHWEST, RICHLAND, WASH.

ACTIVITY BUILDUP + REACTOR, FAST + FAILURE, FUEL ELEMENT + REACTOR, LMCR + REACTOR, TEST + TEST, PROOF + SAMPLING + FFF (TR) + CRUD + THERMAL INSULATION + CREEP + R AND D PROGRAM + COOLANT QUALITY + REACTOR, LMFBR

18-032543 ALSO IN CATEGORY 11 DATED 1968
WOODFIELD FW
FAST FLUX TEST FACILITY DEVELOPMENT MONTHLY PROGRESS REPORT, MAY 1968
BATTTELLE NORTHWEST
BNWL-803 +. 46 PAGES, 11 FIGURES, 2 TABLES, JUNE 10, 1968

SODIUM-METHANE REACTION RATE IS PRIMARILY DEPENDENT ON GAS VOLUME. ***CR-51 WAS FOUND DOWNSTREAM OF THE IRRADIATED 304-SS PIECE IN THE TEST LOOP. ACTIVITY DROPOFF COINCIDED WITH STEP CHANGE IN VELOCITY. ***EBR-11 304-SS ROD THIMBLE VOLUME INCREASED AS MUCH AS 1.17% FOLLOWING 4.8 X 10¹²NDI NVT AT 792 F. ***CYCLIC STRAIN ALLOWED BY ASME CODE NOT CONSERVATIVE FOR IRRADIATED 304-SS AT 1200 F BECAUSE THE MANSON-HALFORD RELATIONSHIP INDICATES SHORTER FATIGUE LIFE FOR CONDITIONS CAUSING FAILURE IN 30,000 OR MORE CYCLES.

AVAILABILITY - F.W. WOODFIELD, BATTELLE-NORTHWEST, RICHLAND, WASH.

CODES AND STANDARDS + REACTOR, FAST + REACTOR, LMCR + RADIATION DAMAGE + REACTOR, TEST + STEEL, STAINLESS + FFF (TR) + FAST NEUTRON + CRUD + IRRADIATION TESTING + FAILURE, FATIGUE + R AND D PROGRAM + EXPANSION + REACTOR, LMFBR

18-032587 ALSO IN CATEGORIES 7 AND 17 DATED 1968
STEWART DC + MACIAS ES + BASILE LJ + MILSTED J
BUILDUP OF RADIOACTIVE PRODUCTS IN THERMAL REACTORS. II. RA-226 TARGETS
ARGONNE NATIONAL LAB., 111.
ANL-7478 +. 133 PAGES, FIGURES, TABLES, REFERENCES, JULY 1968

THIS REPORT CONTINUES A SERIES DESCRIBING THE BUILDUP OF ISOTOPIC L, PHYSICS, AND COMPONENT DEVELOPMENT), EBR-2 OPERATION, (2) GENERAL NEUTRON FLUXES AT VARIOUS LEVELS. REFERENCES ARE GIVEN IN ITS DEVELOPMENT, CHEMISTRY, (3) NUCLEAR SAFETY. WEAR TESTS WERE RUN WITH USED ARE DESCRIBED IN DETAIL. AC-227 IS ALSO OF SOME INTEREST AS AT 300 F, SODIUM LUBRICATES. BUT, AT 500 F, IT WETS SURFACE AND A ALPHA EMITTERS. THIS APPLICATION, HOWEVER, WILL PROBABLY ALWAYS LIQUID SODIUM FLASHING, PRESSURE WAVE PROPAGATION, SUPERHEAT, COOLANT ANY QUANTITY. THERE ARE ALSO SEVERAL INTENSE GAMMA EMITTERS AMONG TRANSFER, AND SODIUM-AIR REACTION. PRESENT SOME PROBLEMS IN HANDLING THE MATERIAL FOR HEAT SOURCE APPLICATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.05 MICROFICHE

CHEMICAL REACTION + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + MOLYBDENUM + REACTOR, SAFETY RESEARCH + SODIUM + REACTOR, TEST + TESTING + TREAT (PRR) + REACTOR COOLANT + METAL, LIQUID + REACTOR, PULSED + LUBRICATION + R AND D PROGRAM + REACTOR, LMFBR

18-032622 DATED 1967
FOURNIER J + LADET J + PUIG I
SAFETY EXAMINATION OF THE RAPSODIE REACTOR
CEN, CADARACHE, FRANCE
CONF-670916 +. 11 PAGES, PP. VI9.1-10 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 15-22, 1967

THE SAFETY EVALUATION TOOK INTO ACCOUNT THE MAIN CHARACTERISTICS AS COMMON TO TYPICAL FAST NEUTRON SODIUM COOLED REACTORS. THRESHOLDS WHICH SHOULD NOT BE EXCEEDED FOR CERTAIN PARAMETERS ARE DEFINED. MALFUNCTIONS WHICH COULD OCCUR ARE ANALYZED SUCCESSIVELY. THIS ANALYSIS SHOWS THAT A GENERALIZED ACCIDENT, INVOLVING THE DAMAGE OF ALL THE CORE IS ALTOGETHER UNLIKELY. ON THE OTHER HAND, DOUBT EXISTS REGARDING THE CONSEQUENCES OF A LOCALIZED ACCIDENT ALTHOUGH ITS DETECTION IS PROBABLE. DESPITE GREAT CONFIDENCE PLACED ON SAFETY DEVICES, ACCIDENT CALCULATIONS HAVE MOSTLY BEEN PERFORMED ASSUMING THEIR FAILURE. THE RESULTS PROVE THAT THE REACTOR IS INTRINSICALLY VERY SAFE. FINALLY THE EVALUATIONS OF CONSEQUENCES OF HYPOTHETICAL ACCIDENTS, SUCH AS SECONDARY REACTIVITY ACCIDENTS OR NA-AIR REACTIONS, HAVE PERMITTED US TO SHOW THAT THE DIFFERENT SAFETY BARRIERS ARE SUFFICIENT.

FRANCE + SODIUM + REACTOR, TEST + SAFETY ANALYSIS + REACTOR, LMFBR

18-032632 ALSO IN CATEGORY 17 DATED 1969
REPORT ON SEFOR OPERATING LICENSE APPLICATION

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032632 *CONTINUED*
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
2 PAGES, LETTER - ACRS TO AEC, MARCH 13, 1969, DOCKET 50-239

ACRS RECOMMENDS OPERATION UP TO 20 MWT BUT INDICATES CONCERN FOR PROVISIONS TO PROTECT CORE INTEGRITY AGAINST PRIMARY LEAKS, INADEQUACY OF SIPHON BREAKER, GUIDELINES FOR ACTIONS IN EVENT OF ANOMALOUS REACTIVITY ON COOLANT TEMPS, COOLANT MONITORING TECHNIQUES AND FREQUENCY, BASES AND ACCEPTABILITY OF CONTINUED OPERATION IN PRESENCE OF KNOWN LOSS OF FUEL CLAD INTEGRITY.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, FAST + REACTOR, LMCR + ACRS + LICENSE APPLICATION + REACTOR, PULSED + SEFOR (RE) + OPERATING LICENSE PROCESS + REACTOR, LMFBR

18-032656 ALSO IN CATEGORY 6 DATED 1967
BILLURIS G + PFLASTERER R + WOLFE B
THE SEFOR EXPERIMENTAL PROGRAM AND ITS INFLUENCE ON REACTOR DESIGN
GENERAL ELECTRIC CO., SUNNYVALE, CALIF.
CONF-670916 +. 10 PAGES, PP. VI8-1-9 OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

DESCRIBES THE PROPOSED SEFOR EXPERIMENTAL PROGRAM, THE APPROACH TO ANALYSIS OF THE EXPERIMENTAL DATA, AND THE VARIOUS FEATURES INCORPORATED INTO SEFOR TO OBTAIN HIGH QUALITY EXPERIMENTAL RESULTS, AND THE APPROACH PLANNED TO ENSURE REACTOR SAFETY DURING THE TRANSIENT TESTS.

REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR DYNAMICS + TESTING + REACTOR, PULSED + SEFOR (RE) + R AND D PROGRAM + REACTOR, LMFBR

18-032806 DATED 1968
EMERGENCY PLAN
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.
50231-23 +. 50 PAGES, FIGURES, TABLES, SEFOR AMENDMENT 21 TO LICENSE APPLICATION, SEPTEMBER 30, 1968, DOCKET 50-231

THOROUGH TREATMENT OF ON- AND OFF-SITE EMERGENCY PLANS. TOPICS - OBJECTIVES, TYPES OF INCIDENTS CONSIDERED, RESPONSIBILITIES, TREATMENT AND/OR RESCUE OF SITE PERSONNEL, NATURAL DISASTERS, FIRES, RADIOACTIVITY INCIDENTS, EVACUATION, TRAINING.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

GUIDE + REACTOR, FAST + REACTOR, LMCR + PROCEDURES AND MANUALS + RADIATION INJURY TREATMENT + RADIATION PROTECTION, ORGANIZATION + EMERGENCY PROCEDURE + RADIATION SAFETY AND CONTROL + INCIDENT, RECOVERY FROM + REACTOR, PULSED + SEFOR (RE) + REPORT, PSAR + OFF SITE + REACTOR, LMFBR

18-032807 ALSO IN CATEGORY 9 DATED 1968
REFLECTOR CONTROL SYSTEM
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-11 +. 160 PAGES, 24 FIGURES, TABLES, SEFOR, PSAR, SUPPLEMENT 11, FEBRUARY 29, 1968, DOCKET 50-231

DESCRIBES CONTROL SYSTEM AND DISCUSSES THE VARIOUS COMPONENTS FROM DESIGN TO INSTALLATION AND TESTING. SYSTEM HAS BEEN SUBJECTED TO MORE THAN 800,000 INCHES OF TRAVEL AT NORMAL DRIVE SPEEDS AND 130,000 INCHES AT SCRAM SPEED, WHICH IS APPROXIMATELY 12 TIMES EXPECTED USAGE, AND HAS NEVER FAILED TO FUNCTION. GUIDE-PAD DEVELOPMENT AND TEST PROGRAM DEMONSTRATED PADS AND RAILS HAVE LIFE EXPECTANCY CONSIDERABLY LONGER, AND AT LOADS HIGHER THAN EXPECTED OPERATING CONDITIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CONTROL SYSTEM + REACTOR, FAST + REACTOR, LMCR + TESTING + REACTOR, PULSED + SEFOR (RE) + REFLECTOR + SYSTEM DESCRIPTION + REPORT, PSAR + REACTOR, LMFBR + CONTROL RODS + CONTROL ROD DRIVE MECHANISMS

18-032808 ALSO IN CATEGORY 10 DATED 1968
ELECTRIC POWER SYSTEMS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-12 +. 60 PAGES, 4 TABLES, SEFOR, PSAR, SUPPLEMENT 12, MARCH 11, 1968, DOCKET 50-231

TOPICS DESCRIBED AND DISCUSSED INCLUDE REQUIREMENTS OF NORMAL AND EMERGENCY ELECTRIC SYSTEMS, GROUNDING AND CATHODIC PROTECTION, COMMUNICATION SYSTEM, DISTRIBUTION, DC SYSTEM, AND PUMP-POWER SUPPLIES.

CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ELECTRIC POWER + REACTOR, PULSED + SEFOR (RE) + SYSTEM DESCRIPTION + REPORT, PSAR + REACTOR, LMFBR

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032809

DATED 1968

IRRADIATED FUEL STORAGE

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-12 +. 28 PAGES, FIGURES, SEFOR, PSAR, SUPPLEMENT 13, MARCH 11, 1968, DOCKET 50-231

DESIGN OF STORAGE EQUIPMENT IS DISCUSSED, AND ADDITIONAL INFORMATION GIVEN CONCERNING EQUIPMENT DESIGN AND SAFETY ANALYSIS. HOWEVER, FOR CLARITY, SOME OF THE INFORMATION THERE HAS BEEN REPEATED IN THIS SUPPLEMENT. THE ANALYSIS OF COOLING HAS BEEN REVISED BASED ON MORE DETAILED CALCULATION OF HEAT GENERATION AND HEAT LOSSES IN THIS SYSTEM.

CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY \$0.65
MICROFICHE

REACTOR, FAST + REACTOR, LMCR + FUEL STORAGE + REACTOR, PULSED + SEFOR (RE) + HEAT TRANSFER ANALYSIS + SYSTEM DESCRIPTION + REPORT, PSAR + COOLING + REACTOR, LMFBK

18-032810

DATED 1968

QUALITY CONTROL FOR SODIUM SYSTEM

GENERAL ELECTRIC

50231-14 +. 12 PAGES, 2 FIGURES, SEFOR, PSAR, SUPPLEMENT 14, MARCH 11, 1968, DOCKET 50-231

DESCRIBES THE QUALITY-CONTROL PROGRAM FOR THE COMPONENTS AND ASSEMBLIES WHICH COMPRISE THE COOLANT SYSTEMS. REQUIREMENTS BEGAN WITH INITIAL PLANT DESIGN AND CARRIED THROUGH ALL PHASES OF MATERIAL PROCUREMENT, EQUIPMENT FABRICATION AND PLANT CONSTRUCTION. FOR THE MAJOR COMPONENTS, INCLUDING HEAT EXCHANGERS AND ELECTROMAGNETIC PUMPS, GE HAS BEEN RESPONSIBLE FOR THE DESIGN SPECIFICATIONS, EQUIPMENT PROCUREMENT AND FOR APPROVAL OF VENDOR FABRICATION PROCEDURES USED FOR QUALITY CONTROL ACCEPTANCE. SOME SODIUM COMPONENTS - PRINCIPALLY PIPING, FITTINGS, TANKS AND VALVES - WERE PROCURED BY THE ARCHITECT ENGINEER, IN THESE CASES BOTH GENERAL ELECTRIC AND THE ARCHITECT-ENGINEER APPROVED THE FABRICATION AND TEST PROCEDURES FOR QUALITY CONTROL.

CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY \$0.65
MICROFICHE

REACTOR, FAST + REACTOR, LMCR + SODIUM + REACTOR COOLANT + MATERIAL + REACTOR, PULSED + SEFOR (RE) + EQUIPMENT + REPORT, PSAR + QUALITY ASSURANCE + MAIN COOLING SYSTEM + REACTOR, LMFBK + PUMPS + HEAT EXCHANGERS

18-032811

DATED 1968

MAIN SODIUM PIPING, PUMPS, AND HEAT EXCHANGERS

GENERAL ELECTRIC

50231-12 +. 150 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 15, MARCH 11, 1968, DOCKET 50-231

COMPLEMENTS INFORMATION IN SECT. 5.2 OF PSAR, AND INCLUDES DESCRIPTIONS, DESIGN CRITERIA, AND TYPICAL CALCULATIONS. ALL MATERIAL IN CONTACT WITH NA OR NA VAPOR IS 304 STAINLESS STEEL OR EQUIVALENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + SODIUM + REACTOR COOLANT + STEEL, STAINLESS + STRESS ANALYSIS + MATERIAL + REACTOR, PULSED + SEFOR (RE) + SYSTEM DESCRIPTION + REPORT, PSAR + MAIN COOLING SYSTEM + REACTOR, LMFBK + PUMPS + PIPES AND PIPE FITTINGS + HEAT EXCHANGERS

18-032813

ALSO IN CATEGORY 12

DATED 1968

PRIMARY SHIELD DESIGN AND EVALUATION

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-8 +. 50 PAGES, FIGURES, TABLES, SEFOR, PSAR, SUPPLEMENT 8, FEBRUARY 1968, DOCKET 50-231

DESCRIBES CORE AND REFLECTOR SHIELD USED TO ATTENUATE RADIATION TO LEVEL WHERE CONCRETE CAN BE USED WITHOUT DAMAGE. SHIELD MATERIALS ARE STEEL, BORON CARBIDE, AND SERPENTINE. TOPICS INCLUDE FABRICATION, INSTALLATION, FIGURES, AND APPENDIX A CONTAINS TEMP.-DISTRIBUTION CALCULATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

CONCRETE + REACTOR, FAST + REACTOR, LMCR + RADIATION DAMAGE + STEEL + THERMAL MECHANICAL EFFECT + REACTOR, PULSED + SEFOR (RE) + SHIELDING + SYSTEM DESCRIPTION + REPORT, PSAR + SYSTEM CAPABILITY + REACTOR, LMFBK

18-032814

ALSO IN CATEGORY 12

DATED 1968

SHIELDING NUCLEAR DESIGN CALCULATIONS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-9 +. 175 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 9, FEBRUARY 1968, DOCKET 50-231

DESCRIBES CALCULATIONS OF NEUTRON AND GAMMA SOURCES AND ATTENUATION. DOSE RATES DURING AND AFTER OPERATION WERE CALCULATED TO SPECIFY SHIELD THICKNESSES AND COMPOSITIONS. ENERGY ABSORPTION RATES WERE CALCULATED TO PROVIDE COOLING REQUIREMENTS. DESCRIPTION OF SHIELD INCLUDED. A MORE DETAILED DESCRIPTION IS GIVEN IN SUPPLEMENT 8.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032814 *CONTINUED*
CHARGE \$2.00)

REACTOR, FAST + REACTOR, LMCR + SOURCE, RADIATION + RADIATION SAFETY AND CONTROL + REACTOR, PULSED + SEFOR (RE) + SHIELDING + MATHEMATICAL STUDY + THEORETICAL INVESTIGATION + REPORT, PSAR + MATHEMATICAL TREATMENT + REACTOR, LMFBFR

18-032815 ALSO IN CATEGORY 5 DATED 1968
CORE DESIGN AND ANALYSIS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-10 +. 175 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 10, FEB. 29, 1968, DOCKET 50-231

CONTAINS DETAILED DESCRIPTION OF MECHANICAL DESIGN AND ANALYSIS OF CORE. EXPANDED VERSION OF INFORMATION IN SECT. IV OF PSAR. THE MECHANICAL ANALYSIS IS DESCRIBED IN SECTION I, DESIGN IN SECTION II. SECTION III CONTAINS ADDITIONAL PHYSICS INFORMATION WHICH SUPPLEMENTS PSAR. A SUMMARY OF CODES USED IN DESIGN OF CORE IS INCLUDED. SUMMARIES OF THE MORE SIGNIFICANT RESULTS OF R AND D EFFORT HAVE EITHER BEEN TRANSMITTED AS SEPARATE SUPPLEMENTS (SUPPLEMENTS 1, 3, AND 4), ARE SUMMARIZED IN APPENDIX OF THIS SUPPLEMENT OR ARE REFERRED TO.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, FAST + HYDRAULIC ANALYSIS + REACTOR, LMCR + THERMAL ANALYSIS + REACTOR, PULSED + SEFOR (RE) + HEAT TRANSFER ANALYSIS + REPORT, PSAR + REACTOR, LMFBFR + FUEL ELEMENTS

18-032816 ALSO IN CATEGORY 11 DATED 1968
CONTAINMENT DESIGN
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-7 +. 14 PAGES, SEFOR, PSAR, SUPPLEMENT 7, FEB. 14, 1968, DOCKET 50-231

A CONDENSATION OF THE MATERIAL IN THE PSAR SECTION VII ON CONTAINMENT DESIGN FOR THE PURPOSE OF CLARITY BY ELIMINATING REPETITION.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

CONTAINMENT DESIGN + REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + REPORT, PSAR + REACTOR, LMFBFR

18-032817 ALSO IN CATEGORY 2 DATED 1968
ADDITIONAL INFORMATION ON SEISMIC DESIGN
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-6 +. 6 PAGES, 1 FIGURE, 2 TABLES, SEFOR, PSAR, SUPPLEMENT 6, FEB. 14, 1968, DOCKET 50-231

RESULTS OF DYNAMIC AND STATIC SEISMIC ANALYSIS GIVEN. TABLE LISTS LOADING OF ALL MAJOR EQUIPMENT (PUMPS, CRANES, TANK, AND HEAT EXCHANGERS). RESULTS FOR 109-FT STACK ARE - PERIOD 0.71 SEC, MAX. DISPLACEMENT 1.82 IN., AND MAX. MOMENT 31.8 FT. K.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

REACTOR, FAST + REACTOR, LMCR + STACK + STRESS ANALYSIS + REACTOR, PULSED + SEFOR (RE) + EQUIPMENT + REPORT, PSAR + EARTHQUAKE ENGINEERING + INTERACTION, SOIL AND STRUCTURE + STRUCTURAL ANALYSIS, DYNAMIC + REACTOR, LMFBFR

18-032818 DATED 1968
FUEL QUALITY CONTROL AND INSPECTION
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-5 +. 50 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 5, JAN. 18, 1968, DOCKET 50-231

CONTAINS CORE-1 FUEL SPECIFICATIONS. DESCRIBES FUEL INSPECTION AND QC REQUIREMENTS.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

CODES AND STANDARDS + DESIGN CRITERIA + REACTOR, FAST + REACTOR, LMCR + PLUTONIUM + REACTOR, PULSED + SEFOR (RE) + REPORT, PSAR + QUALITY ASSURANCE + NITRATE + REACTOR, LMFBFR + FUEL, LIQUID METAL

18-032819 ALSO IN CATEGORIES 5 AND 17 DATED 1967
FLUID FLOW AND VIBRATION TESTS ON 1/2 SCALE VESSEL MODEL
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-4 +. 60 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 4, DEC. 5, 1967, DOCKET 50-231

WATER, AT TEMPERATURES UP TO 265 F, WAS USED TO SIMULATE COOLANT CONDITIONS IN THE SODIUM-COOLED REACTOR. FLOWS WERE VARIED FROM 10 TO 100 PERCENT OF RATED REYNOLDS NUMBER IN THE REACTOR. DATA WAS OBTAINED ON RELATIVE CORE FLOW DISTRIBUTION, FLOW PATTERNS, PRESSURE LOSSES, AND VIBRATION CHARACTERISTICS OF THE CANTILEVER-MOUNTED CORE SHROUD. FLUID FLOW RESULTS PROVIDED INFORMATION

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032819 *CONTINUED*

NECESSARY FOR VERIFICATION OF FINAL DESIGN. VIBRATION RESULTS INDICATED NO SERIOUS EFFECTS OF AMPLITUDE OR STRESS IN THE FULL-SCALE REACTOR.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, FAST + FLOW THEORY AND EXPERIMENTS + REACTOR, LMCR + PRESSURE VESSELS + TEST, PRESSURE VESSEL + SODIUM + TESTING + REACTOR COOLANT + METAL, LIQUID + REACTOR, PULSED + SEFOR (RE) + PRESSURE DROP + FLOW DISTRIBUTION + VIBRATION + REPORT, PSAR + MODEL TESTING + FLOW + REACTOR, LMFBR

18-032820 ALSO IN CATEGORY 9 DATED 1967

SPECIFICATION AND ELEMENTARY DIAGRAMS OF REACTOR SAFETY SYSTEM

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-2 +. 75 PAGES, FIGURES, TABLES, REFERENCES, SEFOR, PSAR, SUPPLEMENT 2, DEC. 5, 1967, DOCKET 50-231

GIVES DETAILED DRAWINGS OF ELECTRICAL MAKEUP OF SAFETY SYSTEM. SYSTEM IS DESCRIBED, SPECIFICATIONS AND QUALITY ASSURANCE ARE GIVEN.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

DESIGN CRITERIA + REACTOR, FAST + REACTOR, LMCR + REACTOR SAFETY SYSTEM + REACTOR, PULSED + SEFOR (RE) + SYSTEM DESCRIPTION + REPORT, PSAR + REACTOR, LMFBR

18-032821 ALSO IN CATEGORY 6 DATED 1967

MOCKUP EXPERIMENTS IN ZPR-3

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-1 +. 40 PAGES, FIGURES, TABLES, 24 REFERENCES, SEFOR, PSAR, SUPPLEMENT 1, DEC. 5, 1967, DOCKET 50-231

ANALYSES OF A SERIES OF CRITICAL EXPERIMENTS AND APPLICATION OF RESULTS TO THE SEFOR DESIGN ARE DISCUSSED. CRITICAL-MASS VALUES WERE DETERMINED FOR 1-, 2-, AND 3-SEGMENT FUEL DESIGNS TO HELP ESTABLISH THE PLUTONIUM ATOM FRACTION. REACTIVITY EFFECTS OF AXIAL FUEL EXPANSION WERE MEASURED WHICH LED TO SELECTION OF THE 2-SEGMENT DESIGN FOR SEFOR. MEASUREMENTS OF THE REACTIVITY WORTH OF THE RADIAL REFLECTOR ESTABLISHED THE ADEQUACY OF THE REFLECTOR CONTROL SYSTEM.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + MOCKUP + REACTIVITY COEFFICIENT + CRITICALITY EXPERIMENT + MEASUREMENT, REACTIVITY + REACTOR, PULSED + SEFOR (RE) + REPORT, PSAR + REACTOR PHYSICS + CRITICAL MASS + REACTOR, LMFBR

18-032839 ALSO IN CATEGORY 17 DATED 1969

ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT MARCH 1969

ARGONNE NATIONAL LABORATORY

ANL-7561 +. 110 PAGES, FIGURES, TABLES, REFERENCES, MARCH 1969

ONE OF A SERIES. TOPICS ARE - LMFBR DEVELOPMENT OF FUEL (PHYSICS, AND COMPONENTS), FAST BREEDER FUEL DEVELOPMENT, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY. OPERATIONAL ACTIVITIES DISCUSSED FOR EBR-2, ZPR-3, AND TREAT. APPROVAL FOR OPERATION OF ZPPR WAS RECEIVED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CRITICAL ASSEMBLY + EBR 1 AND 2 (RE) + REACTOR, FAST + REACTOR, LMCR + REPORT, OPERATIONS + REACTOR, TEST + COMPONENTS + R AND D PROGRAM + ZPR 3 (CAF) + REACTOR, LMFBR + FUEL ELEMENTS

18-032843 ALSO IN CATEGORIES 7 AND 11 DATED 1969

BULBA E + UNDERHILL D

APPLICATION OF A PEBBLE BED HEAT-SINK TO THE FAST FLUX TEST FACILITY CLEANUP SYSTEM

HARVARD UNIVERSITY

NYU-841-17 +. 19 PAGES, 2 FIGURES, 9 REFERENCES, MARCH 1969

CONTAINS MATHEMATICAL EQUATIONS FOR GAS TEMPERATURES FOLLOWING SODIUM RELEASE AND BURNING INTO FFTF VENTILATION SYSTEM. ALLOWS FOR VARIABLE INLET GAS TEMPERATURE BUT DOES NOT ACCOUNT FOR SODIUM DEPOSITION PRIOR TO RELEASES. EXPERIMENTAL WORK ON THE LATTER NEEDED, ALSO ON GAS-FLOW SCHEMES AND ON BEST STRATEGY FOR PLACING SYSTEM IN OPERATION IN ACCIDENTS OF VARIOUS MAGNITUDES.

DEPOSITION + REACTOR, FAST + FIRE + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + SODIUM + REACTOR, TEST + VENTILATION SYSTEM + FFTF (TR) + HEAT SINK + MATHEMATICAL TREATMENT + R AND D PROGRAM + GAS + REACTOR, LMFBR + PACKED BED + THERMAL TRANSIENT

18-032844 ALSO IN CATEGORY 17 DATED 1967

SCOTT FA

SAMPLING FFTF SODIUM COOLANT FOR ANALYTICAL PURPOSES

BATTELLE-NORTHWEST

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032844 *CONTINUED*
BNWL-491 +. 33 PAGES, 1 FIGURE, 3 TABLES, 70 REFERENCES, SEPTEMBER 1967

DESCRIBES CONCEPTUAL DESIGN USING A SMALL BYPASS ROOF. ALSO DESCRIBES AND EVALUATES EXPERIENCE WITH 3 ALTERNATIVES (VACUUM THIEF, DIPPING CUP, DRAIN SAMPLING).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + REACTOR, FAST + REACTOR, LMCR + SODIUM + REACTOR, TEST + SAMPLING + FFTF (TR) + OPERATING EXPERIENCE SUMMARY + MAIN COOLING SYSTEM + COOLANT QUALITY + REACTOR, LMFBFR

18-032877 DATED 1968
RESPONSES TO AEC QUESTIONS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 56 PAGES, 11 FIGURES, 3 TABLES, SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

CONTAINS RESPONSES TO 19 QUESTIONS CONCERNING PIPE BREAK, TORNADO, SPENT FUEL STORAGE TANK PLUG DROP, FASTENERS, PREOPERATIONAL CHECKS, ETC.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, (08 CENTS/PAGE -- MINIMUM CHARGE \$2.00)

NRC QUESTION + REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + REPORT, PSAR + REACTOR, LMFBFR

18-032878 ALSO IN CATEGORY 5 DATED 1968
RESPONSE TO QUESTION 1 - GUILLOTINE PIPE BREAK
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 4 PAGES, 4 FIG., 2 TAB., PAGE 1 THRU 8 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DEVELOP IN MORE DETAIL THE CONSEQUENCES OF THE GUILLOTINE-TYPE BREAK IN THE MAIN PRIMARY COOLANT SYSTEM DISCUSSED IN SECTION 16.3.3.2.3.3 OF THE FDSAR. PROVIDE BASIS FOR CONCLUSION THAT CIRCULATION WOULD BE RESUMED THROUGH THE REACTOR VESSEL AND AUXILIARY COOLANT LOOP FOLLOWING THIS ACCIDENT. (ANSWER ON PAGES 1 THRU 8).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + SODIUM + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBFR

18-032879 ALSO IN CATEGORY 5 DATED 1968
RESPONSE TO QUESTION 2 - MAIN OUTLET THERMAL SLEEVE FAILURE
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 2 PAGES, 1 FIG., PAGE 9 THRU 11 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

IF THE FASTENER OF THE SLEEVE IN THE REACTOR VESSEL MAIN PRIMARY OUTLET NOZZLE SHOULD FAIL, THE SLEEVE COULD MOVE AND CAUSE AN INCREASE IN THE AMOUNT OF COOLANT FLOW BYPASSING THE CORE. DESCRIBE THE EVENTS FOLLOWING MOVEMENT OF THIS SLEEVE. INCLUDE THE METHOD OF DETECTION, MEANS OF SCRAMMING THE REACTOR AND THE TEMPERATURE TIME HISTORY OF THE CORE COOLANT. (ANSWER ON PAGES 9 THRU 11).

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + REACTOR, PULSED + SEFOR (RE) + NOZZLE + RESPONSE TO NRC QUESTION + REPORT, PSAR + DISPLACEMENT + FLOW, BYPASS + COOLING + REACTOR, LMFBFR

18-032880 ALSO IN CATEGORIES 2 AND 11 DATED 1968
RESPONSE TO QUESTION 3 - TORNADO - REFUELING CELL OPERATION
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 1 PAGES, PAGE 12 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

WHAT SAFETY PRECAUTIONS ARE PLANNED FOR ENSURING SAFE OPERATION IN THE REFUELING CELL WITH REGARD TO POTENTIAL TORNADO DAMAGE. ***[ANSWER] FOR DESIGN PROVISIONS, SEE PSAR, SUPPLEMENT 1, PP. 3-61 THRU 3-66, FDSAR, VOL. 1, PAGE 7-1, PAGE 7-9, PAGE 7-16, AND FDSAR SUPPLEMENT 17, ANSWER TO QUESTIONS A-11, D-2, AND D-8.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REFUELING + FUEL STORAGE + STORAGE CONTAINER + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + DAMAGE + DESTRUCTIVE WIND + REPORT, PSAR + REACTOR, LMFBFR

18-032881 ALSO IN CATEGORY 7 DATED 1968
RESPONSE TO QUESTION 4 - RADWASTE LINE BREAK
GENERAL ELECTRIC CO., SAN JOSE, CALIF.

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032881 *CONTINUED*
50231-20 +. 3 PAGES, PAGE 13-17 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DISCUSS THE CONTROL OF THE WASTE GAS BATCH TANK VALVES AND THE MANNER IN WHICH THE CONTROLLER WOULD PREVENT THE CONTENTS OF THE BATCH TANKS FROM BEING DISCHARGED INTO THE DECAY TANK VAULT AND FROM THERE TO THE ENVIRONS IN THE EVENT OF A PIPE BREAK BETWEEN THE PIPE TUNNEL AND THE INLET VALVES TO THE DECAY TANKS. ***ANSWER ON PAGES 13 THRU 15.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + FISSION PRODUCT RELEASE + REACTOR, LMCR + WASTE STORAGE + WASTE DISPOSAL, GAS + WASTE MANAGEMENT + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

18-032882 ALSO IN CATEGORY 5 DATED 1968

RESPONSE TO QUESTION 5 - BLOCKAGE OF MAIN OUTLET NOZZLE

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-20 +. 2 PAGES, 1 FIG., PAGE 16 AND 17 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DISCUSS THE EFFECT OF BLOCKAGE OF THE UPFLOW PASSAGE LEADING TO THE REACTOR VESSEL MAIN COOLANT FLOW EXIT NOZZLE. INCLUDE THE TIME HISTORY OF THE CORE COOLANT TEMPERATURE. ***ANSWER ON PAGES 16 AND 17.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + FLOW BLOCKAGE + REPORT, PSAR + REACTOR, LMFBR

18-032883 ALSO IN CATEGORY 5 DATED 1968

RESPONSE TO QUESTION 6 - SPENT FUEL STORAGE TANK SHIELD PLUG DROP

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-20 +. 1 PAGE, PAGE 18 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

ANALYZE THE CONSEQUENCES OF DROPPING A SHIELD PLUG ON THE SPENT FUEL STORAGE TANK. ***ANSWER ON PAGE 18.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REFUELING + FUEL STORAGE + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

18-032884 DATED 1968

RESPONSE TO QUESTION 7 - BOLT LOADS FOR VESSEL INTERNALS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-20 +. 2 PAGES, PAGE 19 AND 20 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

PROVIDE ADDITIONAL DATA ON SPECIFIC EXPERIMENTS CONDUCTED BY GE AT KAPL IN SUPPORT OF SIR MARK B PROGRAM. ***ANSWER ON PAGES 19 AND 20.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + FASTENER + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + CORE COMPONENTS + R AND D PROGRAM + REACTOR, LMFBR

18-032885 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION 8 - OPERATION OF CONTAINMENT DOORS AND HATCHES

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-20 +. 5 PAGES, PAGE 21 THRU 25 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DISCUSS DESIGN AND OPERATION OF DOORS AND HATCHES OF CONTAINMENT USED FOR PASSAGE OF EQUIPMENT AND PERSONNEL. ***ANSWER ON PAGES 21 THRU 25.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

CONTAINMENT DESIGN + REACTOR, FAST + REACTOR, LMCR + CONTAINMENT AIR LOCK + CONTAINMENT EQUIPMENT HATCH + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

18-032886 ALSO IN CATEGORY 11 DATED 1968

RESPONSE TO QUESTION 9 - ACTIONS REQUIRED TO CLOSE TYPE 3 SYSTEMS

GENERAL ELECTRIC CO., SAN JOSE, CALIF.

50231-10 +. 1 PAGE, PAGE 26 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

ENUMERATE HOW MANY MORE OF THE 78 TYPE-3 CONTAINMENT PENETRATIONS ARE DEPENDENT UPON OPERATIONS

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032886 *CONTINUED*
CARRIED OUT BY OPERATOR IN ORDER TO QUALITY AS A CLOSED SYSTEM. ***SEE COMMENT K-2 OF PSAR, SUPPLEMENT 17.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFB

18-032887 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION 10 - CONSEQUENCES OF TYPE 3 LINE FAILURES
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 8 PAGES, PAGES 27 THRU 34 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

PROVIDE DETAILED DISCUSSION OF 51 PENETRATIONS RANGING IN SIZE FROM 1/2 TO 2 INCHES, WITH REGARD TO FUNCTION AND CONSEQUENCES OF FAILURES. ***ANSWER ON PAGES 27 THRU 34.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFB + PENETRATIONS, PRIMARY CONTAINMENT

18-032888 ALSO IN CATEGORIES 5 AND 11 DATED 1968
RESPONSE TO QUESTION 11 - DYNAMIC LOADS FROM FRED TESTS
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 6 PAGES, 1 TAB., PAGES 35 THRU 40 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

DISCUSS DYNAMIC LOADINGS IMPOSED BY MAXIMUM PLANNED TRANSIENT ON THE CORE SUPPORT STRUCTURE AND REACTOR VESSEL. ***ANSWER ON PAGES 35 THRU 40.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + CONTAINMENT VESSEL LOADING + REACTOR, FAST + REACTOR, LMCR + PRESSURE VESSELS + REACTOR TRANSIENT + STRESS + THERMAL MECHANICAL EFFECT + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + CORE COMPONENTS + REACTOR, LMFB

18-032889 ALSO IN CATEGORIES 6 AND 9 DATED 1968
RESPONSE TO QUESTION 12 - FRED OPERATING VARIABLES
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-20 +. 3 PAGES, 1 FIGURE, PAGE 41 THRU 43 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1968, DOCKET 50-231

PROVIDE INFORMATION ON OPERATIONAL VARIABLES TO BE USED DURING THE FAST REACTIVITY EXCURSION DEVICE EXPERIMENTS AND INDICATE WHICH VALUES HAVE BEEN OBTAINED BY MEANS OTHER THAN DIRECT MEASUREMENT. *** ANSWER ON PAGES 41 THRU 43.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + MEASUREMENT, REACTIVITY + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFB + INSTRUMENTS, MISC.

18-032890 ALSO IN CATEGORIES 6 AND 9 DATED 1968
RESPONSE TO QUESTION 13 - INSTALLATION AND REMOVAL OF TEST DEVICES
GENERAL ELECTRIC
50231-20 +. 4 PAGES, 1 FIGURE, PG. 44 THRU 48 OF SEFOR, PSAR, SUPPLEMENT 18, AUG. 6, 1969, DOCKET 50-231

FROM INFORMATION SUBMITTED, IT APPEARS THAT DUE TO HEIGHT AND OTHER LIMITATIONS, THE EXPERIMENTAL EQUIPMENT FOR EXCURSION MEASUREMENT MUST BE PARTIALLY ASSEMBLED AND DISASSEMBLED IN PLACE. DESCRIBE SEQUENTIALLY THE STEPS INVOLVED AND THE CHECKS THAT WILL BE MADE IN REMOVING AND INSTALLING THE DEVICES. *** ANSWER ON PAGES 44 THRU 48

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + EXCURSION, LARGE + MEASUREMENT, REACTIVITY + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFB + INSTRUMENTS, MISC.

18-032891 DATED 1968
RESPONSE TO QUESTION 14 - CONTROL OF PARTS AROUND THE REACTOR
GENERAL ELECTRIC
50231-20 +. 2 PAGES, 1 FIGURE, PAGES 49 AND 50 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

HOW WILL PROVISION BE MADE TO PROTECT AGAINST LOOSE PIECES FALLING INTO THE DRY-WELL THIMBLE OR REACTOR. *** ANSWER ON PAGES 49 AND 50.

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032891 *CONTINUED*

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
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CONTAINMENT, PRESSURE SUPPRESSION + REACTOR, FAST + FAILURE, OPERATOR ERROR + REACTOR, LMCR + PRESSURE VESSELS
+ REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + OBSTRUCTION

18-032892 ALSO IN CATEGORY 6 DATED 1968

RESPONSE TO QUESTION 15 - PREOPERATIONAL CHECKS OF TEST DEVICES

GENERAL ELECTRIC, SAN JOSE

50231-20 +. 1 PAGE, PAGE 51 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

WHAT PRECAUTIONS WILL BE TAKEN TO ENSURE THAT ASSEMBLY STEPS HAVE BEEN PROPERLY CARRIED OUT AND
EQUIPMENT WILL OPERATE PROPERLY PRIOR TO THE ACTUAL EXCURSION EXPERIMENT. *** (ANSWER) THE FAST
REACTIVITY EXCURSION DEVICE CAN BE TEST-FIRED FROM OUTSIDE CORE USING POSITIONER TO SET INITIAL
POSITION 30 IN. ABOVE CORE MIDPLANE, PROVIDING CAPABILITY OF RUNNING COMPLETE TEST WITHOUT
AFFECTING REACTOR. AN OPERATIONAL CHECKOUT WILL BE PERFORMED TO CHECK PROPER OPERATION OF
SWITCHES AND LATCHES BEFORE EACH TEST. SIMILAR TESTS WILL BE MADE WITH OSCILLATOR ROD TO ENSURE
PROPER OPERATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) +
RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

18-032893 ALSO IN CATEGORY 5 DATED 1968

RESPONSE TO QUESTION 16 - VERTICAL STROKE OF TEST DEVICE POSITIONER

GENERAL ELECTRIC

50231120 +. 1 PAGE, PG. 52 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

DISCUSS HOW STROKE OF EXCURSION DEVICE POSITIONER IS INTERLOCKED WITH DEVICE MOUNTED ON IT IN
ORDER TO AVOID INADVERTENTLY LOWERING THE POISON ROD MORE THAN PLANNED, THEREBY INCREASING THE
LIKELIHOOD OF GREATER REACTIVITY CHANGES THAN ANTICIPATED, AND DAMAGE TO THE DRY WELL THIMBLE.
*** (ANSWER) LOWEST POSSIBLE POINT OF POISON ROD IS SAME WITH EITHER OSCILLATOR OR FRED
INSTALLED. IN OPERATION WITH OSCILLATOR DEVICE, THE POSITIONER USES 20 IN. OR LONG GUIDE BARS
WITH A KING MOUNTED ON THE TOP. FAILURE OF UPPER LIMIT SWITCH WOULD MEAN DEVICE WOULD BE DRIVEN
UP AGAINST RING, AND MOTOR WOULD STALL.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REACTIVITY + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE +
MEASUREMENT, REACTIVITY + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR,
LMFBR + INSTRUMENTS, MISC.

18-032894 ALSO IN CATEGORIES 6 AND 9 AND 11 DATED 1968

RESPONSE TO QUESTION 17 - DESIGN OF PRESSURE VESSELS IN FRED

GENERAL ELECTRIC

50231-10 +. 2 PAGES, 1 FIGURE, PG. 53 AND 54 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

DESCRIBE BLOWOUT PATCH AND ACCUMULATOR ON THE EXCURSION DEVICE. *** ANSWER ON PAGES 53 AND 54.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) +
SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + INSTRUMENTS, MISC.

18-032895 ALSO IN CATEGORY 5 DATED 1968

RESPONSE TO QUESTION 18 - CONTROL OF POISON SLUGS USED WITH TEST DEVICES

GENERAL ELECTRIC

50231-20 +. 1 PAGE, PG. 55 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

DESCRIBE THE METHOD TO ENSURE THAT PROPER POISON ROD FOR OSCILLATOR AND FRED EXPERIMENTS WILL
ALWAYS BE INSTALLED PRIOR TO ACTUAL EXPERIMENT. *** (ANSWER) WITH EXCEPTION OF ROD DIAMETER,
OSCILLATOR AND FRED POISON RODS ARE COMPLETELY DIFFERENT. ATTACHMENT OF OSCILLATOR ROD TO
MECHANISM IS ACCOMPLISHED BY A DIFFERENT GRAPPLING MEANS THAN FRED POISON ROD WHICH IS ATTACHED
TO THE PISTON. PROCEDURAL CONTROL USED TO INSURE PROPER FRED POISON ROD IS INSTALLED.
REACTIVITY WORTH OF POISON ROD WILL BE VERIFIED AS IT IS LOWERED INTO CORE WITH POSITIONER.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT + ADMINISTRATIVE CONTROL + REACTOR, FAST + FAILURE, OPERATOR ERROR + REACTOR, LMCR + REACTOR DYNAMICS
+ TESTING + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + CONTROL
RODS

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032896 ALSO IN CATEGORY 6 DATED 1968
RESPONSE TO QUESTION 19 - BOLT PRELOAD FOR FRED EXPANSION CHAMBER
 GENERAL ELECTRIC
 50231-20 +. 1 PAGE, PG. 56 OF SEFOR, PSAR, SUPPLEMENT 18, AUGUST 6, 1968, DOCKET 50-231

DISCUSS SAFETY FACTOR OF THE SIX STUDS ON THE EXCURSION DEVICE WHEN THE CHAMBER EXPERIENCES ITS MAXIMUM PRESSURE PEAK. *** (ANSWER) THESE STUDS ARE DESIGNED SO THEY WOULD YIELD IF DEVICE WERE OVERPRESSURIZED AND BLOWOUT PATCH FAILED TO OPERATE PROPERLY. THEY ARE SIZED TO BEGIN ELONGATING AT APPROXIMATELY 1500 PSI AND RESULT IN PRESSURE RELIEF. NORMAL PRESSURE IN EXPANSION CHAMBER WILL BE A MAXIMUM OF 200 PSI.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TENSILE PROPERTY + TESTING + FASTENER + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + SAFETY MARGIN + REACTOR, LMFBR

18-032903 ALSO IN CATEGORIES 5 AND 12 DATED 1969
SIMULTANEOUS FAILURES - LOCA AND SAFEGUARD
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 50231 +. 7 PAGES, 1 FIGURE, SEFOR, PSAR, SUPPLEMENT 23, FEB. 20, 1969, DOCKET 50-231.

DESCRIBES MODIFICATIONS MADE TO MITIGATE CONSEQUENCES OF TWO SIMULTANEOUS FAILURES, I.E., A FAILURE IN THE PRIMARY COOLANT SYSTEM FOLLOWED BY A FAILURE OF ONE OF THE SYSTEMS OR COMPONENTS (ENGINEERED SAFEGUARDS FEATURES). GIVES SUMMARY OF PROTECTION AND INHERENT DESIGN FEATURES OF THESE SAFEGUARDS THAT ASSURE OPERABILITY IN THE ENVIRONMENT IN WHICH THEY ARE REQUIRED TO OPERATE FOLLOWING A PRIMARY SYSTEM FAILURE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + MODIFICATION + REACTOR, PULSED + SEFOR (RE) + REPORT, PSAR + FAILURE, SIMULTANEOUS + SYSTEM OPERABILITY IN ACCIDENT + ENGINEERED SAFETY FEATURE + REACTOR, LMFBR

18-032904 ALSO IN CATEGORIES 5 AND 12 DATED 1969
RESPONSE TO AEC QUESTION
 GENERAL ELECTRIC
 50231-26 +. 7 PAGES, SEFOR, PSAR, SUPPLEMENT 22, JANUARY 24, 1969, DOCKET 50-231

SUPPLEMENTARY INFORMATION GIVEN TO PREVIOUSLY ANSWERED QUESTIONS CONCERNING THE CONSEQUENCES OF COOLANT SYSTEM LEAKS AND RESULTING SYSTEM CHARACTERISTICS (SEE COMMENTS 1 IN SUPPLEMENT 18 AND PAGE 53 OF SUPPLEMENT 19), SPECIFIC TOPICS AND SMALL LEAKS IN MAIN COOLANT SYSTEM, AND EVALUATION OF EQUIPMENT NEEDED TO RECOVER FROM A GUILLOTINE PIPE FAILURE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

NRC QUESTION + ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + INCIDENT, RECOVERY FROM + REACTOR, PULSED + SEFOR (RE) + EQUIPMENT + REPORT, PSAR + REACTOR, LMFBR

18-032905 ALSO IN CATEGORIES 5 AND 12 DATED 1969
RESPONSE TO QUESTION 1 - SMALL LEAKS IN MAIN COOLANT SYSTEM
 GENERAL ELECTRIC
 50231-26 +. 3 PAGES, PAGES 1 THRU 3 OF SEFOR, PSAR, SUPPLEMENT 22, JAN. 24, 1969, DOCKET 50-231

DISCUSS CONSEQUENCES AND SYSTEM RESPONSE TO SMALL LEAKS IN THE MAIN COOLANT SYSTEM. *** ANSWER ON PAGES 1 THRU 3.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + MAIN COOLING SYSTEM + LEAK + REACTOR, LMFBR + PIPES AND PIPE FITTINGS

18-032906 ALSO IN CATEGORIES 5 AND 12 DATED 1969
RESPONSE TO QUESTION 2 - RECOVERY EQUIPMENT FOR LOCA PIPE FAILURE
 GENERAL ELECTRIC
 50231-26 +. 4 PAGES, PG. 4 THRU 7 OF SEFOR, PSAR, SUPPLEMENT 22, JAN. 24, 1969, DOCKET 50-231

DISCUSS EQUIPMENT NEEDED TO RECOVER FROM A GUILLOTINE PIPE FAILURE *** ANSWER ON PAGES 4 THRU 7. SEE ALSO COMMENT 1 IN SUPPLEMENT 18.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + INCIDENT, RECOVERY FROM + REACTOR, PULSED + SEFOR

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032906 *CONTINUED*
(RE) + RESPONSE TO NRC QUESTION + EQUIPMENT + REPORT, PSAR + REACTOR, LMFBR

18-032907 ALSO IN CATEGORY 1 DATED 1968
AEC QUESTIONS - 7 TOPICS
GENERAL ELECTRIC COMPANY
50231-21 +. 17 PAGES, 5 FIGURES, SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

RESPONSES TO 7 QUESTIONS CONCERNING EMERGENCY TREATMENT OF INJURED PERSONNEL BOTH ON AND OFF SITE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

NRC QUESTION + RADIATION INJURY TREATMENT + RADIATION PROTECTION, ORGANIZATION + EMERGENCY PROCEDURE +
PERSONNEL EXPOSURE, RADIATION + SEFOR (RE) + RADIOLOGICAL ASSISTANCE + REPORT, PSAR + OFF SITE + REACTOR,
LMFBR

18-032908 ALSO IN CATEGORY 1 DATED 1968
RESPONSE TO QUESTION 1 - INJURED PERSONNEL - ON SITE TREATMENT FACILITIES
GENERAL ELECTRIC CO., SAN JOSE
50231-21 +. 3 PAGES, PG. 1 THRU 3 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT., 1968, DOCKET 50-231

DESCRIBE ON-SITE FACILITIES FOR DECONTAMINATION AND IMMEDIATE EMERGENCY TREATMENT OF INJURED
PERSONNEL. *** ANSWER ON PAGES 1 THRU 3.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + ON
SITE + REACTOR, LMFBR

18-032909 ALSO IN CATEGORY 1 DATED 1968
RESPONSE TO QUESTION 2 - QUALIFICATION OF PHYSICIANS AND HEALTH PHYSICISTS
GENERAL ELECTRIC
50231-21 +. 2 PAGES, PAGES 4 AND 5 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT., 1968, DOCKET 50-231

DESCRIBE QUALIFICATIONS, EDUCATION, AND TRAINING OF PERSONNEL RESPONSIBLE FOR SUPERVISION AND CARE
OF INJURED RADIOACTIVELY CONTAMINATED PERSONS AND OTHER OCCUPATIONAL HEALTH HAZARDS. *** ANSWER
ON PAGES 4 AND 5.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + TRAINING + RADIATION PROTECTION, ORGANIZATION + REACTOR, PULSED + SEFOR (RE) +
RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + CERTIFICATION + STAFFING

18-032910 ALSO IN CATEGORY 1 DATED 1968
RESPONSE TO QUESTION 3 - TRANSPORT OF INJURED PERSONNEL
GENERAL ELECTRIC CO., SAN JOSE, CALIF.
50231-21 +. 1 PAGE, PG. 6 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

DESCRIBE ARRANGEMENTS FOR TRANSPORT OF INJURED PERSONNEL. *** ANSWER ON PAGE 6. TWO VEHICLES
AVAILABLE. EMERGENCY PROCEDURE NO. 006 APPLIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR +
REACTOR, LMFBR

18-032911 ALSO IN CATEGORY 1 DATED 1968
RESPONSE TO QUESTION 4 - HOSPITALS - ACCEPTANCE OF CONTAMINATED PERSONNEL
GENERAL ELECTRIC
50231-21 +. 2 PAGES, PAGES 7 AND 8 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

DESCRIBE LOCATION OF HOSPITAL AGREEABLE TO ACCEPTING PATIENTS FOR FURTHER DECONTAMINATION, AND
CONTAMINATED INJURED PERSONNEL. *** ANSWER ON PAGES 7 AND 8.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + OFF
SITE + REACTOR, LMFBR

18-032912 ALSO IN CATEGORY 1 DATED 1968

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032912 *CONTINUED*
RESPONSE TO QUESTION 5 - MEDICAL PERSONNEL QUALIFICATIONS
 GENERAL ELECTRIC
 50231-21 +. 1 PAGE, PG. 9 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

DESCRIBE QUALIFICATIONS OF PROFESSIONAL MEDICAL PERSONNEL AT THE SUPPORT FACILITY (HOSPITAL OR CLINIC) TO TREAT RADIATION AND CONTAMINATION VICTIMS, INCLUDING NUMBER AND TYPES OF PHYSICIANS AND A DESCRIPTION OF ANY SPECIALIZED TRAINING RELATED TO CONTAMINATION OR RADIATION INJURIES.
 *** ANSWER ON PAGE 9.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + TRAINING + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + OFF SITE + REACTOR, LMFB + CERTIFICATION + STAFFING

18-032913 ALSO IN CATEGORY 1 DATED 1968
RESPONSES TO QUESTION 6 - OFF SITE MEDICAL FACILITIES
 GENERAL ELECTRIC
 50231-21 +. 1 PAGE, PG. 10 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

DESCRIBE LIMITATIONS THAT EXIST REGARDING AVAILABILITY OF OFF-SITE MEDICAL FACILITIES AND SUPPORT.
 *** (ANSWER) NONE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + OFF SITE + REACTOR, LMFB

18-032914 ALSO IN CATEGORY 1 DATED 1968
RESPONSE TO QUESTION 7 - STANDING ORDERS AT OFF SITE HOSPITAL
 GENERAL ELECTRIC CO., SAN JOSE, CALIF.
 50231-21 +. 1 PAGE, PG. 11 OF SEFOR, PSAR, SUPPLEMENT 20, SEPT. 1968, DOCKET 50-231

DESCRIBE WRITTEN PLAN AND STANDING ORDERS IN RECEIVING AREA OF HOSPITAL DETAILING ACTIONS TO BE TAKEN AND PROCEDURE TO BE FOLLOWED WHEN CONTAMINATED PERSON WITH OR WITHOUT INJURY IS BROUGHT TO HOSPITAL. *** (ANSWER) PLANS AND ORDERS ARE BEING DEVELOPED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + OFF SITE + REACTOR, LMFB

18-032915 ALSO IN CATEGORIES 2 AND 16 DATED 1968
AEC QUESTIONS - 11 TOPICS
 GENERAL ELECTRIC
 50231-21 +. 150 PAGES, 25 FIGURES, TABLES, SEFOR, PSAR, SUPPLEMENT 19, OCTOBER 1968, DOCKET 50-231

SEVENTY-THREE PAGES DEVOTED TO RESPONSES TO 11 AEC QUESTIONS ON CONTROL SYSTEM MODIFICATION, AUXILIARY POWER, REFUELING ACCIDENT, THE EXCURSION DEVICE (FRED) LEAKS, TRANSIENT TEST LOADS, NOZZLE FAILURE, FUEL-ROD PIN-HOLE LEAKS, SURVEILLANCE PROGRAM, THERMAL INSULATION, AND EFFECT OF SODIUM IN THE CORE DURING MHA. APPENDIX (43 PAGES) CONSISTS OF SITE METEOROLOGICAL DATA.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

NRC QUESTION + CONTAINMENT VESSEL LOADING + CONTROL SYSTEM + EMERGENCY POWER, ELECTRIC + FAILURE, EQUIPMENT + REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + ACCIDENT, REFUELING + SITE CLIMATOLOGY + ACCIDENT, HYPOTHETICAL + REACTOR, PULSED + SEFOR (RE) + THERMAL INSULATION + NOZZLE + REPORT, PSAR + MAIN COOLING SYSTEM + SURVEILLANCE PROGRAM + LEAK + REACTOR, LMFB

18-032916 ALSO IN CATEGORY 9 DATED 1968
RESPONSE TO QUESTION 1 - CONTROL SYSTEM MODIFICATIONS
 GENERAL ELECTRIC, SAN JOSE
 50231-21 +. 5 PAGES, 2 TABLES, PAGES 1 THRU 5 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

WHAT MODIFICATIONS ARE TO BE MADE TO CONTROL SYSTEM TO ENSURE GREATER FORCE AVAILABLE TO WITHDRAW SEGMENT THAN AVAILABLE TO INSERT. *** ANSWER ON PAGES 1 THRU 5.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
 COPY, \$0.65 MICROFICHE

CONTROL SYSTEM + REACTOR, FAST + REACTOR, LMCR + REACTOR CONTROL + MODIFICATION + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFB + CONTROL ROD DRIVE MECHANISMS

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032917 ALSO IN CATEGORIES 2 AND 10 DATED 1968
RESPONSE TO QUESTION 2 - LOSS OF SITE POWER - TORNADO
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 5 PAGES, 7 FIGURES, PAGES 6 THRU 11 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

WHAT PROVISIONS HAVE BEEN MADE FOR LOSS OF SITE POWER DUE TO A TORNADO PASSING THROUGH OR NEAR THE REACTOR SITE. *** ANSWER ON PAGES 6 THRU 11.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF POWER + ELECTRIC POWER, NORMAL + REACTOR, PULSED + SEFOR (RE) + SAFE FAILURE CRITERION + RESPONSE TO NRC QUESTION + DESTRUCTIVE WIND + REPORT, PSAR + REACTOR, LMFBR

18-032918 DATED 1968
RESPONSE TO QUESTION 3 - REFUELING ACCIDENT
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 3 PAGES, 1 FIGURE, PAGES 12 THRU 15 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL INFORMATION ON CONSEQUENCES OF DROPPING ONE OF SPENT FUEL STORAGE TANK SHIELD PLUGS. *** (ANSWER) SEE COMMENT 6, PSAR SUPPLEMENT 18.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + ACCIDENT, REFUELING + FUEL STORAGE + STORAGE CONTAINER + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

18-032919 ALSO IN CATEGORIES 6 AND 9 DATED 1968
RESPONSE TO QUESTION 4 - DESCRIPTION OF EXCURSION TESTING DEVICE
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 35 PAGES, 13 FIGURES, 1 TABLE, PAGES 16 THRU 51 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL INFORMATION ON THE EXCURSION TESTING DEVICE (FRED) INCLUDING DETAILS OF ITS DESIGN AND HOW IT WILL BE UTILIZED IN THE SEFOR EXPERIMENTAL PROGRAM. *** ANSWER ON PAGES 18 THRU 51.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR DYNAMICS + TESTING + EXCURSION, LARGE + REACTOR, PULSED + SEFOR (RE) + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + EQUIPMENT + REPORT, PSAR + REACTOR, LMFBR

18-032920 ALSO IN CATEGORIES 5 AND 9 DATED 1968
RESPONSE TO QUESTION 5 - COOLANT LEAK EFFECT ON CONTROL
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 4 PAGES, 1 FIGURE, PAGES 52 THRU 56 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

DISCUSS POSSIBILITY OF SODIUM LEAK IN PRIMARY COOLANT SYSTEM THAT MIGHT INTERFERE WITH OPERATION OF REFLECTOR CONTROL SYSTEM. *** ANSWER ON PAGES 52 THRU 57.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT, CONSEQUENCES + CONTROL SYSTEM + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + REACTOR CONTROL + REACTOR, PULSED + SEFOR (RE) + REFLECTOR + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR + CONTROL RODS

18-032921 ALSO IN CATEGORIES 6 AND 11 DATED 1968
RESPONSE TO QUESTION 6 - CORE COMPONENT LOADINGS DURING TRANSIENT TESTS
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 5 PAGES, 1 FIGURE, 1 TABLE, PAGES 57 THRU 62 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL INFORMATION ON DYNAMIC LOADINGS IMPOSED ON CORE SUPPORT STRUCTURE DURING PROMPT TRANSIENT TEST PROGRAM. *** ANSWER ON PAGES 57 THRU 62.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBR

18-032922 ALSO IN CATEGORY 5 DATED 1968
RESPONSE TO QUESTION 7 - COOLING CAPABILITY WITH OUTLET NOZZLE FAILURE

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-032922 *CONTINUED*
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 2 PAGES, PAGES 63 AND 64 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL INFORMATION ON COOLING CAPABILITIES FOR FAILURE OF FASTENER OF THERMAL SLEEVE IN VESSEL MAIN PRIMARY OUTLET NOZZLE. *** ANSWER ON PAGES 63 AND 64.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + FAILURE, PRESSURE VESSEL + REACTOR, PULSED + SEFOR (RE) + NOZZLE + RESPONSE TO NRC QUESTION + REPORT, PSAR + SYSTEM OPERABILITY IN ACCIDENT + MAIN COOLING SYSTEM + SYSTEM CAPABILITY + REACTOR, LMFBK

18-032924 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION 9 - SURVEILLANCE PROGRAM - REACTOR VESSEL
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 2 PAGES, 1 TABLE, PAGES 67 AND 68 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL INFORMATION ON REACTOR VESSEL AND INTERNALS SURVEILLANCE PROGRAM. *** ANSWER ON PAGES 67 AND 68 PLUS PSAR SUPPLEMENT 16, SECT. 7.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

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18-032925 ALSO IN CATEGORY 11 DATED 1968
RESPONSE TO QUESTION 10 - VESSEL - THERMAL INSULATION
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 3 PAGES, 1 FIGURE, PAGES 69 THRU 72 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

PROVIDE ADDITIONAL DETAILS ON DESIGN AND INSTALLATION OF THERMAL INSULATION ON CORE SUPPORT SHROUD. DISCUSS POTENTIAL CONSEQUENCES WITH REGARD TO FLOW BLOCKAGE IF ONE OF PLATES SHOULD COME LOOSE. *** ANSWER ON PAGES 69 THRU 72.

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DESIGN CRITERIA + REACTOR, FAST + REACTOR, LMCR + ACCIDENT, LOSS OF FLOW + PRESSURE VESSELS + THERMAL MECHANICAL EFFECT + REACTOR, PULSED + SEFOR (RE) + THERMAL INSULATION + SYSTEM DESCRIPTION + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBK

18-032926 ALSO IN CATEGORIES 5 AND 11 DATED 1968
RESPONSE TO QUESTION 11 - SODIUM IN CORE - EFFECT ON MHA
GENERAL ELECTRIC, SAN JOSE
50231-21 +. 1 PAGE, PAGE 73 OF SEFOR, PSAR, SUPPLEMENT 19, OCT. 1968, DOCKET 50-231

DISCUSS EFFECT OF SODIUM IN CORE REGION ON MHA ANALYSIS. *** ANSWER ON PAGE 73.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ACCIDENT, CONSEQUENCES + REACTOR, FAST + REACTOR, LMCR + REACTOR STABILITY + SODIUM + ACCIDENT, HYPOTHETICAL + REACTOR, PULSED + SEFOR (RE) + RESPONSE TO NRC QUESTION + REPORT, PSAR + REACTOR, LMFBK + FUEL ELEMENTS

18-032933 ALSO IN CATEGORY 17 DATED 1968
ANL REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT
ARGONNE NATIONAL LABORATORY
ANL-7527 +. 140 PAGES, FIGURES, TABLES, REFERENCES, DECEMBER 1968

ONE OF A SERIES. TOPICS ARE - LMFBK DEVELOPMENT PROGRAM, FUEL DEVELOPMENT STUDIES OF OTHER FAST REACTORS, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY. TESTS WITH EBR-2, ZPR-3, AND ZPPR ARE DISCUSSED. CONVERSION OF ZPR-6 AND 9 TO PU OPERATIONS CONTAINER.

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18-032934 ALSO IN CATEGORY 17 DATED 1969
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18-032934 *CONTINUED*

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SECTION TITLES - PLUTONIUM UTILIZATION, EBWR, LIQUID-METAL FAST-BREEDER REACTORS, GENERAL REACTOR TECHNOLOGY, ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT, AND NUCLEAR SAFETY. SEARCH FOR LOCATION OF FISSION-GAS RELEASE IN EBR-2 CONTINUES. LMFBR TOPICS INCLUDE EBR-2 OPERATION, PHYSICS, COMPONENT AND FUEL DEVELOPMENT, PROGRESS ON FFTF. NUCLEAR SAFETY TOPICS INCLUDE ACCIDENT ANALYSIS, COOLANT DYNAMICS, FUEL MELTDOWN STUDIES WITH TREAT, TREAT OPERATIONS, CONTAINMENT BY ENERGY ABSORPTION, AND PLUTONIUM VOLATILITY STUDIES.

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18-032940 ALSO IN CATEGORY 17 DATED 1968

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18-032949 *CONTINUED*

AND COMPONENT DEVELOPMENT, EBR-2 OPERATION, GENERAL REACTOR TECHNOLOGY, AND NUCLEAR SAFETY.
NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS.

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NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS.

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NUCLEAR SAFETY SUBJECTS - REACTOR KINETICS, TREAT OPERATIONS.

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18-032953 ALSO IN CATEGORY 17 DATED 1968

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KINETICS + REACTOR, RESEARCH + REACTOR, TEST + TREAT (PRR) + ANL + REACTOR, PULSED + R AND D PROGRAM +
REACTOR, LMFBR

18-032990 ALSO IN CATEGORY 17 DATED 1968

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18-032990 *CONTINUED*
AND NUCLEAR SAFETY. EBR-2 AND TREAT OPERATIONS DISCUSSED.

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AND D PROGRAM + REACTOR, LMFBR

18-033274 DATED 1969
ARONSTEIN RE + ASH EB + CASEY OF
DESIGN AND OPERATIONAL CHARACTERISTICS OF A 1000-MWE COMMERCIAL LIQUID METAL FAST BREEDER REACTOR PLANT
ATOMICS INTERNATIONAL, CALIF.
41 PAGES, 16 FIGURES, 2 TABLES, 9 REFERENCES, PAPER PRESENTED AT THE AMERICAN POWER CONFERENCE, APRIL 1969

DESCRIBES A 1000-MWE LMFBR AND INDICATES SIGNIFICANT DIFFERENCES BETWEEN EARLY AND ADVANCED
PLANTS. ATOMICS INTERNATIONAL THINKS THAT AN INTERMEDIATE 300- TO 500-MWE PLANT SHOULD FIRST
BE BUILT RATHER THAN TO USE A ONE-STEP APPROACH TO THE 1000-MWE PLANT. DISCUSSES SUCH A
PROGRAM SCHEDULE AND THE PERFORMANCE AND OPERATIONAL CHARACTERISTICS OF THESE PLANTS. INCLUDES
GENERALITIES ABOUT MAINTAINING REACTOR AND HEAT TRANSFER SYSTEMS, AND THE STEAM GENERATORS.

AVAILABILITY - R.E. ARONSTEIN/E.B. ASH/D.F. CASEY, ATOMICS INTERNATIONAL, 8900 DESOTO AVE., CANOGA PARK, CALIF.

MAINTENANCE AND REPAIR + REACTOR DESCRIPTION + REACTOR, POWER + DESIGN STUDY + REVIEW + R AND D PROGRAM +
REACTOR, LMFBR

18-033333 ALSO IN CATEGORIES 5 AND 6 DATED 1967
PETERSON RE + SIMPSON DE + LITTLE WW
SELECTED SAFETY CONSIDERATIONS IN DESIGN OF THE FAST FLUX TEST FACILITY
BATTELLE MEMORIAL INSTITUTE
BNWL-SA-1236 +. CONF-670916 +. 10 FIGURES, 4 TABLES, 9 REFERENCES, JUNE 1967, PROCEEDINGS OF THE INTERNATIONAL
CONFERENCE ON THE SAFETY OF FAST REACTORS

REVIEW OF SKEWED-TUBE DESIGN. LOW FERTILE CONTENT GIVES LOW DOPPLER COEFFICIENT. FEW CORE LOOPS,
HIGH-WORTH KODS REQUIRE LOOPS OUTSIDE CORE. REFLECTOR INSTEAD OF BREEDING BLANKET MAXIMIZES
NEUTRON FLUX. SUMMARIZES RESULTS OF VARIOUS PRELIMINARY TRANSIENTS OR ACCIDENTS.

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LMFBR

18-033427 ALSO IN CATEGORIES 5 AND 6 DATED 1969
GOLDEN GH
ELEMENTARY NEUTRONICS CONSIDERATIONS IN LMFBR DESIGN
ARGONNE NATIONAL LABORATORY
ANL-7532 +. 68 PAGES, 12 FIGURES, 10 TABLES, 35 REFERENCES, MARCH 1969

THE SUBJECT MATERIAL PROGRESSES FROM BASIC PHYSICS CONCEPTS TO THE DESIGN OF REAL LMFBRs - FLUX
SPECTRA AND CROSS-SECTION VARIATION WITH ENERGY IN FAST REACTORS, ASPECTS OF MULTIGROUP
CALCULATIONS IN DIFFUSION THEORY, THE SIGNIFICANCE OF BREEDING AND CONVERSION RATIOS, SPECIFIC
POWER, AND DOUBLING TIME, RADIAL CORE ZONING, REFUELING (I.E. TIME-DEPENDENT BEHAVIOR), A
DISCUSSION OF REACTIVITY EFFECTS AS RELATED TO SAFETY, WITH SPECIFIC REFERENCE TO SODIUM VOID,
DOPPLER, AND EXPANSION EFFECTS, AND CURRENT DESIGN APPROACHES. PRESENTATION IS APPLICABLE TO
DESIGN OF ANY FAST BREEDER, REGARDLESS OF COOLANT. HENCE, PERTINENT COMMENTS WERE MADE REGARDING
GAS-COOLED FAST-BREEDER REACTORS.

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EFFECT + REACTOR DYNAMICS + REACTOR KINETICS + VOID COEFFICIENT + ANL + DESIGN STUDY + REACTOR PHYSICS + R
AND D PROGRAM + REACTOR, HTGR + REACTOR, LMFBR

18-033428 ALSO IN CATEGORIES 3 AND 13 DATED 1968
LMFBR PROGRAM PLAN. VOLUME 8 - FUEL RECYCLE
LMFBR PROGRAM OFFICE, ARGONNE NATIONAL LAB.
WASH-1108 +. 350 PAGES, FIGURES, TABLES, AUGUST 1968

CONTENTS INCLUDE CHARACTERISTICS OF FUELS, FABRICATION, COSTS, PROCESSING OF CORE AND BLANKET, AND
A THOROUGH DISCUSSION OF THE TASKS, INCLUDING COORDINATION OF RECYCLE DEVELOPMENTS, SYSTEM
DEMONSTRATIONS, HANDLING AND SHIPPING, RECOVERY, FUEL PREPARATION AND FABRICATION, AND GENERAL
STUDIES. APPENDIXES INCLUDE FLOW SHEETS, SUB-TASKS, ECONOMICS, AND A COLLATION OF PROGRAM TASKS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, BREEDER + ECONOMICS + FABRICATION + REACTOR, FAST + RADIOCHEMICAL PROCESSING + RECOVERY PROCESS +
FUEL REPROCESSING + R AND D PROGRAM + REACTOR, LMFBR + FUEL MANAGEMENT + JACOBS + FUEL ELEMENTS

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-033429 ALSO IN CATEGORIES 1 AND 17 DATED 1968
LMFBR PROGRAM PLAN. VOLUME 1 - OVERALL PLAN
ARGONNE NATIONAL LAB.
WASH-1101 +. 350 PAGES, FIGURES, TABLES, AUGUST 1968

CONTENTS - THE OBJECTIVES, SCOPE, BACKGROUND INFORMATION, REQUIREMENTS AND GOALS, PLAN ORGANIZATION, NEEDS, PLAN OF ACTION, SCHEDULING AND PRIORITIES, AND FACILITIES. APPENDIXES CONTAIN COLLECTION OF TASKS AND CONTRACT WORK, FACILITIES TABLE, GLOSSARY, AND PRIVATELY SUPPORTED WORK. NINE OTHER ELEMENTS OF THE PLAN, WHICH ARE PRESENTED IN SEPARATE VOLUMES, ARE PLANT DESIGN, COMPONENTS, INSTRUMENTATION AND CONTROL, SODIUM TECHNOLOGY, CORE DESIGN, FUELS AND MATERIALS, FUEL RECYCLE, PHYSICS, AND SAFETY.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, BREEDER + REACTOR, FAST + SODIUM + REACTOR COOLANT + DESIGN STUDY + REVIEW + R AND D PROGRAM + REACTOR, LMFBR + JACOBS

18-033448 ALSO IN CATEGORY 11 DATED 1968
LMFBR PROGRAM PLAN. VOLUME 6 - CORE DESIGN
LMFBR PROGRAM OFFICE, ARGONNE NATIONAL LAB.
WASH-1106 +. 400 PAGES, FIGURES, TABLE, AUGUST 1968

CONTENTS - SUMMARIES OF CORE DESIGN SYNTHESIS, TECHNOLOGY, UTILIZATION OF FACILITIES WITH MAJOR PORTION OF REPORT DEVOTED TO DESCRIPTION OF THE TASKS IN THE TECHNICAL PROGRAM. APPENDIXES DISCUSS CHARACTERISTICS OF CORE AND FUEL DESIGNS, REQUIREMENTS AND CAPABILITIES FOR IRRADIATION TESTING, CURRENT PROJECTS RELATED TO CORE-DESIGN SYNTHESIS.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

REACTOR, BREEDER + REACTOR, FAST + SODIUM + REACTOR COOLANT + DESIGN STUDY + IRRADIATION TESTING + CORE COMPONENTS + R AND D PROGRAM + REACTOR, LMFBR + JACOBS + FUEL ELEMENTS

18-033481 DATED 1968
WRIGHT JR + MCFALL A
IMPROVEMENTS RELATING TO NUCLEAR REACTORS
UNITED KINGDOM ATOMIC ENERGY AUTHORITY
BRITISH PATENT 1,134,698 +. NOVEMBER 27, 1968

HYDRAULIC COUNTERFLOW BRAKE FOR PREVENTION OF COOLANT VOIDING BY FISSION-GAS RELEASE IN THE INLET PLENUM OF LIQUID-METAL-COOLED REACTORS.

AVAILABILITY - THE PATENT OFFICE, 25 SOUTHAMPTON BLDG., LONDON, W.C. 2, ENGLAND (\$0.49 CENTS/COPY)

REACTOR, LMCR + ACCIDENT, LOSS OF COOLANT + UNITED KINGDOM + PATENT + MAIN COOLING SYSTEM + REACTOR, LMFBR + VALVE, CHECK

18-033535 ALSO IN CATEGORIES 2 AND 11 DATED 1968
LMFBR PROGRAM PLAN. VOL. 2 - PLANT DESIGN
ARGONNE NATIONAL LAB., LMFBR PROGRAM OFFICE
WASH-1102 +. 197 PAGES, 8 FIGURES, 17 TABLES, AUGUST 1968

DISCUSSES DEVELOPMENT OF PLANT CONCEPTS AND TECHNOLOGY FOR PLANT DESIGN. GIVES DETAILS OF OVERALL PLAN. MAIN TOPICS - (1) PLANT DESIGN DEVELOPMENT STUDIES, (2) GENERAL STUDIES - SITING, ETC., (3) DEMONSTRATION PLANT PROGRAM, AND (4) COMPILATION, ASSESSMENT, AND DISSEMINATION OF DESIGN INFORMATION

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

ECONOMICS + DESIGN STUDY + INFORMATION RETRIEVAL + R AND D PROGRAM + REACTOR, LMFBR + JACOBS

18-033542 ALSO IN CATEGORY 5 DATED 1968
KATO WY + RUSCH GK + DATES LR + TILL CE + VAN DOORNINCK J + CHEEVER CL
FINAL SAFETY ANALYSIS REPORT ON THE USE OF PLUTONIUM IN ZPR-6 AND -9
ARGONNE NATIONAL LAB.
ANL-7442 +. 303 PAGES, FIGURES, TABLES, REFERENCES, JULY 1, 1968

THE ZPR-6 AND -9 ARE SPLIT-TABLE CRITICAL FACILITIES USED FOR STUDY OF FAST-REACTOR PHYSICS SINCE JULY 1963. MODIFICATIONS WERE MADE SO THAT PLUTONIUM MAY BE USED AS FUEL TO PROVIDE EXPERIMENTAL PHYSICS DATA FOR THE LMFBR PROGRAM. THE DESIGN-BASIS ACCIDENT RESULTS IN MELTING AND BURNING 60 KG OF PU AND VAPORIZATION OF 22 KG OF SODIUM. OFF-SITE DOSE FROM NOBLE GASES AND RADIOIODINE ARE - 190 REMS AT 225 METERS AND 13 REMS AT 1500 METERS.

CRITICAL ASSEMBLY + PLUTONIUM + REPORT, SAR + ZPR 9 (CAF) + ZPR 6 (CAF) + HAZARDS ANALYSIS + R AND D PROGRAM + REACTOR, LMFBR

18-033932 ALSO IN CATEGORY 1 DATED 1968

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-033932 *CONTINUED*
MEUSER MH + PERRY JA
FABRICATION, CONSTRUCTION, AND INSTALLATION QUALITY ASSURANCE REQUIREMENTS FOR THE FFTF
BATTELLE-NORTHWEST, RICHLAND, WASH.
BNWL-511 +. 56 PAGES, MARCH 15, 1968

STATES GENERAL REQUIREMENTS FOR QUALITY ASSURANCE FOR PACIFIC NORTHWEST LABORATORY (PNL)
CONTRACTORS INVOLVED IN PROCUREMENT, FABRICATION, CONSTRUCTION, INSTALLATION, OR TESTING EFFORT
FOR THE FFTF PROJECT. THE SECTION 1.0, INTRODUCTION, THROUGH SECTION 14.0, PROGRAM
DOCUMENTATION, ARE AEC/PNL BASELINES. A GLOSSARY DEFINING TERMS IS INCLUDED AS APPENDIX A.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + CONTAINMENT CONSTRUCTION + FFTF (TR) + QUALITY
ASSURANCE + R AND D PROGRAM + REACTOR, LMFB

18-033950 ALSO IN CATEGORY 9 DATED 1968
SMAARDYK A + DIVONA CJ + OLP RH + HUTTER E
DESIGN AND ANALYSIS OF INSTRUMENTED SUBASSEMBLY SYSTEM FOR EBR-II
ARGONNE NATIONAL LAB., ILLINOIS
ANL-7423 +. 48 PAGES, 21 FIGURES, 3 TABLES, 7 REFERENCES, JUNE 1

DESCRIBES THE RESEARCH, DESIGN, AND DEVELOPMENT WORK PERFORMED AFTER A FEASIBILITY STUDY OF
INSTALLING AN INSTRUMENTED SUBASSEMBLY SYSTEM CAPABLE OF MEASURING PERTINENT OPERATING PARAMETERS
OF SPECIMEN FUELS AND RELATED COMPONENTS WHILE UNDER IRRADIATION IN THE EBR-II CORE. IMMEDIATE
OBJECTIVE IS CONSTRUCTION OF A PROTOTYPE SYSTEM FOR INSTALLATION AND PERFORMANCE EVALUATION IN
THE EBR-II DURING THE FIRST QUARTER OF CALENDAR YEAR 1969 FOR EVALUATING ADVANCED FUELS OF
INTEREST TO THE AEC LMFB PROGRAM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

EBR 1 AND 2 (RE) + REACTOR, FAST + FUEL BURNUP + INSTRUMENT, IN CORE + REACTOR, LMCR + REACTOR, TEST + ANL +
DESIGN STUDY + IRRADIATION TESTING + R AND D PROGRAM + REACTOR, LMFB + FUEL ELEMENTS

18-033951 DATED 1968
WOODFIELD FW
FAST FLUX TEST FACILITY DEVELOPMENT MONTHLY REPORT, DECEMBER 1967
BATTELLE-NORTHWEST, RICHLAND, WASH.
BNWL-666 +. 47 PAGES, JANUARY 10, 1968

ONE OF A SERIES. TOPICS ARE COOLANT CHEMISTRY DEVELOPMENT - CORROSION AND CHEMISTRY, EQUIPMENT
DEVELOPMENT - FACILITIES AND PROTOTYPES, INSTRUMENTATION, METALS AND MATERIALS DEVELOPMENT -
METALS IRRADIATION AND PROPERTIES - FABRICATION - CODES AND STANDARDS, PROCESS AND MATERIAL
SPECIFICATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CODES AND STANDARDS + FABRICATION + METAL + REACTOR, TEST + FFTF (TR) + CORROSION + COOLANT CHEMISTRY + DESIGN
STUDY + BATTELLE NORTHWEST + R AND D PROGRAM + REACTOR, LMFB + INSTRUMENTS, MISC.

18-033952 DATED 1968
WOODFIELD FW
FAST FLUX TEST FACILITY DEVELOPMENT MONTHLY PROGRESS REPORT
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-747 +. 53 PAGES, MARCH 1968

ONE OF A SERIES. TOPICS ARE COOLANT CHEMISTRY DEVELOPMENT - CORROSION AND CHEMISTRY, EQUIPMENT
DEVELOPMENT - FACILITIES AND PROTOTYPES, INSTRUMENTATION, METALS AND MATERIALS DEVELOPMENT -
METALS IRRADIATION AND PROPERTIES - FABRICATION - CODES AND STANDARDS, PROCESS AND MATERIAL
SPECIFICATIONS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CODES AND STANDARDS + FABRICATION + METAL + REACTOR, TEST + FFTF (TR) + CORROSION + COOLANT CHEMISTRY + DESIGN
STUDY + BATTELLE NORTHWEST + R AND D PROGRAM + REACTOR, LMFB + INSTRUMENTS, MISC.

18-033953 ALSO IN CATEGORY 11 DATED 1967
WOODFIELD FW
FAST FLUX TEST FACILITY DEVELOPMENT MONTHLY PROGRESS REPORT, NOVEMBER 1967
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-639 +. 58 PAGES, DECEMBER 10, 1967

ONE OF A SERIES. TOPICS - COOLANT SYSTEMS DEVELOPMENT, EQUIPMENT, INSTRUMENTATION, METALS AND
MATERIALS, AND PROCESS AND MATERIALS SPECIFICATIONS. FISSION-PRODUCT DIFFUSION THROUGH 304 SS
CLADDING AFTER 1052 HR AT 1200 F WAS LESS THAN 1 MIL. 304 SS IRRADIATED AT 1.7×10^{22} NVI

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-033953 *CONTINUED*
AT 1000 F REVEALED MARKED INCREASE IN 0.2% OFFSET YIELD STRENGTH AND A DECREASE IN DUCTILITY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + METAL + REACTOR, TEST + FISSION PRODUCT TRANSPORT + STEEL, STAINLESS + FFTF (TR) + CORROSION + COOLANT CHEMISTRY + DESIGN STUDY + MATERIAL + IRRADIATION TESTING + BATTELLE NORTHWEST + EQUIPMENT + R AND D PROGRAM + REACTOR, LMFBF + RADIATION EFFECT

18-033964 ALSO IN CATEGORY 6 DATED 1967
HAGAN JW + WORTH GA + PADILLA A
PARAMETRIC SCRAM TRANSIENT ANALYSES FOR THE FFTF FAST TEST REACTOR
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-657 +. 27 PAGES, NOVEMBER 1967

PARAMETRIC ANALYSES OF THE TRANSIENT CHARACTERISTICS FOR VARIOUS ASSUMED RAMP REACTIVITY INPUTS WHICH ARE TERMINATED BY TRIP OF REACTOR SAFETY SYSTEM HAVE BEEN PERFORMED AND RESULTS OF STUDIES ARE DESCRIBED. IN EACH CASE REACTOR WAS ASSUMED AT OPERATING CONDITIONS AT INITIATION OF RAMP. THESE RESULTS WILL BE USED IN EVALUATION OF TRANSIENT SENSITIVITIES TO VARIOUS DESIGN PARAMETERS FOR ESTABLISHING DESIGN TRADE-OFFS AND SYSTEM LIMITS. SYSTEM VARIABLES INCLUDED IN STUDY WERE, MAGNITUDE OF RAMP REACTIVITY INSERTION, SYSTEM RESPONSE TIME, SAFETY ROD ACCELERATION RATE, AND MAGNITUDE OF DOPPLER COEFFICIENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + REACTIVITY EFFECT + REACTOR STABILITY + REACTOR TRANSIENT + REACTOR, TEST + FFTF (TR) + REACTOR CONTROL + RESPONSE TIME + THEORETICAL INVESTIGATION + REACTOR PHYSICS + R AND D PROGRAM + REACTOR, LMFBF

18-033965 ALSO IN CATEGORY 11 DATED 1968
NASH CR + PECKINPAUGH CL
FAST FLUX TEST FACILITY EVALUATION OF CONCEPTUAL FEATURES FOR THE REACTOR REFUELING SYSTEM
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-683 +. 37 PAGES, MARCH 1968

PRESENTS RESULTS OF EVALUATION OF ALTERNATIVE METHODS OF FUEL HANDLING. THE SODIUM-COOLED-CASH CONCEPT APPEARS MOST PROMISING BECAUSE OF CAPACITY TO REMOVE DECAY HEAT, NO INTERFACE PROBLEMS ON REACTOR SYSTEM AS DOES A GAS COOLING SYSTEM, AND EXCELLENT HEAT TRANSFER PROPERTIES OF SODIUM WHICH ALLOW GREATER STRETCH CAPABILITY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + REFUELING + FUEL HANDLING MACHINE + REMOTE MANIPULATING AND VIEWING + R AND D PROGRAM + REACTOR, LMFBF

18-033966 DATED 1967
FINCH LM + PETERSON RE + TUYODA KG
EVALUATION OF FFTF REACTOR ASSEMBLY DEVELOPMENT AND TESTING PROGRAM
PACIFIC NORTHWEST LABORATORY, RICHLAND, WASHINGTON
BNWL-614 +. 293 PAGES, 5 FIGURES, 19 TABLES, NOVEMBER 1967

PRESENTS AN OVERALL REACTOR ASSEMBLY ENGINEERING DEVELOPMENT AND TESTING PROGRAM REQUIRED TO SUPPORT THE FFTF DESIGN AND CONSTRUCTION, WHICH INCLUDES IDENTIFICATION OF DESIGN PROBLEMS WHERE DATA MUST BE OBTAINED BY EXPERIMENTATION, DESCRIPTIONS AND TECHNICAL JUSTIFICATION OF PROPOSED TESTS, AN OVERALL TEST SCHEDULE COMMENSURATE WITH THE FFTF SCHEDULE, AND IDENTIFICATION OF TESTING FACILITIES TO BE USED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + R AND D PROGRAM + REACTOR, LMFBF

18-033968 DATED 1969
APPLEBY ER
POWER REACTOR DESIGNS
BATTELLE-NORTHWEST, RICHLAND, WASHINGTON
BNWL-936(VOL.2) +. 302 PAGES, JANUARY 1969

A COMPILATION OF DOMESTIC AND FOREIGN REACTORS BOTH CONCEPTUAL AND OPERATING. INFORMATION IS DESCRIPTIVE (GIVES OWNER, DESIGNER, TYPE COOLANT, TYPE CLADDING, CONTROL, CONTAINMENT TEMPERATURES AND PRESSURES, ETC.). TYPES OF REACTORS LISTED ARE LIQUID-METAL REACTORS, ORGANIC COOLED, PWR, STEAM OR SUPERCRITICAL-WATER COOLED, AND NUCLEAR AUXILIARY POWER SYSTEM. INDEX IS INCLUDED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-033968 *CONTINUED*

BIBLIOGRAPHY + REACTOR, LMCR + REACTOR, PWR + REACTOR, POWER + DESIGN STUDY + BATTELLE NORTHWEST + INFORMATION RETRIEVAL + DIRECT ENERGY CONVERSION DEVICES + R AND D PROGRAM + COMPARISON, REACTOR CHARACTERISTICS + REACTOR, LMFBR + REACTOR, SCBR

18-033976

DATED 1965

LEIPUNSKI AI + PINKHASIK MS + BAGDASARYEV YE + MILOVIDOV IV + POIDO MS + KAZATCHKOVSKI OD + KRASNOYAROV NV + TROYANOV MF + AFRIKANTOV II + STEKOLINIKOV VV

NUCLEAR POWER PLANT BN-350

INSTITUTE OF PHYSICS AND POWER ENGINEERING, CITY OB OBNINSK

AEC-IR-6663 +. 31 PAGES, 6 REFERENCES, 1965, PRESENTED AT THE ANS MEETING, APRIL 26-28, 1965, DETROIT, MICHIGAN

DESCRIBES THE SYSTEMS OF THE BN-350 SODIUM-COOLED FAST REACTOR RATED AT 1000 MWt/150 MWe AND 1200 TONS OF PROCESS STEAM PER HOUR. LOCATED ON NE SHORE OF THE CASPIAN SEA. ROD-TYPE FUEL ELEMENTS ARE USED (PUO2-UO2, CLAD IN 0.4-MILLIMETER STAINLESS STEEL).

REACTOR DESCRIPTION + REACTOR, POWER + USSR + R AND D PROGRAM + REACTOR, LMFBR

18-033977

ALSO IN CATEGORIES 5 AND 11

DATED 1967

DENIELOU G + ROZENHOLC M + STORRER F + VAUTREY + VELLENEUVE J

PHENIX - MAIN SAFETY CRITERIA - CONSEQUENCES FOR THE REACTOR DESIGN

CEA-CEN/CADARACHE/DRP, CEA-CEN/SACLAY/DRP, EURATOM-CEN/CADARACHE/DRP

12 PAGES, 3 FIGURES, 4 REFERENCES, PAPER PRESENTED AT THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 19-22, 1967

DISCUSSES INFLUENCE OF SAFETY CONSIDERATIONS ON THE DESIGN OF PHENIX PROTOTYPE. DISCUSSES CORE COOLING FOR VARIOUS ACCIDENTS, CONTAINMENT OF FISSION PRODUCTS, AND MEANS FOR DETECTING LOCAL REACTIVITY AND TEMP. MEASUREMENTS.

AVAILABILITY - F. STORRER, EURATOM-CEN/CADARACHE/DRP

DESIGN CRITERIA + FRANCE + SODIUM + DESIGN STUDY + SAFETY PRINCIPLES AND PHILOSOPHY + REACTOR, LMFBR

18-033983

ALSO IN CATEGORY 6

DATED 1968

ROY DH

FAST BREEDER REACTOR STATIC PHYSICS METHODS DEVELOPMENT AND ANALYSIS PROJECT. QUARTERLY TECHNICAL REPORT, APRIL-JUNE 1968

BABCOCK AND WILCOX COMPANY, LYNCHBURG, VA.

BAW-3867-4 +. 47 PAGES, 4 FIGURES, 21 TABLES, SEPTEMBER 1968

ONE OF A SERIES OF SIMILAR REPORTS ON B AND W PROGRAM OF ANALYTICAL METHODS DEVELOPMENT APPLICABLE TO THE PHYSICS DESIGN OF FAST-BREEDER REACTORS. TOPICS - CROSS SECTION PREPARATION AND CODE DEVELOPMENT, BURNUP STUDIES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

COMPUTER PROGRAM + REACTOR, FAST + REACTOR, LMCR + ANALYTICAL TECHNIQUE + REACTOR PHYSICS + R AND D PROGRAM + REACTOR, LMFBR

18-033985

DATED 1967

GAST K + SCHLECHTENDAHL EG

FAST SODIUM-COOLED REACTOR NA-2

KERNFORSCHUNGSZENTRUM, KARLSRUHE, WEST GERMANY

EURFNR-462 + EUR-3706 +. 377 PAGES, OCTOBER 1967

DISCUSSES GROUND WORK FOR DESIGN AND DEVELOPMENT WORK TO BE CARRIED OUT IN NEXT FEW YEARS ON A FAST SODIUM-COOLED PROTOTYPE REACTOR WITH POWER OF 300 MWe. THE PURPOSE IS TO PROVIDE COMPREHENSIVE SURVEY OF VARIOUS INDIVIDUAL PROBLEMS INVOLVED ON THE BASIS OF A CONSISTENT DESIGN, TO MAKE PROPOSALS TO THEIR TECHNICAL SOLUTION AND TO LAY DOWN GUIDE LINES FOR THE WORK STILL NECESSARY. THE PROTOTYPE DESIGN WILL DIFFER FROM DESIGN COVERED HERE OWING TO FURTHER TECHNICAL PROGRESS MADE.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DESIGN STUDY + GERMANY + R AND D PROGRAM + REACTOR, LMFBR

18-033987

DATED 1968

BUTTREY K + HILLIG OR + MAGEE PM + OTTEWITTE EH

LIQUID METAL FAST BREEDER REACTOR (LMFBR) TASK FORCE FUEL CYCLE STUDY

ATOMICS INTERNATIONAL

NAA-SR-MEMO 12604 +. 177P., 9 TABLES, 11 FIGURES, REFERENCES, JAN. 10, 1968

DESCRIBES A SERIES OF CALCULATIONS TO DETERMINE FUEL CYCLE REQUIREMENTS FOR LIQUID-METAL-COOLED FAST BREEDER REACTORS FUELED WITH PUO2-UO2.

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-033987 *CONTINUED*

AVAILABILITY - ATOMICS INTERNATIONAL, A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION

FUEL BURNUP + DESIGN STUDY + MATHEMATICAL STUDY + R AND D PROGRAM + REACTOR, LMFBR + FUEL MANAGEMENT

18-033989

DATED 1967

PFLASTERER GR

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 12, FEB.-APRIL 1967

GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.

GEAP-5498 +. 64 PAGES, 16 FIGURES, MAY 1967

ONE OF A SERIES OF SIMILAR REPORTS. TOPICS - EXPERIMENTAL PROGRAM IN ZPR-2, FUEL DEVELOPMENT, INSTRUMENTATION, TRANSIENT CHARACTERISTICS OF CORE, REACTIVITY CONTROLS, FUEL HANDLING, CORE DESIGN, FUEL FABRICATION, AND PLANT CONSTRUCTION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DOPPLER COEFFICIENT + BOR 1 AND 2 (RE) + FABRICATION + REACTOR, FAST + FUEL HANDLING + REACTOR, LMCR + REACTIVITY COEFFICIENT + REACTOR, TEST + CONTAINMENT CONSTRUCTION + REACTOR CONTROL + DESIGN STUDY + IRRADIATION TESTING + REACTOR, PULSED + SEFOR (RE) + R AND D PROGRAM + REACTOR, LMFBR + FUEL ELEMENTS

18-033990

DATED 1968

PFLASTERER GR

SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 15, NOVEMBER 1967-JANUARY 1968

GENERAL ELECTRIC, SUNNYVALE, CALIF.

GEAP-5595 +. 19 PAGES, FEBRUARY 1968

ONE OF A SERIES OF QUARTERLY REPORTS. TOPICS - EXPERIMENTAL PROGRAM, SPECIAL INSTRUMENTATION, REACTIVITY CONTROLS, FUEL FABRICATION, STARTUP TESTING, TRAINING, AND PLANT CONSTRUCTION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

DOPPLER COEFFICIENT + FABRICATION + REACTOR, FAST + REACTOR, LMCR + REACTIVITY COEFFICIENT + TRAINING + CONTAINMENT CONSTRUCTION + REACTOR CONTROL + REACTOR STARTUP TESTING + REACTOR, PULSED + SEFOR (RE) + R AND D PROGRAM + REACTOR, LMFBR + INSTRUMENTS, MISC. + FUEL ELEMENTS

18-033991

ALSO IN CATEGORY 11

DATED 1967

FIELD JH + JOHNSON ML + NOVAK PE

EVALUATION OF THE EFFECT OF DESIGN AND OPERATING VARIABLES ON SEFOR FUEL AND FUEL CLADDING

GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.

GEAP-5309 +. 72 PAGES, REFERENCES, DECEMBER 1967

FOURTEEN CAPSULE EXPERIMENTS WERE IRRADIATION-TESTED TO INVESTIGATE DESIGN PARAMETERS AND EXPECTED OPERATING CONDITIONS ON SEFOR FUEL. TESTS WITH REFERENCE DESIGN CLADDING THICKNESS AND FUEL-CLADDING GAPS SUBJECTED TO EXPECTED STEADY-STATE AND POWER OSCILLATION CONDITIONS RESULTED IN NO CLADDING DISTORTION. SIMILAR SPECIMENS SUBJECTED TO TREAT TRANSIENTS TO NEAR FUEL MELTING EXPERIENCED MINOR CLADDING RIDGES AT FUEL PELLET INTERFACES. CLADDING MAINTAINED ITS INTEGRITY DURING ALL TESTS INCLUDING NON-STANDARD CONDITIONS OF OPERATION ABOVE MELTING OF FUEL, REDUCED FUEL-CLADDING GAPS, AND SODIUM INGRESS INTO FUEL PRIOR TO TRANSIENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

CLADDING + REACTOR, FAST + FUEL BURNUP + REACTOR, LMCR + REACTOR, SAFETY RESEARCH + TREAT (PRR) + IRRADIATION TESTING + REACTOR, PULSED + SEFOR (RE) + STRUCTURAL INTEGRITY + PERFORMANCE LIMIT + FUEL ELEMENTS

18-033992

DATED 1968

COCHKAN JT

SEFOR CORE CLAMPING MODEL TESTS ONE-SIXTH SEGMENT

GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.

GEAP-5465 +. 62 PAGES, MARCH 1968

DESCRIBES RESULTS OF EXPERIMENTS WITH THE CLAMPING MECHANISM FOR SEFOR CORE. THE CLAMPING SYSTEM TESTED RESULTED IN A TIGHT CORE STRUCTURE WHICH HAD NO LOOSE CHANNELS IN IT BUT WHICH WAS NOT DEFORMED SO AS TO PRECLUDE INDIVIDUAL FUEL ROD REMOVAL FROM OR INSERTION INTO THE CHANNELS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

GUIDE + REACTOR, FAST + REACTOR, LMCR + FASTENER + TEST, PREOPERATIONAL + TEST, SYSTEM OPERABILITY + REACTOR, PULSED + SEFOR (RE) + CORE COMPONENTS + FUEL ELEMENTS

18-033993

ALSO IN CATEGORY 11

DATED 1968

SODIUM-COOLED REACTORS - FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. TWENTY-SIXTH QUARTERLY REPORT, FEB.-APR.

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-033993 *CONTINUED*
1968

GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIFORNIA
GEAP-5631 +. 121 PAGES, REFERENCES, JUNE 1968

ONE OF A SERIES OF REPORTS ON R-AND-D PROGRAM TO DEVELOP FAST REACTORS EMPLOYING CERAMIC FUELS (MIXED PLUTONIUM-URANIUM OXIDE). TOPICS - FUEL-BEHAVIOR STUDIES, PERFORMANCE EVALUATION, FAST-FLUX IRRADIATION TESTING, AND CLADDING DEVELOPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CERAMICS + CLADDING + FUEL BURNUP + OXIDE + PLUTONIUM + URANIUM + IRRADIATION TESTING + PERFORMANCE LIMIT + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

18-033994 ALSO IN CATEGORY 11 DATED 1968
SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. TWENTY-FIFTH QUARTERLY REPORT, NOV. 1967-JANUARY 1968
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.
GEAP-5584 +. 112 PAGES, REFERENCES, MARCH 1968

ONE OF A SERIES OF REPORTS ON R-AND-D PROGRAM TO DEVELOP FAST REACTORS EMPLOYING CERAMIC FUELS (MIXED PLUTONIUM-URANIUM OXIDE). TOPICS - FUEL-BEHAVIOR STUDIES, PERFORMANCE EVALUATION, FAST-FLUX IRRADIATION TESTING, AND CLADDING DEVELOPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CERAMICS + CLADDING + FUEL BURNUP + OXIDE + PLUTONIUM + URANIUM + IRRADIATION TESTING + PERFORMANCE LIMIT + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

18-033995 ALSO IN CATEGORY 11 DATED 1967
SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. TWENTY-FOURTH QUARTERLY REPORT, AUGUST-OCTOBER 1967
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.
GEAP-5541 +. 135 PAGES, REFERENCES, DECEMBER 1967

ONE OF A SERIES OF REPORTS ON R-AND-D PROGRAM TO DEVELOP FAST REACTORS EMPLOYING CERAMIC FUELS (MIXED PLUTONIUM-URANIUM OXIDE). TOPICS - FUEL-BEHAVIOR STUDIES, PERFORMANCE EVALUATION, FAST-FLUX IRRADIATION TESTING, AND CLADDING DEVELOPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CERAMICS + CLADDING + FUEL BURNUP + OXIDE + PLUTONIUM + URANIUM + IRRADIATION TESTING + PERFORMANCE LIMIT + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

18-033996 ALSO IN CATEGORY 11 DATED 1966
SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM. QUARTERLY REPORT NO. 19, MAY-JULY 1966
GENERAL ELECTRIC COMPANY, SAN JOSE, CALIF.
GEAP-5198 +. 95 PAGES, REFERENCES, AUGUST 1966

ONE OF A SERIES OF REPORTS ON R-AND-D PROGRAM TO DEVELOP FAST REACTORS EMPLOYING CERAMIC FUELS (MIXED PLUTONIUM-URANIUM OXIDE). TOPICS - FUEL-BEHAVIOR STUDIES, PERFORMANCE EVALUATION, FAST-FLUX IRRADIATION TESTING, AND CLADDING DEVELOPMENT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

CERAMICS + CLADDING + FUEL BURNUP + OXIDE + PLUTONIUM + URANIUM + IRRADIATION TESTING + PERFORMANCE LIMIT + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

18-033999 DATED 1967
1000-MWE LMFBR FOLLOW-ON STUDY. QUARTERLY PROGRESS REPORT, JUNE 1-AUGUST 31, 1967
BABCOK AND WILCOX COMPANY, LYNCHBURG, VA.
BAW-1312 +. 184 PAGES, 33 FIGURES, 9 TABLES, AUGUST 1967

ONE OF A SERIES OF SUCH REPORTS WITH REVISED OBJECTIVE WHICH IS TO OBTAIN INFORMATION BEYOND CURRENT TECHNOLOGY, FORMULATE R AND D PROGRAMS BEYOND THOSE EXISTING, TO ESTIMATE PROBABILITY OF SUCCESSFUL ACHIEVEMENT, AND TO ASSESS RELATIVE VALUE OF PERFORMING R AND D IN EACH AREA.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00 COPY, \$0.65 MICROFICHE

DESIGN STUDY + R AND D PROGRAM + REACTOR, LMFBR

18-034632 DATED 1969

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-034632 *CONTINUED*

EADO Y

FAST BREEDER DEVELOPMENT PROGRAM IN JAPAN

CENTRAL RESEARCH INSTITUTE OF ELECTRIC POWER INDUSTRY

2 PAGES, NUCLEAR ENGINEERING INTERNATIONAL, 14, PAGES 413-414 (MAY 1969)

IN 1966, SPECIAL COMMITTEE FOR POWER REACTOR DEVELOPMENT PROJECTS STARTED GATHERING TOP-LEVEL PEOPLE FROM UTILITY, MANUFACTURING, ACADEMIC, AND GOVERNMENT GROUPS TO STUDY EBR PROGRAM, SAFETY, FUEL, SODIUM TECHNOLOGY AND HAVE DEVELOPED SCHEDULE FOR COMMERCIAL FAST BREEDER REACTORS IN 1980S. COMMITTEE WAS ENLARGED AND ABSORBED ATOMIC FUEL CORP. TO FORM POWER REACTOR AND NUCLEAR DEVELOPMENT CORP., WHICH IS NOW RESPONSIBLE FOR NATIONAL FBR EFFORT. MOST SERIOUS PROBLEM IS HOW TO GATHER EXPERIENCED EXPERTS WITHOUT JEOPARDIZING MANUFACTURING SUCH AS LIGHT-WATER REACTORS. JAPAN FERMI REACTOR COMMITTEE RUN BY CENTRAL RESEARCH INSTITUTE OF ELECTRIC POWER INDUSTRY HAS STATIONED MORE THAN 20 ENGINEERS IN DETROIT IN EFFORT TO CLOSE EXISTING TECHNOLOGY GAP THROUGH INTERNATIONAL COOPERATIVE ARRANGEMENTS.

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + REACTOR, POWER + JAPAN + REACTOR, LMFBR

18-034634

DATED 1968

BENJAMIN M

INTRODUCTORY SURVEY OF BASIC DESIGN AND FUEL-ELEMENT CONSIDERATIONS FOR ALRGE, FAST SODIUM-COOLED REACTORS

DEPT. OF NUCLEAR ENGINEERING, IOWA STATE UNIV., AMES, IOWA

11 PAGES, 12 FIGURES, 5 TABLES, 14 REFERENCES, NUCLEAR ENGINEERING AND DESIGN, 7, PAGES 399-410 (MAY 1968)

BASIC DESIGN AND FUEL ELEMENT CONSIDERATIONS IN THE DEVELOPMENT OF LARGE, FAST BREEDER POWER REACTORS ARE ANALYZED ON BASIS OF EXPERIMENTAL AND CALCULATED RESULTS. DOPPLER REACTIVITY EFFECT, SODIUM VOID (OR LOSS) REACTIVITY EFFECT AND STRUCTURAL EXPANSION REACTIVITY EFFECT ARE ANALYZED. BREEDING RATIO AND GAIN, AND ALSO DOUBLING TIME OF FUEL PRODUCTION ARE PRESENTED. THERMAL AND STRUCTURAL CONSIDERATIONS OF FUEL ELEMENT DESIGN ARE OUTLINED. FOR ECONOMY, AS WELL AS BREEDING, DEVELOPMENT OF A CERAMIC (ESPECIALLY CARBIDE) PLUTONIUM FUEL SYSTEM WITH HIGHER POWER DENSITY AND SPECIFIC POWER FOR THE LARGE, FAST POWER REACTOR SYSTEM IS DESIRABLE.

REACTOR, BREEDER + CARBIDE + CERAMICS + DOPPLER COEFFICIENT + DOPPLER EFFECT + REACTIVITY EFFECT, EXPANSION + REACTOR, FAST + FISSION GAS RELEASE + REACTOR, LARGE + PLUTONIUM + SODIUM COEFFICIENT + URANIUM + VOID COEFFICIENT + REACTOR, POWER + CORE MELTDOWN + FUEL EXPANSION COEFFICIENT + SAFETY ANALYSIS + REACTIVITY EFFECT, VOID + REACTOR, LMFBR

18-034636

ALSO IN CATEGORY 17

DATED 1968

SUMMARY OF TREAT SODIUM LOOP EXPERIMENTS ON BEHAVIOR OF SINGLE EBR-II MARK I PINS UNDER ACCIDENT CONDITIONS

12 PAGES, 10 FIGURES, 1 TABLE, 31 REFERENCES, NUCLEAR ENGINEERING AND DESIGN, VOL. 7, PAGES 442-454 (MAY 1968)

PRESENTED IS A STUDY, EXPERIMENTALLY AND BY COMPUTATION, OF MECHANISMS OF FUEL FAILURE AND ASSOCIATED MOVEMENT OF FUEL AND COOLANT DURING CORE MELTDOWN ACCIDENT. THESE TWO PHENOMENA COULD LEAD TO EXTENSION OF ANY INITIAL FUEL FAILURE AND WOULD PROBABLY BE FACTORS CAUSING SERIOUS MELTDOWN. *** FUEL FAILURE EXPERIMENTS IN INSTRUMENTED SODIUM LOOPS WERE CONDUCTED AT VARIOUS CONDITIONS OF TEMP. AND FLOW OF SODIUM COOLANT. POWER BURSTS OF THE TREAT REACTOR, INTO WHICH LOOPS WERE INSERTED, WAS ADJUSTED TO SPECIFIED INTENSITIES. RESULTS OF TESTS AND ANALYSIS ARE PRESENTED.

COMPUTER PROGRAM + EBR I AND 2 (RE) + REACTOR, FAST + FAILURE, FUEL ELEMENT + IN PILE EXPERIMENT + REACTOR, LMCR + SODIUM + COMPARISON, THEORY AND EXPERIENCE + REACTOR, TEST + TREAT (PRR) + ANALYTICAL MODEL + CORE MELTDOWN + IN PILE EXPERIMENT + REACTOR, PULSED + MATHEMATICAL TREATMENT

18-034655

DATED 1969

BALENT R + FALCON JA

ATOMICS INTERNATIONAL FAST BREEDER REACTOR PROGRAM

ATOMICS INTERNATIONAL, CANOGA PARK, CALIFORNIA

NP-17784 +. 16 PAGES, 1968

ATOMICS INTERNATIONAL HAS COMPLETED CONCEPTUAL DESIGN OF 500-MWE SODIUM-COOLED DEMONSTRATIONS FBR PLANT. CONTRACT HAS BEEN SIGNED WITH GENERAL PUBLIC UTILITIES CORP (N.Y.) FOR PARTICIPATION IN DEVELOPMENT AND TEST PROGRAM LEADING TO DECISION BY 1970 FOR CONSTRUCTION OF A 350- TO 500-MWE PLANT IN NORTHEASTERN PENNSYLVANIA. SIGNIFICANT STUDIES INVOLVED IN SELECTION OF SYSTEMS AND COMPONENTS ALONG WITH DESCRIPTION OF PLANT ARE REPORTED. OVERALL ENGINEERING AND TEST PROGRAM IS OUTLINED. FUTURE 1000-MWE AND LARGER FBR PLANTS WILL INCORPORATE MANY OF SAME FEATURES OF DEMONSTRATION PLANT.

REACTOR, BREEDER + CONTAINMENT + REACTOR, FAST + REACTOR DESCRIPTION + SODIUM + URANIUM OXIDE + PLUTONIUM OXIDE + REACTOR, POWER + AI + DESIGN STUDY + REACTOR, LMFBR + PUMPS

18-034658

DATED 1967

THE ANALYSIS OF MIXED CARBIDE REACTORS FOR THE AEC LMFBR TASK FORCE STUDIES. FINAL REPORT

WESTINGHOUSE ELECTRIC CORP., MADISON, PA. ADVANCED REACTORS DIV.

WCAP-3791-11 +. 76 PAGES, REFERENCES, OCTOBER 1967

TWO CONCEPTUAL DESIGNS FOR 1000-MWE, MIXED URANIUM-PLUTONIUM CARBIDE REACTORS ARE PRESENTED, ONE IS A REFERENCE REACTOR BASED ON 1980 TECHNOLOGY, AND THE OTHER IS AN ADVANCED REACTOR PREDICATED ON TECHNOLOGY POSTULATED FOR 1995. FOR EACH DESIGN THE PHYSICS PARAMETERS REQUIRED TO CALCULATE FUEL COSTS OVER THE LIFE OF THE PLANT HAVE BEEN CALCULATED, INCLUDING THE FEED AND DISCHARGE

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-034658 *CONTINUED*

MASSSES OF THE URANIUM AND PLUTONIUM ISOTOPES FOR EACH CYCLE, FROM THE FIRST TO EQUILIBRIUM. DESIGN HAS FIVE IDENTICAL MODULES USED TO MAKE THE REACTIVITY EFFECT OF CCSODIUM VOIDING APPROXIMATELY ZERO. NONVENTED FUEL RODS HAVE AVERAGE DISCHARGE BURNUP OF 80,000 MWD/T. CHROMIUM-STABILIZED, MIXED CARBIDE FUEL IS USED AND THE FUEL PELLETS ARE SODIUM-BONDED TO THE STAINLESS STEEL CLADDING. A LIMIT OF 2350 DEG. F FOR FUEL CENTER TEMPERATURE IS USED, ALONG WITH A LIMIT OF 1300 DEG. F FOR THE CLADDING SURFACE TEMPERATURE. THE MIXED-MEAN SODIUM OUTLET TEMPERATURE IS 1000 DEG. F.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL BURNUP + HYDRAULIC ANALYSIS + REACTOR, LMCR + THERMAL ANALYSIS + DESIGN STUDY + REACTOR, LMFBR

18-034662

DATED 1967

SMITH OL + PERRY AM

SOME CALCULATIONS PERFORMED ON A SERIES OF CONCEPTUAL FAST-NEUTRON REACTORS FUELED WITH HIGHLY ENRICHED UO-2 OR PUO-2 AND REFLECTED WITH CARBON, BERYLLIUM METAL, OR BERYLLIUM OXIDE

OAK RIDGE NATIONAL LAB., TENN.

ORNL-TM-1620 +. 36 PAGES, 11 FIGURES, 8 TABLES, FEBRUARY 1967

AS PART OF THE INITIAL DESIGN PHASE OF THE MPRE PROGRAM, THIS REPORT PRESENTS A NEUTRONIC FEASIBILITY STUDY OF A COMPACT, LIGHT-WEIGHT REACTOR FUELED WITH HIGHLY ENRICHED URANIUM OR PLUTONIUM, COOLED WITH BOILING POTASSIUM AND REFLECTED WITH BERYLLIUM OXIDE. AT THE TIME THIS STUDY WAS MADE A SERIES OF BERYLLIUM- AND CARBON-REFLECTED, UO-2-FUELED CRITICAL EXPERIMENTS WAS ANTICIPATED. THE FIRST PART OF THIS REPORT DESCRIBES A SET OF ONE-DIMENSIONAL CALCULATIONS PERFORMED TO GIVE BASIC NEUTRONIC DATA FOR USE IN DESIGNING THE EXPERIMENTS. THE SECOND PART GIVES THE RESULTS OF A GENERALIZED PARAMETRIC INVESTIGATION OF THE BASIC REACTOR CONCEPT.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA., 22151, \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, AIRCRAFT + REACTOR, LMCR + POTASSIUM + REACTOR COOLANT + ORNL + DESIGN STUDY

18-034668

DATED 1969

STOKRS CL + STAKER WP + NOYES RC

THE HIGH PERFORMANCE BREEDER AS A PRIMARY DEVELOPMENT GOAL

COMBUSTION ENGINEERING, INC., WINDSOR, CONNECTICUT

8 PAGES, 6 FIGURES, 5 TABLES, 2 REFERENCE, PRESENTED AT THE AMERICAN POWER CONFERENCE, CHICAGO, ILLINOIS, APRIL 22-24, 1969

CONCLUDES THAT THE PRINCIPAL REQUIREMENTS FOR LMFBR SAFETY ARE - (1) A STABLE CORE WITH PROTECTIVE AND COOLING SYSTEMS OF HIGH RELIABILITY. (2) THE PREFERRED FUEL SYSTEM IS HYPERSTOICHIOMETRIC SODIUM-BONDED MIXED PLUTONIUM-URANIUM CARBIDE. (3) THE PREFERRED CORE CONFIGURATION IS A MODERATELY PANCACKED CYLINDER WITH A HEIGHT-TO-DIAMETER RATIO OF ABOUT 1 TO 4. (4) THE CHARACTERISTICS AND PERFORMANCE EXPECTED OF 1980 REACTORS ARE - FUEL CYCLE COST (0.38 MILL/KWH), DOUBLING TIME (7 YEARS), AND SPECIFIC INVENTORY (3.5 KG FISSILE/MWE). REPL 34668 ABSTCONCLUDES THAT THE PRINCIPAL REQUIREMENTS FOR LMFBR SAFETY ARE - (1) A STABLE CORE WITH PROTECTIVE AND COOLING SYSTEMS OF HIGH RELIABILITY. (2) THE PREFERRED FUEL SYSTEM IS HYPERSTOICHIOMETRIC SODIUM-BONDED MIXED PLUTONIUM-URANIUM CARBIDE. (3) THE PREFERRED CORE CONFIGURATION IS A MODERATELY PANCACKED CYLINDER WITH A HEIGHT-TO-DIAMETER RATIO OF ABOUT 1 TO 4. (4) THE CHARACTERISTICS AND PERFORMANCE EXPECTED OF 1980 REACTORS ARE - FUEL CYCLE COST (0.38 MILL/KWH), DOUBLING TIME (7 YEARS), AND SPECIFIC INVENTORY (3.5 KG FISSILE/MWE).

REACTOR, BREEDER + REACTOR, FAST + DESIGN STUDY + REACTOR, LMFBR

18-034669

ALSO IN CATEGORIES 6 AND 11

DATED 1969

ROSSIN AD + GLASSNER A

REACTOR DEVELOPMENT PROGRESS REPORT

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ANL-7553 +. 123 PAGES, FIGURES, TABLES, REFERENCES, FEB. 1969

ONE OF A SERIES OF SIMILAR REPORTS. TOPICS - LMFBR DEVELOPMENT STUDIES, FAST REACTORS, GENERAL DEVELOPMENT STUDIES, NUCLEAR SAFETY, FUEL-MELTDOWN STUDIES WITH TREAT, CORE STRUCTURAL SAFETY, AND CONTAINMENT RESPONSE TO HIGH-ENERGY EXCURSION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + FUEL MELTDOWN + REACTOR, SAFETY RESEARCH + REACTOR, RESEARCH + TREAT (PRR) + CONTAINMENT INTEGRITY + EXCURSION, LARGE + REACTOR, PULSED + PRESSURE, INTERNAL + CORE COMPONENTS + STRUCTURAL INTEGRITY + R AND D PROGRAM + REACTOR, LMFBR

18-034673

ALSO IN CATEGORIES 7 AND 11

DATED 1969

PATRIARCA P + RUCKER DJ

FUELS AND MATERIALS DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1968

OAK RIDGE NATIONAL LAB., TENN.

ORNL-4390 +. 308 PAGES, FIGURES, TABLES, REFERENCES, MAY 1969

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-034673 *CONTINUED*

ONE OF A SERIES OF SIMILAR REPORTS. TOPICS - FUELS AND CLADDINGS, COOLANTS, RADIATION DAMAGE TO STRUCTURAL MATERIALS, TECHNIQUES OF FABRICATION AND TESTING, ENGINEERING PROPERTIES OF REACTOR MATERIALS, DEVELOPMENT OF FAST REACTOR OXIDE FUELS, AND LMFBR FUELS, FISSION-GAS RELEASE, ZIRCONIUM METALLURGY, ALKALI-METAL CORROSION, TUNGSTEN METALLURGY, AND CLAD MATERIALS FOR SPACE HEAT SOURCES.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

FABRICATION + CLADDING + REACTOR, FAST + FISSION GAS RELEASE + ZIRCONIUM + ORNL + CORROSION + TUNGSTEN + IRRADIATION TESTING + DIRECT ENERGY CONVERSION DEVICES + R AND D PROGRAM + REACTOR, LMFBR + FUEL COOLANT INTERACTION + FUEL ELEMENTS

18-034765

DATED 1969

THE CONTROVERSY OVER THE FUTURE DEVELOPMENT OF THE GERMAN FAST BREEDER

35 PAGES, FIGURES, TABLES, REFERENCES, ATOMWIRTSCHAFT - ATOMTECHNIK, PG. 187-220, APRIL 1969

THE RESULT OF A REAPPRAISAL WAS THAT ON 5/2/69 THE GERMAN MINISTER FOR SCIENTIFIC RESEARCH DECIDED THAT THE STEAM-COOLED, FAST BREEDER PROJECT SHOULD BE STOPPED, THAT THE WORK IN INDUSTRY SHOULD BE BROUGHT TO AN END, AND THAT ONLY A FEW SPECIAL DEVELOPMENTS, MAINLY IN CONNECTION WITH FUEL RODS, SHOULD CONTINUE WITHIN THE BASIC PROGRAM OF THE NUCLEAR RESEARCH CENTRE KARLSRUHE.

GERMANY + R AND D PROGRAM + REACTOR, LMFBR + REACTOR, SCBR

18-034767

DATED 1969

HAFFEL W

THE IMMEDIATE FUTURE OF THE SODIUM-COOLED BREEDER

3 PAGES, 1 TABLE, 6 REFERENCES, ATOMWIRTSCHAFT-ATOMTECHNIK, PAGES 174-176 (APRIL 1969)

AFTER THE DECISION TO DROP THE DEVELOPMENT WORK FOR A STEAM-COOLED, FAST BREEDER REACTOR AS AN INDEPENDENT PROJECT, EFFORT IS CONCENTRATING ON THE DEVELOPMENT LINE OF SODIUM-COOLED FAST BREEDERS. NATURALLY THE FIRST PROTOTYPE REACTOR IS IN THE FOREFRONT OF INTEREST. THE DESIGN WORK IS THE RESPONSIBILITY OF THE CONSORTIUM SIEMENS/INTERATOM-BELGONUCLEAIRE/NERATOM AND IS AT AN ADVANCED STAGE. AT THE END OF 1969 OR THE BEGINNING OF 1970 THE CONSORTIUM WILL BE IN A POSITION TO DELIVER A FIRM QUOTATION.

GERMANY + R AND D PROGRAM + REACTOR, LMFBR

18-035039

ALSO IN CATEGORY 6

DATED 1969

STEWART JE

PARAMETRIC EVALUATION OF THE INFLUENCE OF ²³⁹PU AND FISSION PRODUCT CAPTURE CROSS SECTIONS ON LMFBR PERFORMANCE

BABCOCK AND WILCOX, LYNCHBURG, VA.

BAW-3867-7 +. 69 PAGES, TABLES, REFERENCES, MARCH, 1969

THIS STUDY INVESTIGATED LARGE UNCERTAINTIES, OVER THE ENTIRE SPECTRUM, IN THE CAPTURE CROSS SECTIONS OF ^{PU-239} AND LUMPED PLUTONIUM FISSION FRAGMENTS. EMPHASIS WAS PLACED ON DETERMINING THE EFFECTS OF THE UNCERTAINTIES ON THE DEPLETION AND FUEL MANAGEMENT CHARACTERISTICS OF TWO LMFBR CORE SIZES. COMPARISONS WERE MADE FOR THE APPROACH TO EQUILIBRIUM AND PARTICULARLY FOR THE EQUILIBRIUM CYCLE ITSELF.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

FUEL BURNUP + MEASUREMENT + PLUTONIUM + NEUTRON + CROSS SECTION + REACTOR PHYSICS + R AND D PROGRAM + REACTOR, LMFBR + FUEL MANAGEMENT

18-035869

DATED 1967

BOYER RL

SYSTEM DESCRIPTION REPORT. TASK I, CONCEPT I. 1000-MWE LMFBR FOLLOW-ON STUDY

BABCOCK AND WILCOX CO., LYNCHBURG, VA. ATOMIC ENERGY DIV.

BAW-1316(VOL.2) +. 157 PAGES, FIGS., TABS., REFS., JUNE 1967

THE CONCEPT OBSERVES THE GROUND RULES OF THE LMFBR FOLLOW-ON PROGRAM AND IS BASED ON THE ESTIMATED STATE OF TECHNOLOGY IN 1980. THE DESIGN IS TO PRODUCE 2649 MWT/1033 MWE (39% EFFICIENCY). THE CORE IS CYLINDRICAL, FUEL IS VIBRATORY COMPACTED UO₂-PUO₂, CLADDING IS 16-MIL 19-9 DL. SIX INTERMEDIATE HEAT EXCHANGERS AND 6 PUMPS ARE LOCATED RADIALLY OUTSIDE THE BIOLOGICAL SHIELD PITS. VESSEL IS WELDED STAINLESS STEEL 304 PLATE 1-5/8 IN. THICK. A FUEL MELTDOWN PAN IS PROVIDED (16 FT 9 IN. IN DIAMETER, 3 IN. THICK, AND 18 IN. DEEP). CONTROL IS BY BORON CARBIDE RODS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR DESCRIPTION + ANAL + DESIGN STUDY + R AND D PROGRAM + REACTOR, LMFBR

18-035870

DATED 1969

1000-MWE FAST BREEDER REACTOR FOLLOW-ON STUDY - TASK I FINAL REPORT

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-035870 *CONTINUED*
WESTINGHOUSE ELECTRIC CORPORATION ADVANCED REACTORS DIVISION
WARD-2000-33 +. PAGES, FIGURES, TABLES, REFERENCES, JANUARY 1969

CONCEPTUAL DESIGN STUDIES ARE REPORTED FOR A 1000-MWE LIQUID METAL FAST BREEDER REACTOR. TECHNICAL AND ECONOMIC COMPARISONS (TRADE-OFF STUDIES) LEAD TO CHOICE OF - A FOUR-MODULE CORE, FUEL VENTED TO COOLANT, COMBINED HOT-CELL REFUELING AND CONTAINMENT, TWO HEAT-REMOVAL LOOPS, A PIPED PRIMARY SYSTEM, AND A NON-REHEAT STEAM CYCLE. ALSO DISCUSSED ARE FUEL AND MATERIALS SELECTION, SODIUM PROPERTIES, FUEL ELEMENT DESIGN, AND PRELIMINARY DESIGN-BASIS ACCIDENT ANALYSES. THE RESULTANT COMPOSITE OF ALL OF THESE DESIGN SELECTIONS IS REFERRED TO HEREIN AS THE TASK 1 REFERENCE DESIGN, WHICH FORMS THE BASIS FOR THE DESIGN OF AN INTEGRATED PLANT TO BE PERFORMED SUBSEQUENTLY IN TASK 11.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + ANL + DESIGN STUDY + R AND D PROGRAM
+ REACTOR, LMFBR

18-035871 ALSO IN CATEGORY 11 DATED 1967
PRELIMINARY STUDIES FOR A REFERENCE CONCEPTUAL DESIGN OF A 1000 MWE LMFBR
ARGONNE NATIONAL LABORATORY
CEND-322 +. 200 PAGES, FIGURES, TABLES, REFERENCES, DECEMBER 1967

DESCRIBES THE CONCEPTUAL DESIGN OF THE HEAT-GENERATION SYSTEM FOR A 1000-MWE LMFBR. THE CONCEPT IS BASED ON ESTIMATED TECHNOLOGY THAT MAY BE ATTAINABLE BETWEEN 1975 AND 1985. THE CORE SELECTED FOR ADDITIONAL DESIGN STUDY AND REFINEMENT IS CYLINDRICAL, HAS A TWO-ZONE SEED REGION, 2 FT HIGH AND ABOUT 9 FT IN DIAMETER, WITH A 1.5-FT AXIAL BLANKETS AND A 1-FT RADIAL BLANKET. THE FUEL IS MIXED U AND PU CARBIDES, CLAD IN THIN STAINLESS-STEEL TUBES. FOR REFUELING, AN UNDER-THE-PLUG SCHEME WAS ADOPTED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

FUEL HANDLING + REACTOR DESCRIPTION + DESIGN STUDY + REFUELING + CORE COMPONENTS + R AND D PROGRAM + REACTOR,
LMFBR + FUEL ELEMENTS

18-035883 DATED 1968
1000 MWE LMFBR LIQUID METAL FAST BREEDER REACTOR
ARGONNE NATIONAL LABORATORY
AI-AEC-12765 (VOL.1) +. 50 PAGES, FIGURES, TABLES, REFERENCES, NOVEMBER 15, 1968

DESCRIBES THE CONCEPTUAL DESIGN SELECTED FOR TASK 1 OF THE LMFBR FOLLOW-ON STUDY. DESIGN WILL BE USED AS BASIS FOR MORE DETAILED DESIGN DEFINITION, FURTHER PARAMETRIC STUDIES, AND DEFINITION OF R AND D REQUIREMENTS. HEAT WILL BE REMOVED BY LIQUID SODIUM PUMPED UPWARD THROUGH REACTOR AT INLET OF 760 F AND EXIT AT 1060 F. HEAT IS THEN TRANSFERRED TO SECONDARY SODIUM CIRCUIT FOR STEAM PRODUCTION. 3 LOOPS WITH PRIMARY AND SECONDARY CIRCUIT IN EACH. STEAM AT 2400 PSIG AND 900 F IS GENERATED IN SINGLE WALL SHELL AND TUBE HEAT EXCHANGER OF MODULAR TYPE. CORE FUEL IS PUO2-UO2 AND BLANKETS ARE DEPLETED UO2. CONTROL BY TANTALUM RODS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, BREEDER + REACTOR, FAST + REACTOR, LMCR + REACTOR DESCRIPTION + ANL + DESIGN STUDY + METEOROLOGY + R
AND D PROGRAM + REACTOR, LMFBR

18-036060 ALSO IN CATEGORY 6 DATED 1969
HANSEN NP
SEFOR CORE CLAMPING SYSTEM
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIF.
2 PAGES, 1 FIGURE, ANS TRANSACTIONS, 12(1), PAGES 321-22 (JUNE 1969)

OBJECTIVE WAS TO MAKE AN ACCURATE DOPPLER-COEFFICIENT MEASUREMENT DURING TRANSIENT CONDITIONS. REACTIVITY EFFECTS CAUSED BY FUEL-ROD THERMAL BOWING MUST BE CONTROLLED. A CORE-CLAMPING SYSTEM WAS NEEDED TO PREVENT MOVEMENT OF RODS THAT WOULD CREATE REACTIVITY EFFECTS. THE CLAMPING SYSTEM DESIGNED TO SATISFY NUMEROUS REQUIREMENTS UTILIZES TORSION RODS TO PROVIDE CLAMPING FORCE AND FLEXIBILITY. THIRTY-SIX TORSION RODS SPACED AROUND THE CHANNELS DISTRIBUTE RADIAL FORCE TOWARD CENTER OF CORE. CLAMPING SYSTEM SUCCESSFULLY MET OBJECTIVES.

AVAILABILITY - N.P. HANSEN, GENERAL ELECTRIC COMPANY, 310 DEGUIGNE DRIVE, SUNNYVALE, CALIFORNIA 94086

DOPPLER COEFFICIENT + REACTOR, FAST + REACTOR, LMCR + MEASUREMENT, REACTIVITY + REACTOR, PULSED + SEFOR (RE) +
CORE COMPONENTS

18-036146 DATED 1968
PFLASTERER GR
SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR DEVELOPMENT PROGRAM
GENERAL ELECTRIC COMPANY, SUNNYVALE, CALIFORNIA
GEAP-5705 +. 34 PAGES, 4 FIGURES, 1 TABLE, NOVEMBER 1968

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-036146 *CONTINUED*

ALL CONTRACTORS LEFT SITE BECAUSE CONSTRUCTION WAS COMPLETE. FRED EXCURSION DEVICE WAS TESTED, AS WAS REACTIVITY OSCILLATOR. 350 FUEL RODS DELIVERED AFTER A VACUUM OFF-GAS FURNACE WAS USED TO ELIMINATE HIGH VOLATILITY CONTENT OF PELLETS. SHIPPING CASK FIRE SHIELD WAS DAMAGED WHEN DRIPPED. HAD TO BE REPAIRED. STARTUP TESTING PLANS PROCEEDED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

FABRICATION + REACTOR, LMCR + OSCILLATOR, REACTIVITY + REACTOR STARTUP TESTING + FAILURE, FABRICATION ERROR + SEFOR (RE) + FUEL ELEMENTS

18-036236 ALSO IN CATEGORY 5

DATED 1967

SOFER GA + CHEKRY BH

DESIGN AND SAFETY OPTIMIZATIONS IN THE USE OF PLUTONIUM-URANIUM CARBIDE FUEL IN LARGE FAST BREEDER REACTORS

COMMISSARIAT A L'ENERGIE ATOMIQUE, PARIS

12 PAGES, PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, SEPTEMBER 19-22, 1967. (SEE CONF-670916)

OPERATING CHARACTERISTICS OF A 1000-MW(E) NA-COOLED PUC-UC REACTOR, CONSISTING OF NEUTRONICALLY UNCOUPLED THREE-MODULE CORE, ARE DESCRIBED AND DISCUSSED. ITS SAFETY CHARACTERISTICS ARE ANALYZED AND COMPARED TO A SINGLE-MODULE CORE WITH A L/D EQUAL TO 1. THE INFLUENCE OF NA VS HE AS THE THERMAL BOND BETWEEN FUEL AND CLADDING ON THE SAFETY CHARACTERISTICS OF BOTH DESIGNS IS EVALUATED. FUEL CYCLE COST ESTIMATES FOR MODULAR AND L/D EQUAL TO 1 DESIGNS ARE COMPARED AND DISCUSSED.

CLADDING + REACTOR, FAST + THERMAL CONSIDERATION + DESIGN STUDY + SAFETY PRINCIPLES AND PHILOSOPHY + ECONOMIC STUDY + COMPARISON, REACTOR CHARACTERISTICS + REACTOR, LMFR + FUEL ELEMENTS

18-036465 ALSO IN CATEGORY 11

DATED 1967

EXPLOSION CONTAINMENT IN THE PROTOTYPE FAST REACTOR SINGLE TANK SYSTEM - FURTHER MODEL EXPERIMENTS

SAFEX/P.55 +. 29 PAGES, 20 FIGURES, 6 TABLES, 3 REFERENCES, DECEMBER 1967

FURTHER EXPLOSION TESTS ON 1/25TH SCALE MODEL OF SINGLE TANK PROTOTYPE DESCRIBED CHECKING EFFECTS OF DETONATION OF TNT. DAMAGE RESULTS DESCRIBED FOR ROUND 4, ASSEMBLY D, ROUND 5, ASSEMBLY D, AND ROUND 6 ASSEMBLY D.

AVAILABILITY - UNITED KINGDOM ATOMIC ENERGY AUTHORITY, 11 CHARLES II STREET, LONDON, S.W. 1, ENGLAND

REACTOR, BREEDER + CONTAINMENT + REACTOR, FAST + REACTOR, LMCR + PRESSURE VESSELS + UNITED KINGDOM + EXPLOSIVE, CONVENTIONAL + PFR (LMFR) + MODEL TESTING

18-036467 ALSO IN CATEGORY 11

DATED 1968

STATIC TESTS ON 1/16TH SCALE MODEL OF PFR VAULT ROOF

SAFEX/P.68 +. 25 PAGES, 20 FIGURES, 2 REFERENCES, 1968

RESULTS GIVEN FOR TEST OF SINGLE-TANK MODEL MADE WITH REINFORCED CONCRETE AND BUILT TO CLOSELY RELATE ALL ASPECTS TO SCALED-DOWN QUANTITIES AND DIMENSIONS. DRAWING AND DESCRIPTION OF MODEL GIVEN. TESTS SHOWED MODEL VAULT ROOF WILL SUPPORT 59 TONS (15,000 TONS FULL SCALE) AT A DEFLECTION OF 0.53 INCH (8.5 INCHES FULL SCALE).

AVAILABILITY - UNITED KINGDOM ATOMIC ENERGY AUTHORITY, 11 CHARLES II STREET, LONDON, S.W. 1, ENGLAND

REACTOR, BREEDER + CONTAINMENT + REACTOR, FAST + REACTOR, LMCR + UNITED KINGDOM + PFR (LMFR) + MODEL TESTING + STRUCTURAL ANALYSIS, DYNAMIC

18-036597 ALSO IN CATEGORIES 5 AND 6

DATED 1967

SIMPSON DE + LITTLE WW + PETERSON RE

SELECTED SAFETY CONSIDERATIONS IN DESIGN OF THE FAST FLUX TEST FACILITY

BATTIELLE-NORTHWEST, WASHINGTON

37 PAGES, 10 FIGURES, 9 REFERENCES, FROM THE INTERNATIONAL CONFERENCE ON THE SAFETY OF FAST REACTORS, AIX-EN-PROVENCE, SEPTEMBER 1967

DESCRIBES FFTF. ACCIDENT-SEVERITY CRITERIA DISCUSSED IN TERMS OF MAX. TRANSIENT FUEL TEMPERATURES AND EXTENT OF MELTING. THE MAJOR SAFETY CONSIDERATIONS AT THE CONCEPTUAL STAGE ARE OUTLINED. TOPICS - FFTF DESCRIPTION, SAFETY PHILOSOPHY AND CRITERIA, TRANSIENT AND ACCIDENT ANALYSES, DESIGN-BASIS ACCIDENT, DESIGN CONSIDERATIONS, AND EFFECTS OF SAFETY ON TESTING CONDITIONS.

AVAILABILITY - D.E. SIMPSON + W.W. LITTLE + R.E. PETERSON, PACIFIC NORTHWEST LABORATORY, P.O. BOX 999, RICHLAND, WASHINGTON

ACCIDENT ANALYSIS + DESIGN CRITERIA + REACTOR, LMCR + REACTOR DESCRIPTION + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + SAFETY PRINCIPLES AND PHILOSOPHY + R AND D PROGRAM + REACTOR, LMFR

18-036600 ALSO IN CATEGORY 6

DATED 1968

ANL - CHEMICAL ENGINEERING DIVISION RESEARCH HIGHLIGHTS

ARGONNE NATIONAL LABORATORY

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-036600 *CONTINUED*
ANL-7550 +. 118 PAGES, FIGURES, TABLES, DECEMBER 1968

ONE OF A SERIES OF SIMILAR REPORTS. TOPICS - COMPACT PYROCHEMICAL PROCESS, SALT TRANSPORT PROCESS FOR FAST BREEDER FUELS, FLUORIDE VOLATILITY PROCESS (CONCEPTUAL PROCESS FOR FBR FUELS), MATERIALS CHEMISTRY AND THERMODYNAMICS, XENON TAGGING IN EBR-2 FUEL ELEMENTS, CHEMISTRY OF LIQUID METALS, ENERGY CONVERSION (CELLS WITH LITHIUM ANODES AND SODIUM ANODES), CROSS SECTION MEASUREMENT, AND FLUIDIZED-BED COMBUSTION OF FOSSIL FUELS. ADDENDUM LISTS ABOUT 200 REPORTS PUBLISHED IN 1968 ON THE ABOVE SUBJECTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

BIBLIOGRAPHY + REACTOR, BREEDER + CHEMICAL REACTION + ENERGY SOURCE + REACTOR, FAST + MEASUREMENT + ANL + CHEMICAL KINETICS + METAL, LIQUID + DIRECT ENERGY CONVERSION DEVICES + CROSS SECTION + FUEL REPROCESSING + R AND D PROGRAM + PYROCHEMICAL + REACTOR, LMFBR

18-036612 DATED 1968
1000 MWE LIQUID METAL FAST BREEDER REACTOR FOLLOW-ON STUDY. REPORT OF TASKS II AND III. A REFERENCE CONCEPTUAL DESIGN OF A 1000 MWE LMFBR. VOLUME III. SAFETY AND CONTROL ANALYSES
COMBUSTION ENGINEERING, INC., WINDSOR, CONN. NUCLEAR POWER DEPT.
CEND-337(VOL.3) +. 189 PG., FIGURES, TABLES, REFERENCES, SEPTEMBER 1968

SUMMARIZES A TRADE-OFF STUDY OF CYLINDRICAL AND MODULAR CORES. RESULTS INDICATE THAT CYLINDRICAL CORE HAS ADVANTAGE IN AREAS OF FUEL-CYCLE COST, BREEDING, AND MECHANICAL DESIGN. BOTH TYPES OF CORE MEET REQUIREMENTS FOR SAFETY. CONCLUSION FROM ANALYSIS OF ACCIDENTS INDICATE THAT ACCIDENT WILL BE TERMINATED BEFORE PROPAGATION BEYOND LOCAL REGION IN CORE. THE 70 AEC DESIGN CRITERIA WERE REVIEWED AND MODIFIED FOR APPLICATION TO AN LMFBR. CONTROL REQUIREMENTS ARE DESCRIBED.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

ACCIDENT ANALYSIS + CONTROL + DESIGN STUDY + SAFETY ANALYSIS + REACTOR, LMFBR

18-036614 DATED 1966
DUFFY JG + HESS JW + MATTE J + MUELLER RE + WHITE JR
ALTERNATE REACTOR CONCEPT DESIGN STUDY FOR THE FAST FLUX TEST FACILITY. FIXED SHIELD PLUG, VERTICAL CORE REACTOR CONCEPT
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC., DETROIT, MICHIGAN
APDA-187 +. 50 PAGES, FIGURES, TABLES, REFERENCES, APRIL 30, 1966

DISCUSSES THE DESIGN STUDY OF A FIXED-SHIELD-PLUG, VERTICAL-CORE REACTOR CONCEPT FOR THE FAST FLUX TEST FACILITY. IT IS ONE OF A NUMBER OF ALTERNATIVE TEST REACTOR CONCEPTS TO BE CONSIDERED IN THE DEVELOPMENT OF THE PRELIMINARY DESIGN. MAJOR FEATURES INCLUDE A SODIUM-COOLED VERTICAL CORE OF FREE THERMAL BOWING SUBASSEMBLIES, AND IN-REACTOR FUEL HANDLING BY THREE PANTOGRAPH-TYPE OFFSET HANDLING MECHANISMS WHICH PENETRATE THE FIXED SHIELD PLUG OF THE REACTOR. THE SIGNIFICANT FEATURES AND CHARACTERISTICS OF THIS CONCEPT ARE IDENTIFIED AND COMPARED WITH THOSE OF THE FFTF INTERIM REFERENCE REACTOR DESIGN CONCEPT AS DEFINED IN REPORT BNWL-CC-400.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCR + REACTOR, TEST + FFTF (TRI) + DESIGN STUDY

18-036616 ALSO IN CATEGORY 11 DATED 1969
COLLINS CG + MOTEFF J + CHANDLER BA
EVALUATION OF THE POTENTIAL OF SELECTED ALLOYS FOR USE AS A FUEL CLADDING MATERIAL IN AN LMFBR
GENERAL ELECTRIC CO., CINCINNATI, OHIO. MISSILE AND SPACE DIV.
GEMP-573(REV.) +. 312 PAGES, FIGURES, TABLES, REFERENCES, JUNE 1969

THE EVALUATION IS BASED ON REVIEWS OF THE DATA ON THE SODIUM CORROSION RESISTANCE, NUCLEAR CHARACTERISTICS, IRRADIATION EFFECTS, AND CREEP OR CREEP-RUPTURE PROPERTIES. THE MATERIALS CONSIDERED INCLUDE THE ALLOYS OF IRON, NICKEL, COBALT, VANADIUM, NIOBIUM, MOLYBDENUM, TANTALUM, AND TUNGSTEN. IT IS CONCLUDED THAT THE ALLOYS OF IMMEDIATE POTENTIAL ARE THE AUSTENITIC STEELS AND THE FE-CR-CO-NI ALLOYS, AND THAT THOSE WITH DEVELOPMENT POTENTIAL OFFERING HIGHER CREEP STRENGTHS INCLUDE THE NICKEL ALLOYS WITH LESS THAN ABOUT 50% NICKEL, AND THE COBALT, VANADIUM, MOLYBDENUM, AND NIOBIUM ALLOYS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151, \$3.00
COPY, \$0.65 MICROFICHE

ALLOY + COBALT + CLADDING + METAL + MOLYBDENUM + SODIUM + NICKEL + STEEL, STAINLESS + IRON + MANITOBA + TANTALUM + CORROSION + TUNGSTEN + REVIEW + VANADIUM + CREEP + REACTOR, LMFBR + DUCTILITY + FUEL ELEMENTS

18-036625 ALSO IN CATEGORY 17 DATED 1969
REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT JUNE 1969
ARGONNE NATIONAL LABORATORY
ANL-7581 +. 136 PAGES, FIGURES, TABLES, JUNE 1969

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-036625 *CONTINUED*

ONE OF A SERIES. TOPICS - LMFBR DEVELOPMENT, EBR-2 OPERATIONS, FAST-REACTOR FUEL DEVELOPMENT, FAST-REACTOR ANALYSIS METHODS AND DATA, CHEMICAL SEPARATIONS, REACTOR KINETICS, FUEL MELTDOWN STUDIES, TREAT OPERATIONS, AND CHEMICAL REACTIONS AND ENERGY.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CHEMICAL REACTION + REACTOR, FAST + FUEL MELTDOWN + REACTOR, LMCR + METAL WATER REACTION + REPORT, OPERATIONS + REACTOR, SAFETY RESEARCH + REACTOR, RESEARCH + REACTOR, TEST + TREAT (PRR) + REACTOR, ERDA OWNED + REACTOR, PULSED + R AND D PROGRAM + REACTOR, LMFBR

18-036630 ALSO IN CATEGORY 17 DATED 1969
NOTICE OF ISSUANCE OF OPERATING LICENSE FOR 20-MW(T) OPERATION OF SEFOR
USAEC DIVISION OF REACTOR LICENSING
20 PAGES, LETTER - URL TO GENERAL ELECTRIC COMPANY - AUGUST 21, 1969, DOCKET 50-231

AMENDMENT TO PROVISIONAL OPERATING LICENSE WOULD PERMIT (1) OPERATION AT STEADY STATE UP TO 20 MW(T) AND IN THE PULSED MODE, AND (2) THE RECEIPT POSSESSION AND USE OF UP TO 10 MILLCURIES OF KR-85, I-131, Xe-133, AND CS-137. RELATED SAFETY EVALUATION AND ACRS REPORT ARE ENCLOSED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545, 108 CENTS/PAGE -- MINIMUM CHARGE \$2.00

REACTOR, FAST + REACTOR, LMCR + ACRS + REACTOR, PULSED + SEFOR (RE) + SAFETY EVALUATION + OPERATING LICENSE PROCESS

18-036632 DATED 1969
1000 MWE LIQUID METAL FAST BREEDER REACTOR FOLLOW-ON STUDY - TASK IV REPORT. A RESEARCH AND DEVELOPMENT PROGRAM NEEDED FOR THE C-E REFERENCE CONCEPT
COMBUSTION ENGINEERING, INC.
CEND-346 +. 200 PAGES, FIGURES, TABLES, FEBRUARY 1969

DESCRIBES THE R AND D PROGRAMS REQUIRED FOR THE HEAT-GENERATION SYSTEM OF THE REFERENCE DESIGN. CURRENT TECHNOLOGY AND INFORMATION NEEDED ARE DISCUSSED FOR THE PLANT DESIGN, FUELS AND MATERIALS, CORE DESIGN, COMPONENTS, AND FUEL-HANDLING SYSTEM. REVIEWS ACQUISITION OF PHYSICS DATA AND PERFORMANCE OF CRITICAL EXPERIMENTS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY \$0.65 MICROFICHE

DESIGN STUDY + R AND D PROGRAM + REACTOR, LMFBR

18-036634 DATED 1967
Goulding BJ
SYSTEM DESCRIPTION REPORT
BABCOCK AND WILCOX CO., LYNCHBURG, VA.
BAW-1316(VOL. 5) +. 174 PAGES, FIGURES, TABLES, REFERENCES, AUGUST 1967

DESCRIBES CONCEPT IV OF TASK I OF THE PROGRAM. THE CONCEPT PROPOSES AN INTEGRAL POT ARRANGEMENT WITH AN UNDER-THE-PLUG REFUELING SCHEME. SIX TANDEM PUMP/1HX ASSEMBLIES ARE SUBMERGED IN THE SODIUM POOL. THE ASSEMBLIES ARE LOCATED RADIALLY FROM THE CORE BETWEEN THE CORE VESSEL AND THE REACTOR VESSEL, RESULTING IN A COMPACT BUILDING. DISADVANTAGES OF THE CONCEPT - THE LARGE SODIUM INVENTORY, ON-SITE CONSTRUCTION OF PRESSURE VESSEL, AND NEUTRON SHIELDING PROBLEMS ASSOCIATED WITH THE INTERMEDIATE HEAT EXCHANGERS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

DESIGN STUDY + SYSTEM DESCRIPTION + REACTOR, LMFBR

18-036635 DATED 1967
Goulding BJ
SYSTEM DESCRIPTION REPORT
BABCOCK AND WILCOX CO., LYNCHBURG, VA.
BAW-1316(VOL. 4) +. 162 PAGES, FIGURES, TABLES, JULY 1967

DESCRIBES CONCEPT 3 OF TASK I OF THE PROGRAM. SYSTEMS DESCRIBED INCLUDE THE VESSEL, CONTROL SYSTEM, SHIELDING, CORE, AUXILIARY SYSTEMS, FUEL HANDLING, SAFETY SYSTEM, INSTRUMENTATION, COOLANT SYSTEM, AND STEAM GENERATORS. PLANT CONSISTS OF A SODIUM-COOLED REACTOR OF 2415 MW(T), WITH 43% EFFICIENCY. STEAM AT 2400 PSIG AND 1000 F IS GENERATED IN 3 ONCE-THROUGH STEAM GENERATORS. THE PRIMARY SYSTEM IS SPREAD OUT, HAVING 6 PUMPS 3 IHX DIVIDED INTO 2 LOOPS. CORE CONFIGURATION IS ANNULAR, AND CLADDING IS 304 STAINLESS STEEL CONTAINING COMPACTED UO2-PUO2.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

DESIGN STUDY + SYSTEM DESCRIPTION + REACTOR, LMFBR

CATEGORY 18
SAFETY-ANALYSIS AND DESIGN REPORTS

18-036636
GOULDING BJ
1000-MWE LMFBR FOLLOW-ON STUDY
BABCOCK AND WILCOX CO., LYNCHBURG, VA.
BAW-1316 (VOL. 1) +. 237 PAGES, FIGURES, TABLES, REFERENCES, JUNE 1967

DATED 1967

BRIEFLY DESCRIBES FOUR BASIC PLANT CONCEPTS FOR AN LMFBR. THE STUDY IS POWER-COST ORIENTED, ASSUMING 1980 TECHNOLOGICAL CAPABILITY. FOR THE CONCEPTS AND CONDITIONS STUDIED, POWER-GENERATION COST DEPENDS STRONGLY ON FUEL-CYCLE COST, WHICH DEPENDS ON BURNUP. THE PREFERRED CONCEPT FEATURES AN INTEGRAL-POT PRIMARY SYSTEM WITH AN UNDER-THE-PLUG TYPE OF REFUELING SYSTEM. THE POT CONCEPT HAS ALL PRIMARY COMPONENTS HOUSED IN THE REACTOR PRESSURE VESSEL. FUEL PINS ARE VENTED TO THE SODIUM.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY \$0.65 MICROFICHE

DESIGN STUDY + REACTOR, LMFBR

18-036765
HAEFELE W
ON THE DEVELOPMENT OF FAST BREEDERS
KERNFORSCHUNGSZENTRUM, KARLSRUHE
KFK-881 + CONF-681101-30 +. 47 PAGES, 8 FIGURES, 7 TABLES, 87 REFERENCES, NOVEMBER 1968, PRESENTED AT THE INTERNATIONAL CONFERENCE ON THE CONSTRUCTIVE USES OF ATOMIC ENERGY, WASHINGTON, D.C.

DATED 1968

REVIEWS THE HISTORICAL DEVELOPMENT OF FAST BREEDERS, ENCOMPASSING TOPICS OF FAST-REACTOR PHYSICS, FUEL BURNUP, CLADDING, SAFETY, SODIUM TECHNOLOGY. LISTS THE PRESENT SODIUM-COOLED FAST-BREEDER PROJECTS THROUGHOUT WORLD, PLUS THE PROTOTYPE-REACTOR PROJECTS.

AVAILABILITY - USAEC DEPOSITORY LIBRARIES IN THE U.S. AND OVERSEAS

REACTOR, BREEDER + CLADDING + REACTOR, FAST + FUEL BURNUP + SODIUM + COOLANT CHEMISTRY + REVIEW + REACTOR PHYSICS + R AND D PROGRAM + REACTOR, LMFBR

18-036802
HIGH-TEMPERATURE LIQUID-METAL TECHNOLOGY REVIEW
BROOKHAVEN NATIONAL LABORATORY
80 PAGES, 11 FIGURES, 1 TABLE, APRIL 1969

DATED 1969

SUMMARIZES PROGRESS IN VARIOUS R AND D PROGRAMS ON MATERIAL DEVELOPMENT, CORROSION, HEAT TRANSFER, FLUID DYNAMICS, INSTRUMENTATION, PHYSICAL PROPERTIES, CHEMICAL ANALYSIS PERTINENT TO FAST-BREEDER DEVELOPMENTAL PROGRAM. PROGRAMS REVIEWED IN CATEGORY OF SAFETY INCLUDE IN-CORE FLOWMETER DEVELOPMENT, EFFECT OF FLOWING SODIUM ON CREEP RATE OF FUEL JACKET (VANADIUM ALLOYS), DEVELOPMENT OF HIGH-TEMP.-BOILING DETECTOR, SODIUM-FIRE FISSION-PRODUCT RELEASE, BOILING STUDIES, SODIUM CHEMISTRY, AND NOISE-ANALYSES INSTRUMENTATION.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00
COPY, \$0.65 MICROFICHE

CHEMICAL REACTION + FIRE + FISSION PRODUCT RELEASE + SODIUM + NOISE ANALYSIS + CORROSION + METAL, LIQUID + REVIEW + R AND D PROGRAM + REACTOR, LMFBR + BOILING + INSTRUMENTS, MISC.

18-036929
QUARTERLY PROGRESS REPORT APRIL, MAY, JUNE 1968. SODIUM-HEATED STEAM GENERATOR DEVELOPMENT
BABCOCK AND WILCOX COMPANY
BAW-1280-57 +. 11 PAGES, 1 FIGURE, 1 TABLE, DECEMBER 1968

DATED 1968

COMPARED OPERATING CHARACTERISTICS OF PROTOTYPE STEAM GENERATORS DESIGNED FOR 254-, 400-, AND 500-PSI WATER-SIDE PRESSURE DROP. RESULTS WERE SUBMITTED TO AEC AND LMEC. AFTER LMEC HAS EVALUATED THE MODIFICATIONS THAT WILL BE REQUIRED TO SCTI, A RECOMMENDATION WILL BE MADE FOR REDESIGN AND FABRICATION OF THE PROTOTYPE STEAM GENERATOR. WORK WAS HELD UP ON TASK II (SODIUM-TO-STEAM HEAT-EXCHANGER STUDY). WORK CONTINUES UNDER TASK III TO RECOMMEND A REACTOR VESSEL CONCEPT FOR FFTF. AN ANNULAR DESIGN AND A BOTTOM INLET DESIGN ARE BEING COMPARED.

REACTOR, FAST + REACTOR, LMFBR + PRESSURE VESSELS + SODIUM + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + METAL, LIQUID + R AND D PROGRAM + STEAM GENERATOR + REACTOR, LMFBR + HEAT EXCHANGERS

18-037462
BUKAU F
A PROTECTIVE GAS SYSTEM OF HIGH PURITY FOR THE APPLICATION OF HIGH-MELTING REACTIVE METALS AT HIGH TEMPERATURES
GESELLSCHAFT FÜR KERNFORSCHUNG MBH, KARLSRUHE
KFK-927 + 5 PAGES, DECEMBER, 1969

DATED 1968

SODIUM IS THE REACTOR COOLANT OF PREFERENCE WHICH OFFERS IN REGARD TO HEAT FLOW AND OPERATING TEMPERATURE AN APPRECIABLE POTENTIAL FOR DEVELOPMENT. TO OBTAIN INFORMATION ON THE TECHNICAL LIMITATIONS OF THE USE OF SODIUM, A RESEARCH PROGRAM WAS INSTITUTED, AN EXPERIMENTAL APPARATUS HAS BEEN PLANNED AND ITS CONSTRUCTION ALMOST COMPLETED. THE CRITICAL HEAT FLUX (CHF) MUST BE REGARDED AS AN ABSOLUTE LIMIT OF THE HEAT FLUX. THIS IS DEPENDENT ON VARIOUS PARAMETERS. IN THE

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SAFETY-ANALYSIS AND DESIGN REPORTS

18-037462 *CONTINUED*

CASE OF SODIUM FORCED FLOW OF THE CHF IN THE REGION OF PRESSURE OF TECHNICAL INTEREST IS SO HIGH THAT UP TO THE PRESENT TIME WITH THE USUAL HEATING METHODS IT COULD NOT BE SIMULATED EXPERIMENTALLY.

AVAILABILITY - USAEC DEPOSITORY LIBRARIES IN THE U.S. AND OVERSEAS

REACTOR, LMCK + SODIUM + TESTING + THERMAL EXPERIMENT + COVER GAS + THERMAL CONSIDERATION + HIGH TEMPERATURE + MATERIAL + HEAT TRANSFER, CONDUCTION + R AND D PROGRAM + HEAT FLUX, CRITICAL

18-037489

DATED 1969

ASTLEY EK

FAST FLUX TEST FACILITY QUARTERLY TECHNICAL REPORT, MARCH-MAY 1969

BATTELLE-NORTHWEST, RICHLAND, WASHINGTON

BNWL-1090 +. 198 PAGES, FIGURES, TABLES, REFERENCES, JULY 1969

SUMMARIZES DESIGN PROGRESS DURING PERIOD MARCH THRU MAY. DEFINES CURRENT CONCEPTUAL FEATURES OF DESIGN AND PRESENTS TABULATION OF CURRENT DESIGN DATA. DESCRIBES WORK DONE IN SUPPORT OF DESIGN. METALLURGICAL STUDIES CONFIRM DESIRABILITY FOR USING 304SS FOR VESSEL. ANALYSIS CONCLUDES VESSEL COVER WILL BE RETAINED FOLLOWING THE DBA. DEVELOPMENT CONTINUED ON EDDY CURRENT TECHNIQUE TO DETECT DEFECTS IN TUBING. STUDIES CONTINUED ON DEPOSITION TENDENCIES OF STAINLESS ACTIVATION PRODUCTS IN FLOWING SODIUM LOOPS.

AVAILABILITY - CLEARINGHOUSE FOR FEDERAL SCIENTIFIC AND TECHNICAL INFORMATION, SPRINGFIELD, VA. 22151 \$3.00 COPY, \$0.65 MICROFICHE

REACTOR, FAST + REACTOR, LMCK + PRESSURE VESSELS + REACTOR, TEST + FFTF (TR) + DESIGN STUDY + TEST, NONDESTRUCTIVE + R AND D PROGRAM

18-106947

ALSO IN CATEGORIES 5 AND 7

DATED 1965

UKRENT D

PROCEEDINGS OF THE CONFERENCE ON SAFETY, FUELS, AND CORE DESIGN IN LARGE FAST POWER REACTORS - OCTOBER 11-14, 1965

ARGONNE NATIONAL LABORATORY, ILL.

ANL-7120 +. 910 PPS, OCT. 11-14, 1965

PRESENTS THE TEXT OF 79 PAPERS PRESENTED AT THE CONFERENCE. TOPICS INCLUDE: PHYSICS UNCERTAINTIES AND SAFETY STUDIES; FAST REACTOR CONCEPTS; SAFETY OF LARGE FAST POWER REACTORS; FUELS AND FUEL BEHAVIOR; REACTIVITY COEFFICIENTS; MAXIMUM ACCIDENT AND CONTAINMENT; FISSION PRODUCTS, REACTIONS AND INTERACTIONS; AND SODIUM BOILING.

AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

JACOBS + *REACTOR, FAST + *REACTOR, LMCK + DESIGN STUDY + REACTOR, GCFR + FUEL ELEMENTS + REACTIVITY COEFFICIENT + CONTAINMENT + PRESSURE TRANSIENT + FISSION PRODUCT RETENTION + BOILING + SODIUM + ACCIDENT ANALYSIS

KEYWORD INDEX

INFORMATION AT NSIC IS DIVIDED INTO 21 CATEGORIES. AN ITEM OF INFORMATION MAY BE KEYED TO AS MANY AS THREE OF THESE. A COLLECTION OF SELECTORS OR KEYWORDS IS USED TO DENOTE THE MAIN SAFETY RELATED POINTS COVERED IN AN ARTICLE. THE FOLLOWING INDEX IS AN ALPHABETICAL LISTING OF SELECTORS GIVING REFERENCES TO EACH ARTICLE WHICH WAS KEYED TO IT. THE CATEGORY NUMBER IS GIVEN FIRST, FOLLOWED BY THE ACCESSION NUMBER. THE ACCESSION NUMBERS ARE USED TO LOCATE BIBLIOGRAPHIC ITEMS WITHIN A CATEGORY.

(MONTICELLO-BWR)

17-013314
AARR (RR)
17-016534 17-026300 18-016534
ABLATION
2-029483 9-029484 9-029485 18-029483
18-029484 18-029485
ACCELERATION
2-029438 5-036238 9-029485 9-036238
18-029438 18-029485
ACCIDENT
1-025467 5-032895 18-024515 18-032895
ACCIDENT ANALYSIS
1-010160 1-029722 5-000899 5-007188
5-010430 5-011673 5-026427 5-029375
5-029508 5-029722 5-033427 5-036597
5-036957 5-106947 6-006227 6-010430
6-011162 6-032797 6-033427 6-036597
7-106947 8-036957 9-008500 11-002803
11-018638 11-036957 17-025935 17-040390
18-005832 18-006306 18-007188 18-012011
18-012012 18-012013 18-025935 18-029375
18-029508 18-029722 18-033427 18-036597
18-036612 18-106947
ACCIDENT MODEL
5-031420 5-037850 6-037850 7-029516
7-029940 7-034360 8-031420 12-037850
18-029516
ACCIDENT, COLD COOLANT
11-002803
ACCIDENT, CONSEQUENCES
5-029374 5-029375 5-029446 5-029459
5-032878 5-032882 5-032888 5-032904
5-032905 5-032920 5-032926 9-032920
11-000749 11-029446 11-029488 11-029490
11-029493 11-032888 11-032905 11-032926
12-032904 17-021081 18-029374 18-029375
18-029446 18-029459 18-029488 18-029490
18-029493 18-029519 18-032878 18-032882
18-032888 18-032904 18-032905 18-032920
18-032926
ACCIDENT, CONTROL ROD WITHDRAWAL
6-003915 6-011162 6-029301 6-030252
9-008500 11-002803 18-029301 18-030252
ACCIDENT, CORE DISRUPTIVE
6-003904 6-032797 8-003904
ACCIDENT, CRITICALITY
6-030252 18-030252
ACCIDENT, FUEL SLUMP
18-009964 18-017341
ACCIDENT, HYPOTHETICAL
2-032915 5-029459 5-029509 5-030253
5-030633 5-031574 5-032926 6-011516
6-011996 6-032797 7-030253 11-032926
12-035266 16-032915 18-029459 18-029509
18-030253 18-030633 18-031574 18-032915
18-032926
ACCIDENT, LOSS OF COOLANT
5-011674 5-027438 5-029508 5-031420
5-032878 5-032903 5-032904 5-032905
5-032906 5-032920 5-035345 5-036396
5-036403 6-023870 6-034751 7-032843
8-023870 8-031420 9-008500 9-032920
11-002803 11-032843 11-032905 12-032903
12-032904 12-032906 18-012009 18-012011
18-023870 18-029508 18-032843 18-032878
18-032903 18-032904 18-032905 18-032906
18-032920 18-033481
ACCIDENT, LOSS OF FLOW
5-016164 5-026427 5-029505 5-030633
5-032879 5-032882 5-035345 5-036397
5-036403 6-003901 6-003915 6-011162
6-011996 6-016164 9-008500 9-016164
11-032925 17-011666 17-016534 17-021216
18-016534 18-021211 18-029505 18-030633
18-032879 18-032882 18-032925
ACCIDENT, LOSS OF POWER
2-032917 5-030255 10-020579 10-030255
10-032917 11-002803 12-030255 17-013534
17-020579 18-030255 18-032917
ACCIDENT, LOSS OF PRESSURE
11-002803

ACCIDENT, MAXIMUM CREDIBLE (MCA)

1-007695 2-002801 5-005116 5-006684
5-029375 5-030633 6-003901 6-003915
6-007695 7-034360 8-018634 9-007695
11-002801 11-002803 11-018634 11-020821
17-007695 17-025935 18-005832 18-006306
18-009964 18-019142 18-025935 18-029375
18-030633
ACCIDENT, PROBABILITY OF
2-002801 5-017167 11-002801 11-002803
17-017167
ACCIDENT, REACTIVITY
5-031420 5-032893 5-037850 6-016735
6-021245 6-023870 6-037850 7-021245
8-023870 8-031420 9-008500 9-010439
11-002803 12-021245 12-037850 18-021211
18-023870 18-032893
ACCIDENT, REFUELING
2-032880 2-032915 5-032883 11-029457
11-032880 16-032915 18-017341 18-029457
18-032880 18-032883 18-032915 18-032918
ACCUMULATORS
9-029485 9-029486 18-029485 18-029486
ACOUSTICS
5-036209 9-036209
ACRS
1-021968 17-021968 17-032632 17-036630
18-028964 18-032632 18-036630
ACTINIDE
13-033337
ACTIVATION
5-009456 18-009456
ACTIVATION PRODUCT
7-031338
ACTIVITY BUILDUP
5-009456 5-032542 6-028562 7-029834
17-028562 17-029834 18-009456 18-029510
18-029513 18-032542
ADMINISTRATIVE CONTROL
1-004268 5-032895 13-005315 17-006310
17-016766 17-033982 18-005832 18-025462
18-032895
AEROSOL
7-005008 7-007280 7-014385 7-022479
7-029940 7-029949 7-031505 7-034347
7-034360 8-014385 12-031505 16-034347
AEROSOL PRODUCTION
7-022514 7-027029 7-027030 7-031505
7-034360 8-027030 12-031505
AEROSOL PROPERTIES
5-018209 7-007280 7-022514 7-027029
7-027030 7-029940 7-034360 8-027030
11-018209 12-018209
AGENCY, PROMOTIONAL
1-044436 3-044436
AGGLOMERATE
7-027029 7-034360
AI
1-030761 7-006379 7-024386 7-027283
12-027283 18-030761 18-034655
AIR
5-031420 6-006312 8-006312 8-031420
11-029463 17-006312 17-026164 17-026203
18-029463
AIR CLEANING
7-033156 9-010504 11-002809
AIRBORNE RELEASE
7-029940 7-034347 7-034360 12-029494
12-029514 16-034347 18-016652 18-029494
18-029514
ALLOY
5-005444 5-031448 6-003913 7-005008
7-007280 7-007346 7-015182 7-024369
7-031068 7-031505 9-036775 11-036616
12-024373 12-030879 12-031068 12-031505
13-030879 17-031282 18-031448 18-036616
ALPHA FACILITIES
12-034696 13-034696
ALUMINUM
18-016925
AMERICIUM
13-033337

ANALOG SIMULATION							
5-030633	5-032266	6-032266	6-034390	5-027751	5-032623	5-033984	5-034496
9-023146	9-034390	18-030633		5-034635	5-036209	5-036802	5-037463
ANALYTICAL MODEL				5-037850	5-038234	5-106947	6-032623
5-017167	5-030633	5-031420	5-032266	6-037850	7-106947	8-036802	9-036209
5-032527	5-032623	5-032624	5-034635	12-037850	17-026408	18-036802	18-106947
5-037850	5-037957	5-038234	6-010509	BONUS (ISR)			
6-032266	6-032527	6-032623	6-034390	2-002811	10-004893	10-018293	11-002808
6-037850	7-014385	7-024343	7-027029	11-002810	11-002811	17-003970	
7-031339	7-032253	7-034360	8-014385	BORAX, ALL (S-RR)			
8-031420	9-010150	9-010509	9-012086	17-006457	17-006458		
9-027519	9-032527	9-034390	12-037850	BORON			
17-017167	17-034636	18-030633	18-034636	7-028511	12-024471	13-024471	17-030218
ANALYTICAL TECHNIQUE				17-032324			
2-029436	5-033281	6-033983	7-013907	BUBBLE			
7-018947	7-027028	7-027819	7-027820	5-027751	5-036957	5-040402	6-040402
7-028506	7-029286	7-032250	9-035244	7-029287	8-036957	11-036957	12-029179
11-029436	17-008985	17-018947	17-026164	17-040402			
18-029436	18-033983			BUBBLE NUCLEATION			
ANL				5-027043	5-027751	7-029287	
1-035995	5-031420	5-033427	5-035349	BUCKLING			
6-007108	6-033427	6-035349	6-036600	5-011396	11-011396		
7-028508	8-031420	9-033950	17-007108	BUILDING			
17-032939	17-032940	17-032941	17-032942	11-024156	17-031271	17-036110	17-036111
17-032943	17-032945	17-032946	17-032947	18-024156			
17-032949	17-032950	17-032951	17-032952	BURNOUT HEAT FLUX			
17-032953	18-010107	18-032939	18-032940	5-010430	5-018209	5-026286	6-005759
18-032941	18-032942	18-032943	18-032945	6-010430	11-018209	12-018209	
18-032946	18-032947	18-032949	18-032950	BURST PRESSURE			
18-032951	18-032952	18-032953	18-033427	11-031470			
18-033950	18-035869	18-035870	18-035883	CALIBRATION			
18-036600				7-013907	7-027028	7-027819	7-027820
ANTIMONY				7-028506	7-032250		
7-030790				CANADA			
ARGON				11-002809	17-005375		
7-008142	7-029949	11-006850	11-030609	CARBIDE			
11-031973	12-029494	12-030609	12-034698	7-005008	7-007346	7-015182	7-031068
12-034705	18-011672	18-029494	18-030609	12-031068	18-034634		
ATMOSPHERIC DIFFUSION				CARBON			
2-022328	7-034347	11-022328	16-034347	7-013751	7-018343	7-037186	17-016131
ATMOSPHERIC POLLUTION				17-022914			
1-044436	3-044436			CARBON DIOXIDE			
AUXILIARY COOLING				7-013541			
17-019556				CENTERLINE MELTING			
AUXILIARY COOLING SYSTEM				17-020467	17-020468	17-020469	17-020470
5-029375	5-029459	5-029508	5-030255	17-020472			
9-029473	10-030255	11-029460	11-029492	CERAMICS			
12-030255	17-030433	18-029375	18-029459	7-008031	7-010297	7-029547	7-031068
18-029460	18-029473	18-029492	18-029508	11-033993	11-033994	11-033995	11-033996
18-030255				12-031068	18-007253	18-033993	18-033994
BARIUM				18-033995	18-033996	18-034634	
7-030790	7-034361			CERIUM			
BATTELLE NORTHWEST				7-010558	7-022489	7-034361	
11-033953	17-038113	18-033951	18-033952	CERTIFICATION			
18-033953	18-033968			1-032909	1-032912	18-032909	18-032912
BATTERIES & CHARGERS				CESIUM			
2-029439	10-029439	18-029439		7-010558	7-013907	7-014384	7-018291
BEAM HOLE				7-022489	7-030790	7-032253	7-034361
17-016534	18-016534			7-037183			
BENEFICIAL USE, AGRICULTURE				CHAMBER, FISSION			
1-044436	3-044436			9-023329			
BENEFICIAL USE, COMMERCIAL				CHAMBER, ION			
1-044436	3-044436			9-025615	9-029473	18-029473	
BENEFICIAL USE, RECREATIONAL				CHARCOAL			
1-044436	3-044436			11-002809			
BERYLLIUM				CHEMICAL ANALYSIS			
6-029466	9-036650	9-036766	9-036773	12-037429	17-016131		
9-036775	18-029466			CHEMICAL KINETICS			
BETA EMITTER				6-036600	7-030790	18-036600	
17-025901				CHEMICAL REACTION			
BETHE TAIT				5-000899	5-010430	5-031420	5-033056
1-028163	5-029509	5-032624	6-003904	5-036802	6-006312	6-010430	6-036600
6-011996	6-028163	6-032797	8-003904	7-031340	7-032587	8-006312	8-031420
9-028163	18-029509			8-033056	8-036802	9-037965	11-029457
BIBLIOGRAPHY				11-031340	11-036627	12-019567	12-033056
1-033051	5-010430	5-037838	6-010430	17-006312	17-027844	17-032587	17-036625
6-036600	8-010113	9-033051	12-029527	17-036627	18-010107	18-029457	18-029828
12-030879	12-033051	12-034696	12-034705	18-030210	18-032587	18-036600	18-036625
12-037838	13-030879	13-033337	13-034696	18-036802			
13-034697	17-003970	17-029975	17-032844	CHLORINE			
17-037838	18-032844	18-033968	18-036600	11-011394			
BIG ROCK POINT (BWR)				CHROMATOGRAPHY			
2-002811	10-004893	10-018293	11-002809	9-035244			
11-002810	11-002811	17-003970	17-005375	CHROMIUM			
BLOWDOWN				7-031338	7-031505	9-036775	12-031505
5-036957	8-036957	11-036957		17-031282			
BNL				CLADDING			
6-017522				5-011299	5-029507	5-032836	5-036236
BODEGA BAY (BWR)				7-010297	7-010533	7-024369	7-028511
2-002811	11-002811			7-028987	7-031068	7-031338	7-031340
BOILING				7-032265	7-034673	9-032836	9-034784
				9-036775	11-028987	11-031340	11-031470

11-032836	11-033336	11-033991	11-033993	11-002811	11-006850	11-007912	11-029450
11-033994	11-033995	11-033996	11-034673	11-032885	12-005807	17-005807	17-008056
11-036616	12-006488	12-031068	12-034784	18-029450	18-032885		
12-036795	12-037429	17-006488	17-021335	CONTAINMENT ANALYSIS			
17-022914	17-029794	18-018745	18-024002	8-018634	11-018634	11-018638	11-020821
18-025002	18-029507	18-029828	18-032265	11-029518	18-029518		
18-033991	18-033993	18-033994	18-033995	CONTAINMENT ATMOSPHERE, INERT			
18-033996	18-034673	18-036236	18-036616	11-025992	11-029477	11-030609	11-033016
18-036765				12-030609	12-034698	12-034705	17-011671
COATED PARTICLE				17-032294	17-032324	17-033016	18-025992
7-015212	18-016925	18-024002	18-025002	18-029477	18-030609		
COBALT				CONTAINMENT CONSTRUCTION			
7-031338	11-036616	18-036616		1-033932	2-002811	11-002808	11-002809
CODES AND STANDARDS				11-002811	11-006850	11-007912	18-033932
1-029722	1-033051	2-002801	2-002811	18-033989	18-033990		
5-029722	9-029475	9-033051	11-002801	CONTAINMENT DESIGN			
11-002808	11-002809	11-002810	11-002811	2-002801	2-029433	2-029436	2-029445
11-032543	12-033051	18-029475	18-029722	5-000899	5-029375	5-030633	7-024343
18-032543	18-032818	18-033951	18-033952	7-036473	8-010242	11-002801	11-002808
COMPARISON, FACILITIES				11-002809	11-006850	11-009750	11-016576
7-029548	17-029548			11-025992	11-029433	11-029436	11-029445
COMPARISON, REACTOR CHARACTERISTICS				11-029447	11-029448	11-029454	11-030609
5-036236	9-024332	18-024515	18-033968	11-032816	11-032885	11-036473	12-030609
18-036236				17-010242	18-016576	18-025992	18-029375
COMPARISON, THEORY AND EXPERIENCE				18-029433	18-029436	18-029445	18-029447
1-023168	5-011040	5-032624	5-033984	18-029448	18-029454	18-030609	18-030633
5-035822	5-036209	6-019757	6-027775	18-032816	18-032885		
6-032845	9-007720	9-016943	9-020709	CONTAINMENT EQUIPMENT HATCH			
9-028016	9-028929	9-036209	12-035363	11-006850	11-032885	18-032885	
13-027777	17-007720	17-008985	17-011040	CONTAINMENT FILTERING SYSTEM			
17-019757	17-027775	17-029982	17-032845	2-002811	11-002809	11-002811	
17-034636	18-034636			CONTAINMENT INSPECTION AND MAINTENANCE			
COMPLIANCE				11-002808	11-002810		
12-032303	17-032303			CONTAINMENT INSTRUMENTATION			
COMPONENTS				9-031241	9-036206	11-002809	11-007912
9-038092	12-029527	12-029533	17-029048	CONTAINMENT INTEGRITY			
17-029932	17-032839	18-030399	18-032839	6-034669	11-000749	11-029457	11-029458
COMPUTER CONTROL				11-029464	11-034669	18-029457	18-029458
5-032627	9-023347	9-032627	9-037406	18-029464	18-034669		
17-032627	17-037406			CONTAINMENT ISOLATION			
COMPUTER PROGRAM				7-013541	9-029461	11-002809	11-006850
5-027438	5-031420	5-032266	5-032527	11-029453	11-029458	11-029460	11-029461
5-034497	5-036403	5-037957	6-006673	11-029462	11-029465	12-029495	17-008867
6-010509	6-011516	6-023870	6-032266	18-029453	18-029458	18-029460	18-029461
6-032527	6-033983	7-024343	7-029547	18-029462	18-029465	18-029495	
7-037184	8-023870	8-031420	9-010509	CONTAINMENT LEAKAGE CONTROL			
9-027519	9-032527	9-037406	11-033336	2-002801	7-013541	11-002801	11-007746
12-035263	17-006673	17-017164	17-034636	11-009750	17-007746	17-011994	
17-037406	18-023870	18-033983	18-034636	CONTAINMENT LINER			
COMPUTER, ANALOG				11-029448	18-029448		
5-030633	6-003407	6-010509	9-003407	CONTAINMENT PENETRATION, ELECTRICAL			
9-005333	9-010509	9-012086	9-012130	9-026513	9-029461	9-037487	10-026513
9-013355	9-023146	9-027519	9-030743	11-002809	11-006850	11-025992	11-026513
17-003407	17-010263	18-030633		11-029461	17-032294	17-037487	18-025992
COMPUTER, DIGITAL				18-029461			
5-033984	6-003407	6-003408	6-003904	CONTAINMENT PENETRATION, PIPE			
6-003910	6-006325	6-009822	6-010959	11-002809	11-006850	11-029452	11-029453
6-011153	6-025361	8-003904	9-003407	11-029455	11-029456	18-029452	18-029453
9-027519	9-037406	9-037487	17-003407	18-029455	18-029456		
17-037406	17-037487			CONTAINMENT R AND D			
CONCRETE				5-018209	7-024343	11-018209	12-018209
2-002811	11-002808	11-002810	11-002811	CONTAINMENT REFERENCE MEASURING SYSTEM			
11-029443	11-029447	12-032813	17-019561	11-002810	11-009750		
18-029443	18-029447	18-032813		CONTAINMENT SPRAY			
CONCRETE, PRESTRESSED				11-002809			
5-036957	8-036957	11-002808	11-036957	CONTAINMENT STRUCTURE			
CONDENSATION				2-002801	2-002811	2-029445	11-002801
5-011915	5-017577	5-033984	6-017577	11-002808	11-002809	11-002810	11-002811
17-011605				11-006850	11-007912	11-009746	11-009750
CONGRESSIONAL ACTIVITY				11-025992	11-029443	11-029445	11-029447
1-021968	1-049354	17-014804	17-021968	18-009746	18-025992	18-029443	18-029445
18-014804				18-029447			
CONNECTICUT YANKEE (PWR)				CONTAINMENT SYSTEM, OPERATION			
2-002811	11-002811			2-002801	11-002801	11-002810	
CONTAINMENT				CONTAINMENT VACUUM BREAKER			
1-004268	5-005116	5-026461	5-036468	11-029451	18-029451		
5-106947	6-003912	7-010297	7-106947	CONTAINMENT VESSEL LOADING			
8-036468	11-010356	11-018638	11-025992	2-029434	2-029435	2-029449	2-029450
11-036465	11-036466	11-036467	11-036468	2-032915	5-029446	5-032888	11-029434
12-010356	17-005375	17-019561	17-025901	11-029446	11-029447	11-029449	11-029450
17-026300	17-032939	17-032940	17-032941	11-029452	11-029458	11-029469	11-029470
17-032942	17-032943	18-010356	18-016652	11-032888	16-032915	18-029434	18-029435
18-021211	18-024515	18-025992	18-032939	18-029446	18-029447	18-029449	18-029450
18-032940	18-032941	18-032942	18-032943	18-029452	18-029458	18-029469	18-029470
18-034655	18-034665	18-034667	18-106947	18-032888	18-032915		
CONTAINMENT ABSOLUTE MEASURING SYSTEM				CONTAINMENT, HIGH PRESSURE			
11-007912	11-009750			2-002801	2-002811	5-005116	6-003901
CONTAINMENT AIR COOLING				6-003904	6-005122	6-005807	6-006227
11-029477	18-029477			6-006312	6-006561	6-006673	7-005008
CONTAINMENT AIR LOCK				8-003904	8-006312	9-006561	10-004893
2-002811	2-029450	6-005807	11-002810	10-004894	10-005262	11-002801	11-002803

11-002808	11-002809	11-002810	11-002811	9-027519	9-029120	9-029122	9-029961		
11-006850	11-007912	11-009746	11-009750	9-032807	9-032916	9-032920	9-033128		
11-012319	11-029714	11-033016	12-005262	12-034708	13-034708	16-032915	17-006357		
12-005807	17-003645	17-003970	17-005262	17-014334	17-021352	18-007188	18-024515		
17-005264	17-005375	17-005807	17-006312	18-032807	18-032915	18-032916	18-032920		
17-006317	17-006457	17-006458	17-006555	CONTROLLER					
17-006561	17-006673	17-012319	17-029714	9-022457	17-022457				
17-033016	18-008423	18-009746	CONVECTIVE BOILING						
CONTAINMENT, LOW PRESSURE				5-026286	5-026635	5-027440	5-027751		
2-002801	2-002811	10-004893	11-002801	COOLANT CHEMISTRY					
11-002803	11-002808	11-002810	11-002811	5-011299	5-017167	5-029002	5-031448		
11-009750	17-003970	17-005375							
CONTAINMENT, MULTIPLE				5-040402	6-040402	7-018291	7-024386		
2-002801	2-002811	8-018634	10-004893	7-027818	7-027819	7-027820	7-028511		
10-004894	11-002801	11-002808	11-002809	7-029949	7-030378	7-031340	7-033156		
11-002810	11-002811	11-006850	11-018634	7-034361	9-029002	9-034784	11-031340		
11-020821	11-030609	12-030609	17-003970	11-033953	12-029002	12-034784	17-006457		
18-030609				17-006458	17-006973	17-006976	17-007736		
CONTAINMENT, PRESSURE RELIEF				17-007743	17-008984	17-008985	17-009420		
2-002801	2-002811	11-002801	11-002809	17-013315	17-016131	17-017093	17-017167		
11-002811				17-018606	17-022914	17-029933	17-032146		
CONTAINMENT, PRESSURE SUPPRESSION				17-040402	18-011672	18-024515	18-031448		
2-002801	2-002811	11-002801	11-002808	18-033951	18-033952	18-033953	18-036765		
11-002809	11-002810	11-002811	11-009750	COOLANT COEFFICIENT					
17-003970	18-032891			5-033944	6-028802	6-033944	6-034751		
CONTAINMENT, PRESSURE VENTING				17-028802					
2-002801	2-002811	5-005444	6-003589	COOLANT PURIFICATION SYSTEM					
6-004479	6-005094	6-006357	7-036473	9-023947	9-029128	12-029533	17-008985		
9-003589	9-006357	9-009794	11-002801	17-011605	17-017093	17-019557	17-022914		
11-002803	11-002808	11-002809	11-002811	17-032146	17-036053	18-023947	18-023948		
11-009750	11-036473	12-004440	12-006488	COOLANT QUALITY					
17-003589	17-003637	17-003644	17-003970	5-032542	5-032853	5-038396	6-017131		
17-004028	17-006317	17-006357	17-006488	7-029547	12-032392	12-032853	17-016131		
17-009794				17-017131	17-018681	17-025935	17-030278		
CONTAINMENT, SHOCK GENERATION AND PROTECTION				17-032844	18-025935	18-032542	18-032844		
5-005116	11-002808	11-036825	COOLING						
CONTAINMENT, UNDERGROUND				5-032879	17-037469	18-032809	18-032879		
2-002811	11-002810	11-002811	COOLING, SHUTDOWN						
CONTAMINATION				5-029503	18-029502	18-029503			
7-015538	7-018343	7-025254	12-032303	COPPER					
17-032303				7-027028					
CONTROL				CORE COMPONENTS					
18-019142	18-036612			2-029483	5-028889	5-032888	5-036238		
CONTROL PANEL/ROOM				6-034669	6-036060	9-029484	9-034784		
2-029440	7-029516	9-029440	18-029440	9-036238	9-036239	11-029487	11-029488		
18-029476	18-029516			11-029489	11-029490	11-029491	11-030609		
CONTROL ROD BURNUP				11-032888	11-032924	11-033448	11-034669		
6-003589	9-003589	17-003589							
CONTROL ROD CALIBRATION				11-035871	11-036469	12-030609	12-034696		
5-022456	6-006312	6-010139	8-006312	12-034784	12-035195	13-034696	17-013238		
9-007720	9-010139	9-012130	9-028016	17-013239	17-016032	18-029483	18-029484		
17-006312	17-006977	17-007720	17-022456	18-029487	18-029488	18-029489	18-029490		
CONTROL ROD DRIVE MECHANISMS				18-029491	18-030498	18-030609	18-032884		
1-029865	6-007435	6-007436	6-007441	18-032888	18-032924	18-033448	18-033992		
9-007436	9-007441	9-008052	9-010439	CORE MELTDOWN					
9-012256	9-012383	9-025112	9-029195	5-032624	5-036957	6-023870	8-023870		
9-029484	9-029485	9-032807	9-032916	8-036957	11-002803	11-036957	17-022498		
11-007435	11-007436	11-025992	12-029865	17-034636	18-023870	18-034634	18-034636		
17-005375	17-006555	17-007435	17-007708	CORRELATION					
17-008052	17-008097	17-012383	17-018681	9-029040	17-029040				
17-019557	17-021352	17-021753	18-025992	CORROSION					
18-029484	18-029485	18-029865	18-032807	5-036238	5-036802	5-037838	7-015182		
18-032916				7-024369	7-025254	7-031338	7-031339		
CONTROL ROD INTERACTION				7-031340	7-032699	7-033156	7-034673		
9-023946	9-028016	17-023946							
CONTROL ROD SCRAM MECHANISM				7-037184	7-037186	8-010242	8-036802		
5-036238	9-007721	9-010523	9-012383	9-023947	9-034784	9-036238	11-031340		
9-014190	9-018506	9-029485	9-029486	11-033953	11-034673	11-036616	11-038785		
9-036238	17-007167	17-007721	17-012383	12-019567	12-029533	12-030664	12-030879		
18-029485	18-029486			12-032699	12-034784	12-035044	12-037429		
CONTROL ROD MORTH				12-037838	13-030664	13-030879	17-010242		
5-022456	6-003588	6-005094	6-005807	17-025996	17-028643	17-031284	17-032472		
6-006312	6-007435	6-007436	6-010139	17-037838	17-040390	18-021086	18-023947		
6-016736	8-006312	9-003588	9-005333	18-023948	18-033951	18-033952	18-033953		
9-007436	9-010139	9-010523	9-023946	18-034673	18-036616	18-036802			
9-028016	11-007435	11-007436	12-005807	COUNTER					
17-005807	17-006312	17-007435	17-022456	9-025835	17-026164				
17-023946				COUPLED CORES					
CONTROL ROD, SHIM SAFETY				6-017522 6-023162 18-012009					
9-012383	9-018506	17-012383							
CONTROL RODS				6-017522	6-023162	18-012009			
5-032895	5-032920	6-030351	9-012256	COVER GAS					
9-030351	9-032807	9-032920	17-009420	5-037462	7-029834	9-035244	12-029494		
17-013315	17-014334	17-029835	18-030498	12-029495	17-029834	17-029933	18-029494		
18-032807	18-032895	18-032920							
CONTROL SYSTEM				18-029495	18-029520	18-037462			
2-032915	5-007188	5-032920	6-006357	CREEP					
6-008211	6-010509	6-010978	6-017132	5-032542	7-037186	11-033336	11-036616		
9-006357	9-008211	9-008500	9-010150	12-036795	18-032542	18-036616			
9-010509	9-010978	9-012069	9-012256	CREEP BEHAVIOR					
9-017132	9-019115	9-023146	9-023347	5-031448	11-023857	11-031528	11-038785		
				11-039017	18-031448				
				CREEP PROPERTY					
				11-033336					
				CRITICAL ASSEMBLY					
				1-023168	5-029374	5-033542	6-003657		

6-007670	6-008090	6-010978	6-011424	17-033982	18-005832	18-005877	18-010356
6-016735	6-017008	6-017565	6-029301	18-014800	18-018942	18-019560	18-020046
7-028508	9-010978	9-011424	12-007670	18-020482	18-029433	18-029435	18-029439
12-011424	17-007670	17-029732	17-032839	18-029440	18-029448	18-029467	18-029469
17-032933	18-011429	18-029301	18-029374	18-029470	18-029488	18-029501	18-029514
18-032839	18-032933	18-033542		18-029516	18-031574	18-032818	18-032820
CRITICAL MASS				18-032925	18-033977	18-036597	
6-032821	18-032821			DESIGN STUDY			
CRITICALITY EXPERIMENT				1-006974	1-010160	1-010510	1-011398
1-023168	5-029374	6-010139	6-021245	1-021390	1-030719	1-030761	1-030762
6-032821	7-021245	9-010139	12-021245	1-033429	2-033535	5-033427	5-033977
12-024471	13-024471	17-005375	18-029374	5-035885	5-036236	5-036597	5-106947
18-032821				6-004027	6-006974	6-007247	6-023162
CRITICALITY SAFETY				6-029321	6-033427	6-036597	6-037182
3-014469	3-023755	6-003901	6-032391	7-029411	7-037186	7-106947	8-010113
6-037747	9-023358	12-024471	12-032391	8-018634	9-012383	9-016943	9-027541
13-005315	13-024471	13-036276		9-028929	9-029122	9-029321	9-032590
CROSS SECTION				9-033950	9-034784	9-035244	9-036206
6-011249	6-021244	6-035039	6-036600	10-028896	10-030767	11-018634	11-033448
18-035039	18-036600			11-033535	11-033953	11-033965	11-033977
CRUD				11-035871	11-036825	11-037182	12-034696
5-032542	11-032543	17-012102	18-032542	12-034698	12-034705	12-034784	13-034696
18-032543				13-034697	17-012383	17-030571	17-033429
CRYOGENICS				18-007317	18-010118	18-010318	18-010748
5-017576	6-017576	8-017576		18-012007	18-012008	18-012009	18-012010
CSE				18-012011	18-012012	18-012013	18-016530
5-018209	11-018209	12-018209		18-016652	18-016925	18-017092	18-017113
CURIUM				18-018942	18-019562	18-020046	18-021111
13-033337				18-021112	18-021113	18-021114	18-021115
CVTR (PWR)				18-021116	18-021211	18-022583	18-029812
10-004893	10-018293	11-002808	11-002810	18-030399	18-030571	18-030761	18-030762
17-003970				18-033274	18-033427	18-033429	18-033448
DAMAGE				18-033535	18-033950	18-033951	18-033952
2-032880	7-029548	10-028796	10-029478	18-033953	18-033965	18-033966	18-033968
11-032880	17-028643	17-028796	17-029548	18-033977	18-033985	18-033987	18-033989
17-029835	17-030218	18-029478	18-032880	18-033999	18-034655	18-034658	18-034662
DAMPING				18-034668	18-035869	18-035870	18-035871
2-029437	18-029437			18-035883	18-036236	18-036597	18-036612
DANGER COEFFICIENT				18-036614	18-036632	18-036634	18-036635
6-007436	6-007441	6-016735	9-007436	18-036636	18-036929	18-037489	18-106947
9-007441	11-007436			DESTRUCTIVE TRANSIENT			
DATA COLLECTION				6-007247			
1-044436	1-049354	3-044436	9-036206	DESTRUCTIVE WIND			
DATA PROCESSING				2-029442	2-032880	2-032917	10-032917
5-032526	6-034390	9-023347	9-032526	11-029452	11-029458	11-032880	18-029442
9-034390	17-021352	18-020482		18-029452	18-029458	18-032880	18-032917
DECAY HEAT				DIFFUSION			
3-012533	7-034421			7-015212			
DECOMMISSIONING				DIFFUSION COEFFICIENT			
8-014787	12-032303	17-007710	17-007736	7-015212			
17-008825	17-014787	17-014856	17-014858	DIRECT ENERGY CONVERSION DEVICES			
17-014859	17-016765	17-016766	17-019481	6-036600	7-034673	11-034673	18-014675
17-019558	17-020516	17-020665	17-024647	18-033968	18-034673	18-036600	
17-025315	17-027844	17-029737	17-029975	DISPLACEMENT			
17-030238	17-031271	17-032303	17-032304	5-032879	18-032879		
17-032469	17-035763	17-036110	17-036111	DISPLACEMENT DESIGN			
DECONTAMINATION				2-029483	9-029484	18-029483	18-029484
7-005008	7-006379	7-018343	7-027031	DISSOLUTION			
7-027818	7-037181	7-037186	9-030748	13-028951	13-034697	13-036276	
11-030748	12-006308	12-032303	17-016766	DISTANCE			
17-019861	17-020045	17-022914	17-025901	2-022328	11-022328		
17-032303				DOPPLER COEFFICIENT			
DEFORMATION				1-007695	1-023168	5-030633	5-034635
6-036078	9-036078	17-036078		6-003219	6-003904	6-003911	6-003913
DELAYED NEUTRON				6-003915	6-003916	6-007695	6-008090
1-007695	6-003913	6-006312	6-007695	6-011424	6-023162	6-025361	6-025690
6-032797	8-006312	9-007695	9-034663	6-025692	6-025876	6-029321	6-029466
17-006312	17-007695			6-031349	6-031439	6-032797	6-032821
DEPOSITION				6-033515	6-033964	6-036060	8-003904
7-022489	7-027029	7-027283	7-029940	9-007695	9-010504	9-011424	9-029321
7-031338	7-032699	7-032843	11-032843	11-030609	12-011424	12-030609	12-035196
12-027283	12-032699	18-032843		17-007695	18-007253	18-029466	18-030609
DESALTING				18-030633	18-032821	18-033964	18-033989
7-025254				18-033990	18-034634	18-036060	
DESIGN CRITERIA				DOPPLER EFFECT			
1-004268	1-025467	2-029433	2-029435	1-028163	5-030633	5-032624	5-033427
2-029439	2-029440	3-034707	3-034711	6-004027	6-007670	6-010609	6-016735
5-000899	5-004938	5-031574	5-032266	6-017565	6-028163	6-033427	9-019758
5-033977	5-035885	5-036597	5-038234	9-028163	11-030609	12-007670	12-030609
6-014800	6-029501	6-032266	6-032391	17-007670	17-019758	18-007253	18-012009
6-036597	6-037182	7-004938	7-007346	18-030609	18-030633	18-033427	18-034634
7-029411	7-029516	7-030378	7-031068	DOSE			
7-037181	7-037186	9-010439	9-023327	2-002801	5-029374	5-029375	5-030633
9-026513	9-029440	9-029871	9-029961	7-029516	11-002801	11-029518	18-016652
9-032820	9-035244	9-038092	10-026513	18-029374	18-029375	18-029516	18-029518
10-029439	11-010356	11-026513	11-029433	18-029519	18-030633		
11-029448	11-029467	11-029469	11-029470	DOSE CALCULATION, EXTERNAL			
11-029488	11-031977	11-032925	11-033977	2-022328	11-022328		
11-034453	11-037182	12-010356	12-019567	DOSE CALCULATION, INTERNAL			
12-029514	12-031068	12-032391	12-032931	2-022328	11-022328		
12-034705	12-034707	13-034711	17-019561	DOSE MEASUREMENT, EXTERNAL			

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17-016408	17-020727	17-029794	17-029831	6-005810	6-006227	6-007435	6-007444
17-030571	17-030675	17-031273	17-032619	6-009711	6-018195	6-018946	6-019757
17-032627	17-032629	18-030571		6-037850	7-007346	7-009711	7-010559
EXCURSION, LARGE				7-011795	7-011945	7-018195	7-018947
5-032893	6-010959	6-032797	6-032889	7-019555	7-019756	7-024386	7-028511
6-032890	6-032892	6-032894	6-032896	7-029548	7-029834	7-029949	7-030253
6-032919	6-034669	9-010439	9-032889	7-035866	7-037181	7-037183	7-037184
9-032890	9-032894	9-032919	11-032894	8-007444	8-018195	9-018814	9-019758
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18-032892	18-032893	18-032894	18-032896	11-032629	12-037850	17-003644	17-007435
18-032919	18-034669			17-007444	17-011795	17-011945	17-018857
EXPANSION				17-018946	17-018947	17-019555	17-019756
11-023857	11-031977	11-032543	17-017500	17-019757	17-019758	17-019759	17-019860
18-017500				17-019862	17-021216	17-021741	17-029548
EXPERIMENT				17-029831	17-029834	17-029930	17-029934
3-023229	5-030633	5-032853	5-037463	17-030806	17-032629	17-034636	18-011247
7-027283	11-030609	12-027283	12-030609	18-030210	18-030253	18-032542	18-034636
12-030879	12-032271	12-032392	12-032853	FAILURE, FUEL ELEMENT WATERLOGGING			
12-032876	12-035044	12-036795	13-023229	18-030210			
13-028951	13-030879	13-033337	18-030609	FAILURE, INCIPIENT			
18-030633				5-032526	9-032526	9-037406	17-037406
EXPLOSION				FAILURE, INSTALLATION ERROR			
6-003904	7-029940	7-036473	8-003904	5-020733	17-018032	17-018682	17-020733
8-014787	11-000749	11-036473	13-005315	17-029616			
17-008056	17-010986	17-014787		FAILURE, INSTRUMENT			
EXPLOSIVE, CONVENTIONAL				9-009794	9-012303	9-019758	9-037487
5-036468	8-036468	11-036465	11-036466	17-009794	17-012303	17-013314	17-013315
11-036468	11-036469	18-036465		17-016149	17-018681	17-019758	17-019762
EXTRAPOLATION FACTOR				17-037487			
7-031339				FAILURE, OPERATOR ERROR			
FABRICATION				5-029297	5-032895	17-006317	17-018682
1-029865	1-033932	3-033428	7-015934	17-021736	17-029297	17-029835	17-030806
7-015935	7-015936	7-031068	7-034673	18-032891	18-032895		
8-015934	8-015935	8-015936	9-036239	FAILURE, PIPE			
9-036482	11-029487	11-031973	11-031977	5-029297	5-029459	6-003589	9-003589
11-033336	11-034673	12-015934	12-015935	11-002803	11-036779	17-003589	17-006458
12-015936	12-029865	12-031068	12-032396	17-011138	17-019559	17-025996	17-028643
13-033428	17-020467	17-020468	17-020469	17-029297	17-029616	17-031284	17-032472
17-020470	17-020472	18-021116	18-024002	17-036779	18-029459		
18-025002	18-026596	18-029487	18-029865	FAILURE, PRESSURE VESSEL			
18-031032	18-033428	18-033932	18-033951	5-032922	11-002803	18-032922	
18-033952	18-033989	18-033990	18-034673	FAILURE, SCRAM MECHANISM			
18-036146				9-005263	9-008052	9-012383	12-005263
FAILURE				17-005263	17-006977	17-008052	17-011994
6-005807	9-005263	10-005262	11-029493	17-012383	17-013315		
12-005262	12-005263	12-005807	12-024373	FAILURE, SEQUENTIAL			
17-005262	17-005263	17-005264	17-005807	9-009794	10-020579	17-009794	17-020579
17-018681	17-018682	17-019557	17-019558	17-030806			
17-019559	17-021081	17-021216	18-029493	FAILURE, SIMULTANEOUS			
FAILURE MODE ANALYSIS				5-032903	10-020579	12-032903	17-020579
11-029492	18-029492			18-032903			
FAILURE, ADMINISTRATIVE CONTROL				FAILURE, TUBING			
6-019757	17-013315	17-018682	17-019757	5-020733	8-010113	9-023947	11-031470
17-021216	17-026099			12-019567	17-013536	17-018682	17-020733
FAILURE, CLADDING				17-030192	17-030571	18-017355	18-023947
5-019860	5-029507	5-037850	6-009711	18-023948	18-030571		
6-018195	6-028562	6-037850	7-009711	FALLOUT			
7-018195	7-019555	7-019756	7-037183	7-027029	7-034360		
8-018195	9-009107	9-019862	12-037850	FAST NEUTRON			
17-006914	17-009107	17-018681	17-018682	11-032543	18-032543		
17-019555	17-019756	17-019860	17-019862	FASTENER			
17-028562	18-009964	18-029507		6-032896	11-029469	11-029487	11-029489
FAILURE, COMPONENT				11-029490	11-029491	11-029492	11-029493
6-006357	6-007435	6-009709	9-006357	18-029469	18-029487	18-029489	18-029490
9-022457	11-002803	11-007435	11-011394	18-029491	18-029492	18-029493	18-032884
11-029487	11-029488	11-029490	12-006488	18-032896	18-033992		
17-006357	17-006488	17-006973	17-007167	FAULT			
17-007435	17-007738	17-012102	17-014857	17-029616	17-033489		
17-017093	17-022457	18-029487	18-029488	FERMI (LMFBR)			
18-029490				1-006974	1-007695	1-021968	3-029831
FAILURE, DESIGN ERROR				3-034706	3-034707	5-000899	5-005116
5-018857	5-029297	17-008984	17-016150	5-011040	5-011673	5-011674	5-017167
17-018681	17-018682	17-018857	17-019556	5-026427	5-027595	5-027596	5-027725
17-021741	17-029297	17-029851		5-029606	5-029831	5-029984	5-031574
FAILURE, EQUIPMENT				5-032526	5-032627	5-032629	5-035345
2-032915	3-034706	6-003589	7-019756	5-040402	6-003901	6-003904	6-006227
9-003589	9-019442	12-033050	12-034706	6-006974	6-007695	6-010978	6-011996
13-034706	16-032915	17-003589	17-006555	6-017008	6-017132	6-018195	6-018946
17-006914	17-006976	17-006979	17-007708	6-025692	6-027774	6-027775	6-028801
17-012102	17-013314	17-016148	17-016150	6-028802	6-031354	6-040402	7-011795
17-018681	17-018682	17-019756	17-019762	7-011945	7-013751	7-019594	7-018195
17-021741	17-026398	17-029851	17-030433	7-018947	7-029547	7-029548	8-003904
17-034706	18-029498	18-032915		8-010359	8-015934	8-018195	9-007695
FAILURE, FABRICATION ERROR				9-007720	9-007721	9-007728	9-010523
11-036779	17-018681	17-036779	18-031032	9-010978	9-011441	9-012256	9-012383
18-036146				9-013355	9-017132	9-019500	9-020638
FAILURE, FATIGUE				9-020709	9-023146	9-023331	9-023946
11-032543	17-013315	17-029616	18-032543	9-024332	9-026513	9-027595	9-027596
FAILURE, FUEL ELEMENT				9-027774	9-028016	9-029040	9-029877
3-029831	5-018857	5-019860	5-029831	9-032526	9-032627	9-032628	9-033128
5-030253	5-032542	5-032629	5-037850	9-037406	9-037487	10-004893	10-004894

10-018293	10-020579	10-026513	10-028796	17-032940	17-032941	17-032942	17-032943
11-000749	11-002808	11-002809	11-002810	18-030253	18-032939	18-032940	18-032941
11-012319	11-026513	11-029714	11-032629	18-032942	18-032943	18-034634	18-034673
11-033016	12-015934	12-034706	12-034707	FISSION PRODUCT ACTIVITY, GROSS			
13-034706	17-003645	17-003970	17-006317	7-018947	7-019555	7-034421	17-018947
17-006977	17-006979	17-007695	17-007720	17-019555	FISSION PRODUCT RELEASE		
17-007721	17-007728	17-008867	17-008984	1-044436	1-049354	3-014870	3-044436
17-008985	17-010050	17-010622	17-010624	5-029375	5-036802	6-006561	7-004216
17-011040	17-011138	17-011795	17-011945	7-005008	7-006379	7-006380	7-008031
17-012319	17-012383	17-013238	17-013239	7-010558	7-010559	7-013907	7-014870
17-013314	17-013315	17-013534	17-013535	7-018947	7-019555	7-027283	7-029411
17-013536	17-013537	17-013979	17-014804	7-029516	7-029940	7-029949	7-030790
17-016408	17-017167	17-018032	17-018946	7-032881	7-034361	7-035866	7-037183
17-018947	17-020579	17-020638	17-020727	8-006380	8-036802	9-006561	9-010504
17-021216	17-021968	17-022900	17-022902	11-029518	12-027283	17-004301	17-006561
17-022903	17-022904	17-022905	17-022906	17-018947	17-019555	17-029930	18-029375
17-022907	17-022909	17-023776	17-023946	18-029510	18-029516	18-029518	18-032881
17-025901	17-025979	17-025996	17-026099	18-036802	FISSION PRODUCT RETENTION		
17-026408	17-027595	17-027596	17-027774	1-027827	5-032629	5-106947	6-021245
17-027775	17-027776	17-028643	17-028644	7-010297	7-010558	7-013907	7-015212
17-028796	17-028798	17-028801	17-028802	7-018291	7-019555	7-019756	7-021245
17-029040	17-029548	17-029714	17-029831	7-018291	7-019555	7-022489	7-024386
17-029851	17-029932	17-029933	17-029934	7-022489	7-029411	7-029949	7-037181
17-029939	17-029982	17-029984	17-030192	7-106947	9-010504	11-027827	11-032629
17-030218	17-030278	17-030432	17-030433	11-033336	12-021245	17-019555	17-019756
17-030571	17-031273	17-031274	17-031275	17-032629	18-106947	FISSION PRODUCT TRANSPORT	
17-031282	17-031283	17-031284	17-031286	7-006379	7-006380	7-008031	7-010533
17-032324	17-032471	17-032472	17-032607	7-010558	7-013548	7-013907	7-014384
17-032619	17-032627	17-032628	17-032629	7-014385	7-018291	7-022489	7-024386
17-033016	17-034706	17-035802	17-037406	7-027029	7-027283	7-029949	7-030790
17-037469	17-037487	17-040390	17-040402	7-032253	7-032699	7-033156	7-034361
18-008423	18-010862	18-011295	18-011672	7-037184	8-006380	8-013548	8-014385
18-014804	18-029606	18-030571	18-031574	11-033953	12-027283	12-032699	18-033953
FFTF (TR)				FISSION PRODUCT, IODINE			
1-033932	5-010430	5-011299	5-017339	7-006379	7-010558	7-013548	7-013907
5-028889	5-032266	5-032542	5-032836	7-019555	7-019756	7-024343	7-029940
5-032853	5-033333	5-036238	5-036597	7-034361	7-037183	8-013548	17-019555
6-006312	6-007436	6-007441	6-007670	17-019756	17-030806	FISSION PRODUCT, VOLATILE	
6-010430	6-016736	6-023870	6-032266	7-019555	7-029411	7-029834	7-030790
6-033333	6-033964	6-036597	7-007280	17-019555	17-029834	FISSION RECOIL	
7-007346	7-031338	7-031339	7-032265	7-004216	FLANGE		
7-032843	8-006312	8-023870	9-007436	11-029468	11-029493	18-029468	18-029493
9-007441	9-012256	9-016943	9-023327	FLOOD			
9-023947	9-025177	9-027582	9-028929	2-029512	18-029512	FLOW	
9-029871	9-029961	9-030945	9-031241	5-029503	5-032819	6-028802	6-034390
9-032590	9-032836	9-034784	9-035244	9-034390	17-028802	17-032819	18-029503
9-036206	9-036238	10-028901	10-030092	18-032819	FLOW BLOCKAGE		
11-007436	11-009746	11-032543	11-032836	1-021968	3-029831	5-011673	5-017167
11-032843	11-033953	11-033965	11-036825	5-019860	5-029831	5-029984	5-032627
12-007670	12-024471	12-030664	12-032392	5-032629	5-032882	6-003901	7-019756
12-032853	12-034696	12-034705	12-034784	9-020638	9-032627	9-032628	11-032629
12-035266	12-036504	13-024471	13-030664	17-017167	17-018032	17-019756	17-019860
13-034696	17-006312	17-007670	17-032844	17-020638	17-020727	17-021216	17-021968
17-032939	17-032940	17-032941	17-032942	17-025901	17-025935	17-026099	17-026408
17-032943	18-009746	18-012007	18-012138	17-029831	17-029984	17-030432	17-031273
18-017341	18-023870	18-023947	18-023948	17-031275	17-032472	17-032619	17-032627
18-024515	18-029812	18-032265	18-032542	17-032628	17-032629	18-025935	18-030210
18-032543	18-032843	18-032844	18-032939	18-032882	FLOW DISTRIBUTION		
18-032940	18-032941	18-032942	18-032943	5-032819	5-032853	5-034497	11-030845
18-033333	18-033932	18-033951	18-033952	12-030845	12-032853	17-029851	17-030845
18-033953	18-033964	18-033965	18-033966	17-032819	18-032819	FLOW STABILITY	
18-036597	18-036614	18-036929	18-037489	5-018857	5-038234	6-027775	17-018857
FILTER CHARACTERISTICS				17-027775	17-029851	FLOW THEORY AND EXPERIMENTS	
12-029514	18-029514	5-026461 5-032819 5-034635 12-032392					
FILTER, COMBINATION				17-026300	17-032819	18-029502	18-032819
12-029514	18-029514	FLOW, ANNULAR					
FILTER, LIQUID				12-032392	FLOW, BYPASS		
5-017576	6-017576	8-017576	5-032879 18-032879				
FILTER, TRAP				5-029503	18-029502	18-029503	FLOW, CONVECTION
7-014384	7-027820	7-034361	7-037181	5-036396	5-036397	FLOW, CRITICAL	
7-037186	5-038396						
FILTERS				5-038396	FLOW, CROSS		
2-002811	7-005008	7-007346	9-010504	5-026121	5-026461	5-026634	5-034497
9-036239	11-002809	11-002811	5-034694				
FIRE				5-026121	5-026461	5-026634	5-034497
5-017339	5-036802	6-006312	7-006380	5-034694	FLOW, TWO PHASE		
7-007280	7-013548	7-014385	7-022479	5-016164	5-034496	5-036396	5-036397
7-022514	7-027029	7-027030	7-029940	5-038234	6-010057	6-016164	9-016164
7-029949	7-032843	7-037186	8-006312				
8-006380	8-013548	8-014385	8-027030				
8-036802	9-010504	9-026513	10-026513				
11-000749	11-026513	11-029452	11-029457				
11-029458	11-032843	11-036627	13-005315				
17-006312	17-008375	17-029835	17-036627				
18-005834	18-029452	18-029457	18-029458				
18-032843	18-036802						
FISSION GAS RELEASE							
5-030253	7-008031	7-010533	7-010552				
7-019555	7-029411	7-029834	7-030253				
7-031003	7-034673	7-037181	11-034673				
12-031003	17-019555	17-029834	17-032939				

GAMMA	12-034696	13-034696	17-026301		3-012533	5-003647	5-007556	5-010430
GAMMA EMITTER					5-011915	5-026634	5-027552	5-032266
7-014384					5-034496	5-035822	5-037838	5-037957
GAS					5-038234	5-038396	5-040402	6-006673
5-040402	6-040402	7-032843	9-035244		6-010430	6-011424	6-032266	6-040402
11-032843	17-040402	18-032843			7-007556	9-011424	12-007556	12-011424
GENERATOR, DIESEL					12-029179	12-037838	17-006673	17-016032
2-029442	6-005807	10-030767	12-005807		17-037838	17-040402	18-005832	18-011429
17-005807	17-014857	17-035802	18-029442		18-021086			
GERMANY					HEAT TRANSFER ANALYSIS			
1-010160	1-011398	1-030762	5-007188		3-023755	5-017576	5-026427	5-031420
5-033281	5-033984	6-032797	7-024343		5-032266	5-032629	5-032815	5-036957
7-032250	8-018634	11-018634	11-033336		6-017576	6-032266	8-017576	8-031420
13-033337	17-033982	18-007188	18-010318		8-036957	11-032629	11-036957	17-032629
18-012011	18-012145	18-019142	18-021211		18-032809	18-032815		
18-022105	18-028942	18-030574	18-030762		HEAT TRANSFER COEFFICIENT			
18-033985	18-034765	18-034767			5-031909			
GRAPHITE					HEAT TRANSFER EXPERIMENT			
7-007346	7-013751	7-015212	7-015538		5-011915	5-014676	5-018209	5-026634
12-006488	17-006317	17-006488	17-006914		5-029506	5-031420	5-034496	6-029506
17-018682	17-030218	17-032324			7-029287	8-031420	11-018209	12-018209
GROSS ALPHA					17-014676	18-029506		
17-026164					HEAT TRANSFER, BOILING			
GROSS BETA					5-003647	5-009520	5-011673	5-014676
17-026164	17-026203				5-017577	5-018209	5-026286	5-027438
GROSS GAMMA					5-027440	5-031420	5-034496	5-034635
17-026203					6-017577	7-029287	8-031420	11-018209
GROUND WATER					12-018209	17-013535	17-014676	18-016379
3-014870	7-014870				HEAT TRANSFER, CONDENSATION			
GUIDE					5-017577	6-017577		
2-029483	9-029484	11-029456	17-029835		HEAT TRANSFER, CONDUCTION			
18-029456	18-029483	18-029484	18-032806		5-007556	5-037462	7-007556	9-029122
18-033992					12-007556	18-037462		
HAFNIUM					HEAT TRANSFER, CONVECTION			
9-036775					5-026286	5-031909	18-029502	
HALLAM (LMR)					HEAT TRANSFER, NATURAL CONVECTION			
3-014469	3-014870	3-023755	5-005444		5-017576	6-017576	8-017576	
5-009456	5-011915	5-029375	6-003589		HELIUM			
6-004026	6-004479	6-005094	6-006357		1-007695	6-007695	6-017131	7-028511
6-008211	6-010509	7-014870	7-015538		7-038775	9-007695	11-032986	17-007695
8-014787	9-003589	9-006357	9-008211		17-017131	18-011672		
9-009794	9-010504	9-010509	9-012303		HFIR (FTR)			
11-002803	11-002808	11-011394	12-004440		6-007444	8-007444	17-007444	
12-006488	12-032303	17-003589	17-003637		HIGH TEMPERATURE			
17-003644	17-003970	17-004028	17-005375		3-030256	5-037462	6-009711	7-009711
17-006317	17-006357	17-006488	17-006914		7-031505	9-016943	9-023329	9-023332
17-006973	17-007738	17-008375	17-008825		9-023335	9-023350	9-036200	9-036482
17-009794	17-010050	17-011428	17-012102		9-036650	9-036766	9-036773	9-036775
17-012303	17-014787	17-014856	17-014857		9-038580	11-031443	11-031528	11-031973
17-014858	17-014859	17-016131	17-016765		11-031977	11-032986	11-034453	12-031505
17-016766	17-018595	17-018681	17-018682		12-034708	12-035044	12-036504	13-034708
17-019481	17-019558	17-020516	17-020665		17-016032	18-030256	18-037462	
17-021081	17-024647	17-026164	17-026203		HOT CELL			
17-026398	17-027844	17-029737	17-029931		3-029831	5-029831	11-010356	11-029462
17-030238	17-031271	17-032303	17-032304		12-010356	12-034696	12-034698	13-034696
17-032469	17-035763	17-036110	17-036111		17-022498	17-029831	18-010356	18-029462
18-009456	18-011333	18-018942	18-029375		HOT CHANNEL			
HANFORD (PR)					5-026635	5-029505	18-029505	
7-005008	11-002803	11-002809			HOT CHANNEL ANALYSIS			
HANFORD SITE					5-033281	5-035822		
12-024471	13-024471				HOT CHANNEL FACTOR			
HAZARDS ANALYSIS					5-033281			
3-012530	5-004938	5-029374	5-029375		HOT SPOT			
5-030633	5-033542	7-004938	7-032265		5-029505	18-029505		
18-017341	18-021080	18-029374	18-029375		HRT (RE)			
18-030633	18-032265	18-033542			6-005122			
HEAT CONDUCTANCE, FUEL TO CLAD					HUMBOLDT BAY (BWR)			
17-021335					2-002811	10-004893	10-018293	11-002808
HEAT EXCHANGERS					11-002809	11-002810	11-002811	17-003970
1-029865	2-029442	5-017576	5-020733		HWCTR (TR)			
5-033056	5-035885	6-017576	7-015934		2-002811	11-002808	11-002811	
7-015935	7-015936	8-015934	8-015935		HYDRAULIC ANALYSIS			
8-015936	8-017576	8-033056	9-038092		5-028889	5-032815	5-033984	5-037850
12-015934	12-015935	12-015936	12-019567		6-007247	6-007444	6-010057	6-037850
12-029865	12-032876	12-033056	17-005264		8-007444	9-023359	12-037850	17-007444
17-006317	17-007708	17-007736	17-008867		18-019142	18-021116	18-032815	18-034658
17-009420	17-016456	17-018681	17-018682		HYDRAULIC EFFECT			
17-019558	17-019762	17-020733	17-021081		17-005375	17-016033		
17-021741	17-027844	18-012165	18-029442		HYDRAULIC EXPERIMENT			
18-029865	18-032810	18-032811	18-036929		5-029984	9-023359	17-029984	18-011152
HEAT FLUX, CRITICAL					HYDRIDE			
5-037462	18-037462				5-032629	9-023946	11-032629	17-023946
HEAT GENERATION, INTERNAL					17-031273	17-031275	17-032629	18-012145
5-007556	7-007556	12-007556			18-022105			
HEAT PIPE					HYDRODYNAMIC ANALYSIS			
9-029122					5-034497	5-034694	6-005104	17-032939
HEAT SINK					17-032940	17-032941	17-032942	17-032943
5-029508	7-032843	7-036473	11-032843		18-032939	18-032940	18-032941	18-032942
11-036473	18-029508	18-032843			18-032943			
HEAT TRANSFER					HYDROGEN			
					5-011299	7-013751	17-010986	17-028644

IAEA				17-028562	17-032628	17-037406	18-029480
17-005375				18-030254			
IMPACT SHOCK				INSTRUMENT, FLOW			
5-005116				5-029002	6-006673	9-016943	9-019500
IMPURITY				9-020709	9-023328	9-023329	9-023331
5-029002	7-033156	9-029002	12-029002	9-023332	9-023359	9-025615	9-028929
IN CORE MEASUREMENT				9-029002	9-029877	9-036200	12-029002
9-023327	9-025615	9-036766	9-036775	17-006673	17-013535		
IN PILE EXPERIMENT				INSTRUMENT, FLUIDICS			
5-004938	5-031420	5-037850	6-007670	9-036775			
6-007670	6-009711	6-010058	6-010058	INSTRUMENT, FUEL SCANNING			
6-037850	7-004938	7-007346	7-007346	9-031241	17-026301		
7-009711	7-031338	7-031339	7-032265	INSTRUMENT, IN CORE			
8-031420	9-036766	11-016576	12-007670	5-027595	5-027596	6-007670	6-028802
12-007670	12-037850	17-007670	17-007670	9-003587	9-008456	9-010504	9-016943
17-034636	17-034636	18-016576	18-032265	9-023327	9-023328	9-023329	9-023350
18-034636	18-034636			9-023351	9-025177	9-025615	9-027582
INCIDENT				9-027595	9-027596	9-029120	9-029481
3-029831	5-019860	5-029831	5-032627	9-029871	9-029961	9-032590	9-033950
5-040402	6-018946	6-040402	7-011795	9-036206	9-036239	9-038058	11-023813
7-018947	7-019555	7-019756	9-020638	12-007670	17-007670	17-008097	17-027595
9-032627	9-037406	17-006914	17-010263	17-027596	17-028802	18-021111	18-021112
17-011795	17-018032	17-018946	17-018947	18-021113	18-021114	18-021115	18-021116
17-019555	17-019756	17-019860	17-020638	18-029481	18-033950		
17-021216	17-025935	17-029831	17-029934	INSTRUMENT, INTERLOCK			
17-032627	17-037406	17-040402	18-025935	17-014857			
INCIDENT, CONSEQUENCE				INSTRUMENT, INTERMEDIATE RANGE			
6-034390	9-034390			6-030252	18-030252		
INCIDENT, EQUIPMENT				INSTRUMENT, LIQUID LEVEL			
5-029297	9-022457	17-006457	17-006458	9-011441	9-013337	9-022457	9-023330
17-007167	17-008056	17-022457	17-029297	9-023332	17-006457	17-022457	
INCIDENT, HUMAN ERROR				INSTRUMENT, NUCLEAR			
17-008375	17-021736	17-030806		6-010139	7-010559	9-003587	9-007728
INCIDENT, RECOVERY FROM				9-010139	9-012303	9-023329	9-023351
5-032627	5-032904	5-032906	9-020638	9-029120	9-033128	9-034663	9-036206
9-032627	12-032904	12-032906	17-007167	9-036239	17-005375	17-007728	17-012303
17-016408	17-019759	17-019861	17-020638	17-037469	18-005832		
17-021741	17-022498	17-028644	17-032619	INSTRUMENT, OPERATING REACTIVITY			
17-032627	17-037469	17-040390	18-032806	5-032627	9-020638	9-032627	17-020638
18-032904	18-032906			17-032627			
INCOLOY				INSTRUMENT, PERIOD			
7-028987	11-028987	11-031443	11-031528	9-019758	17-019758		
11-032986				INSTRUMENT, POWER RANGE			
INCONEL				9-020709	9-029473	18-029473	
5-036238	9-036238	11-029491	18-029491	INSTRUMENT, PRESSURE			
INDEPENDENCE				9-023328	9-023330	9-023331	9-023332
9-022457	11-029454	17-022457	18-029454	9-029481	18-029481		
INDIAN POINT 1 (PMR)				INSTRUMENT, PROCESS			
2-002811	10-004893	10-018293	11-002808	9-005333	9-011441	9-013337	9-023330
11-002810	11-002811	17-003970		9-023331	9-023332	9-025177	9-031241
INDUSTRY, UTILITY				9-036206	17-016149		
1-044436	3-044436			INSTRUMENT, PROTECTIVE			
INFORMATION RETRIEVAL				1-029994	7-029994	9-019862	9-023334
2-033535	5-029297	11-033535	17-029297	9-029471	9-029474	9-029475	9-029871
18-033535	18-033968			9-029994	9-036206	9-037406	17-019862
INSTABILITY				17-037406	18-029471	18-029474	18-029475
9-029040	17-029040	18-009964		INSTRUMENT, PULSE			
INSTRUMENT CALIBRATION				9-024332			
5-032836	9-020709	9-023331	9-023946	INSTRUMENT, STARTUP			
9-030945	9-032836	9-036773	11-032836	9-033128	17-011994		
17-006973	17-023946			INSTRUMENT, STARTUP RANGE			
INSTRUMENT, ABNORMAL INDICATION				9-033128	11-012319	17-012319	
5-032627	9-023946	9-030945	9-032627	INSTRUMENT, SURVEILLANCE			
9-032628	9-036766	17-006914	17-012102	1-029994	5-027595	5-029002	7-010559
17-013534	17-013535	17-023946	17-032627	7-029994	9-018814	9-027595	9-029002
17-032628				9-029994	9-035244	9-037487	12-029002
INSTRUMENT, COMPONENT				17-027595	17-037487		
5-036209	6-029501	9-018814	9-023328	INSTRUMENT, TEMPERATURE			
9-026513	9-027541	9-028929	9-029120	5-026427	5-032627	6-011424	6-027774
9-029128	9-033128	9-036209	9-036239	9-011424	9-016943	9-020709	9-023327
9-038058	10-026513	11-026513	18-029501	9-023328	9-023329	9-023330	9-023331
INSTRUMENT, CONTROL				9-023332	9-023334	9-023335	9-023359
2-029440	5-029375	9-008500	9-012069	9-025615	9-027582	9-027774	9-029473
9-023146	9-023334	9-023347	9-029120	9-030748	9-030945	9-031241	9-032627
9-029122	9-029440	9-029482	9-030743	9-033128	9-036206	9-036482	9-036650
9-033128	9-034784	9-036206	11-030609	9-036766	9-036773	9-036775	9-038093
12-030609	12-034784	18-020482	18-029375	9-038580	11-030748	12-011424	17-027774
18-029440	18-029482	18-030498	18-030609	17-032627	18-020482	18-029473	
INSTRUMENT, COOLANT QUALITY				INSTRUMENT, TESTING			
5-031448	9-023947	9-025177	9-029128	9-012130	9-018506	9-023327	9-023332
9-031241	17-030433	18-023947	18-023948	9-029120	9-029961		
18-031448				INSTRUMENT, VOLTAGE			
INSTRUMENT, CURRENT				9-036650			
9-036650				INSTRUMENT, WIDE RANGE			
INSTRUMENT, DETECTION FAILED FUEL ELEMENT				9-025835			
6-007108	6-028562	7-010559	7-035866	INSTRUMENTS, MISC.			
9-009107	9-009717	9-017479	9-019862	1-029994	1-033051	5-032893	5-036802
9-020638	9-023328	9-025615	9-030254	6-011424	6-032889	6-032890	6-032894
9-032590	9-032628	9-034663	9-036206	7-015934	7-015935	7-015936	7-029994
9-037406	17-006976	17-007108	17-009107	8-015934	8-015935	8-015936	8-036802
17-017479	17-019862	17-020045	17-020638	9-011424	9-019442	9-019862	9-023327

9-029120	9-029461	9-029994	9-031241	MAGNESIUM			
9-032889	9-032890	9-032894	9-033051	9-036775			
9-033364	9-035244	9-036206	9-038092	MAIN COOLING SYSTEM			
11-029461	11-032894	12-011424	12-015934	1-006974	1-029865	2-029441	2-032915
12-015935	12-015936	12-033051	17-006977	5-006817	5-029297	5-029503	5-029505
17-019862	17-029048	18-011429	18-019142	5-029508	5-029606	5-030255	5-032905
18-020482	18-029461	18-032889	18-032890	5-032922	5-035345	6-006974	6-027774
18-032893	18-032894	18-033951	18-033952	6-028562	7-015538	7-015934	7-015935
18-033990	18-036802			7-015936	7-018947	7-019555	7-019756
INSURANCE				8-015934	8-015935	8-015936	9-027774
2-002801	11-002801			10-030255	11-006817	11-029453	11-029493
INTEGRITY				11-030609	11-032905	11-036779	12-006817
5-006817	5-029507	11-006817	11-029454	12-015934	12-015935	12-015936	12-029495
11-029491	12-006817	18-029454	18-029491	12-029865	12-030255	12-030609	16-032915
18-029507				17-006973	17-007736	17-009420	17-016033
INTERACTION, FOUNDATION AND STRUCTURE				17-016148	17-018682	17-018947	17-019555
2-029438	18-029438			17-019558	17-019561	17-019756	17-019861
INTERACTION, SOIL AND STRUCTURE				17-020045	17-025901	17-026398	17-026408
2-029444	2-029445	2-029445	2-032817	17-027774	17-027844	17-028562	17-029297
11-029445	11-029449	18-029444	18-029445	17-029932	17-030278	17-032844	17-033489
18-029449	18-032817			17-036779	17-040390	18-014646	18-018942
IODINE				18-019560	18-019891	18-029441	18-029453
3-023229	5-018209	7-004216	7-013548	18-029493	18-029495	18-029497	18-029503
7-013907	7-014384	7-022479	7-022489	18-029505	18-029508	18-029606	18-029812
7-024343	7-029940	7-030790	7-031003	18-029865	18-030255	18-030609	18-032810
8-013548	11-002809	11-018209	12-018209	18-032811	18-032844	18-032905	18-032915
12-031003	13-023229			18-032922	18-033481		
IRON				MAINTENANCE AND REPAIR			
11-036616	18-036616			5-018857	5-027595	5-027596	9-027595
IRRADIATION FACILITY				9-027596	9-037487	11-010356	11-025992
9-031241	17-030806			11-033016	11-036779	12-010356	17-006317
IRRADIATION TESTING				17-006458	17-006555	17-008375	17-010622
6-007670	6-009711	7-009711	7-034673	17-010624	17-016148	17-016149	17-018857
9-030945	9-033950	11-016576	11-032543	17-019558	17-020045	17-022900	17-022902
11-033448	11-033953	11-033991	11-033993	17-022903	17-022904	17-022905	17-022906
11-033994	11-033995	11-033996	11-034673	17-022907	17-022909	17-025979	17-026398
11-036627	12-007670	12-034696	12-035363	17-027595	17-027596	17-029048	17-029835
13-034696	17-006457	17-007670	17-014804	17-029939	17-030218	17-032324	17-032472
17-020467	17-020468	17-020469	17-020470	17-033016	17-033489	17-036779	17-037469
17-020472	17-021335	17-029825	17-029935	17-037487	18-005832	18-007317	18-010118
17-036627	18-014804	18-016576	18-018745	18-010356	18-020046	18-025992	18-033274
18-021115	18-021116	18-022105	18-024002	MAINTENANCE, REMOTE			
18-025002	18-032543	18-033448	18-033950	5-018857	5-027595	5-027596	9-027595
18-033953	18-033989	18-033991	18-033993	9-027596	12-034698	12-034708	13-034708
18-033994	18-033995	18-033996	18-034673	17-018857	17-027595	17-027596	
ITALY				MANGANESE			
5-031574	18-031574			7-031338			
JACOBS				MANITOBA			
1-029865	1-029994	1-033429	2-022328	11-036616	18-036616		
2-033535	3-033428	5-036957	5-106947	MASS TRANSFER			
7-029994	7-031068	7-033156	7-106947	5-037838	7-031338	7-031339	12-037838
8-036957	9-029120	9-029994	11-022328	17-037838			
11-033448	11-033535	11-036957	12-029865	MATERIAL			
12-031068	13-033428	17-033429	18-029865	5-036238	5-037462	6-007108	9-008052
18-033428	18-033429	18-033448	18-033535	9-036238	11-033953	12-030879	13-030879
18-106947				17-007108	17-008052	17-019561	18-007317
JAPAN				18-024002	18-025002	18-032810	18-032811
18-030576	18-034632			18-033953	18-037462		
KARLSRUHE				MATHEMATICAL STUDY			
11-033336	13-033337	18-019142		6-010509	6-021244	9-010150	9-010509
KIWI (S-RR)				12-032814	18-032814	18-033987	
6-005810				MATHEMATICAL TREATMENT			
KRYPTON				1-028163	2-029445	5-017167	5-031909
7-007346	7-010558	7-014384	7-029949	5-034635	5-035822	6-028163	6-031349
7-034361	7-037181	7-037183		6-031439	6-032391	6-032797	7-032843
LAMPRE 1 (RE)				9-023146	9-028163	11-029445	11-032843
6-003657	6-010959	7-007346	9-012256	12-029133	12-032391	12-032814	12-032876
12-032271				17-017167	17-026408	17-034636	18-029445
LASL				18-032814	18-032843	18-034636	
18-014675				MEASUREMENT			
LEAK				6-007670	6-007848	6-008090	6-017008
2-032915	5-029297	5-032905	7-038775	6-035039	6-036600	7-029834	9-036206
11-032905	11-036779	16-032915	17-019556	12-007670	17-007670	17-007848	17-016033
17-026408	17-029297	17-029616	17-030433	17-021352	17-029834	17-029982	17-031275
17-032324	17-033489	17-036779	18-032905	18-035039	18-036600		
18-032915				MEASUREMENT, REACTIVITY			
LEAK RATE				1-023168	5-032893	6-010139	6-017008
7-038775	11-029518	17-032294	18-029518	6-017565	6-032821	6-032889	6-032890
LIAPUNOV'S FUNCTION				6-036060	6-036078	9-010139	9-010523
6-025357				9-012086	9-032889	9-032890	9-036078
LICENSE APPLICATION				17-006976	17-029932	17-029933	17-036078
17-032632	18-032632			18-032821	18-032889	18-032890	18-032893
LICENSE STATUS				18-036060			
17-013314				MEASUREMENT, STRAIN GAGE			
LIQUID FUEL				17-029048			
9-033364	12-032876	17-016032		MEASUREMENT, TEMPERATURE			
LITHIUM				9-005333	9-016943	9-023335	9-032590
9-033375	12-030879	13-030879		9-036775	17-019561		
LUBRICATION				MERCURY			
7-032587	11-029489	17-032587	18-029489	7-027031			
18-032587				METAL			

5-031448	11-033953	11-036616	18-031448	MPBE (MOLTEN PLUTONIUM BURNUP EXPERIMENT)
18-033951	18-033952	18-033953	18-036616	6-010959 7-007346
METAL WATER REACTION				NTR (TR)
5-010430	5-033056	6-005810	6-010430	17-029825
7-015934	7-015935	7-015936	8-006305	N-POWER FORECAST
8-010113	8-010359	8-014787	8-015934	1-044436 1-049354 3-044436
8-015935	8-015936	8-033056	9-029128	N-POWER, SAFETY OF
11-036627	12-015934	12-015935	12-015936	1-044436 1-049354 3-044436
12-033056	17-006317	17-011671	17-013315	NAK
17-014787	17-016534	17-030571	17-036625	1-010510 5-026634 7-008142 7-027031
17-036627	18-006305	18-011665	18-016534	8-010242 11-036779 17-010242 17-011605
18-017355	18-030571	18-036625		17-036779
METAL, ALKALI				NEPTUNIUM
5-017577	5-027751	5-033056	6-017577	13-033337
7-024369	7-027818	7-027819	7-027820	NETHERLANDS
7-029287	7-029949	7-033156	8-033056	17-030806
12-033056				NEUTRON
METAL, LIQUID				6-035039 9-023351 9-023358 9-025835
5-009520	5-011915	5-016164	5-017577	11-031443 11-031528 11-032986 18-035039
5-026121	5-026286	5-026635	5-027043	NEUTRON INTERACTION
5-027552	5-031909	5-032819	5-033056	3-014469 6-033515
5-033984	5-034496	5-034497	5-035885	NICKEL
5-036397	5-036403	5-036802	5-036957	6-029466 7-031505 11-036616 12-031505
5-037838	5-038234	6-016164	6-017577	18-029466 18-036616
6-036600	7-008142	7-014384	7-014385	NITRATE
7-015182	7-015538	7-018291	7-022514	13-028951 18-032818
7-024369	7-024386	7-025254	7-027028	NITRIDE
7-027029	7-027030	7-027031	7-027818	7-007346 7-015182 7-031068 9-037965
7-027819	7-027820	7-028506	7-028508	12-031068 17-029825 18-024002 18-025002
7-029287	7-029949	7-030378	7-032587	18-029828
7-033156	8-010242	8-014385	8-027030	NITROGEN
8-033056	8-036802	8-036957	9-011441	9-029473 11-006850 11-030609 12-029494
9-016164	9-025177	9-027541	9-028929	12-030609 18-029473 18-029494 18-030609
9-029877	9-036200	11-036469	11-036779	NOBLE GAS
11-036957	12-033056	12-037838	17-010242	3-023229 7-010558 7-014384 7-029834
17-011605	17-016131	17-032587	17-032819	7-029949 13-023229 17-029834
17-032933	17-032990	17-036779	17-037838	NOISE
18-016379	18-016925	18-032587	18-032819	5-010430 5-036209 6-010430 6-017131
18-032933	18-032990	18-036600	18-036802	9-032628 9-036209 17-017131 17-032628
18-036929				NOISE ANALYSIS
2-029511	7-034347	16-029511	16-034347	5-036802 6-014800 6-034390 8-036802
18-029511	18-035883			9-023358 9-023359 9-029040 9-034390
18-029511	18-035883			17-029040 18-014800 18-036802
MISSILE GENERATION AND PROTECTION				NOZZLE
2-029442	5-005116	11-000749	11-002808	2-032915 5-032879 5-032922 11-036825
11-029452	11-029458	17-010986	18-029442	16-032915 18-032879 18-032915 18-032922
18-029452	18-029458			NRC QUESTION
MOCKUP				1-032907 2-032915 5-032904 12-032904
5-029984	5-030633	6-017565	6-032821	16-032915 17-025730 17-040390 18-025730
7-031339	11-030845	12-030845	17-017395	18-025880 18-026082 18-030209 18-032877
17-029984	17-030845	18-021111	18-021112	18-032904 18-032907 18-032915
18-021113	18-021114	18-030633	18-032821	NS SAVANNAH (PMR)
MODEL TESTING				10-004893 10-004894 10-018293 11-002810
5-028889	5-032819	5-036468	5-037463	NUCLEAR ROCKET
8-036468	11-036465	11-036466	11-036467	17-026300
11-036468	17-032819	18-032819	18-036465	NUCLEATE BOILING
18-036467				5-014676 5-027043 5-027751 5-034496
MODERATOR				17-014676 18-012165
7-015538	9-009794	17-006914	17-006973	OBSTRUCTION
17-007738	17-009794	17-018682		11-029490 17-028644 17-028798 18-029490
MODIFICATION				18-032891
5-018857	5-029297	5-032903	9-012303	OFF GAS
9-012383	9-019862	9-032916	12-032903	7-010297 7-013751 18-029480 18-031032
17-002202	17-006317	17-007710	17-010622	OFF SITE
17-010624	17-012303	17-012383	17-013315	1-029517 1-032907 1-032911 1-032912
17-018606	17-018857	17-019556	17-019762	1-032913 1-032914 5-029374 5-029375
17-019862	17-021736	17-022900	17-022902	5-030633 11-029518 18-029374 18-029375
17-022903	17-022904	17-022905	17-022906	18-029517 18-029518 18-029519 18-030633
17-022907	17-022909	17-022929	17-029851	18-032806 18-032907 18-032911 18-032912
17-029931	17-029939	17-032933	17-035802	18-032913 18-032914
17-037469	18-011333	18-032903	18-032916	ON SITE
18-032933				1-032908 18-032908
MOLTEN FUEL				OPERATING EXPERIENCE
5-031420	8-031420	9-011441	12-034705	3-034706 3-034707 5-035885 6-028562
18-014675				6-036078 7-027031 7-028508 9-025615
MOLYBDENUM				9-026513 9-032628 9-036078 10-026513
5-005444	7-032587	11-036616	17-032587	11-026513 11-036779 12-030879 12-034706
18-032587	18-036616			12-034707 12-034708 13-030879 13-034706
MONITOR				13-034708 17-022900 17-022902 17-022903
7-010559	9-029472	9-031241	9-037406	17-022904 17-022905 17-022906 17-022907
17-037406	18-029472			17-022909 17-026408 17-028562 17-029934
MONITOR, ENVIRONMENTAL				17-029939 17-032628 17-034706 17-036078
17-026164	17-026203	18-029515		17-036779 17-037469
MONITOR, GAS				OPERATING EXPERIENCE SUMMARY
7-029834	17-029834	18-029480		1-006974 1-007695 6-006974 6-007435
MONITOR, PERSONNEL				6-007695 6-007848 7-015934 7-015935
17-036110				7-015936 8-015934 8-015935 8-015936
MONITOR, STACK				9-007695 9-007720 9-007721 9-007728
18-029479				9-012383 9-019442 9-019500 9-020638
MONITORING SYSTEM, RADIATION				9-020709 9-023330 9-023331 9-023332
7-035866				11-007435 11-031972 12-015934 12-015935

12-015936	17-005375	17-006317	17-006458	9-032628	17-032628		
17-006555	17-006726	17-006914	17-006976	PERSONNEL EXPOSURE, RADIATION			
17-006977	17-006979	17-007435	17-007695	1-032907	7-029516	18-029516	18-032907
17-007720	17-007721	17-007728	17-007736	PERSONNEL PROTECTIVE DEVICE			
17-007743	17-007848	17-008375	17-008867	11-029463	17-008375	18-029463	
17-009420	17-010050	17-010622	17-010624	PFR (LMFR)			
17-011138	17-012102	17-012383	17-013314	1-010510	5-036468	8-036468	11-036465
17-013315	17-016148	17-016149	17-016150	11-036466	11-036467	11-036468	11-036469
17-016151	17-017164	17-017395	17-018606	18-036465	18-036467		
17-018681	17-018682	17-019557	17-019558	PHASE CHANGE			
17-019861	17-020045	17-020638	17-021216	5-032527	6-032527	9-032527	17-031273
17-021736	17-021741	17-029835	17-029851	PIPES AND PIPE FITTINGS			
17-029930	17-029931	17-029932	17-029933	1-029865	1-033051	2-029438	2-029441
17-029935	17-029939	17-030278	17-030571	2-029444	5-006817	5-011396	5-027804
17-032146	17-032844	17-033982	18-030571	5-031448	5-032905	7-018343	7-027031
18-032844				9-033051	9-037965	11-006817	11-011396
OPERATING LICENSE PROCESS				11-029453	11-029460	11-030675	11-031973
17-025730	17-032632	17-036630	18-025730	11-031977	11-032905	12-006817	12-029865
18-032632	18-036630			12-032303	12-033051	17-019557	17-019561
OPERATION				17-019861	17-021741	17-025901	17-030675
8-010242	9-029486	12-017102	17-010242	17-031284	17-032303	18-018942	18-029438
17-017102	18-029486	18-029498	18-029499	18-029441	18-029444	18-029453	18-029460
OPERATOR ACTION				18-029865	18-031448	18-032811	18-032905
17-032294	18-029476			PIQUA (OCR)			
ORGANIC IODIDE				2-002811	10-004893	10-018293	11-002803
7-029949				11-002811	17-003970	17-005375	18-011152
ORIFICE				PLANT PROTECTIVE SYSTEM			
7-027819	7-027820	9-019500	17-028643	9-036206	17-018681	18-024515	
17-029851				PLASTICITY			
ORNL				17-029825			
3-023229	7-034673	11-034673	13-005315	PLUM BROOK (TR)			
13-023229	18-034662	18-034673		11-002810			
ORR (RR)				PLUTONIUM			
9-038093	11-002809			1-015836	3-023229	5-018209	5-033542
OSCILLATION				6-007848	6-016735	6-025876	6-025946
6-025357				6-032391	6-035039	6-037747	7-005008
OSCILLATOR, REACTIVITY				7-007280	7-007346	7-008031	7-015182
6-003588	6-009709	6-014800	6-028801	7-022514	7-024343	7-029286	7-029949
9-003588	9-005263	9-036239	12-005263	7-031068	7-034421	7-036473	11-018209
17-005263	17-006977	17-021753	17-028801	11-033993	11-033994	11-033995	11-033996
18-014800	18-020482	18-036146		11-036473	12-018209	12-031068	12-032271
OUT OF PILE EXPERIMENT				12-032391	12-036795	13-023229	13-033337
5-009520	5-031448	5-038396	7-014384	17-006976	17-007848	17-013537	17-029825
7-014385	7-024386	7-027031	7-028506	17-032933	17-032939	17-032940	17-032941
7-031338	7-031339	7-037186	7-038775	17-032942	17-032943	18-009964	18-010107
8-014385	9-023947	17-017395	18-023947	18-011021	18-012010	18-014651	18-014675
18-023948	18-031448			18-024002	18-025002	18-029513	18-029828
OXIDATION				18-032818	18-032933	18-032939	18-032940
5-003279	5-031420	7-007280	7-013541	18-032941	18-032942	18-032943	18-033542
7-029940	8-031420	9-030748	11-030748	18-033993	18-033994	18-033995	18-033996
OXIDE				18-034634	18-035039		
5-038396	6-029466	7-008031	7-015182	PLUTONIUM DIOXIDE			
7-027029	7-027030	7-027031	7-027819	6-021244	6-025692	7-015182	7-029949
7-027820	7-029286	7-031068	8-027030	7-032250	7-032265	7-034360	11-030609
9-016943	9-036650	9-036773	9-036775	12-030609	12-035363	18-011247	18-030609
11-033993	11-033994	11-033995	11-033996	18-032265			
12-031068	18-029466	18-033993	18-033994	PLUTONIUM OXIDE			
18-033995	18-033996			6-005810	17-020467	17-020468	17-020469
OXYGEN				17-020470	17-020472	17-021335	18-034655
7-022479	7-025254	7-028506	7-037186	POISON, SOLUBLE			
12-029133				9-003921	12-024471	13-024471	
PACKED BED				POLAND			
7-032843	11-032843	18-032843		18-011336			
PARTICLE SIZE				POTASSIUM			
7-006380	7-007280	7-014385	7-022479	5-009520	5-027043	7-008142	12-030879
7-022514	7-034360	8-006380	8-014385	12-032697	13-030879	18-034662	
PARTICLE SIZE DISTRIBUTION				POWER COEFFICIENT			
7-022514				5-011040	5-022456	5-033944	6-006312
PARTITION COEFFICIENT				6-007435	6-009709	6-018946	6-027775
1-027827	11-027827			6-028562	6-033944	6-036078	8-006312
PATENT				9-036078	11-007435	17-006312	17-007435
9-025112	9-029195	9-033375	11-025992	17-011040	17-016534	17-018946	17-022456
12-031467	18-025992	18-033481		17-027775	17-028562	17-036078	18-011295
PATHFINDER (ISR)				18-016534			
2-002811	10-004893	10-018293	11-002808	POWER DISTRIBUTION			
11-002810	11-002811	17-003970	17-005375	1-006974	6-006974	17-006457	17-006914
PBF (S-RR)				17-006977	18-007317		
6-005810				POWER UPGRATING			
PEACH BOTTOM 1 (HTGR)				17-007710	17-019558	17-019762	
2-002811	9-026513	10-004893	10-018293	PRESSURE DROP			
10-026513	11-002808	11-002809	11-002811	5-032819	6-003600	6-003657	7-005008
11-026513	17-003970	17-005375		17-003600	17-032819	18-032819	
PENETRATIONS, PRIMARY CONTAINMENT				PRESSURE PULSE			
11-002808	11-002809	11-002810	11-006850	5-031420	5-033056	8-031420	8-033056
11-007912	11-029448	11-029454	11-031977	9-029481	12-033056	18-029481	
11-032887	18-029448	18-029454	18-032887	PRESSURE RELIEF			
PERFORMANCE LIMIT				5-033056	8-033056	11-029457	12-033056
11-033991	11-033993	11-033994	11-033995	18-029457			
11-033996	17-016534	18-016534	18-033991	PRESSURE TRANSIENT			
18-033993	18-033994	18-033995	18-033996	5-029297	5-031420	5-036957	5-106947
PERIOD METER				7-106947	8-031420	8-036957	11-036957

17-029297	18-106947			9-036766	9-038058	9-038093	11-030609
PRESSURE VESSELS				11-030675	11-032543	11-032836	11-032843
1-029865	1-033051	5-032819	5-032888	11-033448	11-033535	11-033953	11-033965
5-036957	6-007108	7-007346	8-036957	11-033993	11-033994	11-033995	11-033996
9-033051	11-009746	11-023857	11-029467	11-034669	11-034673	11-035871	11-036627
11-029468	11-029469	11-029470	11-030609	12-029865	12-030609	12-036504	12-037838
11-030675	11-030845	11-032888	11-032924	13-033428	17-016456	17-019559	17-020467
11-032925	11-033016	11-036465	11-036825	17-020468	17-020469	17-020470	17-020472
11-036957	11-039017	12-029865	12-030609	17-021335	17-026300	17-029048	17-029825
12-030845	12-033051	17-005375	17-007108	17-030675	17-032587	17-032839	17-032933
17-016408	17-016534	17-019557	17-025979	17-032934	17-032939	17-032940	17-032941
17-030218	17-030675	17-030845	17-032324	17-032942	17-032943	17-032945	17-032946
17-032819	17-033016	18-009746	18-016534	17-032947	17-032949	17-032950	17-032951
18-029467	18-029468	18-029469	18-029470	17-032952	17-032953	17-032990	17-033429
18-029865	18-030609	18-032819	18-032888	17-033982	17-036625	17-036627	17-037838
18-032891	18-032924	18-032925	18-036465	17-040390	18-016379	18-017095	18-017355
18-036929	18-037489			18-018745	18-019891	18-020482	18-021086
PRESSURE, EXTERNAL				18-021111	18-021112	18-021113	18-021114
6-028802	17-028802	17-029825		18-021115	18-021116	18-023947	18-023948
PRESSURE, INTERNAL				18-024002	18-025002	18-025462	18-026596
5-017339	5-029446	5-029507	6-007444	18-029606	18-029722	18-029865	18-030574
6-009711	6-034669	7-009711	8-007444	18-030576	18-030609	18-030633	18-031032
11-029446	11-029457	11-031973	11-031977	18-031448	18-032542	18-032543	18-032587
11-034669	11-039017	17-007444	18-029446	18-032656	18-032839	18-032843	18-032884
18-029457	18-029507	18-034669		18-032933	18-032934	18-032939	18-032940
PROBABILITY				18-032941	18-032942	18-032943	18-032945
1-027827	11-027827			18-032946	18-032947	18-032949	18-032950
PROCEDURES AND MANUALS				18-032951	18-032952	18-032953	18-032990
17-029794	17-029975	18-029497	18-029499	18-033274	18-033333	18-033427	18-033428
18-032806				18-033429	18-033448	18-033535	18-033542
PROMPT NEUTRON LIFETIME				18-033932	18-033950	18-033951	18-033952
6-006312	6-016736	8-006312	17-006312	18-033953	18-033964	18-033965	18-033966
PROPERTY, CHEMICAL				18-033968	18-033976	18-033983	18-033985
7-005008				18-033987	18-033989	18-033990	18-033993
PROPERTY, MECHANICAL				18-033994	18-033995	18-033996	18-033999
5-036238	9-036238			18-034669	18-034673	18-034765	18-034767
PROPERTY, PHYSICAL				18-035039	18-035869	18-035870	18-035871
1-029865	5-031420	5-031448	5-036238	18-035883	18-036597	18-036600	18-036625
5-037838	7-024369	7-027028	7-028987	18-036632	18-036765	18-036802	18-036929
7-031068	8-031420	9-036238	11-028987	18-037462	18-037489		
11-031443	12-024373	12-029865	12-031068	RADIATION DAMAGE			
12-037838	17-016032	17-019561	17-020467	1-006974	6-006974	7-024386	7-028987
17-020468	17-020469	17-020470	17-020472	11-028987	11-032543	11-032986	12-032813
17-037838	18-019560	18-024002	18-025002	13-027777	18-032543	18-032813	
18-029865	18-031448			RADIATION EFFECT			
PROPOSITOR				9-037965	11-029491	11-031443	11-031470
1-044436	1-049354	3-044436		11-031528	11-033953	12-036795	18-029491
PUBLIC RELATIONS				18-033953			
1-044436	1-049354	3-044436		RADIATION INJURY TREATMENT			
PULSED NEUTRON TECHNIQUE				1-032907	17-025730	18-025730	18-032806
9-024332				18-032907			
PUMP, MAGNETIC				RADIATION MONITORS			
1-033051	9-033051	12-033051		7-035866	9-003587	9-012069	9-029472
PUMPS				9-036206	17-026301	18-029472	18-029479
1-029865	1-033051	2-029437	5-030255	RADIATION PROTECTION, ORGANIZATION			
6-007435	7-015934	7-015935	7-015936	1-029517	1-032907	1-032909	18-029517
8-015934	8-015935	8-015936	9-030748	18-032806	18-032907	18-032909	
9-033051	9-038092	10-030255	11-007435	RADIATION SAFETY AND CONTROL			
11-030748	12-015934	12-015935	12-015936	1-029517	7-024386	12-032814	17-016150
12-029865	12-030255	12-032697	12-032931	18-019142	18-029517	18-032806	18-032814
12-033050	12-033051	17-007435	17-013238	RADIATION UNIT			
17-013239	17-013979	17-014334	17-018595	9-036650			
17-018606	17-019558	17-019559	17-019762	RADIOACTIVITY RELEASE			
17-021741	18-025157	18-029437	18-029812	3-014870	7-014870	7-027283	12-027283
18-029865	18-030255	18-032810	18-032811	17-021741			
18-034655				RADIOCHEMICAL ANALYSIS			
PYROCHEMICAL				7-018947	17-018947		
6-036600	12-034708	13-034708	18-036600	RADIOCHEMICAL PLANT SAFETY			
PYROLYTIC				12-024471	13-024471		
7-015212				RADIOCHEMICAL PROCESSING			
PYROPHORIC REACTION				3-023229	3-033428	7-024386	7-029286
5-003279	7-022514			13-023229	13-033428	18-012008	18-033428
QUALITY ASSURANCE				RADIOGRAPHY			
1-021968	1-029865	1-033932	1-044436	18-021086			
3-044436	9-019442	11-029448	11-029487	RADIOISOTOPE			
12-029865	17-021968	17-029794	18-029448	7-013907	7-028511	7-031338	13-033337
18-029487	18-029865	18-032810	18-032818	RADIOLOGICAL ASSISTANCE			
18-033932				1-032907	17-025730	18-025730	18-032907
R AND D PROGRAM				RADIONUCLIDE, INDUCED			
1-029722	1-029865	1-029994	1-033429	7-034421			
1-033932	2-033535	3-033428	5-029606	RAPSODIE (LMCR)			
5-029722	5-030633	5-031448	5-032542	5-027552			
5-032836	5-033333	5-033427	5-033542	RATE			
5-033984	5-036597	5-036802	5-037462	5-029503	7-031339	18-029503	
5-037838	5-037957	6-029321	6-032656	RAVENSWOOD (PMR)			
6-033333	6-033427	6-033515	6-033964	2-002811	11-002811		
6-033983	6-034669	6-035039	6-036597	REACTIVITY COEFFICIENT			
6-036600	7-029994	7-032587	7-032843	5-010430	5-033427	5-033944	5-106947
7-034673	8-036802	9-023947	9-028929	6-003588	6-004027	6-005094	6-006312
9-029120	9-029321	9-029994	9-030945	6-007247	6-010430	6-011249	6-025690
9-032836	9-033950	9-036239	9-036482	6-026783	6-032821	6-033427	6-033944

7-106947 8-006312 9-003588 17-006312
 17-006976 18-032821 18-033427 18-033989
 18-033990 18-106947

REACTIVITY EFFECT
 1-006974 5-007188 5-033427 5-035349
 6-003922 6-006974 6-007108 6-026783
 6-028802 6-029466 6-029501 6-033427
 6-033964 6-035349 9-005333 9-033375
 17-005375 17-006458 17-006976 17-007108
 17-013315 17-016029 17-021736 17-028802
 17-029835 18-007188 18-016379 18-021111
 18-021112 18-021113 18-021114 18-024515
 18-029466 18-029501 18-033427 18-033964

REACTIVITY EFFECT, ANOMALOUS
 5-019860 6-017131 6-018946 6-019757
 6-028562 6-036078 9-019758 9-023946
 9-032628 9-036078 9-037406 17-006914
 17-006973 17-006977 17-007738 17-011994
 17-013315 17-016029 17-016534 17-017131
 17-018946 17-019757 17-019758 17-019860
 17-023946 17-028562 17-032294 17-032628
 17-036078 17-037406 18-016534

REACTIVITY EFFECT, EXPANSION
 1-023168 5-033944 6-006561 6-026783
 6-027775 6-033944 9-006561 9-023946
 17-006561 17-023946 17-027775 18-012009
 18-034634

REACTIVITY EFFECT, SLUMP
 3-029831 5-029831 17-029831 18-030210

REACTIVITY EFFECT, VOID
 5-040402 6-007247 6-040402 17-016029
 17-040402 18-034634

REACTIVITY, EXCESS
 5-005106 5-016164 5-034635 6-003912
 6-005094 6-005106 6-006673 6-011153
 6-016164 9-005333 9-016164 17-006673

REACTOR
 7-031003 10-030767 12-024471 12-029179
 12-030664 12-031003 12-031467 12-032271
 12-034696 12-036795 13-024471 13-030664
 13-034696

REACTOR CONTROL
 1-028163 5-032920 6-021245 6-025357
 6-028163 6-031354 6-033964 6-034390
 7-021245 9-023146 9-027519 9-028163
 9-030743 9-032916 9-032920 9-033128
 9-033375 9-034390 9-036239 12-021245
 17-026300 18-021111 18-021112 18-021113
 18-021114 18-032916 18-032920 18-033964
 18-033989 18-033990

REACTOR COOLANT
 1-033429 5-032819 5-035349 5-036403
 5-037838 6-017522 6-035349 7-010297
 7-032587 11-033448 12-029179 12-037838
 17-032587 17-032819 17-032933 17-032939
 17-032940 17-032941 17-032942 17-032943
 17-033429 17-037838 18-032587 18-032810
 18-032811 18-032819 18-032933 18-032939
 18-032940 18-032941 18-032942 18-032943
 18-033429 18-033448 18-034662

REACTOR DESCRIPTION
 1-004609 1-006038 5-029374 5-029375
 5-030633 5-033333 5-036597 6-003600
 6-032656 6-033333 6-036597 11-009746
 11-030609 11-035871 12-030609 17-003600
 17-003637 17-021351 18-003622 18-005832
 18-005877 18-006307 18-008423 18-008704
 18-009746 18-010748 18-011247 18-011336
 18-012007 18-012008 18-012010 18-012011
 18-012012 18-012013 18-012138 18-012145
 18-012165 18-014651 18-016379 18-016925
 18-017699 18-019562 18-022105 18-022583
 18-025454 18-029374 18-029375 18-030498
 18-030609 18-030633 18-032656 18-033274
 18-033333 18-033976 18-034655 18-035869
 18-035870 18-035871 18-035883 18-036597

REACTOR DYNAMICS
 1-028163 2-032915 5-010430 5-032527
 5-032623 5-032893 5-032895 5-033427
 5-035349 6-010430 6-010509 6-010978
 6-021245 6-023162 6-025357 6-025690
 6-028163 6-032527 6-032623 6-032656
 6-032889 6-032890 6-032892 6-032894
 6-032896 6-032919 6-033427 6-035349
 7-0421245 7-028508 9-010509 9-010978
 9-012086 9-028163 9-030743 9-032527
 9-032889 9-032890 9-032894 9-032919
 11-032894 12-021245 16-032915 18-032656
 18-032889 18-032890 18-032892 18-032893
 18-032894 18-032895 18-032896 18-032915
 18-032919 18-033427

REACTOR FUSE

9-008456

REACTOR KINETICS

1-006974 5-010430 5-032266 5-033427
 6-006974 6-010430 6-014800 6-025690
 6-027073 6-032266 6-033427 7-028508
 11-036627 17-004301 17-021741 17-032945
 17-032946 17-032947 17-032949 17-032950
 17-032951 17-032952 17-032953 17-036627
 18-007317 18-010107 18-014800 18-032945
 18-032946 18-032947 18-032949 18-032950
 18-032951 18-032952 18-032953 18-033427

REACTOR PHYSICS

5-029374 5-033427 6-029466 6-032821
 6-033427 6-033515 6-033964 6-033983
 6-035039 6-036078 7-028508 9-036078
 17-017093 17-020467 17-020468 17-020469
 17-020470 17-020472 17-021335 17-036078
 18-029374 18-029466 18-032821 18-033427
 18-033964 18-033983 18-035039 18-036765

REACTOR POWER

1-044436 3-044436 6-008211 9-008211

REACTOR SAFETY SYSTEM

1-010160 1-028163 1-029994 2-002801
 5-007188 6-010509 6-028163 7-029994
 9-008500 9-010498 9-010509 9-012069
 9-028163 9-029482 9-029485 9-029994
 9-032820 9-037406 11-002801 17-021352
 17-037406 18-007188 18-024515 18-029482
 18-029485 18-032820

REACTOR STABILITY

1-006974 1-007695 5-010430 5-032926
 5-040402 6-003454 6-003915 6-005104
 6-005122 6-006974 6-007695 6-007848
 6-009709 6-009822 6-010430 6-011996
 6-017132 6-017522 6-025357 6-025690
 6-028801 6-030351 6-033964 6-034390
 6-036078 6-040402 9-007695 9-017132
 9-030351 9-034390 9-036078 11-030609
 11-032926 12-030609 17-007695 17-007848
 17-021741 17-026300 17-028801 17-036078
 17-040402 18-007317 18-021111 18-021112
 18-021113 18-021114 18-030609 18-032926
 18-033964

REACTOR STARTUP EXPERIENCE

5-026427 5-027552 6-010139 7-013751
 9-010139 17-006458 17-006976 17-011994
 17-026408 17-036053

REACTOR STARTUP TESTING

6-032845 9-007720 9-023946 17-005375
 17-007720 17-019559 17-021350 17-021353
 17-023946 17-032845 18-033990 18-036146

REACTOR TEST FACILITY

18-016652

REACTOR TRANSIENT

5-010430 5-016164 5-019860 5-029375
 5-029504 5-029506 5-032888 5-034635
 6-003454 6-005810 6-007247 6-010430
 6-014800 6-016164 6-029506 6-031354
 6-033964 9-016164 9-019758 9-029471
 9-029481 11-032888 17-010263 17-019758
 17-019860 17-029933 18-014800 18-019142
 18-020482 18-029375 18-029471 18-029481
 18-029504 18-029506 18-030210 18-032888
 18-033964

REACTOR, AIRCRAFT

18-034662

REACTOR, BREEDER

1-006038 1-006974 1-007695 1-011398
 1-021390 1-023168 1-026406 1-033429
 1-035995 1-044436 1-049354 2-002811
 3-023229 3-029831 3-033428 3-034706
 3-034707 3-034711 3-044436 5-000899
 5-005116 5-007556 5-010430 5-017339
 5-018857 5-027595 5-027596 5-029606
 5-029831 5-029884 5-031574 5-032526
 5-032627 5-032629 5-033427 5-035345
 5-036468 5-037838 5-040402 6-003454
 6-003600 6-003657 6-003901 6-003904
 6-003912 6-004027 6-005122 6-005807
 6-006227 6-006312 6-006561 6-006673
 6-006974 6-007108 6-007435 6-007436
 6-007441 6-007444 6-007695 6-007848
 6-009709 6-009711 6-009822 6-010058
 6-010127 6-010139 6-010430 6-010959
 6-010978 6-011424 6-014800 6-016735
 6-016736 6-017008 6-018195 6-018946
 6-023162 6-025946 6-027073 6-031354
 6-033427 6-036600 6-040402 7-005008
 7-007346 7-007556 7-009711 7-010297

7-010533	7-010552	7-011795	7-013541	5-027438	5-027440	5-027595	5-027596
7-013548	7-013751	7-015182	7-018195	5-027725	5-028889	5-029374	5-029606
7-018947	7-024343	7-025254	7-029411	5-029831	5-029984	5-031574	5-032266
7-029547	7-029548	7-033156	7-037184	5-032526	5-032542	5-032624	5-032627
8-003904	8-006312	8-007444	8-013548	5-032629	5-032815	5-032819	5-032836
8-018195	8-036468	9-006561	9-007436	5-032853	5-032878	5-032879	5-032882
9-007441	9-007695	9-007720	9-007721	5-032883	5-032888	5-032893	5-032895
9-007728	9-008052	9-009717	9-010139	5-032903	5-032904	5-032905	5-032906
9-010523	9-010978	9-011424	9-011441	5-032920	5-032922	5-032926	5-033333
9-012256	9-013355	9-016943	9-018814	5-033427	5-033944	5-033984	5-034635
9-023146	9-023330	9-023331	9-023946	5-035345	5-035885	5-036209	5-036236
9-025835	9-027595	9-027596	9-028016	5-036238	5-036468	5-037838	5-038234
9-029040	9-032526	9-032627	9-032628	5-040402	5-106947	6-003454	6-003600
9-037406	9-037487	10-004893	10-004894	6-003657	6-003901	6-003903	6-003904
11-000749	11-002808	11-002809	11-002810	6-003910	6-003911	6-003912	6-003913
11-002811	11-006850	11-007435	11-007436	6-003914	6-003915	6-003916	6-003922
11-007746	11-007912	11-009746	11-029714	6-004027	6-005122	6-005807	6-006227
11-031972	11-032629	11-033016	11-033448	6-006325	6-006561	6-006673	6-006974
11-036465	11-036466	11-036467	11-036468	6-007108	6-007247	6-007435	6-007436
11-036469	12-005807	12-007556	12-011424	6-007441	6-007444	6-007695	6-007848
12-017102	12-024373	12-034706	12-034707	6-008090	6-009709	6-009711	6-009822
12-034708	12-037838	13-023229	13-033337	6-010058	6-010127	6-010139	6-010430
13-033428	13-034706	13-034708	13-034711	6-010959	6-010978	6-011424	6-011996
17-003600	17-003645	17-003970	17-005807	6-014917	6-016164	6-016735	6-016736
17-006312	17-006317	17-006457	17-006458	6-017008	6-017132	6-017565	6-018195
17-006555	17-006561	17-006673	17-006976	6-018946	6-023162	6-023870	6-025357
17-006977	17-006979	17-007108	17-007167	6-025361	6-025690	6-025876	6-025946
17-007435	17-007444	17-007695	17-007708	6-026783	6-027073	6-028163	6-028562
17-007714	17-007720	17-007721	17-007728	6-029321	6-030351	6-031354	6-032266
17-007746	17-007848	17-008052	17-008056	6-032656	6-032821	6-032845	6-032889
17-008097	17-008867	17-010050	17-010986	6-032890	6-032892	6-032894	6-032896
17-011138	17-011795	17-013314	17-013315	6-032919	6-032921	6-033333	6-033427
17-013537	17-014804	17-016033	17-016148	6-033944	6-033964	6-033983	6-034669
17-016149	17-016150	17-016151	17-016408	6-034751	6-036060	6-036600	6-040402
17-016534	17-017102	17-017500	17-018857	7-005008	7-007280	7-007346	7-007556
17-018946	17-018947	17-021350	17-021351	7-008031	7-009711	7-010297	7-010533
17-021352	17-021353	17-021753	17-022498	7-011795	7-011945	7-013541	7-013548
17-023776	17-023946	17-025996	17-026099	7-013751	7-015182	7-018195	7-018291
17-026300	17-026301	17-027595	17-027596	7-018343	7-018947	7-022489	7-022514
17-029040	17-029548	17-029714	17-029831	7-024343	7-025254	7-027283	7-028506
17-029851	17-029932	17-029933	17-029934	7-028508	7-028511	7-028987	7-029286
17-029939	17-029982	17-029984	17-030192	7-029287	7-029411	7-029547	7-029548
17-030218	17-030432	17-030433	17-032324	7-029834	7-029949	7-030378	7-030790
17-032471	17-032472	17-032607	17-032619	7-031338	7-031339	7-032250	7-032253
17-032627	17-032628	17-032629	17-033016	7-032265	7-032587	7-032843	7-032881
17-033429	17-034706	17-035802	17-037406	7-033156	7-034347	7-034421	7-034673
17-037469	17-037487	17-037838	17-040390	7-035866	7-037183	7-106947	8-003904
17-040402	18-007253	18-008423	18-009746	8-007444	8-013548	8-018195	8-018634
18-010318	18-010862	18-011021	18-011247	8-023870	8-036468	9-003921	9-006561
18-011295	18-011429	18-012138	18-014800	9-007436	9-007441	9-007695	9-007720
18-014804	18-016530	18-016534	18-017092	9-007721	9-007728	9-008052	9-009717
18-017341	18-017396	18-019562	18-019562	9-010139	9-010504	9-010523	9-010978
18-020482	18-024002	18-025002	18-025454	9-011424	9-011441	9-012069	9-012256
18-025462	18-025880	18-029606	18-030574	9-012383	9-013355	9-016164	9-016943
18-031574	18-033427	18-033428	18-033429	9-017132	9-018506	9-018814	9-019500
18-033448	18-034632	18-034634	18-034655	9-020709	9-023146	9-023327	9-023946
18-034668	18-035870	18-035883	18-036465	9-023947	9-024332	9-025112	9-025177
18-036467	18-036600	18-036765		9-025835	9-027519	9-027582	9-027595
REACTOR, BHR				9-027596	9-028016	9-028163	9-028929
2-002801	2-002811	6-007108	7-005008	9-029040	9-029122	9-029321	9-029871
9-013337	9-025615	9-026513	10-004893	9-029961	9-030351	9-030748	9-030945
10-004894	10-026513	11-002801	11-002803	9-032526	9-032590	9-032627	9-032628
11-002808	11-002810	11-002811	11-009750	9-032807	9-032820	9-032836	9-032889
11-026513	11-030675	17-003970	17-005375	9-032890	9-032894	9-032916	9-032919
17-006457	17-006458	17-007108	17-030675	9-032920	9-033364	9-033950	9-034663
18-008704				9-034784	9-035244	9-036206	9-036209
REACTOR, DESALINATION				9-036238	9-036482	9-036650	9-036766
2-022328	11-022328	18-011247		9-036773	9-037406	9-037487	9-038058
REACTOR, ERDA OWNED				9-038580	10-004893	10-004894	10-028901
6-017131	6-025337	6-028562	6-030351	10-030092	10-032808	10-032917	11-000749
7-029834	9-008052	9-017479	9-029122	11-002801	11-002808	11-002809	11-002810
9-030351	9-030748	11-030748	12-032271	11-002811	11-006850	11-007435	11-007436
13-028951	17-007714	17-008052	17-008056	11-007746	11-007912	11-009746	11-010356
17-008375	17-016534	17-017131	17-017164	11-012319	11-018634	11-020821	11-023813
17-017479	17-017500	17-026401	17-028562	11-023857	11-028987	11-029714	11-030748
17-029794	17-029834	17-029835	17-036625	11-030845	11-031443	11-031470	11-031528
18-012138	18-016534	18-017500	18-036625	11-031972	11-032543	11-032629	11-032816
REACTOR, FAST				11-032836	11-032843	11-032880	11-032885
1-004268	1-004609	1-006974	1-007695	11-032886	11-032887	11-032888	11-032894
1-010160	1-010510	1-012341	1-015836	11-032905	11-032921	11-032924	11-032925
1-021390	1-021968	1-023168	1-025467	11-032926	11-032986	11-033016	11-033448
1-026406	1-028163	1-030719	1-032908	11-033953	11-033965	11-033991	11-034453
1-032909	1-032910	1-032911	1-032912	11-034669	11-034673	11-034665	11-034666
1-032913	1-032914	1-033429	1-033932	11-034667	11-034668	11-034669	11-034670
2-002801	2-002811	2-032817	2-032880	11-036779	11-036825	12-005807	12-007556
2-032915	2-032917	3-023229	3-029831	12-010356	12-011424	12-017102	12-024373
3-033428	3-034706	3-034707	3-034711	12-024471	12-027283	12-029527	12-030664
5-000899	5-005116	5-007556	5-009520	12-030845	12-031467	12-032271	12-032392
5-010430	5-011040	5-011673	5-011674	12-032396	12-032813	12-032814	12-032853
5-014917	5-016164	5-017167	5-017339	12-032903	12-032904	12-032906	12-032931
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12-034708	12-034784	12-035195	12-035196	REACTOR, FLUX TRAP			
12-035266	12-036504	12-037429	12-037838	11-016576	17-016534	18-016534	18-016576
13-023229	13-024471	13-028951	13-030664	REACTOR, GCGR			
13-033337	13-033428	13-034696	13-034706	5-106947	7-106947	18-106947	
13-034708	13-034711	16-032915	16-034347	REACTOR, GCR			
17-003600	17-003645	17-003970	17-004301	1-004268	1-027827	1-028163	2-002801
17-005807	17-006317	17-006457	17-006458	2-002811	2-022328	6-028163	7-005008
17-006555	17-006561	17-006673	17-006976	9-025615	9-028163	9-033364	9-038092
17-006977	17-006979	17-007108	17-007167	10-004893	10-018293	11-002801	11-002803
17-007435	17-007444	17-007695	17-007708	11-002808	11-002809	11-002811	11-009750
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17-007746	17-007848	17-008052	17-008056	17-005375	18-008704	18-014675	18-017699
17-008867	17-010050	17-010622	17-010624	REACTOR, GRAPHITE MODERATED			
17-010986	17-011040	17-011138	17-011795	3-012530	3-012533	3-014469	3-014870
17-011945	17-011994	17-012319	17-012341	3-023755	5-009456	5-011915	5-014917
17-012383	17-013238	17-013239	17-013314	6-007247	6-007444	6-008211	6-009711
17-013315	17-013534	17-013535	17-013536	6-010058	6-010609	6-011153	6-011162
17-013537	17-013979	17-014804	17-014917	6-011249	6-011516	6-014917	6-034390
17-016033	17-016148	17-016149	17-016150	7-009711	7-014870	8-007444	8-014787
17-016151	17-016408	17-016534	17-017102	9-008211	9-008456	9-009107	9-010150
17-017164	17-017167	17-018857	17-018946	9-010498	9-010504	9-012086	9-012130
17-018947	17-019561	17-021736	17-021753	9-012303	9-034390	9-037965	11-009750
17-021968	17-022900	17-022902	17-022903	11-011394	11-039017	17-006914	17-006973
17-022904	17-022905	17-022906	17-022907	17-007444	17-007709	17-007710	17-007736
17-022909	17-023776	17-023946	17-025730	17-007738	17-007743	17-008375	17-008825
17-025935	17-025936	17-026099	17-026300	17-009107	17-010050	17-010263	17-011428
17-027595	17-027596	17-028562	17-029040	17-011671	17-012102	17-012303	17-014334
17-029548	17-029714	17-029794	17-029825	17-014787	17-014856	17-014857	17-014858
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17-032627	17-032628	17-032629	17-032632	18-009456	18-010118	18-011333	18-018942
17-032819	17-032839	17-032844	17-032845	REACTOR, HOMOGENEOUS			
17-032933	17-032934	17-032939	17-032940	6-005122	18-011336		
17-032941	17-032942	17-032943	17-032945	REACTOR, HTGR			
17-032946	17-032947	17-032949	17-032950	1-027827	1-044436	3-044436	5-033427
17-032951	17-032952	17-032953	17-032990	6-033427	9-026513	10-026513	11-026513
17-033016	17-033429	17-033489	17-034636	11-027827	18-033427		
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REACTOR, MARITIME				17-032952	17-032953	17-036625
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REACTOR, MOLTEEN SALT				5-029297	5-031420	6-003901
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REACTOR, ORDERED BED, FAST				11-036627	17-029297	17-032587
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12-029514	12-030255	12-030609	13-005315	18-029495	18-029497	18-029498	18-029499
16-029511	17-002202	18-007253	18-009964	18-029500	18-029501	18-029502	18-029503
18-016576	18-019142	18-025880	18-026082	18-029504	18-029505	18-029506	18-029507
18-026090	18-028942	18-029374	18-029433	18-029508	18-029509	18-029510	18-029511
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18-029446	18-029447	18-029448	18-029449	18-030254	18-030255	18-030256	18-032878
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18-029454	18-029455	18-029456	18-029457	18-032883	18-032884	18-032885	18-032886
18-029458	18-029459	18-029460	18-029461	18-032887	18-032888	18-032889	18-032890
18-029462	18-029463	18-029464	18-029465	18-032891	18-032892	18-032893	18-032894
18-029466	18-029467	18-029468	18-029469	18-032895	18-032896	18-032905	18-032906
18-029470	18-029471	18-029472	18-029473	18-032908	18-032909	18-032910	18-032911
18-029474	18-029475	18-029476	18-029477	18-032912	18-032913	18-032914	18-032916
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18-029482	18-029483	18-029484	18-029485	18-032921	18-032922	18-032924	18-032925
18-029486	18-029487	18-029488	18-029489	18-032926			
18-029490	18-029491	18-029492	18-029493	REVIEW			
18-029494	18-029495	18-029497	18-029498	1-012341	1-033429	5-035885	5-036802
18-029499	18-029500	18-029501	18-029502	5-037838	6-025690	6-029321	8-010113
18-029503	18-029504	18-029505	18-029506	8-036802	9-029321	9-033729	9-035244
18-029507	18-029508	18-029509	18-029510	9-036650	9-036775	10-030767	11-009750
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18-029515	18-029516	18-029517	18-029518	17-025935	17-026300	17-029934	17-033429
18-029519	18-029520	18-030209	18-030210	17-037838	18-021080	18-024002	18-024515
18-030252	18-030253	18-030254	18-030255	18-025002	18-025462	18-025935	18-033274
18-030256	18-030609	18-030633	18-033542	18-033429	18-036616	18-036765	18-036802
RESEARCH CONTRACT				RHENIUM			
6-011424	7-033156	9-011424	12-011424	9-016943	9-036482	9-036650	9-036766
18-011429				9-036773	9-036775	9-038093	9-038580
RESONANCE OVERLAP				RHOIDIUM			
6-033515				9-023350			
RESPONSE SPECTRUM				ROSSI ALPHA			
2-029434	2-029435	11-029434	18-029434	6-016736			
18-029435				ROVER PROGRAM			
RESPONSE TIME				6-005810			
5-022456	6-027774	6-033964	9-014190	RUBIDIUM			
9-027774	17-022456	17-027774	18-033964	7-030790			
RESPONSE TO NRC QUESTION				RUTHENIUM			
1-029517	1-032908	1-032909	1-032910	7-010558	7-018291	7-022489	7-034361
1-032911	1-032912	1-032913	1-032914	SAFE FAILURE CRITERION			
2-029433	2-029434	2-029435	2-029436	2-032917	10-032917	17-011994	18-032917
2-029437	2-029438	2-029439	2-029440	SAFETY ANALYSIS			
2-029441	2-029442	2-029444	2-029445	1-027827	1-030762	3-023755	5-029504
2-029449	2-029450	2-029483	2-029511	5-030633	5-032526	5-037463	6-003901
2-029512	2-032880	2-032917	3-030256	6-027073	6-029301	9-032526	9-034784
5-029446	5-029459	5-029503	5-029504	11-027827	12-034784	17-013537	18-017092
5-029505	5-029506	5-029507	5-029508	18-029301	18-029504	18-030633	18-030762
5-029509	5-030253	5-030255	5-032878	18-032622	18-034634	18-036612	
5-032879	5-032882	5-032883	5-032888	SAFETY EVALUATION			
5-032893	5-032895	5-032905	5-032906	11-033016	17-013239	17-013315	17-013979
5-032920	5-032922	5-032926	6-029466	17-016408	17-020516	17-033016	17-036630
6-029501	6-029506	6-030252	6-032889	18-036630			
6-032890	6-032892	6-032894	6-032896	SAFETY MARGIN			
6-032919	6-032921	7-029516	7-030253	6-032896	18-032896		
7-032881	9-029440	9-029461	9-029471	SAFETY PRINCIPLES AND PHILOSOPHY			
9-029472	9-029473	9-029474	9-029475	1-007695	1-025467	1-027827	1-029994
9-029481	9-029482	9-029484	9-029485	5-031574	5-033977	5-036236	5-036597
9-029486	9-030254	9-032889	9-032890	6-007695	6-036597	7-010297	7-010552
9-032894	9-032916	9-032919	9-032920	7-029994	9-007695	9-008500	9-029994
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11-029433	11-029434	11-029436	11-029443	17-025935	17-033982	18-025935	18-031574
11-029445	11-029446	11-029447	11-029448	18-033977	18-036236	18-036597	
11-029449	11-029450	11-029451	11-029452	SAFETY PROGRAM			
11-029453	11-029454	11-029455	11-029456	1-011398	1-021390	1-029722	5-011915
11-029457	11-029458	11-029460	11-029461	5-029722	18-029722		
11-029462	11-029463	11-029464	11-029465	SAFETY REVIEW			
11-029467	11-029468	11-029469	11-029470	1-010160	1-027827	5-007188	7-027030
11-029477	11-029487	11-029488	11-029489	8-027030	11-027827	18-007188	
11-029490	11-029491	11-029492	11-029493	SAMARIUM			
11-029518	11-032880	11-032885	11-032886	9-005333			
11-032887	11-032888	11-032894	11-032905	SAMPLING			
11-032921	11-032924	11-032925	11-032926	5-018209	5-032542	7-029940	11-018209
12-029494	12-029495	12-029514	12-030255	12-018209	17-026164	17-026203	17-030278
12-032906	16-029511	18-029433	18-029434	17-032844	18-032542	18-032844	
18-029435	18-029436	18-029437	18-029438	SAN ONOFRE 1 (PWR)			
18-029439	18-029440	18-029441	18-029442	2-002811	11-002811		
18-029443	18-029444	18-029445	18-029446	SAXTON (PWR)			
18-029447	18-029448	18-029449	18-029450	11-002808	11-002810	17-003970	
18-029451	18-029452	18-029453	18-029454	SCRAM, REAL			
18-029455	18-029456	18-029457	18-029458	6-006357	6-027774	9-006357	9-027774
18-029459	18-029460	18-029461	18-029462	9-029485	17-006357	17-006457	17-027774

17-029930 18-029485

SCRAM, SPURIOUS

17-013535

SCRUBBER

11-002809

SEAL

9-029461	9-030748	11-007912	11-029454
11-029455	11-029460	11-029461	11-029462
11-029463	11-029468	11-029487	11-029493
11-030748	11-033016	17-016148	17-016534
17-030218	17-031271	17-032324	17-033016
17-037469	18-016534	18-029454	18-029455
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18-029468	18-029487	18-029493	

SEDIMENT

17-026164 17-026203

SEFOR (RE)

1-006974	1-023168	1-029517	1-032907
1-032908	1-032909	1-032910	1-032911
1-032912	1-032913	1-032914	2-029433
2-029434	2-029435	2-029436	2-029437
2-029438	2-029439	2-029440	2-029441
2-029442	2-029444	2-029445	2-029449
2-029450	2-029483	2-029511	2-029512
2-032817	2-032880	2-032915	2-032917
3-030256	5-026121	5-026461	5-029446
5-029459	5-029503	5-029504	5-029505
5-029506	5-029507	5-029508	5-029509
5-030253	5-030255	5-030633	5-032815
5-032819	5-032878	5-032879	5-032882
5-032883	5-032888	5-032893	5-032895
5-032903	5-032904	5-032905	5-032906
5-032920	5-032922	5-032926	6-006974
6-010978	6-011424	6-014800	6-025361
6-029321	6-029466	6-029501	6-029506
6-030252	6-032656	6-032821	6-032889
6-032890	6-032892	6-032894	6-032896
6-032919	6-032921	6-036060	7-029516
7-030253	7-032881	9-010978	9-011424
9-012069	9-012256	9-029321	9-029440
9-029461	9-029471	9-029472	9-029473
9-029474	9-029475	9-029481	9-029482
9-029484	9-029485	9-029486	9-030254
9-032807	9-032820	9-032889	9-032890
9-032894	9-032916	9-032919	9-032920
9-036239	9-036650	9-036766	9-036773
9-038058	9-038580	10-029439	10-029478
10-030255	10-032808	10-032917	11-006850
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11-029493	11-029518	11-030609	11-030845
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11-032921	11-032924	11-032925	11-032926
11-033991	12-011424	12-029494	12-029495
12-029514	12-030255	12-030609	12-030845
12-032813	12-032814	12-032903	12-032904
12-032906	12-035195	12-035196	16-029511
16-032915	17-016033	17-025730	17-030845
17-032294	17-032632	17-032819	17-036053
17-036630	18-007253	18-011429	18-014800
18-020482	18-021111	18-021112	18-021113
18-021114	18-021115	18-021116	18-025730
18-025880	18-026082	18-026090	18-026594
18-028964	18-029433	18-029434	18-029435
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18-029464	18-029465	18-029466	18-029467
18-029468	18-029469	18-029470	18-029471
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18-029476	18-029477	18-029478	18-029479
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18-030209	18-030210	18-030252	18-030253
18-030254	18-030255	18-030256	18-030609
18-030633	18-031032	18-032632	18-032656
18-032806	18-032807	18-032808	18-032809
18-032810	18-032811	18-032813	18-032814
18-032815	18-032816	18-032817	18-032818
18-032819	18-032820	18-032821	18-032877
18-032878	18-032879	18-032880	18-032881
18-032882	18-032883	18-032884	18-032885
18-032886	18-032887	18-032888	18-032889
18-032890	18-032891	18-032892	18-032893
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18-032916	18-032917	18-032918	18-032919
18-032920	18-032921	18-032922	18-032924
18-032925	18-032926	18-033989	18-033990
18-033991	18-033992	18-036060	18-036146
18-036630			

SEISMOLOGY5-006684 11-002808
SELF POWERED FLUX DETECTOR
9-023350**SERVOMECHANISM**

9-019115

SHIELDING

1-006974	3-023755	5-030633	6-006974
7-029516	9-030748	11-030609	11-030748
12-030609	12-032813	12-032814	13-036276
17-013536	17-016148	17-029982	17-030218
17-032324	18-011672	18-029516	18-030399
18-030609	18-030633	18-032813	18-032814

SHIPPING CONTAINER

3-012533	3-023229	3-023755	3-034706
3-034711	12-034706	13-032329	13-034706
13-034711	17-034706	18-029499	

SHIPPINGPORT (PWR)

10-004893	10-004894	10-018293	11-002808
11-002810			

SHUTDOWN MARGIN

6-030252	9-007720	9-023358	17-007720
18-030252			

SIMULATION

5-030633	5-032527	5-037463	6-032527
9-023146	9-030743	9-032527	9-037487
17-037487	18-030633		

SITE CLIMATOLOGY

2-029511	2-032915	16-029511	16-032915
18-029511	18-032915		

SITING

2-002801	2-002811	11-002801	11-002811
18-024515			

SITING, REACTOR

2-002801	2-002811	2-022328	11-002801
11-002811	11-022328	11-030609	12-030609
17-033982	18-030609		

SL 1 (BWR)

11-002803

SNOKE

7-014385 8-014385

SNAP (SR)

9-023332 12-032697 12-033050

SNAP 10A (SR)

5-006817 9-019115 11-006817 12-006817

SNAP 2 (SR)

9-019115

SNAP 8 (SR)

6-034390 7-008142 7-027031 9-019115

SOCIO/PHILOSOPHICAL CONSIDERATION

1-044436 3-044436

SODIUM

1-007695	1-010160	1-010510	1-033051
1-033429	1-035995	3-023229	5-000899
5-003279	5-007188	5-009456	5-011915
5-014676	5-016164	5-017339	5-026121
5-026461	5-026635	5-027043	5-027438
5-027440	5-027552	5-027751	5-029002
5-031420	5-031448	5-032623	5-032629
5-032819	5-032878	5-032926	5-033056
5-033977	5-033984	5-034635	5-035345
5-035349	5-035885	5-036397	5-036403
5-036802	5-037462	5-037850	5-038234
5-038396	5-106947	6-003219	6-006312
6-007444	6-007695	6-010057	6-016164
6-017522	6-018195	6-021244	6-021245
6-032623	6-035349	6-037850	7-004216
7-006379	7-006380	7-007280	7-007346

7-008031	7-010297	7-010533	7-010552	6-005104	6-005106	6-005122	6-006331
7-010558	7-010559	7-013541	7-013548	6-010609	6-011153	6-011162	6-011249
7-013751	7-013907	7-014384	7-014385	6-017131	6-019757	6-034390	7-018343
7-015538	7-015934	7-015935	7-015936	7-019555	7-019756	8-006305	9-003407
7-018195	7-018291	7-018343	7-021245	9-003460	9-003587	9-003588	9-005333
7-022479	7-022489	7-022514	7-024343	9-008456	9-009107	9-010498	9-012086
7-024386	7-025254	7-027028	7-027029	9-012130	9-014190	9-018506	9-019758
7-027030	7-027031	7-027283	7-027818	9-019862	9-034390	9-037965	11-002808
7-027819	7-027820	7-028506	7-029287	11-016576	12-025266	13-005315	13-025266
7-029940	7-029949	7-030378	7-030790	17-003407	17-003644	17-006310	17-006317
7-031003	7-031338	7-031339	7-031340	17-006726	17-007709	17-007710	17-007736
7-032253	7-032587	7-032699	7-032843	17-007743	17-009107	17-009420	17-010263
7-033156	7-034360	7-034361	7-037181	17-011671	17-016029	17-017093	17-017131
7-037183	7-037184	7-037186	7-038775	17-018606	17-019555	17-019558	17-019756
7-106947	8-006312	8-006380	8-007444	17-019757	17-019758	17-019759	17-019762
8-010113	8-010242	8-010359	8-013548	17-019860	17-019861	17-019862	17-021741
8-014385	8-014787	8-015934	8-015935	17-022914	17-029931	17-029975	18-006305
8-015936	8-018195	8-027030	8-031420	18-009456	18-016576		
8-033056	8-036802	9-007695	9-010504	STACK			
9-016164	9-016943	9-018814	9-019500	2-002811	2-032817	11-002811	12-029494
9-023947	9-025177	9-027519	9-027541	18-029479	18-029494	18-029520	18-032817
9-029002	9-029877	9-033051	9-034784	STAFFING			
9-036239	10-029478	11-011394	11-030609	1-032909	1-032912	18-032909	18-032912
11-031340	11-031973	11-031977	11-032629	STATISTICAL ANALYSIS			
11-032843	11-032926	11-033448	11-033977	5-033281	5-035822		
11-034453	11-036469	11-036616	11-036627	STEAM			
11-038785	12-006488	12-015934	12-015935	1-007695	5-018209	5-033056	6-007695
12-015936	12-019567	12-021245	12-024373	7-027818	7-037186	8-033056	9-007695
12-027283	12-029002	12-029133	12-029179	9-010150	11-018209	12-018209	12-033056
12-029533	12-030609	12-030879	12-031003	17-007695	18-030574		
12-031467	12-032303	12-032697	12-032699	STEAM GENERATOR			
12-032931	12-033050	12-033051	12-033056	1-029865	1-033051	5-033056	8-010113
12-034705	12-034708	12-034784	12-035044	8-033056	9-022457	9-033051	9-038092
12-035195	12-035196	12-035266	12-037850	11-031972	12-019567	12-029865	12-033051
13-023229	13-030879	13-034708	17-006312	12-033056	17-013536	17-022457	17-029851
17-006317	17-006488	17-007444	17-007695	17-029932	17-030192	17-030571	17-037469
17-007736	17-008375	17-008825	17-010242	18-017355	18-019891	18-020046	18-021086
17-014676	17-014787	17-016032	17-016033	18-029865	18-030498	18-030571	18-036929
17-016131	17-018595	17-018682	17-019556	STEEL			
17-019559	17-019561	17-020045	17-021081	11-002808	11-002809	11-002810	11-023857
17-021335	17-025935	17-026301	17-026408	11-038785	12-032813	17-022914	18-032813
17-027844	17-029048	17-029825	17-029835	STEEL LINER			
17-029930	17-029931	17-029933	17-030278	11-031977			
17-032146	17-032303	17-032587	17-032629	STEEL, STAINLESS			
17-032819	17-032844	17-032933	17-032990	5-006817	5-011396	5-031448	5-032836
17-033429	17-034636	17-036053	17-036627	6-036078	7-028987	7-031068	7-031338
18-005832	18-005877	18-007188	18-009456	7-031340	7-031505	7-032699	7-037186
18-018942	18-019560	18-020046	18-021080	7-038775	9-016943	9-032836	9-036078
18-021086	18-023947	18-023948	18-025157	9-037965	11-006817	11-011396	11-028987
18-025935	18-026596	18-029478	18-029500	11-029489	11-031340	11-031443	11-031470
18-029828	18-030210	18-030609	18-031448	11-032543	11-032836	11-032986	11-033953
18-032587	18-032622	18-032810	18-032811	11-034453	11-036616	11-038785	12-006817
18-032819	18-032843	18-032844	18-032878	12-031068	12-031505	12-032699	12-035044
18-032926	18-032933	18-032990	18-033429	12-035363	12-037429	17-036078	18-024002
18-033448	18-033977	18-034636	18-034655	18-025002	18-029489	18-031448	18-032543
18-036616	18-036765	18-036802	18-036929	18-032811	18-033953	18-036616	
18-037462	18-106947			STOICHIOMETRY			
SODIUM COEFFICIENT				17-020467	17-020468	17-020469	17-020470
1-007695	1-023168	5-034635	6-003407	17-020472	18-029828		
6-007695	6-010978	6-016735	6-017008	STORAGE CONTAINER			
6-025690	6-025692	6-025876	6-034751	2-032880	11-032880	12-025266	12-034708
9-003407	9-007695	9-010978	17-003407	13-025266	13-034708	18-029500	18-032880
17-007695	18-034634			18-032918			
SOIL				STRESS			
17-026203				2-029450	5-029505	5-032888	11-029448
SOL-GEL PROCESS				11-029450	11-029487	11-029489	11-032888
5-034497				11-038785	17-006914	17-007738	17-019557
SOLID STATE DEVICE				17-021741	17-026408	18-029448	18-029450
9-023350	9-027541	9-029122		18-029487	18-029489	18-029505	18-032888
SOLVENT EXTRACTION PROCESS				STRESS ANALYSIS			
3-023229	13-023229	13-027777	13-033337	1-033051	2-029441	2-032817	5-006817
13-034697	13-036276			5-027804	5-036238	9-033051	9-036238
SOURCE, CONTINUOUS				11-006817	11-024156	11-029443	11-039017
12-029494	18-029479	18-029494		12-006817	12-033051	18-007317	18-024156
SOURCE, NEUTRON				18-029441	18-029443	18-032811	18-032817
6-010139	6-028802	9-007728	9-010139	STRESS CORROSION			
17-007728	17-028802	18-010862		11-011394	17-010050	17-029851	
SOURCE, RADIATION				STRESS RUPTURE			
12-032814	18-032814			11-034453	11-038785		
SPACE DEPENDENT DYNAMICS				STRESS STRAIN DATA			
5-010430	6-007444	6-010430	6-010978	11-029491	11-039017	18-029491	
6-025357	6-028801	8-007444	9-010978	STRONTIUM			
17-007444	17-028801			7-010558	7-013907	7-018291	7-030790
SPACECRAFT				7-034361			
6-016736				STRUCTURAL ANALYSIS, DYNAMIC			
SPOKESMAN, AEC				2-032817	5-036238	9-036238	11-036467
1-044436	3-044436			18-032817	18-036467		
SRE (RE)				STRUCTURAL INTEGRITY			
3-012530	3-012533	5-005106	7-006684	2-029483	3-023755	5-032629	6-034669
5-009456	5-019860	6-003407	6-003408	11-029447	11-029469	11-029487	11-029489
6-003460	6-003469	6-003588	6-005093	11-032629	11-033991	11-034669	17-032629

18-029447	18-029469	18-029483	18-029487	17-013315	17-016534	17-029714	17-037487
18-029489	18-033991	18-034669		18-016534	18-029464		
SUPERHEAT				TEST, NONDESTRUCTIVE			
5-038234				1-029865	5-032836	9-032836	11-011394
SUPPORT STRUCTURE				11-032836	12-029865	17-029794	18-029865
2-029437	2-029438	2-029444	11-029456	18-037489			
11-029470	17-031273	18-029437	18-029438	TEST, ON LINE			
18-029444	18-029456	18-029470		5-032526	9-032526		
SURFACE				TEST, PHYSICS			
5-026461				5-022456	5-029506	5-030633	6-011424
SURFACE CONTAMINATION				6-027775	6-028801	6-028802	6-029506
7-019555	17-019555			9-011424	9-029481	9-037965	12-011424
SURVEILLANCE PROGRAM				17-021353	17-022456	17-027775	17-028801
2-032915	11-029487	11-032924	16-032915	17-028802	17-029933	18-011429	18-016379
18-029487	18-032915	18-032924		18-029481	18-029506	18-030633	
SURVEY				TEST, PLANT RESPONSE			
1-015836	5-035885	9-018814	9-033364	6-008211	6-014800	6-031354	9-008211
12-030879	13-030879			9-012130	9-012303	9-023146	9-024332
SURVEY, RADIATION				17-006457	17-006458	17-006973	17-012303
17-025979	17-029982			18-014800			
SYSTEM CAPABILITY				TEST, PREOPERATIONAL			
5-030633	5-032922	11-029487	12-032813	11-023813	12-006488	17-006488	17-018606
18-029487	18-029502	18-030633	18-032813	17-019556	17-019762	17-030571	18-030571
18-032922				18-033992			
SYSTEM DESCRIPTION				TEST, PRESSURE VESSEL			
5-029375	5-029606	5-030633	6-032894	5-032819	17-032819	18-032819	
6-032919	6-036078	6-037182	7-029516	TEST, PROOF			
9-023146	9-023327	9-032807	9-032820	5-032542	9-017479	11-002810	11-023813
9-032894	9-032919	9-036078	9-037406	17-017479	18-032542		
10-032808	11-029451	11-029456	11-029463	TEST, SYSTEM OPERABILITY			
11-029477	11-030609	11-032894	11-032925	6-008211	9-008211	9-023146	9-036239
11-037182	12-029514	12-030609	12-032813	11-029492	17-019762	18-029492	18-029502
17-021352	17-026301	17-030571	17-036078	18-033992			
17-037406	18-014646	18-017113	18-029375	TESTING			
18-029451	18-029456	18-029463	18-029477	1-033051	5-029374	5-032819	5-032836
18-029479	18-029480	18-029498	18-029514	5-032893	5-032895	5-037462	6-018195
18-029516	18-029606	18-030498	18-030571	6-032656	6-032889	6-032892	6-032894
18-030609	18-030633	18-032807	18-032808	6-032896	6-032919	6-034390	7-015934
18-032809	18-032811	18-032813	18-032820	7-015935	7-015936	7-018195	7-025254
18-032894	18-032919	18-032925	18-036634	7-010688	7-032587	7-034347	8-015934
18-036635				8-015935	8-015936	8-018195	9-010523
SYSTEM OPERABILITY IN ACCIDENT				9-028016	9-029471	9-032807	9-032836
5-032903	5-032922	7-029516	12-032903	9-032889	9-032894	9-032919	9-033051
18-029516	18-032903	18-032922		9-034390	9-036482	11-023813	11-023857
TANTALUM				11-029451	11-030845	11-031973	11-032836
5-011299	9-016943	11-036616	18-036616	11-032894	12-015934	12-015935	12-015936
TECHNICAL SPECIFICATIONS				12-019567	12-030845	12-031068	12-032392
5-007188	7-033156	11-029714	11-033016	12-032697	12-033050	12-033051	12-035044
12-036504	17-011428	17-013238	17-013239	12-035195	12-035196	12-037429	16-034347
17-013537	17-013979	17-016408	17-016765	17-030845	17-032587	17-032819	18-029374
17-029714	17-030218	17-032294	17-032324	18-029451	18-029471	18-032587	18-032656
17-033016	17-035802	17-040390	18-007188	18-032807	18-032819	18-032889	18-032892
18-010862	18-011295			18-032893	18-032894	18-032895	18-032896
TELLURIUM				18-032919	18-037462		
7-030790				THEORETICAL INVESTIGATION			
TEMPERATURE COEFFICIENT				1-028163	5-031909	6-009822	6-028163
6-003407	6-010139	6-011153	6-027073	6-033964	6-034751	6-037747	7-015212
9-003407	9-005333	9-010139	17-003407	7-030790	7-032253	7-034360	9-023146
17-006458	17-006977			9-027519	9-028163	12-029133	12-032814
TEMPERATURE GRADIENT				17-021081	18-032814	18-033964	
5-026427	5-032853	7-029547	11-023857	THERMAL ANALYSIS			
11-029443	12-032853	18-029443		5-032266	5-032815	5-033984	5-036957
TEMPERATURE REACTIVITY EFFECT				5-037850	6-005104	6-005759	6-005810
5-019860	6-017522	17-019860		6-006325	6-006331	6-006673	6-007247
TENSILE PROPERTY				6-010057	6-010127	6-032266	6-037850
6-032896	11-029491	18-029491	18-032896	7-010533	7-031068	8-036957	11-036957
TEST INTERVAL				12-031068	12-037850	17-006673	17-016032
11-009750	11-029714	17-029714		18-020046	18-032815	18-034658	
TEST, COMPONENT				THERMAL CONSIDERATION			
9-019115	9-023327	12-006488	17-006488	5-029506	5-036236	5-037462	6-029506
TEST, CONTROL ROD DRIVE				11-029455	17-019557	17-029933	18-029455
9-012383	9-014190	17-012383		18-029506	18-036236	18-037462	
TEST, DESTRUCTIVE				THERMAL EXPERIMENT			
11-036469				5-037462	18-037462		
TEST, DROP				THERMAL INSULATION			
6-030351	9-007721	9-030351	17-007721	2-032915	5-020733	5-032542	11-006850
TEST, FILTER				11-032925	16-032915	17-020733	18-032542
12-029514	18-029514			18-032915	18-032925		
TEST, INSTRUMENT RESPONSE				THERMAL MECHANICAL EFFECT			
6-027774	9-008456	9-016943	9-020709	5-029297	5-029505	5-032888	6-003915
9-025615	9-027541	9-027774	9-029120	6-005104	9-036650	11-029443	11-029487
9-030945	9-036200	9-036239	9-036766	11-032888	11-032925	12-032813	12-035196
9-036773	9-036775	9-038058	17-027774	17-019561	17-021741	17-026408	17-029297
TEST, LEAK LOCATION				17-029851	18-007317	18-024002	18-025002
11-002810	11-009750	11-029464	11-036779	18-029443	18-029487	18-029505	18-032813
17-030192	17-033489	17-036779	18-029464	18-032888	18-032925		
TEST, LEAK RATE				THERMAL POLLUTION			
6-007435	6-007436	9-007436	9-037487	1-044436	1-049354	3-044436	
11-002810	11-007435	11-007436	11-007746	THERMAL PROPERTY			
11-007912	11-009750	11-012319	11-029464	5-027043	7-025254	7-031338	17-029825
11-029714	17-007435	17-007746	17-012319	18-019560			

THERMAL TRANSIENT				18-024002	18-025002	18-029828	18-033993
5-011674	5-014917	5-026634	5-026635	18-033994	18-033995	18-033996	18-034634
5-029504	5-029505	5-031420	6-004026	URANIUM CARBIDE			
6-011516	6-014917	6-027774	6-031354	6-004026	6-004027	6-004479	6-006357
7-032843	8-031420	9-027774	11-032843	6-016735	6-027073	7-006379	7-032253
17-014917	17-027774	18-029504	18-029505	9-006357	12-025266	13-025266	17-005375
18-032843				17-006357			
THERMOCOUPLE				URANIUM DIOXIDE			
9-016943	9-036482	9-036650	9-036766	5-031420	6-025692	7-029949	7-032250
9-036773	9-036775	9-038093	9-038580	7-032265	7-034360	8-031420	12-035266
17-031282				12-035363	18-011152	18-011247	18-032265
THERMODYNAMICS				URANIUM OXIDE			
5-033984	5-037838	7-030790	12-037838	6-003916	17-020467	17-020468	17-020469
17-037838				17-020470	17-020472	17-021335	18-034655
THORIUM				URANIUM SULFIDE			
6-003913	6-009711	7-009711	9-016943	6-003901	6-006227	6-009711	7-009711
9-036650	9-036766	9-036773	9-036775	URANIUM-235			
TITANIUM				12-032396			
7-024369	9-003921	12-024373		URANIUM-238			
TRACER, RADIOACTIVE				6-032391	12-032391		
7-014384	7-034347	16-034347		USSR			
TRAINING				5-033056	8-033056	12-033056	17-020045
1-032909	1-032912	9-037487	17-005375	18-011247	18-033976		
17-007714	17-036053	17-037487	18-025462	VALVE, CHECK			
18-026596	18-032909	18-032912	18-033990	18-033481			
TRANSFER FUNCTION				VALVES			
5-022456	6-003460	6-007848	6-009822	1-033051	9-019500	9-033051	9-038092
6-017132	6-025690	6-028801	6-030351	11-029451	11-029453	11-029458	11-029463
6-031349	6-031439	9-003460	9-017132	11-029490	11-029492	12-029495	12-029527
9-024332	9-027519	9-029040	9-030351	12-033051	12-036504	17-011666	17-018681
17-006457	17-006976	17-007848	17-022456	17-019557	17-019558	17-019559	17-021741
17-028801	17-029040	17-029932	17-029933	17-026398	18-029451	18-029453	18-029458
TRANSPORT PROPERTY				18-029463	18-029490	18-029492	18-029495
7-032699	12-032699			VANADIUM			
TRANSPORT THEORY				7-024369	11-036616	17-038113	18-036616
6-021244				VAPOR PRESSURE			
TRANSPORTATION AND HANDLING				7-029287	9-036650	10-029478	17-011605
3-029831	3-034706	3-034707	3-034711	18-029478			
5-029831	12-034706	12-034707	13-034706	VBWR (BWR)			
13-034711	17-029831	17-034706		2-002811	7-005008	11-002808	11-002810
TREAT (PWR)				11-002811			
5-014917	5-031420	6-003901	6-004026	VENTILATION SYSTEM			
6-004479	6-005810	6-006227	6-006673	2-002811	7-029516	7-032843	11-002809
6-007247	6-007444	6-009711	6-010058	11-002811	11-032843	12-034696	13-034696
6-014917	6-018195	6-034669	7-009711	17-005375	18-005832	18-024515	18-029516
7-018195	7-032587	8-007444	8-018195	18-032843			
8-031420	11-033991	11-034669	11-036627	VESR (ISR)			
17-006673	17-007444	17-014917	17-032587	11-002808	11-002810		
17-032939	17-032940	17-032941	17-032942	VIBRATION			
17-032943	17-032945	17-032946	17-032947	5-032819	9-023359	12-035195	17-016033
17-032949	17-032950	17-032951	17-032952	17-018682	17-019762	17-029851	17-030571
17-032953	17-032990	17-034636	17-036625	17-032819	18-030571	18-032819	
17-036627	18-032587	18-032939	18-032940	VIBRATION ANALYSIS			
18-032941	18-032942	18-032943	18-032945	11-030845	12-030845	17-030845	
18-032946	18-032947	18-032949	18-032950	VOID COEFFICIENT			
18-032951	18-032952	18-032953	18-032990	1-007695	5-010430	5-016164	5-019860
18-033991	18-034636	18-034669	18-036625	5-032527	5-033427	5-034635	6-003469
TRITIUM				6-003910	6-003913	6-003914	6-003915
7-005008	7-014384	17-005375	18-011152	6-003916	6-003922	6-006325	6-007247
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KOEHLER M				LEVINE MM			
1-030762	18-030762			6-003922	18-010748	18-022583	
KOHL AL				LEW DE			
7-027818				6-006357	9-006357	17-006357	17-006973
KONDO S				LEWIS LC			
5-032624				13-028951			
KOONTZ RL				LEWIS RH			
7-006380	7-014385	7-022479	7-027029	9-023350			
8-006380	8-014385			LEYPUNSKII AI			
KOPCHINSKI NL				18-011247			
17-026398				LHIAUBET G			
KOSKY PG				6-032845	17-032845		
5-026286				LIBBY HL			
KOTTOWSKI H				9-028929			
5-009520	5-027751			LICHTENBERGER HV			
KOVACEVICH EA				5-029374	18-029374		
7-032699	12-032699			LINDNER L			
KOVACIC EC				7-032250			
7-013751				LINDSEY PS			
KOWALEW A				17-030571	18-030571		
18-011336				LINK LE			
KRASNOYAROV NV				6-003914	18-012008		
17-004301	18-033976			LIPPS AJ			
KREWER KH				7-037181			

LITTLE WM				11-036779	17-029616	17-033489	17-036779
5-033333	5-036597	6-033333	6-036597	MCCANN PH			
18-012007	18-033333	18-036597		6-003589	9-003589	17-003589	
LOCHBAUM R				MCCARTHY JF			
18-030399				10-020579	17-020579	17-028798	
LOCKHART RM				MCCARTHY WJ			
7-031339				5-031574	7-011945	7-029548	9-020638
LOEWENSTEIN W				17-008867	17-011945	17-020638	17-029548
17-002202	18-002241			17-029932	17-029934	17-032619	18-031574
LOEWENSTEIN WB				MCCLINTOCK DC			
6-009709	6-010139	6-036078	9-010139	11-024156	11-031977	17-019561	18-024156
9-036078	17-036078	18-012009		MCCOOL WJ			
LOGAN D				9-008052	17-008052	17-008375	
9-023359				MCCULLOUGH CR			
LONG JK				2-022328	11-022328		
5-033944	5-035349	6-003657	6-033944	MCDONALD J			
6-035349				17-011666			
LONG KJ				MCDONALD JS			
6-036078	9-036078	17-036078		7-013541			
LORD RM				MC FALL A			
6-003910				18-033481			
LOTTIG RA				MCGINNIS FD			
12-032697				6-007848	11-007746	11-007912	17-006976
LOWE AL				17-007746	17-007848	18-009964	
18-006307				MC GUIRE W			
LOWRIE MS				11-000749			
13-034697				MCHUGH WE			
LUNDHOLM JG				3-029831	5-029831	17-029831	
6-019757	7-019756	9-019758	17-019756	MCKEON JT			
17-019757	17-019758	17-019759		5-027804			
LURIE H				MCKISSON RL			
6-005759				5-020733	7-027028	17-020733	
LUTZ DE				MCNEAL N			
18-016530				5-032623	6-032623		
LYERLY RL				MCNEAL NA			
2-022328	11-022328			5-034635			
LYNNWORTHLC				MCNELLY MJ			
9-038093				6-003916			
LYON RN (ORN)				MCPHEETERS CC			
5-003647				7-027819	7-027820		
MACFARLANE DR				MECHAN WJ			
5-016164	5-026635	5-032623	5-034635	18-012008			
6-016164	6-032623	6-034751	9-016164	MENEGHETTI D			
MACIAS ES				18-012009			
7-032587	17-032587	18-032587		MENELEY DA			
MACLEOD CJ				5-032623	5-034635	6-032623	6-034751
9-033364				18-016530			
MACLEOD RA				MERK W			
7-013751				1-010160	1-011398	18-017396	
MACNEAL N				MESHII T			
6-034751				9-029877			
MACPHERSON RE				MEUSER MH			
7-027030	8-027030	17-025935	18-025935	1-033932	18-033932		
MADELL JT				MEYERS DJ			
6-036078	9-036078	17-036078		12-034696	13-034696		
MADSEN EF				MICHEL A			
17-030278				5-037850	6-037850	12-037850	
MAENE NG				MIELCARSKI M			
6-025692				18-011336			
MAGEE PM				MILLER KJ			
18-033987				5-005444			
MAGNUSON DW				MILLER WE			
6-032391	6-037747	12-032391		12-032396			
MALECHA RF				MILLIOT B			
3-034711	13-034711			5-009520			
MALMBERG T				MILLS RM			
1-010160	1-011398	18-017396		6-003589	6-006357	9-003589	9-006357
MANN LA				17-003589	17-006357	17-006914	17-006973
17-021736				MILOVIDOV IV			
MANNAS DA				18-033976			
17-007736				MILSTED J			
MARKEE EH				7-032587	17-032587	18-032587	
7-034347	16-034347			MINUSHKIN B			
MARKLEY RA				8-010242	17-010242		
12-034698				MISHIMA Y			
MARSH DD				18-030576			
17-019481				MITCHELL W			
MARTEN W				2-022328	11-022328		
17-011666				MOAK DP			
MASON DG				18-029828			
18-021080				MOBERLY JW			
MATRAS S				7-031340	11-031340		
12-037429				MONAMECK JH			
MATTE J				8-018195	7-018195	8-018195	
3-034706	5-027596	9-027541	9-027596	MONSON H			
12-034706	13-034706	17-027596	17-034706	17-002202	18-002241		
18-036614				MONSON HO			
MATTEL J				17-010050			
5-027595	9-027595	17-027595		MONTEITH OB			
MATTHEWS RR				10-028901	10-030092		

MOORE CR					13-033337				
7-013751					OKRENT D				
MOORE EA					1-007695	5-106947	6-007695	7-106947	
9-009107	9-012130	17-009107			9-007695	17-007695	18-002241	18-106947	
MOREWITZ HA					OLP RH				
6-007247	7-027029	7-034360			9-033950	18-033950			
MOREWITZ HA (AI)					OLSON WH				
6-003657					17-007736				
MORGENSTERN F					OLSON WR				
1-030762	18-030762				17-006977	17-011994	17-013534	17-029932	
MORIARTY KJ					17-029933				
9-023330	9-029122				ONEILL GL				
MOSS WD					1-030719	7-010552	18-017092		
7-007280	7-022514				ORLOV VV				
MOTEFF J					17-029930				
11-031528	11-036616	18-036616			OTTER JM				
MOWREY J					6-004479	6-010609			
12-006488	17-006488				OTTEWITTE EH				
MUELLER M					18-033987				
6-009711	7-009711				OWENS JE				
MUELLER RE					17-010050	17-019558	17-029931		
6-027775	6-028802	7-018947	9-007720		OYAMA A				
9-028016	17-007720	17-018947	17-027775		5-032624				
17-028802	17-037469	18-036614			PADILLA A				
MUENCHOW HO					5-032853	6-033964	9-029871	12-032853	
8-010359	12-019567	18-017355			18-033964				
MUNKELWITZ HR					PARK TA				
7-032253					9-023351				
MURBACH EW					PARSICK RJ				
7-014384					18-022583				
MURLEY TE					PASCHALL RK				
6-033515					6-010609				
MURPHY G					PASCOVET A				
5-004938	7-004938				1-004268				
MURPHY PM					PATERS NT				
6-003916					9-019500				
MURPHY WF					PATRIARCA P				
11-031470					7-034673	11-034673	18-034673		
MYD HO					PAUL RC				
17-022914					12-032396				
NASH CR					PEARSON E				
11-033965	18-033965				18-011333				
NEBIKER RP					PEARSON EN				
17-011671					6-003407	6-006357	9-003407	9-006357	
NELSON CL					17-003407	17-006310	17-006357	17-006973	
17-007714					PECKINPAUGH CL				
NELSON CT					6-005093	6-005104	6-006331	9-012130	
7-013907	7-022479	7-027029			11-033965	17-018606	18-033965		
NEUENHALN P					PEDERSEN LT				
9-030743					5-018209	11-018209	12-018209		
NEWCOMB L					PEELGREN ML				
12-024373	18-010118				7-013541				
NEWMAN L					PEKRUL PJ				
7-028506					6-034390	9-034390			
NICHOLS JR					PELHAM DD				
12-032271					17-006973				
NICHOLSON JO					PELLET MR				
17-007710					6-003588	9-003588	9-005333		
NICHOLSON RB					PEPPLER W				
5-031574	6-011996	6-030351	9-030351		5-033984				
18-031574					PERKS MA				
NIJSING R					6-003910				
5-037957					PERRET JD				
NIKULIN MP					17-006973	17-016029			
17-004301	17-029930				PERRY AM				
NIMS JB					18-034662				
5-031574	5-032526	6-011996	9-032526		PERRY JA				
17-029933	18-031574				1-033932	18-033932			
NOBLE LD					PERSIANI PJ				
6-009822					6-036078	9-036078	11-009746	17-036078	
NOHOFILOV EV					18-009746				
5-034694					PETERS NT				
NORMAN EC					5-029606	17-011138	18-029606		
7-015182					PETERSON RE				
NOVAK PE					5-033333	5-036597	6-033333	6-036597	
9-036650	9-036766	9-036773	9-038580		9-029871	18-033333	18-033966	18-036597	
11-033991	18-033991				PFLASTERER GR				
NOVICK M					6-014800	6-029321	9-029321	18-014800	
6-007848	17-006976	17-007848	17-016148		18-020482	18-026596	18-031032	18-033989	
17-016149	17-016150	17-016151	17-029935		18-033990	18-036146			
18-009964					PFLASTERER R				
NOYES RC					6-032656	18-032656			
5-011915	6-005759	18-034668			PHELPS RD				
NUGENT G					6-011249				
18-022583					PHILLIPS JL				
O NEIL GL					5-018857	17-018857			
18-012012					PHILLIPS KE				
OAKLEY DJ					18-012009				
11-036825					PINKHASIK MS				
OCMSSENFELD W					5-033056	8-033056	12-033056	17-004301	

17-020045	17-029930	18-033976		6-003904	8-003904		
PIPER HB				RILEY DR			
11-002803	11-002808			9-027582			
PITKANEN PH				RITZ WC			
6-006227				12-034698			
PLANTING PJ				ROACH PF			
7-031340	11-031340			5-029002	9-029002	12-029002	
PLANCHAN JD				ROBIN M			
7-030378	18-019560			12-031467	18-019562		
POIDO MS				ROBINSON GC			
17-004301	18-033976			11-009750			
POLLOCK BD				ROBINSON L			
7-014384	7-027283	12-027283		6-003901			
POLOMIK EE				ROBINSON LE			
5-026121	5-026461	11-030845	12-030845	5-014917	6-009711	6-014917	6-018195
17-030845				7-009711	7-018195	8-018195	17-014917
PONTIER R				RODDIS LM			
17-019559	17-021353			18-025454			
POPLAVSKY VM				RODGERS SJ			
5-033056	8-033056	12-033056		5-031448	18-031448		
POPPER GF				ROQUES C			
9-016943	9-023328	9-025835		11-023857			
POREMBKA SW				ROSE PF			
7-028987	11-028987			7-037184			
PORGES KG				ROSEN SL			
7-010559	9-009717			7-008142			
PORGES KGA				ROSENHOLC M			
9-034663				17-021351			
POTTER PE				ROSSIN AD			
7-029286				6-034669	11-034669	18-034669	
POTURAJ-GUTNIAK S				ROUX DP			
18-011336				9-023358			
POUCHER FW				ROVIG MD			
7-027818				9-029122			
PRACH TF				ROY DH			
7-028506				6-033983	18-033983		
PRICE DE				ROYDEN HW			
18-029828				6-010609			
PRICE JE				ROZENHOLC M			
17-004028				5-033977	11-033977	17-019559	18-019562
PROBERT P				18-030498	18-033977		
18-021086				RUBINSTEIN HJ			
PROBERT PB				6-006357	9-006357	17-006357	17-006973
11-031972				RUCKER DJ			
PROCTOR JF				7-034673	11-034673	18-034673	
5-005116				RUMPF HK			
PROUTEAU LP				5-017339			
9-020709				RUSCH GK			
PUIG I				5-033542	18-033542		
18-032622				RUSSELL CE			
QUINN DJ				11-030845	12-030845	17-030845	
5-036397				RUTHER			
QUINN EP				7-024369			
11-030845	12-030845	17-016033	17-030845	RUTHER WE			
RADCLIFF JG				12-037429			
9-018506				RYAN JD			
RALEIGH HD				5-027596	9-027596	17-027596	
17-003970				RYDEN J			
RAMEY, JT				5-032836	9-032836	11-032836	
18-025462				RYGAERT J			
RAMSEY EB				7-029940			
12-034696	13-034696			SALING JM			
RANDALL RL				3-029831	5-029831	17-029831	
6-034390	9-023359	9-034390		SALVADORES M			
RASTOIN J				6-021244			
18-019562				SANATHANAN CI			
RATEICK MC				6-034751			
3-029831	5-029831	17-029831		SANATHANAN CK			
RAVEN AD				5-032623	5-034635	6-032623	
9-033375				SAUVAGE M			
REED GL				11-010356	12-010356	17-017395	18-010356
5-011915				SAVAGE MM			
REEMSnyder DC				7-015934	7-015935	7-015936	8-015934
12-033050				8-015935	8-015936	12-015934	12-015935
REGIMBAL JJ				12-015936			
9-029871	9-029961			SAVINO JM			
REICKS GH				5-017576	6-017576	8-017576	
9-011441				SAVOLAINEN AW			
REWARD A				2-002801	11-002801		
7-029547				SAVOLAINEN JE			
REYNES L				7-025254			
5-027552	5-029297	11-031973	17-019556	SCHIKARSKI W			
17-019557	17-029297			7-024343	8-018634	11-018634	11-020821
REYNOLDS AB				SCHLAPP FW			
1-023168				3-014870	7-014870		
RICE R				SCHLECHTENDAHL E			
5-029374	18-029374			1-030762	18-030762		
RICE RE				SCHLECHTENDAHL EG			
17-029835				5-033984	12-029179	18-021211	18-033985
RILEY D				SCHLESINGER G			

7-006379
 SCHMEIDER H
 41-038785
 SCHMIDT G
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 SCHNEIDER G
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 SCHOMER RT
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 SCHONFELD E
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 SCHRODER R
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 SCHROEDER R
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 SCHULTHEISS GF
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 SCOTT CC
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 SCOTT FA
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 SEABORG, GT
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 SEBILLEAU F
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 SEELEY GT
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 SEGAL BM
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 SEIM OS
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 SEMERIA R
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 SESONSKE A
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 SHAVER RE
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 SHAW H
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 SHOUDY AA
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 SIEGEL R
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 SIMMONS W
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 SLIPER EG
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 SMITH JP
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 SMITH OL
 18-034662

SMITH RJ
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 SMITH RR
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 SNEPVANGERS JJM
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 SOFER GA
 5-036236 6-003913 6-027073 18-036236
 SOMMER W
 7-010297 7-010533
 SOWA E
 6-003901
 SOWA ES
 6-018195 7-018195 8-018195
 SOWDEN RG
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 SPALDING TR
 17-017164
 SPARKS DW
 6-003454
 SPEIGLER P
 18-011333
 SPIEGLER P
 5-009456 18-009456
 SPINRAD BI
 6-003914
 SPRINGER TM
 9-003587
 STACHURA SJ
 5-032527 6-004026 6-032527 9-032527
 STAKER MP
 18-034668
 STARR C
 17-003644
 START GE
 7-034347 16-034347
 STEARNS JD
 9-037965 17-016131
 STECK RB (COMPILED BY)
 6-004027
 STEELE LE
 11-002808
 STEELE OP
 9-019115 9-023335
 STEKOLINIKOV VV
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 STEPHANY W
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 STEPHENSON JW
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 STEWART OC
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 STEWART JE
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 STEWART R
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 STIEVENART M
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 STORRS CL
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 STRAHL H
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 STRAUSS SD
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 STRAWSON DG
 18-012010
 STROEHR KW
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 STROMM HE
 11-031470
 STURIALE LA
 5-035885
 SUDDATH JC
 3-012530 3-012533
 SUNDERMAN IO

9-009794	17-009794	17-020665	17-024647	11-002809			
SUTHERLAND JD				WALKER CS			
12-034698				11-002809			
SWANSON CD				WALKER LP			
9-023327	9-031241	9-036206		5-005116			
SZAWLEWICZ SA				WALLIN WR			
1-029722	5-029722	18-029722		6-036078	9-036078	17-036078	
TANG IM				WALTERS RJ			
7-030790	7-032253			6-003589	6-006357	9-003589	9-006357
TAUBE M				17-003589	17-006357	17-006973	
6-026783	7-005008	18-011336		WALTHER H			
TAYLOR D				7-015212			
18-025157				WARBERG HS			
TEBO FJ				5-027596	9-027596	17-027596	
11-007746	11-007912	17-007746		WARE WR			
TEITEL RJ				12-017102	17-017102		
1-006038				WARNSING R			
TESSIER JH				5-027751			
6-003454				WATROUS JD			
TESZLER O				17-006973			
6-006357	9-006357	17-006357	17-006973	WATSON CD			
TEVEBAUGH AD				3-023229	13-023229	13-034697	13-036276
5-031420	8-031420			WEAVER LM			
THALGOTT FW				5-005106	6-005106	6-006357	6-011153
5-029374	6-003657	7-029834	17-029834	9-006357	17-006357	17-006973	
18-009964	18-029374			WEBB J			
THIE JA				7-029411			
6-011996				WEBER M			
THOMAS CE				17-006458			
18-006307				WEHMEYER DB			
THOMPSON DH				6-025692			
5-004938	7-004938	17-011605	18-016530	WEIGEL C			
THURNAY K				12-033050			
6-032797				WEILLS JT			
TILL CE				6-003914	18-012008		
5-033542	18-033542			WEISSENFLUH JA			
TOBIAS ML				18-006307			
6-005122				WELLS AC			
TOTH WL				7-031505	12-031505		
5-027595	9-027595	17-027595		WENSCH GW (USAEC)			
TOYODA KG				17-003645			
18-033966				WEST GA			
TRENT DS				13-034697			
5-028889				WHEELER RG			
TRICE WG				5-018209	11-018209	12-018209	
18-012008				WHITE JR			
TROYANOY MF				18-036614			
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