DR-1306

AI-AEC-12791 (VOL. I)

MASTER

1000 MWe

L M F B R

Liquid Metal Fast Breeder Reactor

FOLLOW-ON STUDY

ATOMICS INTERNATIONAL

TASK II REPORT

**Conceptual Systems Design
Descriptions
Volume I**

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Prepared for ARGONNE NATIONAL LABORATORY

United States Atomic Energy Commission Division of Technical Information

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1000-Mwe LIQUID METAL FAST BREEDER REACTOR
FOLLOW-ON STUDY
CONCEPTUAL SYSTEM DESIGN DESCRIPTIONS
VOLUME I

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INTRODUCTION

Atomics International (AI) is one of the five contractors on the 1000-Mwe Liquid Metal Fast Breeder Reactor (LMFBR) Follow-On Study Program which is being managed by Argonne National Laboratory (ANL) for the Atomic Energy Commission (AEC). This work is part of the AEC's continuing LMFBR design studies outlined in Volume II, Plant Design, "LMFBR Program Plan" (WASH 1102). The overall objectives of this program are to identify the research and development (R&D) necessary to lead to safe, reliable, and competitive LMFBR central station power plants in the 1980's, and to establish the relative ranking of the needed R&D. The study objectives are being met through the development of a series of industry-favored reference plant designs, and the evaluation of these designs to determine the R&D program necessary to proceed with the final engineering and construction of the proposed plants. The Conceptual System Design Descriptions, prepared for the AI Task III final reference design, are given in the three volumes of this document.

The Follow-On Studies were divided by ANL into four tasks. Task I comprised the concept selection and the establishment of all significant design criteria. Because of their importance in setting the basic plant configuration and its performance characteristics, the major trade studies performed on the Company program were reexamined in Task I, in light of the criteria for the Follow-On Study Program. These trade studies included:

- 1) Pool vs loop
- 2) Fuel handling
- 3) Vented vs nonvented fuel
- 4) Number of loops.

The selection of mixed-oxide fuel and a regular core geometry was based on previous AI work, and was not reevaluated. The next step was to define a preliminary reference design for a 1000-Mwe plant whose major design and performance characteristics were similar to those for the current AI Demonstration Plant reference design. The results of the Task I effort were reported in AI-AEC-12765, "ANL 1000-Mwe LMFBR Follow-On Study Task I Report."

Tasks II and III were done concurrently in the AI study. The Task III work comprised a safety evaluation, engineering studies, economic evaluation, and selection and analysis of a Task III final reference design. This work is described in AI-AEC-12792, "1000-Mwe Liquid Metal Fast Breeder Reactor Follow-On Study Conceptual Design Report." The Conceptual System Design Descriptions for the Task III final reference design, given in this report, were prepared in Task II.

The Task IV report, AI-AEC-12793, describes the information desired before final design and construction of the proposed plant can begin, and outlines the R&D which can most effectively provide the needed information.

The System Design Descriptions (SDD) are working technical reference documents which define the system requirements, and describe the system, subsystems, and major components. They provide technical guidance for personnel working on the program, serve as a means of monitoring and controlling the design, and are official reference documents which describe the plant. SDD's evolve through stages, from conceptual to final, as the design progresses. Their evolution provides detailed documentation of the technical development history and the design philosophy leading to the final plant configuration.

During the early phases of a program, the SDD's are oriented toward the general plant concept and its overall design, safety, operating, and maintenance philosophy. Only the overall plant and major plant systems (segments) are covered during this phase. As the design progresses, more detail is included in the SDD's, and SDD's are prepared for lower-level systems. For the Follow-On Study, SDD's were prepared for the overall system, the major plant segments, and selected major systems and components. The SDD relationship and breakdown are shown in the SDD tree given in Figure 1 of the Overall System Conceptual SDD (SDD-096-10-000). The criteria for selecting which SDD's were to be prepared were:

- 1) Is R&D likely to be required?
- 2) Is the description needed for preparation of the plant capital cost estimate?

The Conceptual System Design Descriptions in this document describe the Task III final reference conceptual design. They have been reproduced exactly as they are prepared for program use, and are working technical documents.

ATOMICS INTERNATIONAL*A Division of North American Rockwell Corporation***SUPPORTING DOCUMENT****NUMBER**

SDD-096-10-000

REV

LTR A

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System Design Description

PROGRAM TITLE

ANL 1000-Mwe Follow-on Study

KEY NOUNS Overall System Conceptual
System Design Description**DOCUMENT TITLE**Overall System Conceptual System Design
Description - 1000-Mwe LMFBR Follow-on Study**ORIGINAL
ISSUE DATE** 4-22-68**GO NO.**

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PAGE 1 OF**TOTAL PAGES** 67**REL. DATE** 5-23-68**ABSTRACT**This Conceptual System Design Description
presents the requirements for, and a summary
description of, the 1000-Mwe LMFBR Follow-on
Study**PREPARED BY** **DATE** **DEPT/GRP NO.**

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1.0 INTRODUCTION

1.1 System Functions

The primary function of the system, hereinafter referred to as the Plant, is to operate as a safe, reliable, and economical energy producer in a typical utility system. In addition, the Plant shall provide:

- 1) Design, construction, and licensing experience on a large size (1000-Mwe) plant required for demonstration of capital costs of commercial plants in the 1000- to 1500-Mwe range
- 2) Statistically significant data from operating experience on reliability, maintainability, and operability of the plant under typical central station power plant operating conditions
- 3) Statistically significant data on fuel performance and fuel cycle cost trends required for utility acceptance of LMFBRS
- 4) Construction and operating experience leading to improved commercial LMFBRS through simplified design, standardization of components, and higher reliability and availability.

1.2 Summary Description of the System

As currently envisaged, the 1000-Mwe plant shall be a self-sustaining (first unit) power station in a large (>10,000-Mwe) electric utility system. This Plant will be one of an early series of large plants built, following the Demonstration Plant. After an initial period of operation and shakedown, the Plant is expected to become a firm energy source, with power costs competitive with contemporary light water reactors.

The Plant is conveniently divided into the following major segments:

Reactor	Auxiliary Systems
Heat Transfer	Electrical
Energy Conversion	Reactor Support System
Control and Instrumentation	Land, Structures, and Conventional Systems

The key design and performance characteristics for each of these segments, as well as the underlying considerations which lead to the specific selections, are presented in detail in Section 2.0. The following description is a brief summary of Section 2.0, providing an over-view of the plant. The principal design and performance characteristics are given in Table 1.

TABLE 1
1000-Mwe LMFBR DESIGN AND PERFORMANCE
CHARACTERISTICS

Normal power (Mwe)	1000
(Mwt)	2400
Number of coolant circuits	3
Reactor inlet temperature (°F)	780
Reactor outlet temperature (°F)	1140
Number of core elements	
Fast region fuel inner zone	133
Fast region fuel outer zone	141
Radial blanket	138
Control rods	15
Fast region fuel height (in.)	43
Axial blanket length [in. (upper/lower)]	12/15
Radial blanket length [in. (inner/outer)]	56.5/43
Equilibrium fissile mass (kg)	~2744
Breeding ratio	~1.27

The nuclear steam generator portion (i.e., nuclear island) of the plant utilizes a sodium-cooled FBR, fueled with mixed oxides of plutonium and uranium. Heat from the reactor core is transported by liquid sodium through primary and intermediate coolant circuits to a steam generator complex, where steam is generated at conditions similar to those for modern fossil-fueled plants. The nominal net output of the Plant described is 1000 Mwe, with a net station heat rate of about 8175 Btu/kwh.

The reactor is housed in a steel-lined concrete cavity below the operating floor of the reactor containment building. The vessel is surrounded by a containment jacket. The cavity is capped with a double-rotatable shield plug which forms a continuation of the operating floor over the reactor and provides radiation protection for operating personnel. A carbon steel dome is provided over the loading face shield for containment. The reactor vessel is suspended in the cavity from a support ring at the top of the vessel. The

reactor has a high-pressure inlet coolant plenum which supplies coolant to the fast region and the blanket region. Coolant flow is proportioned with fixed orifices in the elements. Sodium flow is upward through the reactor to a free surface pool above the core. The control rods are cooled by thermosyphon from a low-pressure region below the high-pressure plenum. Coolant is supplied from bypass lines from the main lines. The core, which is made up of an assembly of fuel and blanket elements is the heat source. The fuel elements have axial blankets at each end. Two rows of radial blanket elements are used. Separate shim control rods and safety control rods provide reactivity control, and are driven from the reactor loading face.

Heat generated in the reactor is transported to the steam generators by three parallel heat transfer circuits. Each circuit has a primary and a secondary sodium loop. The primary loops are common to the reactor, but the secondary loops are not interconnected, and are physically and hydraulically isolated from the primary system.

The primary sodium flows through the core and then through the sodium-to-sodium intermediate heat exchanger(s), where the heat absorbed in passing through the core is transferred to secondary sodium. The primary sodium becomes radioactive in passing through the core, and therefore a secondary sodium loop is employed to transport the heat to the steam generators. The secondary sodium does not become radioactive, thus allowing normal access to the steam generating facilities. In addition, with this arrangement, any pressure effects and/or sodium-water reaction products which may result from a steam generator tube leak are not transmitted to the reactor.

The steam generator is arranged in three banks, one for each heat transfer circuit, with isolation valves for each bank. By arranging the steam generator equipment in this manner, a failure in one bank of heat exchangers may be isolated to leave two loops available for continued operation or removal of decay heat from the reactor, thus providing minimum interruption of power production and protection against overheating.

Steam is delivered to the turbine plant at 2400 psig/1000°F/1000°F. The main turbine-generator unit is a 1000 Mwe, 3600 rpm tandem compound unit, with 33-1/2 in. last stage buckets.

Plant control is designed for load following between 25 and 100% of nominal plant rating. The plant control system is similar to that used in the Hallam Nuclear Power Facility (HNPF). The turbine-generator demand is met by varying sodium flow and control rod positions while maintaining the reactor ΔT approximately constant. The reactor system has a faster response to load changes than the conventional part of the plant, and therefore is not limiting.

The reactor plant includes facilities for the handling of new and spent fuel. New fuel elements are transferred into reactor fuel transfer positions inside the reactor vessel with an ex-vessel fuel handling machine while the plant is shut down. A second fuel handling machine, mounted on the double rotating shield plug, is used to transfer fuel from the reactor fuel transfer positions to core positions while the plant is shut down. Spent fuel from the core is placed in reactor transfer by the same machine. Spent fuel is removed from the in-vessel transfer positions to an ex-vessel storage tank for decay by the ex-vessel fuel handling machine. Spent fuel is prepared for shipment while the plant is operating, using the ex-vessel fuel handling machine to transport spent fuel from the ex-vessel storage tank to a shielded cell where it is placed in the shipping cask.

The plant design incorporates features which permit the addition of a future second unit with minimum cost and disturbance to the existing unit. The major structures consist of the turbine-generator deck, adjoining steam generator structure, reactor containment building, and supporting facilities.

The Plant requirements and performance/design characteristics shall be set forth in accordance with the System Design Description Breakdown indicated in Figure 1. Components and subsystems which are not included are presumed to be either sufficiently conventional to permit their design and fabrication through the use of existing or conservatively modified standards and specifications, or which have little influence on the public safety (licensing) and operational safety (protection of utility investment) of the plant.

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Figure 1. 1000-Mwe LMFBF Follow-On Study Conceptual System Design Description Tree

1.3 System Design Requirements

1.3.1 Performance

Power Output – The plant shall have a nominal net full power rating of 1000 Mwe. The rated power (100% steam flow at 2400 psig/1000°F/1000°F) for the plant shall correspond to the guaranteed gross output of the turbine-generator at 1.5 in. Hg absolute turbine back pressure and 0% makeup rate.

Plant Loading – The plant shall be operated as a low incremental cost station power plant (plant shall normally be loaded at rated power for 20 to 24 hr/day throughout the year), but shall have provisions for following normal load changes between 25 and 100% of rated power at 2%/min. The plant shall be designed for one such load swing per day.

Maximum Power Ramp – The plant shall be capable of sustaining an emergency change in power demand of at least 25%/min between 50 and 100% of rated power without causing a reactor trip. The plant shall be designed to accommodate four (4) such power ramps per year.

Maximum Refueling Interval – The plant shall have the capability for refueling at intervals of not less than six calendar months.

1.3.2 Availability

The generating capacity of this system will represent a significant portion of any electric system generation capability, and therefore must have availability and reliability (forced outage) factors consistent with other large generating units within typical interconnected utility grids.

Operating availability* which is defined as the percentage of time a unit is available for service whether operated or not is commonly used to describe the availability of conventional plants. This availability definition is unsuitable for engineering studies to determine trade offs between availability and other plant costs because reduced-power operation is not taken into account. Evaluation of plant availability and unavailability on an energy basis is a more suitable method. The plant energy unavailability (1-availability) is defined by the following equation:

*Analytical Report of Equipment Availability for the Seven-Year Period 1955-61, EEI Publication No. 63-42 (May 1963)

$$\text{Unavailability} = \frac{(T_s + T_f + T_E) P_o + T_i (P_o - P_i)}{8760 P_o}$$

where

T_s = total scheduled downtime during year (hr)

T_f = total downtime due to forced (unplanned) outages (hr)

T_E = total downtime due to unplanned extension of scheduled outages (hr)

P_o = design (100% steam flow) full power plant net output (Mwe)

T_i = time of partial power operation because of equipment failure (hr)

P_i = average plant net power output capability over period T_i (i.e., period when output is limited due to equipment failure) (Mwe)

The energy availability goal is 88.9%. The corresponding operating availability goal for a three-circuit plant is about 92%. The plant energy availability and downtime goals at plant maturity (seven years after initial commercial operation) are:

	(%)
1) Overall Availability	88.9
2) Forced Outage Rate	5.4
3) Unscheduled Extension of Refueling Shutdown (including unscheduled maintenance)	1.0
4) Schedule Downtime for Turbine Maintenance	3.8
5) Schedule Downtime for Refueling and Con- current Maintenance	0.9
Total	100.0

The unscheduled (Items 2 and 3) downtime goals of 5.4 and 1.0% are tentatively apportioned to the system segments as follows:

Segment	(%)
Reactor	1.1
Heat Transfer	3.2
Energy Conversion	0.9
Control and Instrumentation	0.3
Auxiliaries	0.4
Support Equipment (including refueling)	0.4
Structures and Services	0.1
Total	6.4

The unavailability factor for a given segment is defined as the percentage of a full-power year of energy that is lost, due to shutdowns or periods of limited power output which are attributable to failures in that segment.

1.3.3 Maintainability

The plant shall be designed such that scheduled maintenance can be accomplished to the greatest extent practicable while the plant is at power. Maintenance actions requiring either a hot or cold shutdown shall consider redundancy (to rotate maintenance actions) or elimination of the requirement through increased component reliability or redesign. Parts or assemblies requiring repair or replacement shall be repaired or replaced with like materials or items. Where practical, defective parts shall be repaired within the support facility.

The general provisioning policy shall be to maintain a material supply sufficient to support the plant for three months of continuous operation, for those items which are not readily available. The identification and quantity of spare parts, repairable parts, and bulk materials shall be determined after establishment of a scheduled maintenance plan.

1.3.4 Useful Life

The design service life of the plant is 30 years. During the first 20 years of service life, the plant shall meet all performance and availability requirements specified under paragraphs 1.3.1 and 1.3.2. During the last 10 years of service life, the plant shall suffer no more than a 10% reduction in availability, and no more than a 5% reduction in steaming rate.

1.3.5 Environment

Until an actual site has been selected, the following environmental conditions shall be considered design requirements, governing the conceptual design.

1.3.5.1 Topography and Site Access — The site is located on the bank of a river, 1500 ft above sea level; it is essentially level, and is clear of any obstructions or former structures. Vehicular, rail, and barge access are available to the site, with good secondary roads and railroad service available at the site boundary. Barge access, suitable for shipment of large components, such as the top shield and reactor vessel, is available in proximity to the location of the reactor building.

(SRO) AEC SENIOR REACTOR OPERATOR LICENSE
(RO) AEC REACTOR OPERATOR LICENSE

*SOME RO LICENSES DESIRABLE BUT NOT MANDATORY

TOTAL STAFF: 78
TOTAL SRO: 13
TOTAL RO: 5*

Figure 2. Organization of Operating Staff

1.3.5.2 Temperature – Summer: 90°F day dry bulb, 78°F day wet bulb
Winter: 0°F dry bulb

1.3.5.3 Hydrology – The plant is located above flood zone. Ground water is encountered at a minimum of 60 ft and a maximum of 250 ft below grade. Adequate ground water for sanitary supply and plant makeup is available within these levels.

1.3.5.4 Geology – Soil conditions at the site permit conventional excavation (no blasting) to a depth of 100 ft without requiring dewatering or sheet pilings. Allowable soil bearing values at 5 ft below grade is at least 6000 lb/ft².

1.3.5.5 Cooling Water – Plant cooling water is fresh river water at 55°F, and supply is adequate for all plant requirements.

1.3.5.6 Site Exclusion – The site exclusion requirements shall be based on population density, as per TID-7025.

1.3.5.7 Meteorology and Climatology – Site dose calculations shall be based on Pasquill Type F condition and a wind speed of 1 m/sec. For the 30-day dose calculation, a reduction of a factor of three in the dose shall be assumed for directional wind changes, and a further factor of two reduction shall be assumed for changes in wind velocity.

The site is in a 25-lb/ft² wind-pressure-map area, as defined by the Uniform Building Code. The snow load for this site is 15 lb/ft².

1.3.5.8 Seismology – Earthquake conditions at the site conform to seismic Zone 1, as defined by the Uniform Building Code, and there are no ground faults within the site boundary.

1.3.5.9 Labor – Labor availability for plant construction is adequate, but shall require transportation of 30 to 50 miles from the site to the nearest large population centers.

1.3.6 Personnel

Operation requires a multi-disciplined staff and several levels of technological training. The organization of such an operating staff can take several forms, depending upon the normal operating structure of the particular utility. A typical organization is presented in Figure 2, to indicate the overall operating personnel requirements.

1.3.6.1 Manpower Requirements — The operating and maintenance manpower requirements for a single-unit 1000-Mwe LMFBR are estimated at 78. These requirements are based on the principle that all of the services would be provided at the station, without relying upon centralized facilities. Several of the personnel, particularly in specialist areas which will be delineated later, may not be required for the full term of plant operation, but will be needed for the first two years of operation and during any major plant changes, such as power escalations or operation with core loadings of new design.

1.3.6.2 Superintendent's Office — The plant superintendent and his assistant have overall responsibilities for the operation of the plant and matters related to plant safety and licensing. The assistant superintendent should have a substantial background in nuclear technology and plant operations, and should hold a senior reactor operator license.

1.3.6.3 Technical Staff — The technical staff is responsible for the overall review and evaluation of the plant operation, to keep management apprised of safety, operational, and plant equipment problems. The staff should be competent to react to any plant problem, provide analysis and recommendations to management, and lead outside analytical help if required.

1.3.6.3.1 Nuclear Engineer — The nuclear engineer provides analytical support in areas of reactor physics, thermal/hydraulics, and reactor control. He maintains surveillance of reactor data to determine proper operation and to obtain the required input for fuel management analyses. He provides Operations with the reactivity status of the plant, so they can recognize deviations from normal. The nuclear engineer should be a licensed senior reactor operator.

1.3.6.3.2 Health Physicist — The health physics function deals with all of the routine survey and monitoring requirements of a nuclear plant. These include pre-operational radiation surveys of the site and surroundings, continuing assessment of solid, liquid, and gaseous radioactive plant effluents, and the internal monitoring, surveying, and record keeping for protection of plant personnel. The health physicist and technician required for this function prepares general plant radiological safety practices and procedures for operations personnel as the need arises.

1.3.6.3.3 Chemist — The chemist provides surveillance of all plant fluids to assure proper purity and treatment. He assists in plant materials' problems, radiochemical analyses, and cleanup of contaminated areas or equipment.

1.3.6.3.4 Instrument Engineer — The instrument engineer is responsible for the maintenance of plant instrumentation. The instruments are maintained, calibrated, and repaired, as required, by a staff of instrumentation technicians under his direction. The group is responsible for procurement and setup of special instrumentation, as required.

1.3.6.3.5 Plant Engineer — The plant engineer is responsible for surveillance of all plant equipment, maintenance of operational histories of plant equipment, and evaluation of plant performance and efficiency.

1.3.6.3.6 Safety Specialist — For the period required to achieve reliable full-power operation, with all systems performing to design standards, a safety specialist should be available to review the plant performance and compare it to design objectives. The purpose of this review is to assure that the plant is operating to the level of safety expected, and to investigate the safety implications of those systems or components which do not operate precisely as expected, even though overall operation may not be affected. The safety specialist will probably be required for two years.

1.3.6.3.7 Training Specialist — For the period required to achieve a full complement of licensed operating personnel, a training specialist should be available to take the lead in operator training and licensing. He will give the classroom instructions, outline the training assignments, conduct written and oral examinations, and handle licensing paperwork.

1.3.6.4 Operations Staff — The plant operations staff consists of an operations supervisor, shift supervisors, and senior control operators, all of whom should be licensed senior reactor operators. Also required, as shown in Figure 2, are control operators, equipment operators, and equipment attendants. The control operators should be licensed reactor operators, as should some of the equipment operators who do not require licenses but who are needed to fill licensed operator vacancies. This operations staff is responsible for the operation of the plant according to established procedures, including periods of reactor shutdown and refueling.

The shift supervisor is responsible for the safe and proper operation of the plant. Normal operation of the plant requires two control operators in constant attendance, one each at the reactor and turbine controls. An assistant control operator is required, to provide surveillance of auxiliary indicators and recorders and to log the data required to maintain adequate records of plant performance.

Roving attendants are provided for necessary data taking and surveillance of equipment and instrumentation of the turbine plant, reactor plant, and miscellaneous outside equipment. An instrument technician is available on each shift to make adjustments and to perform preventive maintenance on the instrumentation systems.

1.3.6.5 Maintenance Staff — The maintenance staff consists of a supervisor and a foreman, directing the work of mechanics, electricians, and machinists. Their responsibility is to keep the plant and its equipment in good operating condition, and to provide skills for modifications and special tests or operations.

1.3.6.6 Administrative Services Staff — This staff includes a supervisor over the office, store facilities, plant protection, and janitorial staffs.

1.3.6.7 Reactor Safety Committee — The Reactor Safety Committee, comprised of specialists in areas of plant design and operation, is charged with the independent review and evaluation of planned reactor operation and of special tests or operations. Deviations from normal operation, and incidents having safety significance, are immediately brought to the attention of this committee. They should also review and approve procedures, license changes, and emergency plans.

Those parts of the organization that should be represented on the committee are the operating staff, the technical staff, and station management.

1.3.7 Safety

1.3.7.1 Nuclear Safety — The plant shall be designed in accordance with the AEC's General Design Criteria for Nuclear Power Plant Construction Permits (10 CFR 50) and shall meet principles of nuclear safety and accident prevention set forth in Safety Criteria to be established.

1.3.7.2 Radiation Control — The exposure levels in occupied areas during reactor operation and shutdown shall be limited to values consistent with the Code of Federal Regulations, Title 10, Part 20.

1.3.7.3 Reactor Site – The reactor site and siting shall be consistent with the requirements of Code of Federal Regulations, Title 10, Part 100.

1.3.7.4 Radioactive Effluent Control – The operations personnel and the general public shall be protected from radiation and radioactive effluents in accordance with the Code of Federal Regulations, Title 10, Parts 20 and 100. Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, holdup capacity shall be provided as required for retention of gaseous, liquid, or solid effluents.

1.3.7.5 Industrial Hygiene – Air concentrations of toxic materials shall not exceed permissible levels, in accordance with applicable State and Federal Regulations. Equipment necessary to provide good industrial hygiene practices shall be provided. Waste disposal facilities, including both radioactive and non-radioactive materials, to prevent environment pollution shall be provided.

1.3.7.6 Fire Protection – Fire protection shall be provided in accordance with the codes and standards of the National Fire Protection Association.

1.3.8 Design

The following broad criteria shall guide the design of the plant in the following order of priority:

- 1) The estimated energy costs must be competitive with contemporary light water reactors of the same size range.
- 2) Plant design shall be based on a realistic safety approach which would assure the licensing of the plant with minimum delays in schedule.
- 3) Operating conditions, hardware configuration, and material selection shall be based on expected state-of-the-art or conservative extrapolation from the Demonstration Plant Program and development programs which are defined. Primary emphasis will be on reliability and availability.
- 4) The plant must be of a design concept (including all critical components) which can be extrapolated from the Demonstration Plant Program and additional Research and Development work which are defined.

1.3.8.1 Reactor Segment – The reactor shall be of the sodium-cooled fast spectrum type, operating on a U-Pu fuel cycle. The reactor shall be designed for operation with nonvented, stainless steel clad (Type 304 or Type 316), mixed U-Pu oxide fuel. Provisions shall be incorporated to permit future modification for operation with vented fuel. Reactor performance shall be limited by the following conditions:

- 1) Fuel at hot spot shall be below the melting point at the maximum anticipated overpower condition.
- 2) Core region target average burnup shall be 67 Mwd/kg (U + Pu).
- 3) Power coefficient of reactivity shall be negative throughout the core under all credible conditions over the power operating range.
- 4) In-core integrated average cladding temperature at the hot spot shall be limited to 1300°F at 100% power.
- 5) It is assumed that fuel will be reprocessed off-site, in a central aqueous reprocessing plant.

The reactor shall incorporate neutron and gamma ray attenuating and shielding provisions, as required to assure the integrity of all permanently installed components for 30 years. Considerations shall be given to designs which permit removal, repair and/or replacement of critical components. Such capability shall not compromise the reliability of the plant during normal operations; but, at the same time, it shall not require more than one year for such low-probability operations.

The plant shall be conservatively designed to produce 100% steam flow at rated steam conditions (~2400 Mwt).

1.3.8.2 Heat Transfer Segment – The heat transfer segment shall remove the heat generated in the heat source and, via primary and intermediate sodium loops, transfer the heat (power) to steam. The heat transfer segment shall also provide a positive and reliable means of removing decay heat from the reactor under any credible set of emergency conditions. The latter requirement shall be met in conjunction with the auxiliaries segment.

The heat transfer segment shall consist of three similar circuits, connected in parallel for core heat removal. Each circuit shall have a reactor

coolant loop (primary loop) and an intermediate coolant loop (secondary loop). Each primary coolant loop shall employ one or more main coolant pumps, one or more intermediate heat exchanger units, and the required flow control devices for normal control, low-flow control, and loop isolation. Each secondary coolant loop shall employ one or more main coolant pumps, one or more steam generator units or banks, and the required flow control devices for normal control, low-flow control, and steam generator isolation.

The temperature gradients across each loop shall be nominally constant, over the desired load-following power range. Variations in power demand shall be accommodated through variation in the main coolant circulation rate, with some trim in temperatures. Valve operators, other than the drive train, shall be accessible for maintenance without required sodium decay. Design of all components and subsystems shall be with conservative engineering factors of safety, to assure rated steam flow corresponding to the ~ 2400 -Mwt rated power output, but at no time should be less than that required to meet the steam flow corresponding to the turbine name-plate rating at design pressure.

Capability shall be provided for positive and reliable means of removing decay heat under all anticipated abnormal and credible accident conditions, including isolation from main condensers, loss of normal power to reactor recirculation pumps, and partial loss of coolant from the system between leak discovery and isolation. All components containing radioactive sodium shall be located within nitrogen-filled and appropriately shielded cells.

The reliability and maintainability for the heat transfer system shall be consistent with the availability allocation under Paragraph 1.3.2. Steam generator modules shall be grouped into multiple banks, with primary emphasis on reliability and maintainability. Provision shall be made for isolation and sodium drainage of any one of these banks with the remainder of the plant in operation.

1.3.8.3 Energy Conversion Segment - The power generation segment shall convert the heat energy in the superheated steam from the heat transfer segment into electrical energy, provide a heat sink to reject the low-level heat from the turbine exhaust, and provide for economical feedwater regenerative heating and treatment. The turbine exhaust back pressure shall be 1.5 in. Hg nominal. The main turbine-generator set shall be conservatively rated for a nominal net electrical output of $\sim 1,000,000$ kwe with 2400 psig/1000°F/1000°F steam conditions.

1.3.8.4 Control and Instrumentation Segment — The plant control and protective system segment shall provide for:

- 1) Regulation of the plant power level in accordance with requirements under Section 1.3.1
- 2) Control of the plant, by automatic and/or manual means, as required, during startup and normal shutdown operation
- 3) Protect the plant from damage as a result of abnormal operating conditions.

The plant control system shall provide the functions necessary to establish the set point levels for power generation in the reactor and to control the transfer of this energy to steam, in accordance with electric power demand. As power demand changes, signals from the generator will be transmitted to the reactor for a corresponding change in the generation of fission energy. Under conditions of startup, and normal and emergency shutdown, the control system shall maintain coolant flows and temperatures within design limits, maintain steam pressure and temperatures within design limits, and control condensate levels in the condenser hot well and feedwater heaters.

The following specific requirements shall further guide the design of the plant control system:

- 1) Steam pressure shall not exceed 130% of rated at any time. Accumulated time in excess of 105% of rated power shall not exceed 12 hr/year.
- 2) Main or reheat steam temperature shall not exceed rated +25°F more than 400 hr/year; shall not exceed rated +50°F more than 15 min continuously, nor more than 80 hr/year. It shall be a design objective that no "steady-state point"* shall exceed rated +100°F under any operating or casualty condition.
- 3) The plant shall be capable of maintaining "steady-state intervals"* of at least 4 hr (provided feedwater temperatures do not vary more than 10°F).

*A point in time is called steady state for the heat transfer system if, during the preceding and subsequent 1/2 min, steam temperatures (at the main and reheat headers) do not vary by more than 7°F, steam pressure by more than 10 psi, and power by more than 1/4%.

- 4) Under normal loading following, steam temperatures shall not vary more than 25°F, and initial pressure not more than 100 psi.
- 5) No reactor trip shall be required in the event of a turbine trip. In the event of a turbine trip, the reactor power shall be reduced at its maximum permissible rate to the steam dump power capability.
- 6) Provisions shall be made for testing of the plant to ASME Power Test Codes.

The reactor control system shall provide the functions necessary to control the fission energy generation in the reactor and to respond to the demand signals from the plant control center (plant load computer). The neutron flux from the reactor shall be monitored from source level to 150% of full nuclear power. Steady-state and transient temperature measurements shall be monitored and maintained within design limits. The reactor control shall have the capability of controlling the reactor from a cold startup at a rate of temperature increase as required by the turbine plant ($\sim 100^\circ\text{F/hr}$).

The plant protective system shall prevent any damage which would result in uncontrolled release of radioactivity from the reactor system. In addition, it shall provide the functions necessary to protect the plant, equipment, and personnel during abnormal operations. The protective action shall be applied automatically, and shall either reduce power to a safe operating level or shut the plant down, whenever unsafe conditions are detected or exist. This system shall protect the plant against damage from thermal shock and excessive pressure at all times. Protection of the core shall receive first priority. The protective system shall not unnecessarily interfere with normal operation of the plant, and shall minimize unnecessary shutdowns. No single-component failure in the plant protective system shall prevent the necessary protective action from taking place or cause a plant shutdown. Interlocks shall be provided to prevent automatic and manual misoperation.

*An interval of time is called a steady-state interval if, during the preceding and succeeding 1/2 hr, steam temperatures do not vary by more than 25°F, pressure by more than 50 psi, and power by more than 1%.

1.3.8.5 Auxiliary Systems Segment – The auxiliary systems segment shall provide for makeup and drainage of primary and secondary main coolant, maintenance of the condition of the coolant and working fluid, maintenance of the condition of the inert gas over the reactor and intermediate coolant streams, maintenance of a suitable inert atmosphere in the reactor cavity and primary equipment cells, secondary component cooling and preheating during normal operations, secondary cooling of reactor under emergency conditions, and safe treatment and disposal of radioactive wastes. As noted previously, the reactor cover gas system shall include provisions to permit future modification for operation with vented fuel.

1.3.8.6 Electrical Segment – The electrical segment shall distribute, control, and meter the power generated by the power generation subsystem to the external power transmission system, and to the plant auxiliary electrical distribution systems. The electrical segment shall provide for startup of the entire system from a cold start, or from normal shutdown conditions. In addition, the electrical segment shall provide emergency, uninterruptible, and transient-free power as required. The electrical segment shall include necessary protective systems, such as grounding, lighting protection, and cathodic protection.

The in-plant electrical subsystem shall supply power for the control and operation of the plant at the following voltage levels:

4160 v, 3 phase, 60 cps, grounded wye

277/480 v, 3 phase, 4 wire, 60 cps

120/208 v, 3 phase, 4 wire, 60 cps

250 v dc

Motor Voltage: 250 hp and above, 4160 v – 3 phase;

1/2 to 250 hp, 480 v – 3 phase;

below 1/2 hp to 208, 3 phase

Any miscellaneous special voltages required for special equipment, devices, or circuits

The in-plant electrical subsystem shall include the necessary switchgear, transformers, motor control centers, panelboards, metering, controls, alarms, and related equipment. The in-plant electrical subsystem shall also include the communications systems. Emergency power shall be provided at all voltage

levels, as required. Uninterruptible and transient-free power shall be provided, as required.

1.3.8.7 Reactor Support Systems Segment – The reactor support systems segment shall provide on-site plant support functions, including:

- 1) In-vessel and ex-vessel fuel handling
- 2) Spent and fresh fuel storage, cooling, and shipping preparation
- 3) Reactor plant equipment decontamination and maintenance
- 4) Miscellaneous conventional power plant services
- 5) Reactor plant cranes
- 6) Reusable fuel shipping containers, exclusive of spent fuel shipping casks
- 7) Special tools and service equipment. Capability shall be provided for safe, reliable, and efficient handling and storage of all removable core elements, such as fuel elements, control rod absorber elements, neutron source, in-core instrumentation elements, moderator elements, reflector elements, and dummy core elements.

Handling of any elements which are within the reactor vessel shall be effected only while the reactor is shut down. Handling of any element outside the reactor vessel will be permitted while reactor is in operation, including fuel preparation for shipment to a separations plant. Storage facilities and capacities shall be determined on the basis of core element decay time storage and handling requirements.

Special consideration shall be given in the design to preventive maintenance of equipment while not in use, to assure its highest availability. To the extent practicable, equipment shall be accessible to permit operation and maintenance functions to be conducted with minimum interference to normal reactor operations.

1.3.8.8 Land, Structures, and Conventional Systems Segment – This segment shall comprise the fixed installations and associated services necessary to house the Plant and personnel. The site is defined as that area within the

control of the plant operator, and includes the exclusion area and approaches to the site.

The fixed installations shall:

- 1) Protect the plant equipment and personnel from damage during all credibly occurring natural phenomena
- 2) Provide an environment for reliable operation, maintenance, and storage of equipment, and for the comfort and safety of facility personnel
- 3) Control the release, within the acceptable limits, of any material or radiation that may be injurious to the health and safety of the populace.

The reactor containment building shall serve to protect the public from uncontrolled fission product release. The containment structure, including access openings and penetrations, shall accommodate, without damage, the maximum transients of pressure and temperature associated with credible accidents. The design leak rate from the containment structure shall be consistent with 10 CFR 100, based on analyses of the design basis accident (DBA). Provision shall be made to test the structure at equivalent design conditions at the completion of building construction. Temporary closures for penetrations, and protection for installed pressure-sensitive equipment, shall be provided for the construction test. Provision shall also be made for periodic leak rate testing of the completed structure at required pressures and temperatures.

If a ventilation stack is required, particular care shall be taken in its design and construction, to insure that it is as short and unobtrusive as possible and does not detract from the general appearance of the plant.

The plant shall be arranged on the site in such a manner as to permit orderly expansion by the addition of future plants of the same or larger capacity. Attention shall be given to arrangement of buildings to facilitate future expansion and common use of facilities by subsequent plants.

1.3.8.9 Seismic Design Criteria

1.3.8.9.1 Basic Requirement — Those systems and components which are essential to the prevention of accidents which could affect the public health and safety,

or to mitigation of their consequences, shall be designed, fabricated, and erected to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena, such as earthquakes.

1.3.8.9.2 Seismic Classifications — Each system, component, and structure in the plant shall be reviewed and classified according to one of the following classifications:

1.3.8.9.2.1 Class I — Those systems, components, and structures, or parts thereof, whose failure might cause or increase the severity of a loss of coolant flow accident, or result in the uncontrolled release of excessive amounts of radioactivity; or which are vital to a proper shutdown of the plant and the removal of decay heat. Class I items shall be further assigned to one of the following subclassifications:

1.3.8.9.2.1.1 Class IA — Those Class I systems, components, and structures whose design, development, and testing should be conservative enough to permit installation and use at any site without requiring significant redesign and development for changes in seismic loads.

Examples of items that may fall into this Class IA include, but are not limited to, the following:

- 1) Reactor vessel, internals, and support
- 2) Core and core support structure
- 3) Top Shield
- 4) Control rods and drive system, including equipment necessary for scram operation
- 5) Fuel assemblies
- 6) Primary sodium piping system, including valves
- 7) Primary sodium pumps
- 8) Intermediate heat exchanger
- 9) Decay heat removal systems
- 10) Reactor and process control and protective instrumentation

11) Fuel handling machines

12) Steam generators

1.3.8.9.2.1.2 Class IB — Those Class I systems, components, and structures which do not require a significant development program, and which have design requirements that are dependent on site conditions, thereby requiring specific final design for each site.

Examples of items that may fall into this Class IB include, but are not limited to, the following:

- 1) Reactor building and all containment structures
- 2) Fuel storage and handling facilities
- 3) Control room
- 4) Reactor building ventilation system
- 5) Inert gas systems
- 6) Secondary sodium system including valves and pumps
- 7) Pipe supports for primary and secondary sodium piping
- 8) Steam generator support structure
- 9) Emergency electrical power system

1.3.8.9.2.2 Class II — All other portions of the plant not falling into Class I.

1.3.8.9.3 Requirement for Class I Structures.

1.3.8.9.3.1 Types of Earthquakes

Design Earthquake (DEQ): The maximum expected horizontal and vertical ground motions.

Maximum Hypothetical Earthquake (MHEQ): The maximum hypothetical horizontal and vertical ground motions.

1.3.8.9.3.2 Functional Requirements

Design Earthquake: All Class I structures shall be designed to withstand, without impairment of function, the loads and forces resulting from the DEQ ground motions, when combined with all other simultaneously occurring loads and forces.

Maximum Hypothetical Earthquake: All Class I structures shall be designed to withstand, without gross failure or loss of capability to achieve a safe shutdown of the plant, the loads and forces resulting from the MHEQ ground motions, when combined with all other simultaneously occurring loads and forces.

1.3.8.9.3.3 Design Seismic Motions

1.3.8.9.3.3.1 Class IA Structures

Design Earthquake: A horizontal ground or equipment support acceleration of 0.33 g, plus a vertical ground or equipment support acceleration of 0.22 g.

Maximum Hypothetical Earthquake: A horizontal ground or equipment support acceleration of 0.5 g, plus a vertical ground or equipment support acceleration of 0.33 g.

1.3.8.9.3.3.2 Class IB Structures:

Design Earthquake: The maximum probable horizontal ground motion at the specific plant site, plus a vertical ground motion that is $2/3$ the maximum horizontal ground motion. This maximum probable horizontal ground motion shall be determined by reviewing the geologic structure and seismic history of the plant site and the surrounding region, and utilizing the more severe of either the maximum recorded ground motion at the site or the maximum estimated ground motion at the site which would have resulted (but may not have been measured) from recorded earthquakes in the region.

Maximum Hypothetical Earthquake: The hypothetical maximum estimated horizontal ground motion at the site, plus a vertical ground motion that is $2/3$ the horizontal ground motion. The hypothetical maximum estimated horizontal ground motion shall be determined by reviewing all active faults in the region which could produce ground motion at the site, and locating the epicenter of the maximum intensity earthquake for each such fault at a location along the fault line with respect to the plant site which would produce the maximum ground motion at the site.

1.3.8.9.4 Seismic Forces and Stresses

1.3.8.9.4.1 Determination of Forces and Stresses — Stresses resulting from seismic loadings for both the DEQ and the MHEQ shall be determined for the more severe of the following methods of determining seismic forces:

- 1) Static lateral seismic forces in conformance with the Uniform Building Code (UBC) for the seismic zone established by the UBC for the plant location. Static vertical seismic forces that are $2/3$ of the lateral forces.
- 2) Lateral seismic forces, as determined by response spectra methods for the maximum estimated horizontal ground or support acceleration. Vertical seismic forces, as determined by response spectra methods for a maximum vertical ground or support acceleration that is $2/3$ the maximum horizontal ground acceleration.

1.3.8.9.4.2 Damping Coefficients — The following damping coefficients, expressed as percentage of critical damping, shall be used in response spectra calculations:

Welded Exposed Steel Structures	1%
Riveted or Bolted Exposed Steel Structures	2.5%
Reinforced Concrete Structures	5%

1.3.8.9.4.3 Combination of Seismic Stresses — The stresses due to horizontal earthquake motions and those due to vertical earthquake motions shall be added directly (in terms of absolute values).

1.3.8.9.4.4 Allowable Stresses — Allowable stress values for each Class I structure under the combined loading of earthquake plus all other simultaneously occurring loads and forces shall be selected to assure that the functional requirements previously described are satisfactory.

1.3.8.9.4.5 Basis for Determining Allowable Stresses — The determination of allowable stresses for each system, component, or structure shall take into consideration such factors as: materials of construction, basic design code, fabrication and quality assurance procedures, and inspectability and testability of the component or structure during the plant lifetime.

1.3.8.9.5 Requirements for Class II Structures

1.3.8.9.5.1 Minimum Requirements — All Class II structures shall be designed in accordance with the applicable provisions of the Uniform Building Code, latest edition, for the seismic zone established by the UBC for the specific plant location.

1.3.8.9.6. Local Codes — If local, state, or municipal government building codes are more stringent than the UBC, the more stringent local code shall govern.

1.3.8.9.7 Seismic Instrumentation — Strong motion accelerograph shall be installed at the plant site to accurately measure and record the ground accelerations associated with a strong earthquake. These data will be used to assess possible hidden structural damage.

1.3.8.10 Applicable Standards, Codes, and Specifications — The following standards, codes, and specifications form a part of the Design Criteria to the extent applicable:

1. USAEC CFR Title 10, PART 20
Standards for Protection Against Radiation
2. USAEC CFR Title 10, PART 50
Licensing of Production and Utilization Facilities
3. USAEC CFR Title 10, PART 100
Reactor Site Criteria
4. National Fire Protection Association Standards
5. Uniform Building Code
6. ASME, PTC 6(1064), Steam Turbines
7. ASME, PTC 19(1955), Instruments and Apparatus Supplement
8. ASME, PTC 32.1 (in preparation) Nuclear Steam Supply Systems
9. ASME, PTC 32.2 (in preparation) Nuclear Reactor Fuel

Specifications and Standards for design, fabrication, construction, quality control, inspection, proof-testing, operation, maintenance, and repair will be identified in the appropriate system design descriptions and specifications for subsystems and components.

1.3.8.10.1 Standards/Specification Application Criteria

When the need for a specification or standard is observed, a study will be made to determine whether a particular standard is mandatory for regulatory or technical reasons. When no mandatory standard is required, a survey shall be made of the following sources to see if a particular applicable standard exists. If no suitable specification or standard can be found, a new Company specification will be prepared.

Standards/Specification Sources

1. USAEC-RDT
2. SOCIETY
3. FEDERAL
4. STATE (AND LOCAL)
5. MILITARY

1.3.9 Document Requirements

Manuals and instructions in support of plant construction, plant testing, plant operation, plant maintenance, and personnel training shall be provided as follows:

1.3.9.1 Plant Construction Manual — This document shall provide construction sequence and interface control, and program the total construction phase.

1.3.9.2 Plant Testing Manual — This document shall identify and describe all testing conducted on-site, subsequent to completion of construction and prior to acceptance by the Customer. It shall include all pre-operational tests.

1.3.9.3 Power Plant Manual — This document shall provide a description of the plant, operating procedures for normal and emergency conditions, and routine test and calibration procedures.

1.3.9.4 Plant Maintenance Manual — This document shall define the scheduled maintenance program and the instruction necessary to accomplish this maintenance. It will also contain a trouble-shooting guide and fault isolation techniques. Operation and maintenance instructions, supplied with certain components and equipment, shall be included.

1.3.9.5 Health and Safety Manual — This document shall prescribe rules, regulations, and procedures which shall provide a general basis for working safely with radiation and/or radioactive materials.

1.3.9.6 Fuel Accountability Manual — This document shall specify the procedures for procurement, utilization, control, transfer, scrap recovery, losses, and return of fuel material to the government.

1.3.10 Spares

The need for spare parts and components shall be established by trade studies, which will weigh the cost of keeping spares on hand against the potential delays in plant operation if spares are not immediately available. Careful review and evaluation of equipment failure rate data and component procurement times will form key inputs for these trade studies.

2.0 DETAILED DESCRIPTION

(The detailed description is covered in Section IV
of AI-AEC-12792, Vol. IV)

3.0 PRINCIPLES OF OPERATION

The operating principles outlined in this section are intended primarily as guidance to the plant designer and may only be representative of procedures to be followed upon plant maturity.

The plant, because of its low incremental power costs, is expected to operate as a base load installation during a significant portion of its lifetime. It will, however, be designed for load following over the range of 25% power to 100%. Operation at less than 25% power will be in a load forcing mode and at off normal system temperatures.

3.1 Normal Startup

Normal startup of the plant is exemplified by return to power after a shutdown for refueling. In this case, sodium will be circulating in at least two of the main heat transfer loops at approximately 500°F; feedwater will be circulating through the evaporator to the flash tank (Steam Dump and Startup System SDD096-64-100), and steam from the flash tank will be circulating through the superheater and reheater, from whence it will be "dumped" directly to the condenser, bypassing the turbine. The auxiliary boilers in the Steam Dump and Startup System will be providing steam to the flash tank as required to maintain system temperatures. Startup from a completely "cold" (preheat) temperature is described in detail in SDD-069-64-100.

After the reactor startup interlocks are satisfied and the required minimum sodium flow (>25%) is established, control rods will be withdrawn (safety rods first followed by shim safeties) to achieve reactor criticality. Reactor power will then be increased manually until neutron flux is about 1% to 2% of full power. At this point, the reactor may be placed on automatic flux control, if desired, to raise the reactor outlet temperature at the design rate (nominally 100°F/hr.). Reactor power and sodium system temperatures will be raised until adequate superheat is available to roll the turbine. At this point the turbine may be rolled by opening the intercept valves and admitting steam to the high pressure section. After synchronization, the turbine-

generator is placed on the line and the bypass steam flows to the condenser are gradually reduced until all steam is flowing through the turbine. When plant power level reaches 25%, the system may be placed on load following control.

3.2 Normal Operation

Normal operation of the plant is defined as operation in the load following mode in the range from 25% to 100% of full power. The systems primarily involved in normal operation of the plant are the reactor (SDD-096-20-000), the heat transfer system (SDD-096-30-000), the turbine generator plant (SDD-096-40-000), and the plant control and instrumentation system (SDD-096-50-000).

Variable load operation is obtained through changes in the main steam flow to the high pressure turbine at constant pressure and temperature. This is accomplished by scheduling the major process variables as a function of load, i. e., for each value of desired load there is a corresponding value of feedwater flow, steam generator sodium inlet temperature, secondary sodium flow, reactor outlet temperature, and primary sodium flow. Reactor power is scheduled on the basis of primary flow. As load decreases feedwater flow is decreased. To maintain constant steam temperature to the turbine under these conditions the secondary hot leg temperature must be reduced; and the secondary flow is reduced to provide a near constant heat transfer distribution in the steam generator. The reactor outlet temperature and primary sodium flow are scheduled down to adjust for the reduced heat transfer to the secondary sodium system required in the IHX.

For steady operation at part load, or for slow transients such as those occurring during normal startup or shutdown, there is very little feedback trim required. For load following operation ($\pm 2\%$ min), or for a rapid load reduction (25% step), however, additional trim controls are required due to the system capacitance. There is also a rate of change limit ($50^\circ\text{F}/30\text{ min}$) imposed on the desired reactor outlet temperature to avoid thermal stress problems in the heat transfer system. Therefore, for a fast load reduction

the heat transfer to the secondary system at the higher reactor outlet temperature is reduced by trimming the primary flow. This reduces the heat transfer to the secondary system by adjusting the temperature difference at the cold end of the IHX. The reactor outlet temperature is indicated as constant for this step reduction. For load changes in general it will follow the steady state schedule but at a maximum allowable rate of 50°F/30 min. A reactor power trim based on controlling this temperature is therefore provided.

During normal operation as defined above, the following auxiliary systems will generally be operated continuously:

- 1) Sodium Overflow/Degasifier System
- 2) Helium Supply System
- 3) Helium Purification System (Purification portion operational only in the event of failed pins)
- 4) Nitrogen System
- 5) Cavity Cooling System
- 6) Vault and Component Cooling System
- 7) R/A Vent System

The following auxiliary systems will generally require only intermittent operation during normal operation of the plant, as briefly described below.

1) Sodium Supply/Purification System

Primary and secondary sodium will be cold trapped only when oxygen content of the sodium exceeds the desired value. Sodium will be added to or subtracted from the system only with the plant shut down.

2) Electrical Preheat System

Portions of the electrical preheat system will be activated continuously; for example, the heaters on the sodium supply tanks. However, much of the system will be required only for initial fill; for example, the heaters on the reactor vessel and the main heat transfer piping.

3) R/A Liquid Waste System

The R/A liquid waste system will be normally activated during and following spent fuel shipping preparation and/or decontamination of contaminated equipment. The system will dispose of the resulting liquid wastes either by direct discharge to the sanitary sewer system or by segregated decay storage for subsequent controlled release to the sanitary system or through tank trucks for off-site disposal. Release to the sanitary system will be regulated within limits set forth in 10CFR20.

4) Steam Dump/Startup System

This system must be in a state of operational readiness during normal operation, but will be required to operate during startup, shutdown, following reactor or turbine trip, and rapid load reductions ($> 25\%/min$).

5) Steam Generator Rupture Relief System

This system is required to operate only when a rupture disc in one or more of the steam generator modules bursts, but must be operational when plant is on the line.

Other systems, not classified in the auxiliary segment, which must be operational to permit normal operation of the plant include:

- 1) Container/Top Shield System (Top Shield not required to rotate during plant operation).
- 2) Energy Absorbing System - Must be in a state of operational readiness during plant operation.
- 3) Plant Protection System
- 4) Radiation Detection and Monitoring System
- 5) Electrical System

6) Site/Structures/Services System

7) Reactor Building System

The In-Plant Fuel Handling System and the Fuel Storage/Shipping System are not required to operate to permit normal plant operation.

3.3 Normal Shutdown

Reactor plant shutdown is normally for the purpose of re-fueling, turbine maintenance, and maintenance on other plant components which can only be worked on during shutdown. The major activities for normal shutdown will be as follows.

3.3.1 Reactor Power Reduction

The reactor power will be reduced by driving in the control rods. Reactor outlet temperature, and reactor ΔT will also be reduced until adequate superheat can no longer be maintained (5 to 10% power) at which time the turbine will be tripped. These operations are achieved while maintaining sodium flow through the reactor at 25% or greater. Upon turbine trip, the intercept valves will close, the steam dump valves on the superheaters and reheaters will open, and the steam will be reduced in pressure and exhausted to the condenser. The sodium flow through the reheaters will be reduced to zero and the reheaters will be isolated.

Feedwater will continue to be pumped to the evaporator by the decay heat condensate pump and the decay heat boiler feed pump. Power for the electric drive on the decay heat condensate pump will be provided by the emergency power bus.

3.3.2 Reactor Outlet Temperature Reduction

With the outlet temperature reduced to the point where reactor trip will not cause temperature shock ($\sim 600^\circ\text{F}$), the reactor is tripped and reactor ΔT reduced to isothermal refueling temperature ($\sim 500^\circ\text{F}$). This

refueling temperature can be maintained partly by the preheating system and partly by sodium pumping (10-15% flow range). During this time (after turbine trip) heat is removed through the steam dump system on pressure control. The steam produced is used for feedwater heating and auxiliary turbine power.

3.4 Reactor Trip

Reactor trip is a rapid shutdown of the reactor plant. In the power range, measured high flux, high temperature, low flow, low reactor sodium level, or manual initiation will cause trip. When trip occurs, the safety rods and the shim-safety rods will be released, power to the sodium pump drives will be cut off, and the throttle valves in the primary and secondary heat transfer circuits will start closing to a predetermined limit. Turbine trip will also take place. Upon turbine trip, the block and intercept valves will close, the steam dump system will maintain pressure control. Excess steam above the steam dump capacity will be exhausted to the atmosphere through the power relief valves.

Following reactor trip, the throttle valves will control the sodium flow in the primary and secondary circuits to hold the rate of decrease of reactor outlet temperature at approximately 100°F/hr. Sodium flow will be continued through the superheaters and evaporators, but the reheater will be valved off. After the main turbine is tripped, feedwater will be supplied by the decay heat condensate pump and the decay heat boiler feed pump. When the main heat transfer loops are nearly isothermal, which will take approximately 4 hr, the reactor will be ready for restart, using the normal startup procedure.

3.5 Turbine Trip

A trip of the turbine generator does not require reactor trip, although it is possible that the reactor will trip in the ensuing upset of the electrical system. If the reactor does not trip, the automatic control system will begin lowering the reactor power automatically at the maximum controllable rate (25%/min.). Upon turbine trip, the turbine intercept valve will close and the steam

dump valves will open and dump to the condenser. The remainder of the steam flow will be exhausted to the atmosphere through the power relief valves. The steam dump and power relief valves are sized such that the safety valves are not required on turbine trip. Reactor power will continue to drop until it matches the capability of the steam dump system. The reactor can then continue to be shut down, or the turbine generator may be restarted and placed on the line. The steam bypass around the turbine to the condenser will be gradually reduced, if the turbine is to be restarted, until all the steam is flowing to the turbine and the plant returned to power.

3.6 One-Circuit Shutdown

Loss of sodium or feedwater flow is not expected to cause a reactor trip, if only one of the three main heat transfer circuits is affected. Instead, only the affected circuit (primary loop, secondary loop, and feed-water/steam loop) will be taken out of service. In a scheduled shutdown of one circuit, the flow of primary and secondary sodium and the flow of feedwater to the evaporator of the circuit to be shut down will be reduced. At the same time, the power demand from the turbine generator must be reduced sufficiently to compensate for the loss of steam from the loop being shut down. When the flow of sodium in the affected circuit reaches about 15%, valves in the primary and secondary loops and the feedwater systems will be closed, isolating the loop. At the same time, the power demand will be reduced to about 60%. When the shutdown circuit is isolated, the power demand can be reset to about 70%, approximately the power capability of the remaining two loops. Temperatures in the shutdown circuit will gradually decay. If the circuit can be restored to service before the hot leg temperature drops more than about 50°F, then a reactor shutdown, prior to returning the third circuit to service, will not be required. However, if the circuit is out of service for a period of time permitting its hot leg temperature to drop more than 50°F, it will be necessary to shut the reactor down, allow its temperature to drop

to match that of the shutdown circuit, restore the shutdown circuit to service, and then restart the reactor. The startup procedure will be as outlined in Section 3.1. When one loop is out of service, faults in either one of the two operable circuits will cause complete reactor shutdown.

3.7 Emergency Operations

Emergency conditions are defined as sudden events requiring immediate action. Most of the emergencies which may arise will be handled automatically by the reactor protective system. The remainder require operator action. Major emergencies, the expected indication, and immediate action required are listed in Table 3.7-1.

3.8 Refueling

Refueling is planned on a one-year cycle with one-half of the core, one-fifth of the inner radial blankets and one-seventh of the outer radial blankets being replaced each cycle. A like number of blanket elements will also be rotated 180° to provide more uniform plutonium buildup associated with the sharp flux gradient in the blanket region of the reactor.

Refueling is accomplished during a plant shutdown. The sequence and operations during a refueling are presented in SDD-096-80-000. Spent fuel from the reactor region is lifted vertically with the in-vessel fuel handling machine and raised into the sodium pool above the reactor. The top plugs are rotated into position over a transfer location and the spent fuel is placed in an empty (sodium-filled) canister in one of the in-vessel transfer positions. The spent fuel is removed from the in-vessel transfer positions to the ex-vessel storage tank by the ex-vessel machine during the same refueling shutdown. It is allowed to decay in the ex-vessel storage tank to a low enough power level to permit shipment. From this tank it is removed for shipment. Blanket elements are removed from the reactor in the same manner.

TABLE 3.7-1

EMERGENCY CONDITIONS

Emergency Condition	Expected Indication	Required Action
1) Fuel pin cladding failures	Excessive fission product activity in reactor cover gas	Activate cover gas purification system
2) Inadequate coolant flow to one or several fuel elements	Excessively high fuel channel exit temperatures from affected fuel elements	Manual reactor trip
3) Unscheduled positive re-activity insertion	Increase in reactor neutron flux	Automatic reactor trip
4) Low reactor coolant level (may be caused by leakage in primary piping or vessel)	Alarm on low reactor coolant level, followed by reactor trip if level continued to drop	Operator action to stop drop in level, followed by automatic reactor trip if operator action is unsuccessful.
5) Heat sink failure	Rise in secondary system Na cold leg temperature	Manual reactor trip (unless automatic trip on reactor Na high temp occurs first)- Place emergency feedwater system in service
6) Inadequate core coolant flow	Excessively high fuel channel exit temperatures	Automatic reactor trip
7) Ex-core fuel handling machine loss-of-cooling while handling a fuel element	Loss of cooling alarm	Insert fuel element into in-vessel or ex-vessel storage
8) Radiological release	Containment building continuous air monitor alarm	Evacuate building - Shut reactor down if it is source of activity. Shut tight H&V isolation valves

TABLE 3.7-1 (Continued)

EMERGENCY CONDITIONS		
Emergency Condition	Expected Indication	Required Action
9) Sodium fire		
a) Primary system (open vault; reactor not in operation)	Smoke and radiation detector alarms	Close vault. Smother fire by blanketing with nitrogen-- isolate and drain leaking systems
b) Secondary System	Smoke alarms and/or visual observation of loss of level in secondary system	Isolate and drain leaking system--extinguish fire with dry fire extinguishers
10) Loss of all vault cooling	Alarm on vault nitrogen tempera- ture	Reduce reactor temp at maxi- mum controllable rate
11) IHX secondary to primary leakage	Loss of Na level in secondary system and increase in level of primary system	Shut down affected circuit and isolate the primary loop - drain primary and secondary loops
12) Steam generator water-to- sodium leak	Rise in hydrogen and oxygen concentration in secondary system	Shut down affected loop -- isolate and drain steam generator
13) Steam generator tube rupture	Activation of relief system and leak detectors. Automatic one loop shutdown	Isolate and drain steam generator

New fuel and blanket elements are stored in canisters in the ex-vessel storage tank. Prior to loading into the reactor, an element within a canister is first placed into an in-vessel transfer location with the ex-vessel fuel handling machine. The in-vessel machine is used to transfer the new element from the canister to the reactor, and a spent element from the reactor is subsequently loaded into this canister.

The fuel elements have an average reactor residence time of two years; the inner radial blanket elements five years; the outer radial blanket elements seven years; and the shim-safety rods eight years. After each one year cycle, an average of 137 fuel elements, 13 inner blanket elements, 10 outer blanket elements, and one control rod absorber column are exchanged, for a total of 161 reactor elements. In addition, 23 blanket elements are rotated 180°. The refueling time and associated unavailability breakdown are given in SDD-096-80-000 and SDD-096-81-000. The downtime per refueling cycle is estimated at about 312 hours. There is no scheduled unavailability contribution due to fuel handling because refueling can be accomplished within the scheduled annual turbine maintenance time.

4.0 SAFETY PRECAUTIONS

4.1 Hazards

The major hazards which could arise, and which will be the subject of formal casualty studies, include:

- 1) Inadequate sodium flow to one or several fuel elements,
- 2) A large unscheduled reactivity insertion,
- 3) Low reactor coolant level,
- 4) Primary piping system leakage,
- 5) Heat sink failure,
- 6) Inadequate total core coolant flow,
- 7) Protective system failure,
- 8) Fuel handling machine loss-of-cooling, and
- 9) Excessive radiological release in containment building.

4.2 Safety Precautions

The safety precautions for the plant will be governed primarily by the "General Design Criteria for Nuclear Power Plant Construction Permits," 10 CFR 50, and associated federal regulations. Appropriate state, local government, insurance, and utility regulations and practices must also be observed. The safety considerations particular to fast reactors and the use of sodium as a coolant must be taken into account, in design and in plant operation. Studies of plant safety will be prepared and presented to the AEC Division of Reactor Licensing in the Preliminary and Final Safety Analysis Reports.

5.0 MAINTENANCE

The overall maintenance program will be based on the plant availability goals set forth in Section 1.3.2. Maintenance frequency and type will be planned to permit achievement, by each major plant segment, of the availability goals specified. In some cases, this may require stocking of complete spare components (e.g., main heat transfer pump internals). In other cases, it may require the capability for on-line maintenance (e.g., individual channels of the protective system through accessibility or redundancy). In still other cases, inspection and required maintenance will be performed on a periodic basis with the plant shut down (e.g., the steam turbine).

Based on the current conceptual design, the normal maintenance methods and policies for the various major plant system and components are as follows.

5.1 Reactor System

5.1.1 Vessel and Internals

The reactor vessel is not expected to require maintenance during its lifetime. However, the current design has provisions for external inspection and repair of the vessel. The vessel internal permanent structure will be made for complete draining of the reactor vessel. The reactor support/plenum assembly, reactor radial restraint assembly and radial neutron shield will be replaceable. The reactor vessel jacket can be pumped out in the event of a sodium leak.

5.1.2 Fuel and Blanket Assemblies

Fuel and blanket assemblies will be removable and replaceable using the fuel handling equipment. In the event that complete unloading of the entire reactor is required, special provisions will be supplied at that time. Provisions will be made for handling a reasonable range of abnormal conditions such as stuck or damaged fuel elements.

5.1.3 Control Rods

Spare control rod drive systems and poison elements will be supplied, as required, to meet the reactivity control system reliability requirements. Control rod assemblies will be removable and replaceable with the fuel handling equipment.

5.1.4 Top Shield

The rotating shields will incorporate replaceable bearings and seals, for which spares may be provided. Capability will be provided for monitoring seal leakage, during reactor shutdown and with the containment dome removed. Capability will also be provided for the replacement of fuel element coolant outlet failed thermocouples during reactor shutdown.

5.2 Heat Transfer System

5.2.1 Primary Loop Block Valves

Failure rate for these valves is expected to be very low, since the valves will be closed only in the event that it is necessary to isolate one of the heat transfer circuits. Valve operator drives will be maintainable from the reactor building floor during plant operation. Repairs to the valves and the portion of the drive train in the shielded vaults will require 10 to 15 days for decay of sodium activity and a reactor shutdown. Removal of a block valve or replacement of the gates will require draining the affected primary loop and the reactor vessel below the level of the valve. Decay heat removal will be provided by the other loops. The valves will be checked for operability during each scheduled reactor shutdown.

5.2.2 Primary Throttle Valve

Valve operator drives will be maintainable from the reactor building floor during plant operation. Repairs to the primary throttle valve will require entry to the primary vault after 10 to 15 days of sodium activity decay. The decay can take place with the reactor in operation on the remaining two

heat transfer circuits and with the block valves on the affected loop closed. Actual repairs to the valve will require a complete reactor shutdown. The affected loop will be drained behind closed block valves, to effect repairs in the throttle valve.

5.2.3 Intermediate Heat Exchanger (IHX)

It is expected that, if repairs are required to the IHX, they will either be repairs to a leaking tube or a repair of a possible leak in the shell. A leak in the shell will result in secondary sodium leaking into the primary vault (but no sodium fire, because of the nitrogen atmosphere in the vault), and will require a shutdown of the affected loop, a 10 to 15 day decay period, and then a reactor shutdown while the vault is entered, the solidified sodium cleaned up, and the shell repaired. If significant quantities of fission products are present in the primary system, it may be necessary to isolate the IHX and subject the primary coolant side to clean sodium flushing or chemical cleaning to reduce the radiation to working levels.

Tube repairs will be accomplished by plugging tubes in place, using semi remote leak detecting and tube welding equipment.

5.2.4 Primary Pump and Drive System

Repairs to the pump drive system and accessible pump sub-components, such as the shaft gas seal, can be made with the pump shut down, affected heat transfer circuits isolated, and with the reactor operating at part load on the remaining two heat transfer circuits. Repairs requiring removal of the pump internals will be accomplished by removing the internals as a unit, using either a shielded cask or a plastic bag. Spare pump internals, if available, may then be installed. The reactor will not be operated during this effort. Decontamination of the faulty pump internals is performed in the equipment wash cell prior to the repair.

5.2.5 Primary Piping and Pump Casing

A leak in the primary piping or pump casing has a very low probability of occurrence, and will not be accompanied by a sodium fire, because of the nitrogen atmosphere in the galleries. Such a leak can be repaired by closing the block valves on the leaking loop, draining it, allowing the sodium to decay for 10 to 15 days while operating the reactor on the remaining two circuits, then shutting down and entering the vault, cleaning up the spilled sodium, and repairing the leak. Flushing of the loop may be required to reduce the level of fission product activity.

5.2.6 Secondary Block and Throttle Valves

Secondary block and throttle valves are accessible at all times. Repair can be accomplished with the reactor operating at part load on the remaining two heat transfer circuits.

5.2.7 Secondary Pump and Drive System

The secondary pump will be repaired in the same manner as the primary pump. The reactor will be operated at part load on the remaining two heat transfer circuits while the repairs are being made.

5.2.8 Steam Generators

The steam generators are expected to be highly reliable, due to their design simplicity and the inspectability of the tube sheet welds. However, tube failures, permitting water to come into contact with secondary sodium, may take place. The affected secondary loop will be shut down, and drained on both the sodium and water sides. If required, the module containing the leaking tube will be cut out and replaced with a spare module. Double block valves are provided on the steam side to permit two-thirds power operation with the remaining two heat transfer circuits. The secondary loop will then be refilled, and the sodium circulated through the system cold trap as necessary to reduce oxygen and hydrogen contamination to an acceptable level. Steam generators must be also cleaned periodically on water side.

5.2.9 Secondary Piping, Expansion Tank, and Steam Generator Shells

The piping, expansion tank, and steam generator shells are designed for long life and high integrity. In the unlikely event of a sodium leak, the affected system will immediately be shut down, while the reactor continues operation on the remaining two heat transfer loops. The system will be drained as rapidly as possible and the probable fire extinguished with calcium carbonate or other "dry" fire fighting material.

5.3 Turbine Generator

The turbine generator and its associated equipment (condenser, condensate pump, feedwater heaters, feedwater pumps, etc.) are conventional equipment, and will be maintained in accordance with standard utility practice. The schedule for inspection of the turbine stages will be coordinated with the reactor refueling schedule to make optimum use of the time during which the entire reactor plant must be shut down.

5.4 Control and Instrumentation

Adjustment and on-line maintenance of control and instrumentation equipment will generally be accomplished as follows.

5.4.1 Plant Control System

Except for the last control element in a chain (for example, a control valve), the plant control system will be capable of being maintained with the reactor at power. Individual portions of the control system will be placed on manual or semi-automatic control while adjustments and repairs are being made.

5.4.2 Plant Protective System

By utilizing two out of three coincidence channels throughout the plant protective system, it will be possible to maintain any channel, replacing components with spares, while the reactor is on the line. Failure of a sensor will generally not require reactor shutdown. Installed spares are provided to always maintain the two out of three coincidence protective function.

5.4.3 Nuclear Instrumentation

All flux monitoring channels will be triplicated, making it possible to perform maintenance on one channel (exclusive of sensor) without shutting the reactor down.

5.4.4 Radiation Monitoring

In general, individual components of the radiation monitoring system will be repaired while performing "hand" monitoring, using substitute instruments. Reactor shutdown will not be required.

5.5 Auxiliary Systems

These systems include the sodium supply and purification system, the inert gas supply systems, the radioactive vent system, the liquid waste system, the component cooling system, the electrical preheat system, the cavity liner cooling system, and the steam generator rupture relief system.

5.5.1 Sodium Supply and Purification System

Two redundant primary purification systems are included in the plant design. These systems include a cold trap, a plugging meter, pumps, valves, and piping. One entire system may be removed from service for replacement of a cold trap or repair of a component, while the other is in use. Maintenance will be performed by allowing 10 to 15 days for decay, and then entering the purification system vault. Cold traps containing significant quantities of fission products will be remotely disconnected from the piping system and placed into a disposal cask. The secondary purification systems, one for each secondary loop, will be nonradioactive. Any one of these systems can be removed from service for a short period of time without shutting down the associated secondary loop.

5.5.2 Inert Gas Supply Systems

These systems must be in essentially continuous operation. Bypass lines placed in service manually are included around all control elements to permit maintenance or replacement of the control element without interruption of service.

5.5.3 Radioactive Vent System

This system, consisting of pressure controllers, compressors, and decay tanks, will normally be accessible on a limited time basis. Pressure controllers will generally have manual bypass lines to permit maintenance. Compressors and decay tanks will be redundant, to permit maintenance on one unit while the other is in use.

5.5.4 Liquid Waste System

In general, operation of the liquid waste system can be temporarily interrupted for repairs without affecting plant operation.

5.5.5 Component Cooling System

One redundant 100% pump and one redundant 100% heat exchanger are provided for both the kerosene system and the closed loop cooling water system to permit maintenance without system shutdown.

5.5.6 Electrical Preheat System

Redundant heaters, operating at partial voltage, are supplied on most piping and equipment which normally contains or could contain sodium. Repairs are expected to be very infrequent. Sodium activity decay will be required for repairs of heaters on equipment containing primary sodium. In the secondary system, heaters can be replaced by shutting down the affected loop while running the reactor at part load on the remaining two circuits. The remaining parts of the preheat system are on vessels and auxiliary lines not normally used during reactor operation and, therefore, required maintenance can be accomplished without reactor shutdown.

5.5.7 Cavity Liner Cooling System

The main circulating pumps and atmospheric boiler for this system are on standby during operation of the plant and can be tested and maintained at any time. These components are located outside the reactor building to facilitate maintenance at all times.

5.5.8 Steam Generator Rupture Relief System

This system is passive except in case of a steam generator module sodium-water leak. Maintenance will primarily consist of periodic inspections to verify integrity of the system.

5.6 Electrical System

The conventional electrical system is maintained in accordance with standard utility practice.

5.7 Support Equipment

Support equipment consists primarily of the in-core and ex-core fuel handling machine, and other machines, such as the plug handling cask and shipping cask. This equipment is required to be used periodically during refueling operations. When not in use, the equipment will be thoroughly checked and repairs made as necessary to assure trouble-free operation. In general, these repairs can be made by direct contact maintenance. Decontamination of machine parts occasionally may be required.

5.8 Site and Structures

The reactor building will periodically be checked for leakage rate. This can be accomplished during routine reactor shutdowns. Other building maintenance will be in accordance with the established practice of the utility.

5.9 Unusual Maintenance

Unusual maintenance is defined as those activities necessary to restore the plant to operation following low probability serious malfunctions. These malfunctions include: partial fuel meltdown, damaged reactor vessel internals, intermediate heat exchanger (IHX) tube leak, steam generator tube rupture and primary sodium spill.

Table 5.9-1 is a summary of reactor outage time and manhours (minimum and maximum) expected for each of these malfunctions. In the

TABLE 5.9-1

UNUSUAL MAINTENANCE SUMMARY

Item	Nominal Expected		Repair Time (Manhours) (3)	Minimum Expected ⁽⁴⁾		Repair Time (Manhours)
	Total Outage (1)	Equivalent Zero Power (2)		Total Outage	Equivalent Zero Power	
a) Partial Fuel Meltdown	20 mo	20 mo	8000	13 mo	13 mo	7000
b) Damaged Reactor Vessel Internals	21 mo	21 mo	2500	7 mo	7 mo	2500
c) IHX Tube Leak	5 wk	4 wk	1000	3 wk	3 wk	1000
d) Steam Generator Tube Rupture	3 wk	1 day	2300	3 wk	1 day	2300
e) Primary Sodium Spill	1 wk	6 days	700	0	0	0

(1) Total outage time of the affected loop, assuming major maintenance equipment is engineered but not procured.

(2) Includes two loops operating at 70% power.

(3) Total manhours to repair.

(4) Minimum expected assumes special tooling and/or spare parts available on site.

maximum-expected-time cases, an estimated normal inventory of tools, equipment, and spare parts is assumed. In the minimum-expected-time cases a total inventory of spare parts, tools, and special equipment is assumed. These assumptions serve to illustrate the potential outage time saving possible by stocking long-delivery-time items, or considering possible availability of the special tools and equipment from a pool on the basis of multiple reactors in operation. Addition of equipment in the initial design, and in some cases where considerations permit omitting steps in the nominal procedure, can reflect a saving in outage time.

A brief discussion of each of the selected unusual maintenance items follows:

a. Partial Fuel Meltdown

The top shield design provides for the removal of a seven element cluster through a minor modification to the in-core fuel handling machine port. This opening can be positioned over any core position by rotation of the inner and outer top-shield plugs.

The basic recovery procedure for a partial fuel meltdown is as follows:

- 1) Locate and examine the top and accessible sides of the damaged fuel elements. This can be done by using the in-core fuel handling machine to determine freedom from adjacent elements, and using remote viewing equipment when the reactor is partially drained.
- 2) Concurrently with the above, initiate procurement of the special tools and equipment necessary to remove and examine the fuel element cluster and provide for disposal of the cluster.

- 3) Unload fuel elements from the core, except for the damaged elements; replace them with the spacer elements. This step may be eliminated if the cause of the damage can be established and there is confidence that the undamaged fuel elements are acceptable for further service.
- 4) Concurrently with the preceding step, continue the examination of other reactor internals with remote viewing equipment and prepare and check out delivered radioactive equipment maintenance cask for removal of the seven element cluster.
- 5) Initiate the visual examination of the undamaged fuel elements in the radioactive maintenance cell.
- 6) Following unloading of undamaged fuel elements, remove the seven element cluster to storage. If no temporary storage is available, the cask may be used until the element is placed in the radioactive maintenance cell for examination or preparation for shipment.
- 7) Upon completion of all desired examination, a review of data is scheduled to establish the corrective action requirements.
- 8) Prepare corrective action procedures while waiting for delivery of material and, upon delivery, take the corrective action.
- 9) Reload the core and restart plant.
- 10) Place the damaged element in the special canister and ship off-site for examination, reprocessing, or disposal.

It is estimated that a reactor shutdown of from 13 to 20 months would be required to accomplish recovery from a partial fuel meltdown. The elapsed time is dependent on the nature of the damage, the fuel examination requirements, the availability of materials and equipment, and the corrective action requirements. Approximately 8000 maintenance manhours are estimated to be required to accomplish these tasks.

The special equipment needed for recovery from a fuel meltdown is estimated to require from three months to a year for delivery. The equipment includes remote inspection equipment, special tooling for core in-cell examination and cluster removal, an inner plug cask, and a shipping canister for the cluster. The inclusion of an in-cell fuel element cooling device will allow an earlier start of nondestructive fuel element examination and the evaluation indicates this would reduce the total down time of the plant.

b) Damaged Reactor Vessel Internals

The reactor vessel internals can be removed through the opening provided by the removal of the inner top shield plug. The most critical component is the reactor support/plenum assembly. Other internals which may require removal for repair or replacement are the reactor radial restraint assembly and the radial neutron shield.

The basic recovery procedure for replacing the removable support plenum assembly consists of the following:

- 1) The initial step for recovery from a damaged removable support plenum assembly consists of the following essentially concurrent activities: unload the core; inspect the reactor internals; evaluate inspection results; and engineer, procure, and deliver special tools and equipment.
- 2) The corrective action involves the preparation of procedures while waiting for materials; the erection of special equipment and shielding; the removal of the top shield inner plug;

the removal, repair, and/or replacement of the damaged internals; inspection of the repair and reinstallation of the internals; reinstallation of the top shield inner plug, and cleanup of the area.

3) Reload the core and restart the plant.

It is estimated that a reactor shutdown of 7 to 21 months would be required to replace the removable support plenum assembly. The elapsed time is dependent upon the nature of the damage and the availability of materials and equipment. Approximately 2500 maintenance manhours are required to accomplish the tasks.

The delivery time of the special equipment required for the replacement of the removable support plenum assembly is estimated to be 6 to 15 months. If the equipment is available at the time of the incident, approximately 14 months can be saved on the outage.

c. IHX Tube Leak

The design of the IHX provides for ease of access for tube repairs with either contact or remote maintenance methods. Access is provided to both upper and lower tube sheets.

The procedures for recovery from an IHX tube leak consist of:

- 1) Making preparations for, and removing, the access plugs to the IHX.
- 2) Removal and reinstallation of insulation, heaters, and manhole cover heads.
- 3) Leak checking and tube repair.
- 4) Preheat and fill of the primary and secondary loops, and a normal cold startup.

Recovery from an IHX tube leak requires an estimated five weeks outage for the maximum case and three weeks for the minimum case. The estimated manhour requirement is about 1000 hours for either case. The difference in the outage estimates is due to the inclusion of an auxiliary cask capable of permitting remote repairs without waiting for sodium decay.

d. Steam Generator Tube Rupture Involving the Relief System

The design of the steam generator complex includes hydrogen-in-sodium detection devices to allow detection of leaks and isolation of affected loop prior to actuation of the relief system. The purpose of the devices is to provide tube leak indication before the leak becomes of sufficient size to operate the relief system.

The following describes the recovery procedure after a tube rupture involving the relief system operation:

- 1) Operational tasks including the restart of the plant to two loop operation, draining the steam generator and sodium loop, pumping reclaimable sodium from the relief system tanks, and cleanup of the sodium by use of cold traps;
- 2) The corrective action that consists of draining non-reclaimable sodium from the relief system tanks, inspection of rupture discs, cleaning relief system piping and equipment, removal and replacement of the defective module, inspection and checkout of the module installation, and replacement of discarded sodium; and
- 3) The restart that consists of the preheat, fill and cold trapping of the affected loop and the trip, cool, and restart from an isothermal condition.

The time for maximum and minimum cases for recovery from a steam generator tube rupture when using the relief system is approximately

three weeks. The reason for the similarity in time is the sodium purification requirement which is governing in either case. The estimated maintenance manpower requirement is 2300 hours for either case.

e. Primary Sodium Spill

The heat transfer system design features a secondary enclosure which precludes any serious sodium fire due to the inert atmosphere within the vaults. However, leakage of sodium from a valve stem freeze seal, for example, could result in small quantities of sodium leaking into the vaults and a possible fire if O_2 concentration in the vaults were allowed to increase above normal values.

Recovery from a primary sodium spill includes the following tasks:

- 1) Operating tasks of transferring the plant to two loop operation and draining the affected loop;
- 2) Following the decay period, the corrective action consists of opening the vault, covering the sodium with calcium carbonate, installing of temporary shielding and equipment, correction of the leak cause, removal and reinstallation of insulation, and closeup of the vault; and
- 3) Restart consisting of preheating and filling the affected loop, and bringing the loop back into service as described in Section 3.6.

The minimum case has no time assignment, on the premise that the cleanup is included in the outage for the repair of the leak causing failure (e.g., repair of cooling system or valve freeze seal). The maximum case requires additional time beyond the necessary loop decay and component repair.

6.0 TEST REQUIREMENTS

6.1 Objectives

The objectives of the plant startup test program are to prepare the nuclear island for operation, determine that the nuclear island meets performance requirements and operates at design conditions. Additional objectives are:

- a) To train personnel in the techniques of plant operation and maintenance.
- b) To supply manuals for operation, maintenance, and health and safety.
- c) To obtain data which will verify plant safety characteristics.
- d) To obtain plant performance data which will assist in improving its performance and aid in the design of subsequent plants.

6.2 Scope of Work

The nuclear island startup test program includes operator training; preparation of operations, maintenance, health and safety, and test manuals; and the conduct of the pre-critical fuel loading, low power, and rise-to-power tests.

6.3 Definitions

1. Pre-critical Phase

The phase encompassing those tests performed on individual plant systems and equipment to verify operability, maintainability, and performance characteristics. These tests are done prior to loading fuel into the reactor and after completion of construction, construction testing, and quality assurance inspection of the completed system.

2. Fuel Loading Phase

The phase which starts when the first fuel element is loaded into the reactor and which ends when the full operational reactor loading has been achieved.

3. Low-Power Phase

The phase following achievement of the full operational reactor loading, during which reactor tests are performed at a power level sufficiently low to minimize temperature effects associated with reactor power that might affect the experimental results. The highest power to be allowed during this phase and the means for measuring it will be established during the detailed planning of the startup test program.

4. Rise-to-Power Phase

The phase during which reactor power will be increased in steps to full design power. This phase starts upon completion of the Low-Power Phase and will end when the nuclear island has demonstrated satisfactory performance at full design power.

6.4 Activities

The following major activities comprise the startup testing program at the plant site.

6.4.1 Pre-Critical Tests

As each plant system is completed, it will be turned over to the plant operating crew for checkout. Each system (for example the cooling water and kerosene systems, the inert gas supply systems, the radioactive vent system, the radioactive liquid waste system, the electrical system, etc.) will be checked for operability and filled with fluid as appropriate in accordance with written procedures. Concurrently with the preceding, site assembly of the fuel handling equipment will be performed and the equipment will receive a "cold" checkout.

After the plant system operability checkout is completed, the plant startup test program shown in Table 6.4-1 will be started. Containment leak rates will be measured and the fuel handling equipment will be used to load the reactor core with "dummy" fuel elements.

The primary and secondary sodium storage tanks will be preheated, purged, and filled with sodium. The primary and secondary sodium purification systems will similarly be preheated and filled and placed in service to purify the sodium in the storage tanks. The secondary systems will be preheated, purged, filled and circulation established. The reactor vessel and primary systems will be purged and pre-heated and the fuel handling equipment, including the top shield, will be tested in position over the core to assure that the rise in temperature has not introduced problems. The reactor and primary systems will then be filled with sodium and circulation established.

Measurements of system hydraulics, pipe displacements, equipment vibration, and flow decay, calibration of instruments, checks for gas bubbles, and checks on valve and pump performance will be carried out with the dummy elements in the core. The isothermal temperature of the heat transfer systems will then be raised to 780°F and a hot sodium circulation test run and the hydraulics tests repeated.

The fuel handling equipment will be checked in sodium at operating temperature and the plant protective and plant control systems will be checked out using the signals from the heat transfer system.

At the completion of the hot sodium circulation test, the systems will be drained to permit equipment inspections and maintainability checks. For example, a primary pump may be removed, cleaned, inspected, and replaced. Remote visual examination of selected core internals may be accomplished. The heat transfer systems will then be refilled, the nuclear instrumentation checked out, and new fuel placed in ex-vessel storage in preparation for reactor core loading.

TABLE 6.4-1
PRECRITICAL TESTS

Test	Purpose	Results Used for		Special Equipment Requirements
		Operation	SAR Verif.	
1. Containment leak-rate and building deflection measurements	Verify requirements	X	X	Precision barometers, manometers, etc.
2. Water system tests, including steam dump and startup system	Verify requirements	X	X	
3. Load dummies and control rods	Checkout core clamp, prepare for loading, checkout fuel handling system	X		Core element dummies
4. Preheat and fill sodium systems	Checkout preheat systems, pipe displacements, fill systems, and cleanup systems	X		
5. Preliminary system hydraulics	Check pump performance, flow restrictions, flow coastdown, gas bubbles, instrument calibration	X	X	Bore scope to view sodium pool and inside pump cases, core element dummies
6. Component vibration tests	Verify acceptability	X		Vibration detection instruments
7. Valve characteristics	Verify acceptability	X		Special pressure instruments for check valve tests
8. Hot circulation test (780°F)	Checkout sys., pressure integrity, pipe displacement meas., fuel handling (at 500°F)	X		
9. Drain sodium systems	Checkout drain sys., component inspect., one-loop shutdown capability	X		
10. Refill sodium systems	Prepare for fuel loading, check one-loop fill procedure			
11. Rod drop tests	Verify drop times, snubber performance	X	X	
12. Gas bubble injection	Determine gas paths, detectability		X	Gas injection capability

6.4.2 Fuel Loading Test

The control rods and special flux detectors will be installed and fuel will be loaded to achieve the operational core loading. The wet critical loading will be checked. The excess reactivity of the operational core loading will be determined.

6.4.3 Low-Power Tests

With the core loaded, core pressure drop will be measured as a function of flow. Control rods will be calibrated and "zero power" reactivity coefficients will be measured. These will include the isothermal temperature coefficients, sodium flow coefficients, cover gas pressure coefficients, and sodium void coefficients. Flux and spectrum measurements will be made to determine the radiation level at the grid plate and vessel. The low-power test program is summarized in Table 6.4-2.

6.4.4 Rise-to-Power Tests

During this phase, the plant power level will be increased in steps. At each step, selected tests will be performed. They will be chosen, as appropriate, from those in Table 6.4-3. The schedule of power and temperature increases will be approximately as follows:

<u>Power Level</u>	<u>Reactor T</u>	<u>Heat Sink</u>
5%	15°F	Steam Dump
10%	35°F	Steam dump
15%	50°F	Steam Dump & T-G
15%	110°F	T-G
30%	220°F	T-G
60%	360°F	T-G
100%	360°F	T-G

T-G = Turbine-Generator

TABLE 6.4-2
LOW POWER TESTS

Test	Purpose	Results Used for		Special Equipment Requirements
		Operation	SAR Verif.	
1. Control rod calibration	Check calculations and acceptability of rod worths, reactivity rates and shutdown margins	X	X	
2. System hydraulics	Determine acceptability, reference point	X	X	
3. Isothermal temperature coefficient	Check calculations, verify acceptability	X	X	
4. Flow coefficient of reactivity	Check for flow-induced reactivity changes	X		
5. Pressure coefficient of reactivity	Check for gas in core	X	X	
6. Void coefficients for small voids	Check calculation of central void worth		X	Special assembly
7. Axial fission distribution	Verify that peak-to-average is acceptable			Special assembly(ies), special cask, glove box
8. Axial/radial flux and spectra meas. to grid plate and vessel	Check calculations and verify acceptability			Special assembly(ies), special cask, glove box
9. Transfer function	Verify delayed-neutron fraction	X	X	Reactivity oscillator
10. Neutron noise analysis	Measure neutron lifetime		X	In-core neutron detector

TABLE 6.4-3
RISE-TO-POWER TESTS*

Test	Purpose	Results Used for		Special Equipment Requirements
		Operation	SAR Verif.	
1. Power and flow coefficients and time constants	Check calculations, verify acceptability	X	X	
2. Core temperature distribution	Determine radial power distribution and specific power, verify orificing, coolant distribution and convection cooling	X	X	
3. Temperatures at shield, vessel, grid plate, etc.	Verify acceptability, post DBA cooling, etc.	X	X	
4. Vessel, top shield and grid plate radiation level meas.	Verify acceptability and streaming through top shield	X	X	Removable tabs
5. Control rod calibration	Measure change in worth with power/temperature	X	X	
6. Core stability (transfer function)	Verify stability, determine individual coefficients and time constants	X	X	Hi-temp reactivity oscillator
7. System hydraulics and heat transfer	Verify that required temperatures, flows, pressure differentials, etc., are met at all loads, perform plant heat balance	X	X	Calorimeter (steam)
8. Coolant chemistry	Verify acceptability			
9. Critical component stress levels	Verify acceptability	X	X	Hi-temp strain gauges installed
10. Steam generator stability	Verify acceptability at low loads	X	X	
11. Control system performance	Adjust parameters, verify performance under transients	X		
12. Off-normal transients (turbine trip, one-loop shutdown)	Verify control and protective system performance and system transients	X	X	
13. Reactor trip tests	Verify transients, check and adjust convective flow control system, check control rod drop time at high temperatures	X	X	
14. Vault temperature and radiation	Verify acceptability	X	X	Radiation monitors in vaults
15. Plant radiation levels	Verify acceptability	X	X	
16. Irradiated fuel handling	Verify system performance	X	X	
17. One hundred hour operation	Final plant acceptance test			

*These tests will be performed at several power levels and flowrates but not all tests at each level.

It is expected that the plant power would be leveled about every 5-10% to check the status of all equipment and ascertain plant normalcy before proceeding. The final test will be a one-hundred hour plant acceptance run prior to turning the plant over to the utility operator.

REV
LTR A

PROGRAM TITLE
ANL 1000 Mwe Follow-on Study

DOCUMENT TITLE	Reactor Segment Conceptual System Design Description - 1000 Mwe LMFBR Follow-on Study
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This conceptual SDD presents the design requirements for, and a brief description of, The Reactor Segment for the 1000 Mwe IMFBR Follow-on Study

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REACTOR SEGMENT, CONCEPTUAL

1. INTRODUCTION

This SDD covers the core (including fuel, blanket, reflector and control elements), the reactor vessel and top shield, and the primary containment barrier up to and including the reactor cavity liner, on a segment level.

1.1 Segment Functions

1.1.1 Principal Performance Function

The principal performance function of the components and systems comprising the reactor segment is to generate the energy (heat) to be supplied to the systems of the heat transfer segment at a rate demanded by plant load.

1.1.2 Other Performance Functions

Other performance functions of the reactor segment are the following:

- 1) Achieve a net gain (breeding) of fissionable material.
- 2) Provide for fuel transfer and remote handling of in-core components.
- 3) Provide for containment and distribution of coolant to transfer the energy from the heat source to the heat transfer systems.
- 4) Provide mechanical support for the reactor element transfer systems.
- 5) Protect the public from the results of any credible malfunction and a Design Basis Accident (DBA).

1.2 Summary Description of the Segment

1.2.1 Reactor System

The reactor system shown by Figures 1, 2, and 3; is the primary heat source for the plant and generates 2400 Mw thermal at nominal rating. It has a core of mixed plutonium and uranium oxide fuel surrounded by a depleted UO_2 breeding blanket, all suitably contained and subdivided to permit adequate heat removal by the circulating liquid sodium coolant. The core operates in a fast neutron flux, and power is regulated by moving neutron absorbing control elements into and out of the core. The core, blanket, reflectors, etc., are housed in a stainless steel reactor vessel which provides containment for the

sodium coolant and mechanical support for the core and associated components. The upper end of the vessel is closed by a double-rotating shield plug which functions as a part of the fuel handling system. At nominal design conditions, sodium enters the vessel near the bottom at 780°F , flows upward through the core and exits from the vessel outlet plenum at 1140°F .

1.2.2 Reactivity Control

Reactivity control is maintained by two banks of control rods, an inner bank of safety rods and an outer bank of shim-safety rods. The shim-safety rods cover all normal operational requirements and can shutdown the reactor from any operating condition. Safety rods are used only to control operating anomalies outside of those expected during normal operation. The maximum worth of a shim-safety rod is about \$1.35. Actuation of the rods is by electric powered gear drives. All rods can be tripped from any position by de-energizing a magnetic holding latch on each rod.

1.2.3 Fuel Handling

Reactor downtime resulting from fuel handling is minimized by refueling on a one year cycle with one-half of the core, one-fifth of the inner blanket and one-seventh of the outer blanket elements being replaced each cycle. A like number of blanket elements will be rotated 180° to even the plutonium buildup associated with a steep flux gradient through the blanket.

1.2.4 Containment

A series of atmospheric and radiological boundaries are established to confine the reactor fluids and radioactive materials. The various boundaries are defined on Figure 4.

1.2.4.1 Primary Coolant Boundary

The primary coolant boundary, within the reactor cavity, consists of the reactor vessel and the sodium piping within the cavity. The reactor vessel is backed up by an independent secondary jacket. This is an external open topped tank around the reactor vessel and sodium piping. The tank extends to a level 5 feet above the top of the core and blanket elements. In the event

of a breach of the reactor vessel this containment tank provides the capability for keeping the core covered which permits continued circulation of coolant through the primary system and core.

1.2.4.2 Cover Gas Boundary

The cover gas, which will contain sodium vapor plus any fission gases that may have leaked from the fuel boundary, is contained by the upper portion of the reactor vessel, the gas supply and return lines within the reactor cavity, and the reactor top shield. During fuel transfer operations, the fuel handling machines, a part of the reactor support equipment (SDD-096-80-000) become an extension of this boundary.

1.2.4.3 Accident Containment

The overall plant containment objective is achieved by providing two accident containment barriers. The primary containment boundary is the reactor cavity liner and the liner of the primary equipment vaults, plus a containment dome over the top of the reactor. The second and final containment barrier is the containment building.

Post accident leakage from the primary containment barrier to the containment building will not exceed 10% of the combined cavity and gallery volume per day. Post accident melt-through of the containment is prevented by providing a 2-inch thick carbon steel cavity bottom which is cooled on the under side.

1.3 Requirements

1.3.1 System Performance Requirements

1.3.1.1 Steady State Rated Power

The reactor shall generate 2400 Mw of thermal power under the following criteria:

There shall be a high probability that the amount of fission products released from the fuel pins will be less than the coolant cleanup capacity.

1.3.1.2 Fuel Burnup

The reactor fuel shall be capable of a batch average discharge burn-up of 67.36 Mwd/kg(U+Pu), (peak burnup roughly estimated to be 115 Mwd/kg). A sealed plenum for fission product gas shall be provided in the upper extension of the cladding commensurate with the fuel volume and burnup.

1.3.1.3 Power Coefficient

The overall and local power coefficient shall be negative over the full range of normal operation.

1.3.1.4 Operating Reactivity Control (Shim-Safeties)

The reactivity control requirement for all normal reactor operation shall include the following parameters:

- a) Fuel burnup
- b) Doppler
- c) Sodium density and expansion
- d) Ta burnup (shim-safeties only)
- e) Shutdown margin*
- f) Control margin
- g) Ta worth uncertainty
- h) Contingency for analytical uncertainties

The worth of any element of the operating reactivity control system shall not exceed (to be established - approximately \$1.35).

1.3.1.5 Safety Reactivity Control

The reactivity control philosophy is based upon the following AEC criteria.

1) Two independent systems shall be capable of shutting the reactor down and holding it subcritical from any hot operating or hot standby condition including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

2) One of these two systems shall have a shutdown margin greater than the highest worth rod, and

3) One system shall be capable of shutting the reactor down under any condition with appropriate margin for contingencies.

The two independent systems are (1) the delatching of all rods, and (2) driving of all rods into the core. For this plant both systems have the same reactivity worth. The shim-safety rods are those that are in the core during reactor operation (see 1.3.1.4), the safety rods are those that are out of the core during normal operation. To meet the above criteria the following

* To be determined.

items must be considered in determining the reactivity worth of both the shim-safeties and the safeties:

- a) Fuel burnup
- b) Doppler
- c) Sodium density and expansion
- d) Ta burnup
- e) Shutdown margin (1%K)
- f) Maximum sodium void worth
- g) Control margin
- h) Tantalum worth uncertainty
- i) Contingency for analytical uncertainties

The worth of the safety rods (those out of the core during normal operation) is found by the difference between the above list and the requirements of 1.3.1.4.

1.3.1.6 Refueling Interval

The normal refueling interval for the reactor shall be one year. This is not to preclude refueling at any desired interval (less than one year) that operating considerations dictate.

1.3.2 Environmental Requirements

1.3.2.1 Operating Sequence of Events

The system is exposed to the radiation, thermal, mechanical and pressure environment produced during plant preheat, startup, operation (including load swings), refueling and maintenance shutdown and restarts throughout its design life of 30 years. In addition various abnormal events, protected and unprotected, may occur during this period which will severely alter the normal environments. These events shall be carefully analyzed and their effects on system performance and design established with due allowance for design margin to meet the performance, reliability and safety requirements.

1.3.2.2 Number of Cycles

The system shall be designed on the basis of the following operational cycles over a 30-year lifetime:

Preheat	3
Startup	3
Load Swings (10% load change)	9,000
Refueling shutdown	30
Maintenance shutdown	30

1.3.2.3 Loads**1) Seismic Load****a) Design Earthquake (DEQ)**

The system, including the cavity liner and the biological shield shall be designed to withstand, without impairment of functions, a horizontal ground acceleration of 0.33g, plus a vertical ground acceleration of 0.22g in combination with the system operational loads.

b) Maximum Hypothetical Earthquake (MHEQ)

The system, including the cavity liner, shall be designed to withstand without gross failure which could impair the capability of primary containment and achievement of a safe plant shutdown, a horizontal ground acceleration of 0.5g plus a vertical ground acceleration of 0.33g in combination with the operational loads.

2) Design Basis Accident Loads

The system shall be designed to contain the energy and products released during a Design Basis Accident (DBA) of 13,000 Mw-sec*, with due consideration for volume allowance of the primary galleries.

1.3.3 Reliability/Availability Requirements

The unavailability objective to meet the design target of 89% mature plant availability shall be as follows:

	<u>Forced</u>	<u>Scheduled</u>	<u>Total</u>
Vessel Assembly			
Reactivity Controls			
Top Shield			To be Determined
Fuel and Blanket			
Fixed Reactor Shielding			
Reactor Segment Items			
Not Included in above.			
Reactor Segment - Total			

* Based on Task I reference design.

1.3.4 Maintenance Requirements

1.3.4.1 Maintenance Capability

Capability for equipment maintenance shall be incorporated into the design of the reactor system to the extent specified in paragraphs 1.3.4.4. through 1.3.4.7.

1.3.4.2 Inspection and Quality Control

The reactor system shall be fabricated and erected with sufficient inspection and quality control to assure a high probability of continuous plant operation for the specified lifetime.

1.3.4.3 Proof Test

The reactor system shall incorporate capabilities for the pre-operational test program. The design must interface with the test program as it is formulated to assure the proper functioning of all components prior to fuel loading.

1.3.4.4 Routine Maintenance

Routine maintenance operations shall be done with the reactor shutdown and shall include a regular inspection program of all items located above the top shield and sealed from the primary system. Included are replacement of thermocouples, drive units, limit switches and electrical leads.

1.3.4.5 Special Maintenance

Special maintenance operations shall involve items requiring interruption of the shield and/or seals of the primary system. They shall be done with the reactor shut down and include replacement of seals and control rod drive extensions.

1.3.4.6 Major Equipment Maintenance

The reactor system shall be maintainable to the extent of being able to repair or replace at least the following components with a shutdown of less than one year.

- 1) Top Shield Bearings
- 2) Core Radial Constraint
- 3) Core Grid Plate
- 4) A partial Core Meltdown up to a Seven-Element Cluster

1.3.4.7 Reactor Vessel Inspection

A design objective shall be to incorporate features which allow remote equipment access to the reactor vessel external surfaces for inspection by visual or other non-destructive methods.

1.3.5 Operational Requirements

1.3.5.1 Normal Power Operation

The reactor system shall be capable of generating thermal energy at a rate of 2400 Mw and shall have the capability for following load automatically between 25% and 100% rated power. The reactor system shall also be capable of continuous base loaded operation up to 100% rated power.

1.3.5.2 Abnormal Shutdown

The reactor system shall accommodate load rejection of any type including turbine trip from full power.

1.3.6 Safety Requirements

The reactor system design shall satisfy the intent of the AEC 10CFR, Part 50, "General Design Criteria for Nuclear Power Plant Construction Permits".

1.3.6.1 Specific Safety Requirements

- 1) The reactor design shall minimize the possibility of fuel element-to-element failure propagation.
- 2) A negative power coefficient (prompt and total) shall exist over all regions of the core in which significant damage could occur due to power oscillations.
- 3) Gas entrainment shall be limited to prevent significant cladding damage and a positive reactivity insertion greater than the integrated Doppler effect from normal operation to the onset of fuel damage.
- 4) The shim-safety rods shall be sufficiently reliable, isolated, redundant and rapid to prevent significant damage to the core from all credible accidents.
- 5) Sufficient instrumentation shall be provided to assure early detection of all system malfunctions that could lead to significant core damage.

6) All credible accident perturbations shall not exceed the integrated Doppler effect from operating conditions to the onset of fuel damage.

7) No credible accident within a radiological boundary or involving such a boundary shall be allowed to propagate through an adjacent radiological boundary. (See Figure 4 for definition of boundaries).

8) During normal operation, the primary and secondary containment boundaries shall not be violated.

9) The primary and secondary containment boundaries shall be designed to limit the consequences of hypothetical accidents to the recommended limits set forth in 10CFR, Part 100.

10) All engineered safeguards required for plant safety shall have demonstrated performance capability, reliability, and where practicable shall be inspectable and testable.

11) The failure of any system or component shall not generate missiles which could result in a violation of any containment boundary.

1.3.7 Design Requirements

1.3.7.1 Applicable Codes and Standards

- 1) ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels
- 2) ASME Interpretations of ASME Boiler and Pressure Vessel Code, Case 1331-3.
- 3) ASME American Standard Code for Pressure Piping, USAS B31.7, Nuclear Power Piping.

1.3.7.2 Materials

1.3.7.2.1 Structural Components in Sodium

Structural components normally in contact with sodium or sodium vapor shall be fabricated of type 304 stainless steel except where other materials are required for special purposes, such as for wear surfaces. When other materials are used, due consideration shall be given to material transfer and corrosion characteristics.

1.3.7.2.2 Structural Components Not in Sodium

Selection of materials for structural components not normally in contact with sodium or sodium vapor shall be made on the basis of compatibility, adequacy and economics.

1.3.7.2.3 Moderating Materials

Graphite and hydrogenous materials shall not be permitted in contact with the primary coolant and cover gas.

1.3.7.3 Specific Design Requirements

1.3.7.3.1 Core Radial Constraint

The core element array shall be constrained radially, sufficiently to prevent an unacceptable positive contribution to the power coefficient as a result of any core element displacement including thermal bowing of the element housings.

1.3.7.3.2 Core Support Grid

The core element array shall be supported on a grid plate structure, having its lattice spacing within a tolerance compatible with the nuclear and safety requirements.

1.3.7.3.3 High Pressure Plenum

Inlet coolant shall be distributed to the fuel and radial blanket elements from a high pressure plenum designed for normal operation at 800°F and 150 psi for 30 years.

1.3.7.3.4 Core Element Holddown

Holddown of core elements subjected to the high pressure coolant shall be of the hydraulic "piston" type. Holddown of all other core elements shall be by gravity.

1.3.7.3.5 High Pressure Plenum Leakage

Leakage of coolant from the high pressure plenum through the upper grid plate shall not exceed 2% of the total flow.

1.3.7.3.6 Grid Plate Structure Neutron Exposure

The design of the grid plate structure shall be based on an integrated fast neutron ($E > 1 \text{ mev}$) exposure of 5×10^{22} nvt accumulated over a period of 30 years.

1.3.7.3.7 Coolant Distribution Through Core Elements

Coolant flow from the high pressure plenum shall be controlled by fixed orifices in the fuel and blanket element nozzles. Orifices shall be sized in rings according to fuel enrichment and radial lattice position.

1.3.7.3.8 In-Vessel Neutron Shield

Neutron shielding shall be provided between the reactor core and the fuel storage region so that the internal heat generation rate in the reactor cavity structure shall not exceed 150 Kw.

1.3.7.3.9 In-Vessel Core Element Transfer

Positions shall be provided in the vessel for 15 transfer canisters. The spacing and general arrangement of the transfer positions shall be compatible with fuel handling equipment requirements. In addition the transfer positions shall have the capability of storing canisters containing core elements during reactor operation without excessive temperatures being experienced.

1.3.7.3.10 Removal of Fuel Element Cluster

Capability shall be provided to allow removal of a 7-element cluster directly from the core through the top shield.

1.3.7.3.11 Reactor Vessel Operating Conditions

The reactor vessel including all internals shall be designed to withstand the following conditions without failure:

(1) Thirty years operation under all pressure, mechanical and thermal loads arising from all normal steady state and transient operating conditions.

(2) Overpressurization of the reactor cover gas to 10 psig.

(3) Reactor sodium fill.

(4) Overpressure resulting from breakage of the vent line of the IHX. This results in the addition of the elevation head of the sodium in the IHX, approximately _____ feet above the operating level in the reactor to the normal pressure in the reactor.

1.3.7.3.12 Reactor System Allowable Stresses

All reactor components except fuel elements shall be designed to the following allowable stress categories for 304 stainless steel. The fuel pin cladding and fuel element housing design allowables for 316 stainless steel are given in SDD-096-28-001.

(1) Normal Operation

Allowable stresses, taking into account stress to rupture at 250,000 hours, stress to produce 1% creep at 250,000 hours, and the effect of irradiation to 10^{23} nvt, are as follows:

<u>Temperature</u>	<u>Allowable Stress</u>
800°F	9000 psi
900	8800
1000	7200
1100	3900
1200	2400

(2) Short Term Mechanical Loads

Allowable stresses as given in Section VIII of the ASME Code.

1.3.7.3.13 Pressure Test

The reactor system coolant and cover gas boundaries (see Figure 4) shall be subjected to an integrated leak test. The total leak rate shall not exceed *% per day at * psi differential.

1.3.7.3.14 Reactor Vessel Closure

The upper closure of the reactor vessel (top shield) shall normally function as a radiation shield and cover gas seal and shall be a component of the reactor shielding.

1.3.7.3.15 Top Shield Shielding Criteria

The radiation level above the operating floor shall not exceed 1.25 mr/hr over the bulk shielding. Streaming through penetration annuli shall not exceed 25 mr/hr.

1.3.7.3.16 Top Shield Fuel Handling Functions

The fuel handling requirements for the top shield are given in SDD-096-81-000.

1.3.7.3.17 Reactor Cover Gas Leakage

The leakage of reactor cover gas through the primary containment boundary (See Figure 4), during normal reactor operation, including fuel handling, shall not exceed 0.5 SCF per day.

* To be determined

1.3.7.3.18 DBA Leakage

The top shield and containment dome shall be designed to minimize leakage after a DBA. Post accident leakage from the entire primary containment barrier shall not exceed 10% of the combined cavity and equipment gallery volume per day.

1.3.7.3.19 Top Shield Holddown

The top shield and the plugs in all penetrations through it shall be held down to prevent the ejection of missiles as a result of a DBA.

1.3.7.3.20 Secondary Containment Tank

The primary coolant boundary shall be backed up by an independent secondary containment tank that maintains the sodium level in the reactor sufficiently high, in the event of a breach of the primary vessel, to keep the core covered and to provide the capability for continued circulation of coolant.

1.3.7.3.21 Reactor Outlet Nozzles

The elevation of the inlets to the reactor outlet nozzles shall be as low as is required to provide continued circulation through the reactor in the event of a primary coolant boundary leak and loss of sodium into the containment tank.

1.3.7.3.22 Containment Tank Design

The secondary containment tank shall have no penetrations and shall be supported completely independent of the reactor vessel.

1.3.7.3.23 Containment Tank Pump-Out

Provision shall be made for pumping sodium out of the containment tank.

1.3.7.3.24 Reactor Insulation

The reactor and its contiguous piping shall be insulated to limit the sensible heat loss to the reactor cavity to * Kwt.

1.3.7.3.25 Reactor Cavity Operating Conditions

The reactor cavity shall be part of the primary containment barrier and shall be designed to withstand the following conditions without failure.

* To be determined

(1) Thirty years operation under all pressure, mechanical and thermal loads arising from all steady and transient operating conditions.

(2) Leak and pressure testing to 20 psig.

In conjunction with the post DBA cooling system, it shall contain the results of a DBA including the effects of missiles and molten fuel.

1.3.7.3.26 Reactor Cavity Blast Protection

The reactor cavity shall be adequately protected from blast loading and missiles resulting from a DBA.

1.3.7.3.27 Reactor Cavity Cooling

The reactor cavity heat removal system for normal operation shall have a capacity of at least * Kw.

1.3.7.3.28 Reactor Cavity Meltdown Pan

The reactor cavity bottom head shall be flat to prevent reassembly of molten fuel into a critical mass.

1.3.7.3.29 Post-DBA Heat Removal

Post-accident heat removal capability of at least * Mw shall be provided by a cooling system closely coupled to the reactor cavity bottom head. The reactor cavity bottom head shall be designed to prevent fuel melt-through.

1.3.7.3.30 Reactor Coolant Gas Pockets

Provision shall be made to eliminate the collection of gas in the high pressure plenum. All gas pockets throughout the reactor shall be eliminated by careful design.

1.3.7.3.31 Reactor Coolant Gas Entrainment

Entrainment of gas in the reactor coolant shall be minimized by immersing the lower part of the reactor top shield in the sodium pool to produce a very small sodium/cover gas interface and to prevent vortexing in the outlet plenum.

1.3.7.3.32 Reactor Alignment

The vertical alignment of the core and smaller rotating shield plug shall be based solely on reactivity control element alignment requirements. In-core fuel handling alignment shall be optimized by calibration of the double rotating plug indexing system.

* To be determined

1.3.7.3.33 Core Element Bowing Constraint

Thermal bowing of core elements shall be partially constrained in a manner and degree consistent with limits on reactivity changes, misalignment of control rod drives, creep distortion of the core element housing, and fast neutron induced metal swelling.

1.3.7.3.34 Core Instrumentation

Access and mechanical support, consistent with other system requirements, shall be provided for the core instrumentation specified in SDD-096-50-000.

1.3.7.3.35 Leak Detection

Means shall be provided for detecting significant sodium leaks from the primary vessel and all primary piping below the elevation of the top of the secondary jacket vessel.

1.3.7.3.36 Reactor Preheat

Provision shall be made for heating the dry reactor vessel and the contiguous piping to at least 350°F at a rate of about 3°F per hour.

1.3.7.3.37 Removable Reactor Internal Structures

Provision shall be made in the design for removal and replacement of the reactor grid plate and/or core constraining device through the opening for the smaller rotating shield plug in a shutdown period not to exceed one year.

1.3.7.3.38 Sodium Frost Buildup

Provision shall be made in the design for minimizing sodium frost buildup in the annuli surrounding the rotating plugs.

1.3.7.3.39 Fuel Element Management

Means shall be provided by design to prevent inadvertent placement, in a particular lattice position, of fuel elements with other than the correct enrichment and orificing.

1.3.7.3.40 Material Surveillance

Provision shall be made for placing specimens of reactor structural materials at various locations within the reactor to subject these specimens to the reactor environment including sodium, temperature, and radiation for surveillance of their mechanical and physical properties over the life of the plant.

1.3.7.3.41 Control Rod Drive Interference

The reactor system arrangement shall preclude any interference between control rod drives and fuel handling machines.

1.3.7.3.42 Reactivity Control Elements

Space shall be provided within the core element array to accommodate the required reactivity control elements.

1.3.7.3.43 Top Shield Penetration Plug Holddown Lock

All normally removable top shield penetration plugs shall be equipped with a mechanical locking device that automatically engages and remains locked unless and until specifically actuated.

1.3.7.3.44 Thermo-Convection Coolant Supply

A separate coolant feed into the reactor vessel shall be provided for the thermo-convection cooling system. This system provides for removal of heat from the control rods, reflector elements, neutron shielding, and the reactor vessel and its internal structure.

1.3.7.3.45 Structure Temperature Monitors

Thermocouples shall be provided as required to adequately monitor the temperature gradients in the reactor and top shield structures.

1.3.8 Interface Requirements**1.3.8.1 Segment Interfaces**

This system segment interfaces with other system segments as follows:

- 1) Heat Transfer Segment (SDD-096-30-000)
 - a) Reactor sodium supply and return piping interfaces and interactions.
- 2) Control and Instrumentation Segment (SDD-096-50-000)
 - a) Reactor control rod command and monitoring.
 - b) Reactor nuclear instrumentation.
 - c) Reactor sodium temperature instrumentation.
 - d) Reactor component temperature instrumentation.
 - e) Reactor sodium level instrumentation.
- 3) Auxiliary Systems Segment (SDD-096-60-000)
 - a) Reactor sodium fill and drain system interfaces and interactions.

- b) Reactor cover gas system interfaces and interactions.
- c) Cavity liner cooling system.
- d) Reactor preheat system.
- 4) Reactor Support Systems Segment (SDD-096-80-000)
 - a) In-vessel fuel handling.
 - b) Ex-vessel fuel handling and storage.
 - c) Handling and storage of control rod and miscellaneous core components.

2. DETAILED DESCRIPTION

2.1 Reactor

2.1.1 General Characteristics

The reactor is the primary heat source for the plant, generating 2400 Mw of thermal power at nominal rating. It consists of a core and breeding blanket of mixed plutonium and uranium oxide fuel, suitably contained and subdivided to permit adequate heat removal by the circulating liquid sodium coolant. The core operates in a fast neutron flux, and power is regulated by moving neutron absorbing control elements into or out of the core. The arrangement is shown in Figure I. The core and blanket are housed in a stainless steel reactor vessel, which provides containment for the sodium coolant. The open upper end of the vessel is closed by a thick, double-rotating plug shield which also functions as part of the fuel handling system.

Sodium enters the vessel at the lower end at 780°F, flows up through the core, and exits from the vessel at 1140°F. The reactor outlet temperature can be increased slightly for short durations to utilize the inherent design margins in the turbine generator.

Secondary containment of primary sodium is provided by the jacket vessel. The jacket vessel provides a void volume around the reactor vessel which is less than that of the volume of sodium in the reactor vessel above the inlets to the outlet nozzles, thus precluding uncovering the core if there should be a sodium leak in the reactor vessel. A gas containment dome over the top shield provides the primary containment barrier against any possible fission gas release from the reactor following a DBA.

The major reactor performance characteristics are tabulated in Table 1.

TABLE 1
Mid-Equilibrium Cycle
Reactor Performance Characteristics

Thermal-Hydraulic Performance

Total reactor thermal power, Mwt (nominal)	2400
Core power fraction	0.906*
Radial blanket power fraction	0.048*
Axial blanket power fraction	0.046*
Total reactor flow rate (10^6 lb/hr)	75.6
Core flow rate (10^6 lb/hr)	69.8
Radial blanket flow rate (10^6 lb/hr)	5.8
Number of coolant circuits	3
Peak linear rod power, kw/ft	16.03
Peaking factors	
axial	1.23
radial	1.22
circumferential (7th row)	1.05
elemental (7th row)	1.08
Reactor mix-mean temperature rise, °F	360
Element outlet temperature, °F	1151
Maximum nominal clad inside temperature	1221
Average coolant velocity highest power element ft/sec	27.5
Core pressure drop, psi	87

*The cost iteration between thermal hydraulic analysis and nuclear analysis ended with new power fractions that aren't significantly different.

The reactor core and blanket consists of an assembly of several types of hexagonal elements arranged as shown in Figures 1 and 2. These elements are supported by a structure consisting of two parallel grid plates 30 inches apart, in holes arranged on a triangular lattice spacing of 5.667 inches (cold). The region between these plates serves as the inlet plenum for distribution of sodium coolant to the fuel and blanket elements. The reactor core and blanket assembly include fuel elements with axial blanket sections on

each end, radial blanket elements, reactivity control elements, and reflectors as shown in Figure 2 and tabulated below:

TABLE II
Reactor Elements

<u>Type</u>	<u>Number</u>
Inner Fast Fuel Region (Zone 1)	133
Outer Fast Fuel Region (Zone 2)	141
Inner Radial Blanket	66
Outer Radial Blanket	72
Shim-Safety Rods	9
Safety Rods	6
Reflector	<u>78</u>
Total	501

The fast fuel or core region is 42.8 inches high with a 12-inch upper blanket and a 15-inch lower blanket for a total of 69.8 inches. The inner radial blanket is 56.3 inches and the outer radial blanket is 42.8 inches. (All dimensions above are "cold"). The core region gas plenum extensions are the longest, 38 inches, resulting in fuel pins approximately 108 inches long. The total length of the fuel and blanket elements, including end hardware is 180 inches.

2.1.2 Physics and Nuclear Characteristics

The following table gives the physics and nuclear characteristics of the reactor core and blanket.

**PHYSICS AND NUCLEAR
CHARACTERISTICS**

**SUMMARY OF REACTOR CHARACTERISTICS AT THE MIDCYCLE
EQUILIBRIUM CONDITIONS FOR THE FINAL REFERENCE DESIGN**

Fuel Pin Diameter OD Clad, inches	.3053
Core Height, cm	109.88
Equivalent Core Diameter, cm	259.31
Ratio of Height to Diameter	.424
Number of Fuel Elements	274
Number of Control Subassemblies	15
Lattice Pitch, cm	14.526
Median Energies in Core Regions	
Fission Energy kev	158.99
Absorption Energy kev	35.81
Flux Energy kev	119.45
Total Reactor Breeding Ratio	1.271
In-Core Breeding Ratio	.944
Doppler Constant $T \, dk/dt$	-0.00795
Maximum Radial Sodium Void, %	8.37
Total Sodium Void, %	8.07
Total Fissile Mass, kg	2744
Core Regions	2191
Radial Blankets	377
Axial Blankets	176

Region	Core Region		Radial Blkts		Axial Blankets
	Inner	Outer	Inner	Outer	
Power (Mw)	1172.1	1029.6	86.16	36.95	75.12
Avg. Enrichment atom %	10.5	13.14	3.09	2.43	1.33
Avg. Discharge Burnup Mwd/MT	67360.	67360.	15010.	10340.	3340
Cell Volume Fractions					
Fuel	.439	.439	.556	.584	.439
Sodium	.326	.326	.281	.269	.326
Stainless Steel	.235	.235	.163	.147	.235
Isotopic Mass kg					
U-235	11	13	12	11	23
U-238	7582	7706	8115	7094	12938
Pu-239	881	1167	187	167	153
Pu-240	361	504	10	8	4
Pu-241	47	73	0	0	0
Pu-242	25	38	0	0	0
Total Heavies kg	8907.	9500	8324	7280	13118
Total Heavies + F kg	9260	9824	8377	7311	13132
<u>Calculated Rod Worth</u>	<u>Reactivity Worth (\$)</u>		<u>Number of Rods</u>	<u>Maximum Rod Worth (\$)</u>	
Total control system (delatched or driven)	24.87		15	2.39	
Shim-safety bank	10.53		9	1.23	
Safety bank	14.34		6	2.39	

2.1.3 Core Element Design

The criteria for the design of the core are:

1. No fission product release greater than coolant cleanup capacity during normal and anticipated abnormal reactor operation.
2. No adverse insertion of reactivity into the core because of fuel movement during normal and anticipated abnormal reactor operation.
3. The peak-to-average power ratios and hot channel factors give a nominal fuel pin rating of 16.03 kw/ft. This rating meets the above criteria. A summary of the core element design data is given in the SDD-096-24-000.

2.1.4 Control Rod Design

The reactivity control system comprises two banks of control rods. One bank is designated shim-safety rods and the other safety rods. All rods in both banks can be scrammed from any position.

The shim-safety bank is used for all normal operating control requirements and is capable of mitigating or terminating all normal operating anomalies. It consists of nine rods located in a nearly circular ring at the interface of the two fast fuel regions as shown in Figure 2.

The safety bank will be used to cover anomalies outside those expected during normal operation, such as a stuck shim-safety rod. The six safety rods are positioned in a ring in the inner fuel zone of the core.

Each control rod neutron absorber consists of a cluster of stainless steel clad tantalum pins that is moved vertically within its circular guide tube.

During reactor operation, the safety rods are held with the lower end of the absorber at the top of the upper axial blanket.

All control rods are cooled by sodium from the low pressure region of the reactor vessel using thermal convection circulation. There are no control rod followers. Details of control rod design are present in SDD-096-22-000.

2.2 Fuel Elements

2.2.1 General Description

The fast fuel element is described in detail in SDD-096-28-000. The conceptual design is shown in Figure 5.

The fuel element enclosure consists of a hexagonal housing fitted with appropriate end hardware. It encloses and supports the fuel pin bundle, provides it with metered coolant flow for the necessary heat removal. The fuel pin bundle consists of an array of 217 cladding tubes which enclose mixed oxide fuel pellets and contain the fission gases.

2.2.2 Interface Details

The fuel elements are supported vertically on the lower grid plate and are constrained laterally at the upper grid plate and above the core centerline. Details concerning the lateral displacement of the fuel elements resulting from thermal expansion and bowing are discussed in Section 2.6, "Core Constraint".

The lower end hardware of the fuel element performs the following functions:

- a. Positions and supports the element vertically and laterally.
- b. Provides the required amount of coolant flow.
- c. Provides hydraulic holddown to counteract the drag uplift due to coolant flow through the element.
- d. Provides mechanical lead-in guidance for insertion of elements into the core lattice.
- e. Provides a positive mechanical go/no-go system to insure that only a correct element is installed in a given lattice position.
- f. Provides blades that turn the element to properly orient the hexagonal housing as it enters the reactor matrix.

The upper end hardware performs the following functions:

- a. Provides a pickup ring for the fuel handling machine gripper.
- b. Provides for centering the in-vessel fuel handling machine guide tube.
- c. Presents a bearing surface to the fuel handling machine guide tube to prevent lifting the element when an adjacent fuel element is raised.
- d. Presents camming surfaces to the blades at the lower end of the housing, to orient the fuel element with the hexagonal matrix.
- e. Provides for minimum disturbance of the outlet coolant stream to obtain maximum thermocouple effectiveness.

The hexagonal housings relate to each other in the following ways:

- a. Provide a clearance between adjacent elements to make it possible to assemble and disassemble the reactor matrix.
- b. Provide a spacer pad on the element housings to make it possible to achieve a "tight" core.

2.2.2.1 Coolant Flow Distribution

The required coolant flow to each element is provided by orificing its lower end hardware. The calculation of flow distribution is based on attaining uniform mixed-mean element outlet temperature by matching the individual element flow rate to its power. This, of course, can only be ideally achieved. Practically, each radial ring of the core and blanket matrix has an individual orifice calibration. The lower extension has orifices, in the form of side openings, that meter the flow.

2.2.2.2 Upper Grid Plate Leakage

The element lower extensions penetrate the upper grid plate through holes having a maximum radial clearance of 0.005 inch. The resulting coolant leakage from the high pressure plenum is calculated as 0.62 lb per sec. per penetration.* There are 412 penetrations through the upper grid plate from the high pressure plenum so that the leakage will be about 257 lb per sec. or 1.22 percent of the total primary system sodium flow. This leakage will be distributed through the nominal 0.050 in. gap between the fuel element housings.

2.2.2.3 Core Matrix Assembly

The normal fuel handling procedure is to remove only one element from the reactor matrix and to replace it with a new element before removing another. To simplify the fuel handling machine requirements, the entering fuel element is free to rotate about its axis and align its hexagonal housing with the hexagonal opening in the reactor matrix. Interaction of the blade at the lower end of the hexagonal housing of the element being loaded into the core with the camming surface at the upper end of the adjacent element causes the element to rotate about its axis and so align itself with the hexagonal opening in the reactor matrix as it is lowered with the fuel handling machine. Once oriented, there is a clearance until the element is within about an inch of being seated. At this point the spacer pads come together and hold the element snugly in the reactor matrix.

The control rod guide tubes, being shorter, have no camming surface. If the blade of a fuel element should happen to be over the control rod guide tube no orientation could occur. A second blade is provided to avoid this situation. The elevation of this blade is a little different to prevent locking on dead center.

*Values for Task I design.

The spacer pads are located on the corners of the hexagonal housing to give high rigidity to the assembled reactor matrix. They are hard-faced weld overlays, compatible with the environment, and are held to a tolerance of ± 1 mil across the housing flats.

There are different enrichment and orificing combinations of high-flow elements which must be managed. Installation in the correct position is guaranteed by employing a mechanical go/no-go system. The lower end of each different type of fuel element is of different O.D. and has a different size key. These are so arranged that only the correct type of element can seat in the grid plate hole of a particular zone. The fuel handling machine can release a fuel element only when the element is seated.

2.3 Radial Blanket Elements

The radial blanket elements are described in detail in SDD-096-28-000. In general, the fuel element description (Section 2.2) is applicable to the radial blanket elements. The conceptual design is shown in Figure 6.

There are two types of radial blanket elements, designated as the inner row and outer row. The power split in the radial blanket is approximately 66/34, inner to outer row. The average power generated in an inner row element is about 1.12 Mwt and in an outer row element it is about 0.578 Mwt. The radial blanket is cooled from the high pressure plenum. The upper grid plate coolant leakage for the inner and outer row radial blanket elements has been included with that for the fast elements given in Section 2.2.2.2.

The go/no-go system that guarantees locating elements correctly in the reactor core matrix, described in Section 2.2.2.3, is extended to the blanket region. The O.D. of the nozzle section of the outer row blanket elements is smaller than that of all other elements to prevent placing a high power element in an outer row blanket position. A diametral pin across the bottom end of these elements prevents seating them in a high flow position.

2.4 Control Rods

2.4.1 General Description

The Reactivity Control System, described in detail in SDD-096-22-000, is an array of control-safety assemblies used to control the reactivity of the core. Each assembly consists of (a) an absorber guide tube which is positioned

vertically in the core, (b) an absorber element which is moved within the guide tube, and (c) an electromechanical drive unit to adjust the position of the absorber element in the core. The drive unit has two speed capability, normal and fast setback (drive-in only) and can release the absorber element and allow it to free fall to its full-in position.

2.4.2 Component Details

2.4.2.1 Absorber Guide Tube

The absorber guide tube is supported by its nozzle on the upper grid plate. The guide tube incorporates a snubber section just above the upper grid plate to decelerate the absorber element following a reactor scram. There is a radial clearance between the guide tube and the adjacent fuel element housings, permitting the guide tube to be relatively unaffected by the fuel element displacement. The upper ends of the absorber guide tube are at a lower elevation than those of the fuel element housing, thereby placing the pickup ring in the absorber guide tube approximately 3 inches lower than that in the fuel element housing to eliminate handling errors. The absorber is captured in the guide tube and its handling head is always below the core matrix-top shield parting plane.

2.4.2.2 Absorber Elements

The absorber sections are of the multiple pin type, with NaK bonded stainless steel clad tantalum pins. Each element has a snubber ram at the lower end, a handling head at the upper end, plus appropriate hardware required to make up an assembly.

2.4.2.3 Rod Drive

The rod drive is electromechanical, using two electric motors, a stepper motor for normal drive speed and a dc motor for fast setback drive speed. The motors, mounted in a series arrangement, drive a ball screw linear actuator. A trip magnet is attached to the lower end of the actuator screw shaft through a load cell coupling. A drive extension passes through the top shield to the grapple at the top of the absorber element in the core matrix. The drives move the absorbers through a 43-inch normal stroke within the guide tube and can release them for free drop into the core.

The lower end of the drive must disengage from the absorber and retract about 7 inches to allow the top shield to be rotated for fuel transfer. This capability is provided by 7 inches over-travel beyond the normal stroke.

2.5 Miscellaneous Core Elements

Several other types of miscellaneous elements are included in the reactor matrix. These are the following:

- 1) Reflector Elements
- 2) Neutron Source Element
- 3) Materials Surveillance Elements

2.5.1 Reflector Elements

The reflector elements form the outer row of elements in the reactor matrix. They 5.50 inch across flats hexagonal stainless steel bars extending upward from the top of the upper grid plate to the top of the core elements. These bars are part of the radial neutron shield and as such attenuate the fast flux by a factor of about 5. Cooling is provided by grid plate leakages and natural convection in the gaps between elements.

2.5.2 Neutron Source Element

At the present time there is no requirement for a neutron source.

2.5.3 Materials Surveillance Elements

The reactor shall include provisions for internal placement of representative samples of the structural materials used. The samples will be removed at intervals during the life of the plant for examination to check for indications of material degradation which could lead to subsequent failure.

2.6 Core Constraints

The mechanical support and constraint for the reactor matrix assembly is shown in Figures 1 and 2. The reactor elements are supported at their lower end on spherical seats on the lower grid plate. They pass through holes in the upper grid plate with a radial clearance of 2.5 to 5 mils. Radial support is provided in a clamping plane which is located above the central plane of the core. The individual elements make contact in this plane through weld-overlay spacer pads, machined to within a tolerance of $\pm .001$ inch. A peripheral structure constrains the reactor matrix radially in this plane. The clamping-plane must be located above the core center to guarantee a negative contribution to the power coefficient as a result of element bowing.

The core radial support structure consists of a short cylinder secured at its lower end to the upper grid plate through a heavy flange. Another flange at its upper end is the clamping plane radial support and conforms, internally, to the peripheral contour of the reactor matrix. The upper end of the cylinder is slotted axially into equal radial segments, one for each reflector element. These segments are flexible "fingers" exerting inward radial forces to hold the reactor matrix together.

2.7 Core Support Structure

2.7.1 General Description

The components that provide the required support for the core matrix are the grid plate support structure and the grid plate structure.

The grid plate support structure is considered a non-removable vessel internal structure. It supports the reactor grid plate and internal structure. The core centerline is 18 inches eccentric to that of the reactor vessel. The grid plate support structure is stiffened, using integral vertical web members, so that its deflection does not exceed 0.03 inch. Together with the grid plate structure, it forms the plenum for high pressure coolant. The maximum exposure of this structure to fast neutrons ($E > 1 \text{ Mev}$) after 30 years does not exceed 1×10^{21} nvt.

The grid plate structure provides mechanical support for each reactor element. It is a replaceable component; it can be removed from the reactor through the opening of the smaller rotating plug. The two grid plates are coupled together by a cylindrical shell and by 4 inch I.D. x 1/2-inch thick tubes at each of the 15 control rod positions. A seal is made between the upper grid plate and the support structure flange. There will be leakage of high pressure coolant to the lower part of the reactor vessel through the annular gap between the lower grid plate structure and the support structure. This leakage is considered as part of the low pressure coolant supply. The maximum exposure of the upper grid plate to fast neutrons ($E > 1 \text{ Mev}$) after 30 years does not exceed 4×10^{21} nvt.

Vertically segmented cylindrical plates surround the reflector elements to provide an additional 7.5 inch thickness of radial neutron shielding. This serves to decouple the reactor core and any fuel that may be stored in-vessel, and to limit the maximum exposure of the reactor vessel wall to fast neutrons ($E > 1 \text{ Mev}$) after 30 years to less than 10^{19} nvt.

2.7.2 Inlet Plenum Hydraulics

The mean friction pressure loss in the inlet plenum is 1.5 psi. The core element nozzles are adequately rigid to preclude vibration.

2.7.3 Core Element Orificing

High pressure sodium enters the outer annular region of the inlet plenum through three symmetrically located 42-inch diameter pipes. It is distributed to the fuel and blanket elements through fixed orifices in the element nozzles. The orifices are specifically sized holes in the vertical wall of the tube of the element nozzles. The individual holes are small compared to the nozzle cross-section, and are dispersed circumferentially and axially, to eliminate the possibility of significant flow channel blockage from solid foreign matter.

2.7.4 Thermo-Convection Cooling

The control rods and any core elements that may be stored are cooled by flow from the region below the grid plate. This cooling depends, for its driving force, on the density difference of the sodium due to temperature differences. It is self-regulating, in that high heat loads increase sodium temperatures resulting in increased buoyant forces due to density differences. The buoyant forces in turn cause higher sodium flow rates which provide greater cooling.

The displaced heated sodium flowing from these elements is discharged into the outlet plenum and on through the primary system. An equal amount of "inlet" sodium is supplied to the reactor vessel low pressure region through a 14-inch diameter inlet pipe. This flow is regulated with a valve to control the outlet temperature from the elements cooled by this method. Leakage from the high-pressure plenum to the lower part of the reactor vessel through the annular gap between the grid plate structures and the grid plate support structure supplements this low-pressure coolant supply. It is necessary that the temperature of the reactor vessel wall be monitored to assure against adverse axial temperature gradients.

2.8 Reactor Vessel

The reactor vessel, shown on Figure 1 and described in detail in SDD-096-21-000, is an open top, stainless steel vessel with an inside diameter of 29 ft and an overall height of approximately 48 ft.

The general arrangement of the reactor vessel and its internals is determined principally by fuel handling requirements. Several factors affect the reactor arrangement. One is that of interference between the control rods and the ex-vessel fuel handling machine, or between the in-core and ex-vessel fuel handling machines. This is resolved by locating the core and in-vessel fuel transfer positions side-by-side rather than concentrically. Secondly, since it is desirable to have all control rod drives on the same plug, the smaller rotating plug is made large enough to achieve this. This also provides a hole through the large plug, large enough for removal of the grid plate structure.

The vessel has nozzles for the following service connections:

1. Three 38-inch ϕ inlets for primary system sodium.
2. Three 44-inch ϕ outlets for primary system coolant.
3. One 14-inch ϕ inlets for low pressure coolant for control channels.
4. One 14-inch ϕ inlet from pump overflow.
5. One 8-inch outlet for reactor overflow.
6. Two 2-inch ϕ cover gas lines.
7. One 2-inch ϕ reactor sodium drain line.

The inlet nozzles are located above the reactor matrix and internal lines feed the high pressure coolant down to the distributor plenum.

The outlets have internal dip tubes so that the suction is taken about two feet above the top of the core matrix.

The vessel is suspended from the reactor cavity, at an elevation about two feet below the floor line, by a flange at its top rim. The reactor top shield forms the vessel closure.

The reactor vessel assembly includes the attachment points for support and alignment of the core support components and any flow baffles, shock liners or other flow controls devices required within the vessel.

2.9 Top Shield

The top shield is the upper closure of the reactor vessel. It provides radiation shielding for the operating floor area, a seal for the reactor cover gas, a heat flow barrier, and mechanical support for the control rod drives and the fuel handling machines. It is a double rotating plug, as shown in Figures 1 and 3 and acts as an integral part of the fuel handling system.

The larger of the two rotating plugs is concentric with the reactor vessel and turns on a thrust bearing approximately 25 ft in diameter. The smaller plug is 15 ft 6 inches in diameter, is mounted on a separate thrust bearing, and has its axis 40.5 inches from the center of the larger plug. The smaller plug has a penetration near its periphery for the in-vessel fuel handling equipment. Gear drives are provided to rotate each plug so that the fuel handling machine can be indexed over any core or storage position. Positive indexing is provided by an electromechanical system similar to that used for large tape-controlled machine tools.

The overall thickness of the top shield is approximately 11 feet. In the current reference design the shielding consists of an approximately 15-inch thick structural top plate and approximately 5 feet of serpentine rock. Below are nearly 2 feet of reflective insulation plates and a 2.5 feet gas space.

The seals for the two large plugs are lip seals which are internally pressurized to provide positive sealing during reactor operation. Friction can be reduced during rotation of the plugs for fuel handling by reducing the internal pressure of the seals.

Penetrations through the top shield are provided for fuel handling, control rod actuators and instrumentation. The larger plug has the following penetrations:

1. A transfer cell access port for the ex-vessel fuel handling machine.
2. Two ports for sodium level probes.

The smaller plug has the following penetrations:

1. One access port for the in-vessel fuel handling machine.

2. Fifteen ports for control rod channel and instrument carrier tubes.

3. Ports for instrument carrier tubes (non-control channel).

Three contain neutron flux monitor thimbles.

In addition, several, as yet undetermined, access ports will be provided for general access to the vessel interior for boroscope viewing, emergency handling, and sodium or cover gas sampling.

The upper structure of the top shield is the mounting face for the control rod drives and fuel handling machine valves. Susceptibility to sodium frosting and plugging of the annuli around the rotating plugs is reduced by minimizing the sodium-surface/cover-gas interface and by continuous sweeping of the annuli with clean helium. The minimum sodium/gas interface is achieved by immersing the bottom of the top shield in the sodium pool. This also reduces gas adsorption and entrainment in the sodium. The bottom plate is sloped slightly to prevent entrapment of gas bubbles, beneath it.

2.10 Vessel Jacket

The vessel jacket is an independent secondary envelope for containment of the primary sodium in the event of a leak or rupture in the reactor vessel or its contiguous inlet piping. The tank is basically a standpipe that encloses the reactor vessel and inlet piping in a fairly close-coupled envelope that minimizes the void volume between these two containment boundaries. The containment tank has no penetrations below a minimum sodium level. It is supported on its flat bottom head completely independent of the reactor vessel. Sufficient vertical clearance is provided to allow for the thermal growth of the reactor vessel.

The 12-inch space between the tank and the reactor vessel is provided for use of vessel inspection equipment. Any permanently installed trackways or other devices required by the vessel inspection system will be a part of the containment tank.

The containment tank extends to five feet above the top of the core elements. Its basic function is to keep the core and outlet pipe intakes submerged in the event of a vessel or pipe rupture.

The containment tank and the reactor cavity have a common nitrogen atmosphere. Insulation is applied to the outer surface of the containment tank bottom and wall. Leak detectors and pump-out capability are provided as shown in Figure 1.

2.11 Reactor Cavity

The reactor cavity has a 35 ft ID and extends approximately 53 feet below the operating floor, as shown in Figure 1. The cavity wall provides structural support for the reactor vessel and top shield. It is part of the biological shield and forms part of the primary containment barrier for the DBA. There are no penetrations through the cavity wall below those for the primary sodium outlet piping.

Cooling coils attached to the concrete side of the liner remove heat generated in the biological shield and conducted through the reactor insulation. A DBA can result in a large quantity of fuel being deposited at the bottom of the cavity in a molten state. The decay heat will be conducted to cooling coils attached to the underside of the cavity liner. This heat will be absorbed pressurized water in the cavity bottom cooling coils. A completely redundant parallel system, sharing only the bottom of the cavity liner, will be provided for high reliability.

2.12 In-Vessel Fuel Transfer Rack

The reactor will be refueled on a one-year cycle with one-half of the core, one-fifth of the inner blanket and one-seventh of the outer blanket being replaced each cycle. A like number of blanket elements will also be rotated 180 degrees to allow leveling of the plutonium buildup associated with the sharp flux profile at the edge of the core.

The fast core elements have an average reactor residence time of 2 years: the inner radial blanket 5 years; the outer radial blanket 7 years; and the shim-control rods 8 years. This means that during a one-year cycle refueling, theoretically, 137 fast elements; 13 inner blanket elements; 10 outer blanket elements; and two control rods, for a total of 162 core assemblies, are exchanged. In addition, 23 blanket elements are rotated 180 degrees. Fifteen in-vessel transfer storage positions have been provided for fuel transfer and added flexibility.

The reference fuel handling procedure requires that spent fuel be removed from the core and placed in canisters located in the transfer positions which are closed at the bottom and open at the top. Spent fuel is normally removed from the vessel during the same refueling shutdown that it is removed from the core.

The canisters are supported in the fuel transfer rack shown in elevation in Figure 1 and in plan in Figure 2. The 15 positions are arranged in an arc having a center in common with the reactor vessel and the large rotating shield plug.

The transfer rack consists basically of 8.5 inch I.D. perforated tubes held in a plate structure. The position of the top end of any fuel canister in storage must be predictable to within .75 inch.

Each canister is a tube, 7.75 inches in diameter, which is closed at the bottom and has a heavy ring welded to the top end. The ring is machined with an internal pickup lip for the grapple of the ex-vessel FHM and internal passages to siphon off the top few inches of sodium upon removal from the sodium pool. When the canister is to be used for shipping spent fuel, or other radioactive elements, a seal cap can be welded to the top ring.

2.13 Instrumentation

Means will be provided to measure the following quantities:

1. Sodium temperature at the outlet from all fast fuel, radial blanket and reactivity control elements.
2. Temperatures at selected points of structural components.
3. Neutron flux during initial startup and subsequent refueling operations.
4. In-core neutron flux of the initial core for physics purposes.
5. Neutron flux during reactor operation.
6. Liquid level of reactor sodium.
7. Reactor cover gas radioactivity.
8. Delayed neutron flux.

More detailed description and requirements for reactor instrumentation are given in SDD-096-50-000, "Instrumentation Segment".

2.13.1 Core Element Temperature

Thermocouples are installed over each of the 27⁴ fast fuel elements and the 13⁸ radial blanket elements. Their hot junctions are positioned about 7 inches above the element coolant outlets. The thermocouples are carried through the top shield in $\frac{1}{2}$ -inch OD guide tubes that are closed at their lower ends to form thimbles. The guide tubes are supported by the instrument carrier tubes and penetrate the top shield in a one-half revolution helix to reduce local radiation streaming to 25 mr per hour. They are fanned out near their bottom ends, in gentle sweeps, where required, and are held in position by the instrument support structure as shown in Figure 1. This structure, which is 13 inches above the element outlets, ties the lower ends of the carrier tubes together. Each guide tube accommodates two 1/8-inch OD sheathed thermocouples and is sealed at the upper end with compression fittings. The thermocouples can be replaced during reactor shutdown. A liquid metal thermal bond can be provided in this guide tube to improve response but in this case thermocouple replacement will be considerably more complicated.

The outlet temperatures of the 15 control rod channels are measured by thermocouples built into the control rod drive extension. The sensing end is attached to the pull shaft just above the latch and therefore is fixed in relation to the top end of the absorber element.

2.13.2 Structural Component Temperature

The temperature of reactor internal structures will be measured where such information is required to assist in initial reactor operation. The thermocouples will have an expected life, limited by their radiation environment, of about 1 to 2 years and will not be replaced. They will be routed out of the reactor vessel through conduits inside the primary sodium piping. The conduits will be taken out through 4 to 6 inch diameter standpipes which terminate in a seal plate. The standpipes are connected to the reactor cover gas system. The number of thermocouples and their location has not been determined.

Top shield temperatures also will be required, but their location and number have not been determined. The thermocouples will be taken out through the top of each of the two rotating plugs.

The temperature of the reactor vessel, vessel supports and piping will be measured by thermocouples mounted externally. Thermocouples will also be provided on the containment tank and on the cavity liner.

2.13.3 Neutron Flux Measurements

Three categories of instruments are furnished to measure neutron flux during power operation, refueling and initial core physics measurements.

During power operation neutron flux will be monitored by instruments in ex-vessel thimbles. Three thimbles about 14-inches in diameter and spaced roughly 120° apart will be provided. Internal cooling will not be required as the detectors are rated at 570°F.

During refueling operations the neutron flux will be monitored by instruments in thimbles extending through the top shield to within 7 inches of the fuel elements. Cooling is not required because the detectors are rated at 700°F and are removed upon reactor startup.

Special provisions must be made to make the necessary physics measurements on the initial core. Fuel element operating instrument carrier tubes can be replaced with special instrument thimbles for core mapping. These thimbles extend into the centers of special fuel elements that have their seven center pins removed to make room for the thimble.

2.13.4 Liquid Level Measurements

The requirements for monitoring the reactor liquid level are given in SDD-096-50-000. Two penetrations of about 6 inch diameter will be provided in the large rotating plug as indicated in Figure 3.

2.13.5 Ex-Vessel Instrumentation

Reactor cover gas radioactivity and delayed neutron flux will be monitored by instrument systems off the reactor.

3. PRINCIPLES OF OPERATION

3.1 Startup

Startup of the reactor is accomplished by first withdrawing the safety rods and then the shim-safety rods to achieve criticality. Sodium flow through the reactor will be at least 25% of full flow before initiating startup. Sodium temperature will be approximately 500 F.

After achievement of criticality, the reactor power may be increased by further withdrawal of the shim-safety rods. The rate of power increase will be limited such that reactor sodium outlet temperature rises at a rate less than 50°F per half hour. When reactor outlet temperature reaches 1140 F and reactor inlet temperature reaches 780 F, the startup phase is considered complete and the normal operation phase is entered.

3.2 Normal Operation

Normal operation is defined as operation between 25% and 100% of full power, with reactor outlet temperature less than or equal to 1140 F. Reactor outlet temperature will be programmed to be somewhat lower at 25% load than at 100% load, to keep secondary sodium system temperatures constant, and to minimize the need for desuperheating at low loads.

Load changes can be made by automatic positioning of the shim-safety rods in coordination with changes in primary and secondary sodium flow.

During steady-state operation, reactor temperatures must be monitored and if desired minor adjustments made in shim-safety rod positions and control rod (thermosyphon) cooling supply to optimize temperature distribution within the reactor.

3.3 Shutdown

3.3.1 Normal Shutdown

A normal shutdown of the reactor is accomplished by gradually reducing reactor power, allowing reactor outlet temperature to decrease at a rate of 50°F per 1/2 hour, and then initiating a scram when reactor delta T has become negligible.

3.3.2 Emergency Shutdown

An emergency shutdown of the reactor is a trip (all rod latches trip). This can be initiated manually or automatically by the protective system, if some plant parameter exceeds a preset limit.

3.4 Special or Infrequent Operation

3.4.1 One or More Control Rods Out of Service

It is expected that the reactor may occasionally be operated with one (or more) of the control rods disabled. The criteria for the maximum allowable number of disabled rods has not yet been established. However, all safety rods must be removed from the core.

3.4.2 One Heat Transfer Circuit Out of Service

The reactor can continue to be operated with one of the three heat transfer circuits out of service. The only effect this has is to limit allowable reactor power to approximately 70% of full power.

3.5 Emergency Operation

Emergency conditions (sudden events requiring immediate action) which can be anticipated are listed below. In most of these cases, the plant protection system is expected to take action.

3.5.1 Gross Failure of Fuel Pin Cladding

Required action is immediate reactor shutdown and activation of the reactor cover gas purification system if not already in service.

3.5.2 Inadequate Coolant Flow to One or Several Fuel Elements

Required action is manual reactor trip.

3.5.3 Excessively High Reactor Flux Level

Required action is manual reactor trip.

3.5.4 Drop in Reactor Vessel Sodium Level

Required action is to restore sodium level (for example, by adding sodium to the system). Further action is to shut the reactor down, either in the normal manner or by means of a reactor trip, depending on the rate of loss of sodium level.

Under emergency conditions, reactor shutdown can be initiated from the cable spreading room (outside the reactor building), from the pre-amp room (inside the reactor building), and near the reactor loading face. These emergency shutdown stations will be under close administrative (keylock) control.

In the unlikely event of a DBA the primary containment boundary and the post DBA cooling plate perform to assure safe containment of the system.

4. SAFETY PRECAUTIONS

4.1 Hazards

The major hazards associated with the reactor segment are:

A. Loss of coolant flow to the entire reactor, either as a result of:

1. A leak in the reactor vessel.
 2. A leak elsewhere in the primary coolant system.
 3. A pipe rupture in the primary coolant system.
- B. Loss of coolant flow to one or several fuel elements.
C. Large gas bubbles transported through the reactor core.
D. Excessive leakage of fission gases.

4.2 Precautions

To minimize the preceding hazards, the following actions should be taken:

- A. Reactor sodium level should be continuously monitored.
- B. Leak detectors in the vessel containment jacket should be periodically monitored. Redundant leak detectors should be installed to assure operating leak detectors during the plant lifetime.
- C. Activity of the nitrogen atmosphere in the primary equipment vaults should be monitored continuously, as an indicator of primary sodium piping or equipment leakage.
- D. The frequency and magnitude of primary system thermal transients should be minimized.
- E. Primary piping and equipment and the reactor vessel should be periodically examined for potential failure mechanisms.
- F. Caution should be exercised to prevent introduction of impurities such as oxygen and hydrocarbons into the primary sodium system.
- G. Upon removal from the reactor, selected fuel elements should be examined for evidence of possible fouling or flow blockage.
- H. Prior to each reactor startup, the primary sodium systems should be operated at full flow until all gas is swept from the system.

I. Seals on the reactor top shield and the reactor containment dome should be checked periodically and replaced as necessary.

5. MAINTENANCE

5.1 Normal/Routine Maintenance

In general, maintenance on the equipment within the reactor segment must be performed with the reactor shut down. The following tasks will be performed periodically, during scheduled plant shutdowns:

- A. Vessel inspection.
- B. Control rod drop time checks.
- C. Control rod component replacement.
- D. Top shield seal replacement.
- E. Containment dome seal replacement.

Top shield bearing replacement will be performed as an as-needed basis.

5.2 Unusual Maintenance

A. Partial Fuel Meltdown

The top shield design provides for the removal of a seven element cluster through a minor modification to the in-core fuel handling machine port. This opening can be positioned over any core position by rotation of the inner and outer top-shield plugs. The basic recovery procedure for a partial fuel meltdown is as follows:

1. Locate and examine the top and accessible sides of the damaged fuel elements. This can be done by using the in-core fuel handling machine to determine freedom from adjacent elements, and using remote viewing equipment when the reactor is partially drained.
2. Concurrently with the above, design and initiate procurement of the special tools and equipment necessary to remove and examine the fuel element cluster and provide for disposal for the cluster.
3. Unload fuel elements from the core, except for the damaged elements; replace them with the spacer elements supplied for the initial core loading. This step may be eliminated if the cause of the damage can be established and there is confidence that the undamaged fuel elements are acceptable for further service.

4. Concurrently with the preceding step, continue the examination of other reactor internals with remote viewing equipment, and modify the radioactive equipment maintenance cask for removal of the seven element cluster.
5. Initiate the visual examination of the undamaged fuel elements in the radioactive maintenance cell.
6. Following unloading of undamaged fuel elements, remove the seven element cluster to storage. If no temporary storage is available, the cask may be used until the element is placed in the radioactive maintenance cell for examination or preparation for shipment.
7. Upon completion of all desired examinations, a review of data is scheduled to establish the corrective action requirements.
8. Prepare corrective action procedures while waiting for delivery of material and, upon delivery, take the corrective action.
9. Reload the core and restart plant.
10. Place the damaged element in the special canister and ship off-site for examination, reprocessing, or disposal.

It is estimated that a reactor shutdown of from 9 to 20 months would be required to accomplish recovery from a partial fuel meltdown. The elapsed time is dependent on the nature of the damage, the fuel examination requirements, the availability of materials and equipment, and the corrective action requirements.

The special equipment needed for recovery from a fuel meltdown is estimated to require from three months to a year for delivery. The equipment includes remote inspection equipment, special tooling for core in-cell examination and cluster removal, an inner plug cask, and a shipping canister for the cluster. The inclusion of an in-cell fuel element cooling device will allow an earlier start of nondestructive fuel element examination and the evaluation indicates this would reduce the total down time of the plant.

B. Damaged Reactor Internals

The reactor design provides for the removal of the inlet plenum assembly through the opening provided by the removal of the inner top shield plug. All the reactor internals requiring removal to repair are designed to be removable without the removal of the outer shield plug. Inspection and repair of the vessel walls is accomplished from the outside of the vessel by using the remotely operated cart.

The basic recovery procedure for replacing the inlet plenum consists of the following:

1. The initial step for recovery from a damaged inlet plenum consists of the following essentially concurrent activities; unload the core; inspect the reactor internals; evaluate inspection results; and engineer, procure, and deliver special tools and equipment.

2. The corrective action involves the preparation of procedures while waiting for materials the placement or erection of special equipment and shielding, the removal and reinstallation of the inner top shield plug and reactor inlet plenum, inspection of the installation, and cleanup of the area.

3. Reload the core and restart the plant.

It is estimated that a reactor shutdown of 7 to 21 months would be required to replace the reactor inlet plenum. The elapsed time is dependent upon the nature of the damage and the availability of materials and equipment.

The delivery time of the special equipment required for the replacement of the reactor plenum is estimated to be 6 to 15 months. If the equipment is available at the time of the incident, approximately 14 months can be saved on the outage.

6. APPENDICES

6.1 Figures

Figure 1	Reactor Vertical Section	Dwg No. N09620021
Figure 2	Reactor Plan View	Dwg No. N09620022
Figure 3	Reactor Plan Section	Dwg No. N09620023
Figure 4	Plant Engineering Safety Features	
Figure 5	Fuel Element Assembly	Dwg No. N09624006
Figure 6	Blanket Element Assemblies	Dwg No. N09624007

REFERENCE DRAWINGS
 NO3620022 REACTOR-PLAN VIEW
 NO3620023 REACTOR-PLAN SECTION

REVISIONS			
SYM	ZONE	DESCRIPTION	DATE
A		SEE REVISION RECORD	12/1/68
		to 5/1/71	12/1/68

SDD-096-20-000
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REFERENCE DRAWING
N09620021 - REACTOR - VERTICAL SECTION
N09620023 - REACTOR - PLAN SECTION

Figure 2

ATOMICS INTERNATIONAL A DIVISION OF NORTH AMERICAN AVIATION, INC. 3935 BRIDGE AVENUE, CANOGA PARK, CALIF.			
REACTOR PLAN VIEW			
1000 MW(4) LMFBR FOLLOW-ON STUDY			
DRAWN	DATE	BY	NO
CHECKED	DATE	BY	NO
DESIGNED	DATE	BY	NO
APPROVED	DATE	BY	NO

REV	DATE	DESCRIPTION	APPROVED
B	3-10-77	SEE REVISION RECORD	AS SHOWN

- CORE LEGEND**
- FAST FUEL - 274
 - ⊗ BLANKET FUEL - 158
 - REFLECTOR - 78
 - △ SHIM SAFETY-CONTROL ROD
 - ⊙ SAFETY-CONTROL ROD (15 TOTAL)

REACTOR DRAWINGS
 N09620021 REACTOR-VERTICAL SECTION
 N09620022 REACTOR-PLAN VIEW

Figure 3

ATOMIC INTERNATIONAL	
A DIVISION OF CHRYSLER CREDIT CORPORATION	
1000 N. W. 10th Ave., Fort Lauderdale, Fla. 33304	
PROJECT: REACTOR PLAN VIEW	
DATE: 3-10-77	REV: 1
N09620023	

151520023001

151520023001

7-08-291-29E

Figure 4 Plant Engineering Safety Features

Figure 5

ATOMICS INTERNATIONAL A DIVISION OF NORTH AMERICAN AVIATION, INC. 3800 BAYVIEW AVENUE, CANOGA PARK, CALIF.	
FUEL ELEMENT ASSEMBLY 1000 MW LMFBR FOLLOW ON STUDY	
DRAWN: [] CHECKED: [] ENGINEER: [] APPROVED: []	DATE: [] SCALE: [] SHEET NO: [] TOTAL SHEETS: []

190042960N1

190042960N1

REVISIONS			
SYN	ZONE	DESCRIPTION	DATE
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		ED 51226	10/16/68

WEIGHTS	
INNER ROW DEPLETED U ASSEMBLY	144 KGS 562 LBS.
OUTER ROW DEPLETED U ASSEMBLY	115 KGS 491 LBS.

Figure 6

ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.
3800 BROADWAY, KANSAS CITY, MISSOURI 64111

BLANKET ELEMENTS ASSEMBLY
1000 MW LMFBR FOLLOW-ON STUDY

DESIGNER	DATE	BY	NO
ED 51226	10/16/68	ED 51226	007

PROJECT NO. 1000 MW LMFBR

1000 MW LMFBR FOLLOW-ON STUDY
Page 52

ATOMICS INTERNATIONAL*A Division of North American Rockwell Corporation***SUPPORTING DOCUMENT**

NUMBER

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PROGRAM TITLE

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ABSTRACT

This SDD presents the design requirements for,
and a brief description of, the Reactor Vessel
Assembly for the 1000 Mwe IMFBR Follow-on
Study

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REACTOR VESSEL ASSEMBLY

1. INTRODUCTION

1.1 System Functions

The functions of the Reactor Vessel Assembly are to:

1. Provide support for the fuel elements, blanket and reflector elements, control channels, etc., in the configuration required for reactor operation.
2. Provide space and support for transfer and storage of spent fuel and blanket elements and other core elements within the reactor vessel.
3. Provide primary containment of the reactor coolant within the reactor cavity and guide the coolant flow to and from the heat generating components within the vessel.
4. Provide radiation shielding within the vessel as required to properly de-couple any stored elements from the core and to limit the heat generation in the surrounding biological shield.

1.2 System Summary Description

The Reactor Vessel Assembly includes the primary core and coolant enclosing vessel and all components located within, attached to, and supported from the vessel, exclusive of the core, blanket and reflector elements.

The vessel shown in elevation in Figure I is made of austenitic stainless steel and is 29 feet in diameter by approximately 48 feet high. It has an ellipsoidal bottom head and an open upper end which is closed by the rotating top shield (not part of this system). The vessel is supported from the top by a ring flange.

There are three main coolant inlet and outlet connections plus several smaller auxiliary service connections. Sodium enters the vessel at 780°F, flows up through the core and exits from the vessel at 1140°F.

The reactor vessel internals include the following components:
(See Fig. I)

1. All coolant flow piping, shock liners, flow guides, and baffles, etc., located within the vessel but not an integral part of the vessel.
2. The grid plate support, and high pressure plenum structure.

3. The upper and lower grid plate assembly.
4. The radial neutron shield and radial core constraint structure and associated components or mechanisms.
5. The fuel transfer and storage support structure and the handling canisters.

The core is located eccentrically within the vessel and the fuel transfer positions are arranged in an arc which is concentric about the vessel center as shown on Figure III.

The reactor vessel supports and locates the core through the lower support structure - inlet plenum assembly and the radial neutron shield assembly. The fuel transfer support structure is mounted on the lower support structure. The core radial support structure, which is actually part of the radial neutron shield assembly, assures a tight core matrix during operation. Except for the lower support structure, all of the above components are removable from the vessel.

1.3 System Requirements

1.3.1 Performance Requirements

1.3.1.1 Coolant Flow

The system shall accommodate a total reactor coolant (sodium) flow rate of 75.58×10^6 lb/hr at nominal temperatures of 780°F inlet and 1140°F outlet with a pressure drop across the fuel and blanket elements of approximately 85 psi. Pressure drop attributed to coolant intake, distribution to core and blanket elements and exit from the vessel shall be minimized.

Low pressure coolant shall be supplied to the control channels, reflector elements, neutron shields, vessel walls and other internal structure by leakage from the high pressure plenum plus externally controlled flow from the primary coolant circuits. This coolant shall flow upward, through or along the components with thermal convection as the primary driving force.

1.3.1.2 Structural Alignment

The system shall be designed to assure adequate alignment of the core matrix for reliable control rod operation and for fuel and blanket element transfer and removal by the fuel handling system.

1.3.1.3 Load Support

- a. Contained Weights
 - Sodium - (to be determined)
 - Core Components - (to be determined)
- b. Pressures
 - See Paragraph 1.3.2.3

c. Piping Reactions - (to be determined)

1.3.2 Environmental Requirements

1.3.2.1 Operating Sequence of Events

The system is exposed to the radiation, thermal, mechanical, and pressure environments produced during plant preheat, startup, operation (including load swings), refueling and maintenance shutdown and restarts through its design life of 30 years. In addition various abnormal events, protected and unprotected, may occur during this period which will severely alter the normal environments. These events shall be carefully analyzed to insure that design margin to meet performance and safety requirements is provided.

1.3.2.2 Atmospheres

Reactor Coolant - Sodium @ 10 ppm oxygen plus small concentrations of other impurities.

Reactor Cover Gas - Helium with sodium vapor and fission product gases.

Reactor Cavity - Nitrogen

1.3.2.3 Pressures

Reactor Coolant

High Pressure - 120 psig

Low Pressure - Reactor cover gas plus sodium static head
(0 to 38 ft)

Reactor Cover Gas

Normal - +0.3 psig

Transient - +5.0 psig (8 hrs max)

Reactor Cavity

Normal 0 to 1/2" H₂O gage negative

During inner containment barrier leak rate tests the reactor cavity will be pressurized to 10 psig. During these tests the differential across the reactor vessel will be limited to +0.5 psi. The vessel shall be designed to withstand a 10 psi external pressure differential at 500°F max.

1.3.2.4 Temperatures

The temperature of the vessel and the components within the vessel will be essentially equal to that of the sodium at any given location. The sodium enters the vessel and is fed to the core element inlets and discharges from the top of the core elements. Heat generated in the vessel walls

and structural components within the vessel will be removed by thermal convective flow of sodium supplied by leakage from the high pressure plenum plus controlled low pressure feed of sodium into the bottom area of the vessel. Inlet and outlet sodium temperatures under various conditions will be as follows:

1. Steady State
Inlet sodium - 780°F
Outlet sodium - 1140°F
2. Transients
See SDD-096-20-000

1.3.2.5 Radiation Flux

1. Radial - The inner surface of the thermal shield, at 67.5" radius from core centerline, is exposed to the following neutron spectrum

<u>Neutron Spectrum</u>	<u>Flux</u> <u>n/cm²-sec</u>	<u>Dose</u> <u>nvt/30 yrs</u>
Fast (> 1.0 Mev)		
Epithermal (27.0 ev-1.0 Mev)	to be determined	
Thermal (< 27.0 ev)		

<u>Gamma Energy Spectrum</u>	<u>Flux</u> <u>mrem/hr</u>	<u>Dose</u> <u>rad</u>
Blanket		
Capture , Na and Stl.	to be determined	

2. Axial - The top surface of the core support structure, 93" from the core mid-plane, is exposed to the following radiation spectrum.

<u>Neutron Spectrum</u>	<u>Flux</u> <u>n/cm²-sec</u>	<u>Dose</u> <u>nvt</u>
Fast		
Epithermal	to be determined	
Thermal		

<u>Gamma Energy Spectrum</u>	<u>Flux</u> <u>mrem/hr</u>	<u>Dose</u> <u>rad</u>
Blanket		
Capture , Na and Stl.	to be determined	

1.3.3 Reliability/Availability Requirements

a. Objective

The 30-year unavailability objective for the vessel assembly is 0.005%.

b. Design Considerations

1. Provide design margins that meet the stated reliability objectives.

2. Consider methods that minimize thermal shocks and gradients to reduce the likelihood of fatigue failures and minimize component distortions that can cause core element misalignment and/or jamming.

3. Consider methods that minimize the possibility of collection of gas in the high pressure inlet plenum which could subsequently slug through the fuel or blanket channels.

1.3.4 Maintenance Requirements

1.3.4.1 Core Support Structure

Design of the core support structure, made up of an assembly of the upper and lower grid plates, the reflector elements, plus the radial neutron shield-core clamping assembly shall be such that replacement can be accomplished within a shutdown period of one year.

1.3.4.2 Vessel Inspection

The design of the vessel and its inlet and outlet connections shall allow inspection of all outer surfaces by remote methods.

1.3.5 Operational Requirements

1.3.5.1 System Preheat

The preheat system concept has not been selected. It may or may not be part of the vessel assembly.

1.3.5.2 Temperature Information

Thermocouples shall be provided on the vessel components as required to monitor heating rates and temperature profiles during reactor preheat and to monitor axial temperature profiles on the vessel during operation.

1.3.6 Safety Requirements

1.3.6.1 Coolant Boundary Integrity

The system shall provide a high integrity coolant boundary with a capability to function throughout its design lifetime with an extremely low probability of damage from any condition or operating environment that may reasonably be postulated.

1.3.6.2 Sodium Drain

The major objective in the arrangement and location of the primary coolant piping and all auxiliary service connections in or around the vessel shall be to preclude the possibility of loss of sodium to the extent that some circulation through the fuel and blanket elements could not be

maintained. There shall be no gravity drain lines installed on the reactor vessel.

1.3.6.3 Fuel Element Location

A positive means of precluding incorrect fuel or blanket element location with respect to enrichment and orificing shall be provided.

1.3.7 Design Requirements

1.3.7.1 Applicable Codes

1. Reactor Vessel - ASME Section III plus Code Case 1331 except as specified in Para. 1.3.7.4
2. USA Standard Code for Pressure Piping, USAS B31.7, Nuclear Power Piping
3. Reactor Internals - (See SDD-096-21-300).

1.3.7.2 Seismic Design

All components of the system shall be designed as Class 1A structures (for installation at any location) as specified in SDD-096-10-000.

1.3.7.3 Design Margins

1. Life - All components of this system shall be designed for a 30-year operational life.
2. Stress - The system shall be analyzed for operational stress and deflection within the intent of the codes listed in 1.3.7.1.
3. Analysis - The system shall be analyzed to meet the stated requirements on the basis of the following margins.
 - a) Transients - The effects of various transients shall be considered during the analysis of the system design, and where applicable, margins shall be applied to assure a safe/reliable Reactor Vessel Assembly.
 - b) Operational Tolerances - Operational tolerances shall be considered during the analysis of the system design, and where applicable, appropriate margins shall be included in the design.
 - c) Uncertainties - The level of uncertainty associated with each analysis method shall be considered and factored into the design to assure that the safety and reliability margins are truly experienced.

1.3.7.4

Materials

1. Sodium Contact - All materials in contact with sodium shall be type 304 stainless steel except where special requirements, such as wear or adhesion characteristics, require special materials. In the selection of special materials due consideration shall be given to mass transfer and sodium corrosion.
2. Allowable Stresses (Type 304 Stainless)
 - a) Normal Operation - Allowable stresses, taking into account stress to rupture in 250,000 hours, stress to produce 1% creep at 250,000 hours, and the effect of irradiation to 10^{23} nvt, are as follows:

<u>Temperature</u>	<u>Allowable Stress</u>
800°F	9000 psi
900	8800
1000	7200
1100	3900
1200	2400

1.3.7.5

Spares

No requirement for spare parts is anticipated.

1.3.7.6

Instrumentation

Instrumentation shall be provided to measure the temperature at critical areas of the vessel and internal structure during initial reactor operation. The number of thermocouples and their location has not been determined.

1.3.8

Interface Requirements

The Reactor Vessel Assembly, as part of the Reactor Segment, has major interfaces with the following components or assemblies within the segment:

1. SDD-096-28-000 - Fuel and Blanket Elements
2. SDD-096-22-000 - Reactivity Controls
3. SDD-096-23-000 - Reactor Shielding

The principle interface outside of the reactor segment involves the "Heat Transfer Segment" (SDD-096-30-000). Other external interfaces may occur with the auxiliary systems under SDD-096-60-000.

1.3.8.1 System Piping

The reactor vessel shall have nozzles for the following primary and service system piping:

- a) Three 38" $\varnothing$ primary inlet pipes
- b) Three 44" $\varnothing$ primary outlet pipes
- c) One 14" $\varnothing$ inlet for control rod and blanket coolant
- d) One 14" $\varnothing$ inlet from pump overflow
- e) One 8" $\varnothing$ outlet for reactor overflow
- f) Two 2" $\varnothing$ cover gas lines
- g) One 2" $\varnothing$ sodium drain line

1.3.8.2 Reactor Cavity

The entire weight of the vessel and its contents are transferred to the reactor support ring at the top rim flange of the vessel. This area shall also include a seal between the vessel and the support ring.

2. DETAILED DESCRIPTION OF SYSTEM

The reactor vessel provides the basic containment for the core and associated elements and is the containment boundary for the sodium coolant in the reactor segment. As shown in Figure 1, the vessel is 29 feet in diameter and approximately 48 feet tall. The shell courses containing the inlet and outlet nozzles and the bottom head are approximately 2-inch thick and the remainder of the shell is approximately 1-inch thick. The vessel is suspended from the reactor cavity structure by a rim flange at its top end. The top closure is provided by the rotating top shield defined in SDD-096-23-000.

There are three 38-inch diameter inlet and three 44-inch diameter outlet nozzles for the primary coolant loops and several smaller connections for auxiliary services. All of the reactor vessel nozzles are located above the reactor core section such that a break in the nozzle or adjacent piping will not cause the coolant to drain to a level that would preclude maintenance of circulation through the core. Internal feed lines conduct the high pressure coolant lines from the inlet nozzles to the coolant distribution chamber made up of the grid plates and the lower support structure. Slip joints take care of the differential thermal expansion between the vessel nozzles and the plenum. Dip tubes extending downward from the outlet nozzles place the suction at an elevation that is containable by the sodium leak container (See SDD-096-20-000) to assure coolant flow capability in the event of pipe or vessel leak.

One 14-inch diameter feed line supplies a regulated quantity of low pressure coolant to the volume beneath the grid plate support structure. This sodium, plus leakage from the high pressure circuit, circulates upward by thermal convection to remove heat generated in the control rod absorbers and the vessel and internal structure.

A 2-inch line is routed within the vessel to its lowest point to allow complete pumpout. Other service lines provided are reactor overflow, pump overflow inlet, and cover gas inlet and outlets. An overflow weir which extends around the I.D. of the vessel provides inherent control of liquid level within the vessel under heatup or steady-state operation. A continuous overflow will be maintained and circulated through a degasifier system.

Suitable thermal barriers will be provided at areas of the vessel and its nozzles and internal piping subject to thermal shock during temperature transients and to reduce thermal shorting between the cold inlet coolant and the hot coolant in the upper areas of the vessel.

The reactor vessel internals, which includes; a) lower support structure, b) grid plate assembly, c) radial neutron shield and core constraint assembly, d) reflector elements, e) fuel transfer rack and storage canisters, f) flow baffles and thermal barriers not constructed as an integral part of the vessel; are described in SDD-096-21-300.

3. PRINCIPLES OF OPERATION

3.1 Startup

Startup of the reactor vessel assembly is primarily related to its function as a container and flow guide for the primary reactor coolant. Prior to admitting sodium to the vessel assembly, the vessel and its internals must be preheated from ambient temperature to approximately 350°F. This may be accomplished utilizing a gas heating and circulating system, or electrical heaters temporarily installed within the vessel, or permanently installed electrical heaters on the vessel jacket, or some combination of the preceding. During the preheat operation, temperatures of the vessel and its various internal components must be monitored carefully and the rate of heating controlled to prevent severe thermal gradients and to maintain uniform temperatures.

Following preheat, the vessel assembly may be filled with sodium through the drain line and/or via the main heat transfer loops.

3.2 Normal Operation

During normal plant operation, sodium level within the vessel assembly will be maintained at the normal operating level automatically by the overflow and degasifier system. Vessel and internal temperatures, sodium level, and cover gas pressure are monitored to assure operation within design limits.

3.3 Shutdown

Normally, it will not be required that the vessel be drained of sodium. Therefore, once preheated and filled, a "shutdown" of the vessel assembly should not be required during the plant lifetime.

Scheduled shutdown of the reactor occurs at one year intervals, for fuel and blanket element changing. During these scheduled shutdowns, vessel temperatures will be varied from normal operating (780°F inlet, 1140 outlet) to 500°F isothermal at a controlled rate, and then will be returned to the normal operating conditions after fuel and blanket elements have been changed.

3.4 Special or Infrequent Operation

The vessel may be operated in a completely or partially drained condition on an infrequent basis.

3.4.1 Partially Drained Condition

The reactor vessel may be partially drained without losing convective heat transfer system flow, and therefore the fuel need not be removed from the reactor. This may be required if there is a major leak in one of the primary reactor coolant loops, or if major maintenance (such as removal and replacement of a blocking valve) is required on a single loop.

3.4.2 Completely Drained Condition

Complete drainage of the reactor vessel may be required if maintenance or internal inspection of vessel is desired. The vessel drain line permits complete drainage, which can only be done after fuel is removed from the reactor. It is advisable to keep the vessel at preheat temperature when in this condition, and to maintain the atmosphere within the vessel free of oxygen and other impurities in order to avoid formation of sodium oxides.

3.5 Emergency

An emergency will be created if the reactor vessel should develop a leak. The resultant loss in sodium level will result in a low sodium level alarm and then a reactor scram. The volume between the vessel and the vessel jacket is such that the sodium level will remain about two feet above the sodium outlet nozzles after the sodium levels in the vessel and in the annulus between the vessel and the jacket have equalized.

The action to be taken is to assure that the reactor is shut down, monitor sodium level, and be prepared to add sodium to the vessel if necessary to maintain the sodium level above the minimum allowable level.

4. SAFETY PRECAUTIONS

4.1 Hazards

Possible hazards associated with the reactor vessel assembly are:

- a) Leakage of sodium from the vessel.
- b) Application of excessive external pressure to the vessel.
- c) Excessive misalignment between the reactor core and the top shield; caused by lateral displacement due to thermal effects or mechanical loads, failure of supporting structure, or improper function of the core clamping system. This can cause jamming of the reactor control rods and loss of ability to control the reactor.
- d) Rupture of a high pressure feed line. This can result in a major reduction of flow through the core and subsequent core damage.
- e) Collection of de-entrained gas in the core inlet area and subsequent passage through the core in large bubbles. This can severely damage the core.

4.2 Precautions

Operational precautions which should be taken to mitigate the hazards listed in 4.1 above, are as follows:

- a) Vessel sodium leakage - The presence of sodium within the vessel jacket should be checked periodically by means of the leak detector. Periodic inspections of the vessel exterior wall should be made using remote viewing equipment. Thermal shocks on the vessel should be minimized in magnitude and frequency, consistent with plant operational requirements. Surveillance samples of structural materials should be monitored during the life of the plant to verify that material properties are not degrading excessively.
- b) Excessive external vessel pressure - Careful attention must be paid to equalizing pressures in the reactor vessel and the reactor cavity.
- c) Reactor core/top shield misalignment - The likelihood of this occurring can be minimized by avoidance of excessive thermal shocks due to rapid sodium temperature changes, and by maintaining the top shield temperature within operating limits.
- d) Sodium inlet line rupture - Stresses in this line should be kept low by avoidance of rapid temperature changes on both the reactor inlet and outlet.

e) Gas collection in core inlet area - Prior to reactor startup, possible entrained gas should be swept out of this area by running sodium flow up to 100%. All sodium levels (reactor vessel and pump) should be maintained at their design level during reactor operation. Normal sodium flow in the pump and reactor overflow lines should be maintained.

5. MAINTENANCE

5.1 Normal or Routine Maintenance

The external surfaces of the reactor vessel and its contiguous piping will be examined periodically (frequency not now established) by visual and/or ultrasonic methods. This examination will search for defects that could lead to subsequent sodium leakage.

The system used for the examination can be extended to enable the repair of defects.

5.2 Unusual Maintenance

a) Reactor Vessel Repair

If the reactor vessel should develop a leak, it is possible to repair it using remote welding and inspection equipment, operating in the space between the vessel and vessel jacket. The major steps in this maintenance operation are:

- 1) Remove energy absorbing structure and containment dome.
- 2) Unload fuel and blanket elements from the reactor.
- 3) Drain sodium from reactor vessel and containment jacket.
- 4) Locate leak using remote inspection equipment in annulus between vessel and jacket.
- 5) Repair leak and inspect repair using remote equipment.
- 6) Refill vessel and reload reactor.

b) Vessel internals repair or replacement

The grid plate structure, neutron shields, and reflector elements can be removed as a unit from the reactor vessel through the 15'6" diameter opening provided by removal of the "small" rotating shield plug. Access to other vessel internals such as the fuel transfer storage racks and the sodium piping is through smaller openings in the top shield.

If the grid plate assembly and associated structure must be removed and replaced, the major steps are:

- 1) Fabricate shielded cask for small rotating shield plug and grid plate structure.
- 2) Fabricate cover for 15'6" opening in top shield.
- 3) Remove fuel and blanket elements from reactor.
- 4) Remove control rod drives and drive extensions.
- 5) Remove small rotating shield plug, using special cask. Install cover over opening.
- 6) Store small rotating shield plug, remove cover from top shield opening, and reposition special cask.
- 7) Unbolt grid structure and remove into special cask.
- 8) Replace components in reverse sequence.

6.

APPENDICES

6.1

Figures

Figure 1 - Reactor Vertical Section (N09620021)

Figure 2 - Reactor Plan View (N09620022)

Figure 3 - Reactor Plan Section (N09620023)

REVISIONS			
SYN	ZONE	DESCRIPTION	DATE
A		SEE REVISION RECORD	12/1/68
			12/1/68

REFERENCE DRAWING
 NO9620021 - REACTOR - VERTICAL SECTION
 NO9620023 - REACTOR - PLAN SECTION

FIG. 2

ATOMIC INTERNATIONAL A DIVISION OF - ATOMIC SERVICES DIVISION, INC. 8000 11TH AVENUE, SAN DIEGO, CALIF.			
REACTOR PLAN VIEW 1000 MW (4) LMFBR FOLLOW-ON STUDY			
DRAWN CHECKED ENGINEER APPROVED	DATE 12/1/68	SHEET NO. 1/16	PROJECT NO. NO9620022

12/1/68

12/1/68

REVISIONS			DATE	APPROVED
SYN	ZONE	DESCRIPTION		
A		SEE REVISION RECORD	12-18-67

- CORE LEGEND**
- FAST FUEL - 274
 - ⊗ BLANKET FUEL - 138
 - REFLECTOR - 78
 - ⊙ SHIM SAFETY-CONTROL ROD
 - ⊗ SAFETY-CONTROL ROD (15 TOTAL)

REACTOR DRAWINGS
 NO9620021 REACTOR-VERTICAL SECTION
 NO9620022 REACTOR-PLAN VIEW

Fig. 3

ATOMIC INTERNATIONAL	
A DIVISION OF NORTH AMERICAN ROCKET CORPORATION	
2000 BROAD AVENUE, CANBERRA, AUSTRALIA	
REACTOR PLAN SECTION	
2000 MW(L) LMFBR FOLLOW-ON STUDY	
UNAPP	DWG NO
CHG	REV
DATE	SCALE
12-18-67	1/2"
NO9620023	

11/200296001

11/200296001

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1. INTRODUCTION

This SDD covers all components within the reactor vessel except for the fuel, blanket, and control elements. It includes grid plates, core clamps, reflector elements, inner thermal and neutron shields, in-vessel fuel storage rack and any shock liners, flow baffles, orifices and strainers not constructed as an integral part of the vessel.

1.1 System Functions

1.1.1 Principal Performance Functions

The principal performance functions of the Reactor Vessel internals are to

- 1) Provide mechanical support and constraint of the core and blanket elements,
- 2) Provide distribution of coolant to the fuel and blanket elements,
- 3) Provide transfer positions for spent fuel within the vessel prior to removal from the vessel,
- 4) Provide radiation shielding, within the vessel as required to: a) properly decouple any stored elements from the core, b) limit the neutron flux at the reactor vessel, and c) limit the heat generation in the surrounding biological shield.

1.2 Summary System Description

The reactor vessel internals are part of the reactor system shown by Figures 1,2 and 3. The reactor, which generates 2400 MW thermal at nominal rating, has a core of mixed plutonium and uranium oxide fuel surrounded by a depleted UO_2 breeding blanket, all suitably contained and subdivided to permit adequate heat removal by the circulating liquid sodium coolant.

The reactor vessel internals support the core within the vessel at a position eccentric to the vessel center and provide transfer positions for spent fuel in an arc which is concentric about the vessel center as shown by Figure 2.

The lower support structure and the grid plate structure provide the basic support for the matrix of the core, blanket, and reflector elements. The matrix of elements is supported and constrained laterally by the neutron shield, which includes the radial core constraint devices, in a manner that assures a tight core matrix during reactor operation. The lower support structure includes reactor coolant feed lines which connect the reactor vessel inlet nozzles to the high pressure plenum formed by the lower support structure and the grid plate structure.

During refueling, new fuel and spent fuel is contained in canisters placed in a transfer rack mounted on the lower support structure. The canisters are not a permanent part of the installation as they are also used as the container for the fuel when it is transferred into and out of the vessel and during ex-vessel storage and shipment.

This system will also include any thermal barriers, flow baffles, orifices, strainers and internal piping not constructed as an integral part of the vessel. Except for the lower support structure and fuel transfer rack, all components shall be removable.

1.3 System Requirements

1.3.1 Performance Requirements

1.3.1.1 Structural Support and Alignment

The system shall be designed to assure adequate support and alignment of the core matrix for reliable control rod operation, to insure that reactivity changes due to load and thermo-mechanical deflections have direction and magnitude compatible with safety requirements, and to allow fuel, blanket, and control element insertion or removal from the matrix without undue force requirements.

The core array consists of a vertical array of 274 fuel elements, 15 control rod elements and 138 blanket elements as shown on Figure 3. These are surrounded by a single row of 78 reflector elements which are a part of this system. Each fuel and blanket element weighs approximately 500 lbs and each control absorber element weighs approximately 400 lbs.

The core element array shall be supported on a grid plate structure, having its lattice spacing within a tolerance compatible with the nuclear and safety requirements and shall be constrained radially, sufficiently to prevent an unacceptable positive contribution to the reactor power coefficient as a result of any core element displacement including thermal bowing of the element housings.

Fifteen in-vessel transfer positions for core elements shall be provided as shown on Figure 3. Normally these will be used for temporary storage prior to transfer of elements out of the vessel or into the core. If required a spent element can be stored through an operating cycle.

1.3.1.2 Coolant Flow & Distribution

The system shall accommodate a total reactor coolant (sodium) flow rate of 75.58×10^6 lb/hr at nominal temperatures of 780°F inlet and 1140°F outlet with a pressure drop across the fuel and blanket elements of about 85 psi. Pressure drop attributed to feed of coolant from vessel inlets to the high pressure plenum and to distribution of coolant within the plenum shall be minimized.

Low pressure coolant shall be supplied to the control channels, reflector elements, neutron shields, stored elements, vessel walls and other internal structure by leakage from the high pressure plenum plus externally controlled feed into the plenum under the core support structure. This coolant shall flow upward, through or along the components, using thermal convection as the primary driving force.

1.3.1.3 Neutron Reflector and Shielding

A single row of solid stainless steel reflector elements shall be provided to reflect neutrons back into the blanket region, thereby increasing breeding and lowering radiation damage and heating outside the core. Additional neutron shielding shall be provided around the core so that 1) the decay heat generation rate in a spent fuel element stored in the reactor vessel shall not exceed 3 KW due to fission products born in storage, 2) the neutron flux at the reactor vessel is limited to 10^{23} nvt fast ($E > 1$ Mev) in 30 years of operation and 3) reduce the generation of heat in the reactor cavity walls.

1.3.2 Environmental Requirements

1.3.2.1 Operating Sequence of Events

The system is exposed to the radiation, thermal, mechanical, and pressure environments produced during plant preheat, startup, operation (including load swings), refueling and maintenance shutdown and restarts through its design life of 30 years. In addition various abnormal events, protected and unprotected, may occur during this period which will severely alter the normal environments. These events shall be carefully analyzed to insure that design margin to meet performance and safety requirements is provided.

1.3.2.2 Atmosphere

All of the reactor vessel internals operate submerged in the sodium coolant which will contain 10 ppm oxygen plus small concentrations of other impurities.

1.3.2.3 Pressures

1. High Pressure - The high pressure sodium will be delivered to the reactor at approximately 120 psig during full power operation.
2. Low Pressure - The pressure will be equal to the reactor cover gas pressure (@.3 psig) plus sodium static head (zero to 38 ft).

1.3.2.4

Temperatures

1. Steady State

Inlet sodium - 780°F
Outlet sodium - 1140°F

2. Transients

See SDD-096-20-000

1.3.2.5

Radiation Flux

1. Radial - The inner surface of the thermal shield will be exposed to the following neutron spectrum.

Neutron Spectrum

$\frac{\text{Flux}}{n/cm^2}$ -sec

$\frac{\text{Dose}}{nvt/30 \text{ yrs}}$

Fast ($E > 1 \text{ Mev}$)

Epithermal ($1 \text{ Mev} > E > 27 \text{ ev}$) (to be determined)

Thermal ($E < 27 \text{ ev}$)

2. Axial - The top surface of the upper grid plate will be exposed to the following neutron spectrum.

Neutron Spectrum

$\frac{\text{Flux}}{n/cm^2}$ -sec

$\frac{\text{Dose}}{nvt/30 \text{ yrs}}$

Fast ($E > 1 \text{ Mev}$)

Epithermal ($1 \text{ Mev} > E > 27 \text{ ev}$) (to be determined)

Thermal ($E < 27 \text{ ev}$)

1.3.3

Reliability/Availability Requirements

a. Objective

The 30-year unavailability objective for the vessel internals is (to be determined).

b. Design Considerations

1. Provide design margins that meet the stated reliability objectives.
2. Consider methods that minimize thermal shocks and gradients to reduce the likelihood of fatigue failures and minimize component distortions that can cause core element misalignment and/or jamming.
3. Consider methods that minimize the possibility of collection of gas in the high pressure inlet plenum which could subsequently slug through the fuel or blanket channels.

1.3.4

Maintenance Requirements

1.3.4.1 Component Replacement

The grid plate structure plus the neutron shields and reflector elements shall be designed as a unitized assembly which can be detached from the lower support structure and removed from the vessel through the 15'-6" diameter opening provided in the top shield when the small rotating plug is removed. Devices for holddown of the assembly, for attachment of hoisting slings, etc., shall be designed for operation remotely from the top face of the shield. The design should allow the replacement to be accomplished in a time period that will not require the total shutdown period to exceed one year.

1.3.5 Operational Requirements

1.3.5.1 Reflector Rotation

The reflector elements shall be capable of being rotated or replaced, as required, to prevent irradiation induced swelling distortions from interfering with maintenance of a tight core, proper alignment of reactor elements, or removal and replacement of fuel, blanket and control elements.

1.3.5.2 Preheat

All components shall be designed to allow a primary system pre-heat rate of 3°F per hour

1.3.6 Safety Requirements

1.3.6.1 Gas Collection

Gas de-entrainment, collection in the inlet area of the reactor, and subsequent slugging through the core can cause cladding damage and significant positive reactivity insertion. The design shall consider methods to collect and continuously vent any gas in the fuel and blanket element intake area.

1.3.6.2 Fuel Element Location

The grid plates, in conjunction with the fuel and blanket elements of SDD-096-28-000 shall provide a positive means of precluding incorrect fuel or blanket location with respect to enrichment and orificing.

1.3.6.3 Fuel Element Motion

The reactor support system shall provide constraint to assure controlled core element motion during temperature or power change such that the reactivity is negative, or that any positive contribution does not exceed the net negative insertion due to Doppler plus sodium coefficients.

1.3.7 Design Requirements

1.3.7.1 Applicable Codes and Standards

To the extent applicable the reactor vessel internals shall be designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code Section VIII, Unfired Pressure Vessels. No code stamp shall be required.

1.3.7.2 Seismic Design Criteria

All components shall be designed as Class 1A structures (for installation at any location) as specified in SDD-096-10-000.

1.3.7.3 Design Margins

1. Life - All components of this system shall be designed for a 30-year operational life.
2. Stress - The system shall be analyzed for operational stress and deflection within the intent of the code listed in 1.3.7.1 (see para. 1.3.7.3).
3. Analysis - The system shall be analyzed to meet the stated requirements on the basis of the following margins.
 - a) Transients - The effects of various transients shall be considered during the analysis of the system design, and where applicable, margins shall be applied to assure a safe/reliable design.
 - b) Operational Tolerances - Operational tolerances shall be considered during the analysis of the system design, and where applicable, appropriate margins shall be included in the design.
 - c) Uncertainties - The level of uncertainty associated with each analysis method shall be considered and factored into the design to assure that the safety and reliability margins are truly experienced.

1.3.7.4 Materials

- 1) Sodium Contact - All materials in contact with sodium shall be type 304 stainless steel except where special requirements, such as wear or adhesion characteristics, require special materials. In the selection of special materials due consideration shall be given to mass transfer and sodium corrosion.

2) Allowable Stresses (304 stainless steel)

- a) Normal Operation -- Allowable stresses, taking into account stress to rupture in 250,000 hours, stress to produce 1% creep at 250,000 hours, and the effect of irradiation to 10^{23} nvt, are as follows:

<u>Temperature</u>	<u>Allowable Stress</u>
800°F	9000 psi
900	8800
1000	7200
1100	3900
1200	2400

- b) Short Term Mechanical Loads - Allowable stresses shall be in accordance with the ASME Code Section VIII, Unfired Pressure Vessels.

1.3.7.5 Design Features1.3.7.5.1 High Pressure Components

All components subject to loading by the high pressure coolant shall be designed to withstand 150 psig at 800°F for 30 years.

1.3.7.5.2 High Pressure Leakage

Core bypass leakage of high pressure coolant shall not exceed 2% of the inlet flow.

1.3.7.5.3 Fuel Storage

In-vessel transfer positions for 15 elements shall be provided. The spacing and general arrangement of the positions shall be compatible with the fuel handling requirements and shall result in a K_{eff} no greater than 0.9 including the effect of coupling with the core when the reactor is at full power.

1.3.7.5.4 Core Element Lattice

The core elements shall be positioned on a nominal 5.667-inch triangular lattice spacing.

1.3.7.5.5 Core Element Holddown

Holddown of core elements subjected to high pressure coolant shall be of the hydraulic piston type. Holddown of all other reactor elements shall be by gravity.

1.3.8 Interface Requirements

The reactor vessel internals, a part of the reactor vessel assembly (SDD-096-21-000) of the reactor segment (SDD-096-20-000) has major interfaces with the following components or assemblies within the reactor segment.

1. SDD-096-28-000 - Fuel and Blanket Elements
2. SDD-096-22-000 - Reactivity Controls

2. DETAILED DESCRIPTION

2.1 Components

The reactor vessel internals include the following components shown on Figures 1 and 3.

- a) Lower Support Structure
- b) Grid Plate Assembly
- c) Radial Neutron Shield and Core Constraint Assembly
- d) Reflector Elements
- e) Fuel Transfer Rack and Storage Canisters
- f) Flow baffles and thermal barriers not constructed as an integral part of the vessel.

2.2 Description of Components

2.2.1 Lower Support Structure

The lower support structure is a stiffened circular plate structure which spans across the vessel near the bottom. It is attached to a support flange on the inside diameter of the vessel and in turn provides the basis support for the grid plate structure, reflector elements, neutron shields and core constraint mechanisms, and the fuel transfer rack. The coolant feed lines which carry the high pressure coolant from the three vessel inlet nozzles to the outer barrel of the high pressure plenum are constructed as an integral part of this structure as shown on Figure 1.

The high pressure plenum outer barrel, together with the grid plate structure, contains the high pressure inlet coolant which flows into the grid plate structure and then into the lower extensions of the fuel and blanket elements.

The center of the core matrix is positioned 18 inches from the center of the reactor vessel and 15 fuel transfer positions are arranged in an arc, concentric about the vessel center, as shown on Figure 3.

The lower support structure is stiffened, using integral vertical web members, so that its deflection does not exceed .030 in. The maximum exposure of this structure to fast neutrons ($E \geq 1$ Mev) after 30 years does not exceed 1×10^{21} nvt. It is considered a non-removable component.

2.2.2 Grid Plate Structure

The grid plate structure provides the mechanical support for the individual reactor elements plus the neutron shields and core constraint device. It is considered a replaceable component and can be removed from the reactor through the opening provided when the small rotating top shield plus is removed.

It consists of an upper and lower grid plate spaced 30 inches apart and coupled together by an outer cylindrical cylinder and 4-inch ID x 1/2-inch thick tubular stays at each of the 15 control rod positions. A radial extension of the upper grid plate forms a mounting and sealing flange which mates with a flange on the plenum barrel of the lower support structure. A lower extension of the outer cylinder has a relatively close fit into an inner hub of the lower support structure. The bottom-most plate shown as part of the grid plate structure on Figure 1 carries go/no-go cams which prevent full seating of any element except one having the correct enrichment and/or orificing for that position.

The upper and lower grid plates are drilled in a 5.667-inch triangular pitch pattern for the core elements. The lower extensions of the fuel and blanket elements have a close fit through the upper grid plate and a ball-seating into a conical socket in the lower grid plate. The design of the lower socket exposes the bottom end of the fuel and blanket element extensions to the low pressure below the grid plate so that the high pressure within the lower extension creates a hydraulic holddown force on the element.

The lower extensions of the control element guide tubes are inserted into the tubular stays described above and are thereby isolated from the high pressure coolant. The reflector elements are ball socketed into the upper surface of the top grid plate.

The outer cylinder of the grid plate structure has large openings to allow flow into the grid plate chamber with minimum pressure drop. Placement of these openings will be designed to minimize the effects of inlet velocity on the flow into peripheral core elements. There will be leakage of high pressure coolant to the upper part of the vessel through the clearance between the upper grid plate holes and the core element lower extensions; and to the lower part of the reactor vessel through the annular gap between grid plate structure and the support structure. This leakage is considered as part of the low pressure coolant supply.

The maximum exposure of the upper grid plate to fast neutrons ($E \geq 1$ Mev) is 1×10^{21} nvt in 30 years of operation.

2.2.3 Reflector Elements

The reflector elements form the outer row of elements in the reactor matrix. They are 5.50-inch OD circular bars extending upward from the top of the upper grid plate to the top of the core matrix. These elements are a part of the radial neutron shielding and as such attenuate by a factor of about 5. They are also an integral part of the radial core constraint. Cooling is provided by grid plate leakage and flows upward along the elements by natural convection.

2.2.4 Radial Neutron Shield

The radial neutron shield consists basically of three 2.5-inch thicknesses of stainless steel plate, with $3/4$ -inch separation for coolant flow, surrounding the reflector elements and core matrix. This limits the exposure of the reactor vessel wall to neutrons during reactor operation. Cooling is by thermal convection circulation of low pressure coolant.

2.2.5 Core Constraint System

The lateral support and constraint for the reactor matrix is shown in Figure 1. The reactor elements are supported at the lower extension on conical seats in the lower grid plate. The extensions pass through holes in the upper grid plate with a radial clearance of 2.5 to 5.0 mils. Radial constraint is provided in a clamping plane which is above the central plane of the reactor core. The clamping plane must be located above the core center to insure a negative contribution to the power coefficient as a result of element bowing. The individual elements make contact in this plane at weld-overlay spacer pads at the corners of the element housings.

The core radial support structure consists of the section of the middle neutron shield up to the clamping plane. It is secured to the upper grid plate by a heavy flange and is slotted axially, nearly to the lower flange, to provide a ring of cantilevered fingers, one for each reflector element. A pressure shoe attached to the top end of each finger bears against a reflector element. The reflector element is seated in a socket in the upper grid plate and is held in a vertical position by a bolt attaching it to the finger.

The unsprung position of the top end of the fingers is nominally that established by the lower flange which is at inlet temperature. When the reactor is generating significant power the element housings will be at a higher temperature in the clamping plane. The core matrix expands by exerting pressure against and deflecting the clamping fingers. This arrangement thereby maintains the core matrix in a tightly packed configuration during temperature and power changes.

At the top end of the matrix a rigid restraint ring is attached to the top of a neutron shield. This ring allows outward movement of the top ends of the core and blanket elements (reflector elements not restrained) in accordance with designed clearance and thermal expansion of the restraint ring which is at reactor outlet temperature.

2.2.6 Fuel Transfer Rack & Canisters

As shown on Figures 1 and 3, 15 fuel transfer positions are arranged in an arc which is concentric about the vessel centerline. The ex-vessel fuel handling machine services these positions through an access port in the large rotating plug of the top shield at the same radius from the vessel center. The in-vessel fuel handling machine is located over these positions by combined rotation of the large and small rotating shield plugs.

Each transfer position consists of a tubular member which will support a 7.75-inch OD fuel storage canister. The channel has flow openings at the bottom (and possibly perforations along its length) to allow coolant to circulate upward around the canister by natural convection. A supporting and positioning structure maintains the top end of the channel at a fixed position to which the fuel handling machines can be aligned.

The fuel elements will be placed in 7.75-inch OD canisters which are closed at the bottom and open at the top for transfer into or out of the vessel. The top end of the canister has a top fitting with an internal gripping lip for the ex-vessel fuel handling machine, an internal flow passage siphons off the top 1 to 2 inches of sodium as the canister is removed from the pool, and is designed to accept a weld-on seal cap for shipment of spent fuel from the reactor site.

The elevation of the core elements in the transfer rack must be the same as in the core matrix because the in-core fuel handling machine grapple is interlocked to release only at a pre-established elevation.

Heat generated by the spent fuel is transferred to the canister wall by natural circulation of the contained sodium and to sodium which originates as leakage from the high pressure plenum to the lower chamber of the vessel and then circulates upward through the storage channels by thermal convection.

The spent fuel canisters are not a permanent part of the reactor installation. In a typical fuel change cycle a canister containing a new fuel element is placed in a transfer position by the ex-vessel fuel handling machine. The in-core FHM then removes a spent element from the lattice, transfers it to an empty canister in the transfer rack, and then moves the new element into the core lattice. The provision of 15 transfer positions allows flexibility in sequencing the fuel change operations and also allows up to 14 core elements to be held in the vessel until the next refueling shutdown should it be desirable to do so.

REVISIONS
DATE APPROVED
DATE REVISION
SDD-096-21-300
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REFERENCE DRAWINGS
N03620022 REACTOR-PLAN VIEW
N03620023 REACTOR-PLAN SECTION

FIG. 1

ATOMIC INTERNATIONAL
A DIVISION OF BETHLEHEM STEEL CORPORATION
REACTOR - VERTICAL SECTION
1000 MW LMFBR FOLLOW-ON STUDY
DRAWN BY: J. L. HARRIS
CHECKED BY: J. L. HARRIS
DATE: 10/1/71
SCALE: 1/4\"/>

2.2.7 Flow Baffles and Thermal Barriers

Flow baffles will be provided as required to assure proper distribution and circulation of both the high pressure and the low pressure coolant within the vessel. Specific requirements and/or designs for these are still to be defined.

Thermal barriers will be provided as required to protect the reactor vessel and structural components within the vessel from thermal shock under conditions of rapidly changing sodium temperature. The thermal barriers around the three high pressure coolant feed lines and the 14-inch low pressure feed line reduce the thermal shorting between the cold inlet coolant and the hot pool. These barriers consist of approximately 1/2 in. thick stainless steel plates spaced 1/4 to 1/2 in. away from the surface to provide a stagnant layer of sodium.

2.2.8 Temperature Instrumentation

The temperature of the reactor internal structure will be measured where such information is required to assist in initial reactor operation. The thermocouples will have an expected life, limited by their radiation environment, of 1 to 2 years and will not be replaced. They will be routed out of the reactor vessel through conduits inside the primary sodium piping. The number of thermocouples and their locations have not been determined.

3. PRINCIPLES OF OPERATION

See Section 3 of SDD-096-21-000

4. SAFETY PRECAUTIONS

See Section 4 of SDD-096-21-000

5. MAINTENANCE

See Section 5 of SDD-096-21-000

6. APPENDICES**6.1 Figures**

Figure 1 - Reactor Vertical Section (N09620021)

Figure 2 - Reactor Plan View (N09620022)

Figure 3 - Reactor Plan Section (N09620023)

REVISIONS		DATE	APPROVED
SYM	DESCRIPTION		
A	SEE REVISION RECORD

REFERENCE DRAWING
 N09620021 - REACTOR - VERTICAL SECTION
 N09620022 - REACTOR - PLAN SECTION

FIG. 2

ATOMICS INTERNATIONAL <small>A DIVISION OF - CATERPILLAR AVIATION, INC. 3500 WEST 10TH AVENUE, DENVER, COLORADO 80202</small>	
REACTOR PLAN VIEW 1000 MW(4) LMFBR FOLLOW-ON STUDY	
DESIGNED CHECKED DRAWN APPROVED	DATE SIZE SCALE SHEET NO.
N09620022	

REVISIONS			
SYN	ZONE	DESCRIPTION	DATE
A		SET REVISION RECORD	12-10-67

- CORE LEGEND**
- FAST FUEL - 274
 - ⊗ BLANKET FUEL - 138
 - REFLECTOR - 78
 - ⊙ SHIM SAFETY-CONTROL ROD
 - ⊗ SAFETY-CONTROL ROD (15 TOTAL)

REACTOR DRAWINGS
 NO9620021 REACTOR VERTICAL SECTION
 NO9620022 REACTOR PLAN VIEW

FIG. 3

ATOMICS INTERNATIONAL <small>A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION</small> <small>4900 BILDED AVENUE, CANADA PARK, CALIF.</small>	
REACTOR PLAN SECTION <small>2000 MW (L) LMFBR FOLLOW-ON STUDY</small>	
DRAWN: J. J. F. (J. J. F.) CHECKED: [] APPROVED: []	DWG. NO: NO96.20023 SCALE: 1/2" = 1'-0" SHEET NO: []

ATOMICS INTERNATIONAL*A Division of North American Rockwell Corporation***SUPPORTING DOCUMENT****PROGRAM TITLE**

ANL 1000 Mwe Follow-on Study

DOCUMENT TITLEReactivity Control Conceptual System Design
Description - 1000 Mwe LMFBR Follow-on Study**ABSTRACT**

This SDD lists the requirements for the Reactivity Control System (RCS) for the 1000 Mwe LMFBR and describes the Conceptual Design. The conceptual design described in this document is similar to previous designs developed by Atomics International.

NUMBER

SDD-096-22-000

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System Design Description

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CLASSIFIER****DATE****DISTRIBUTION****REV****SUMMARY OF CHANGE****APPROVALS AND DATE**

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NAME**LOC**

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1.0 INTRODUCTION

In this SDD the following definitions will be used:

Control Rod

General term including both shim-safety and safety rods. It is used in this SDD when there is no need to differentiate between the two types of rods.

Shim-Safety Rods

These rods are used to cover all normal operational requirements of the reactor, and to terminate or mitigate all credible accidents. They combine the functions of shim control of reactivity and quick release for emergency shutdown of the reactor (trip). Since both functions are combined, the design of the shim-safety is subject to all safety, operating requirements, and criteria applicable to either function.

Safety Rods

These rods will be used to cover anomalies outside those expected during normal operation, such as a refueling error and a stuck rod. An important feature of their design is a quick release disconnect, or trip latch, to drop the rods for emergency shutdown. They are always used in conjunction with another rod bank so that the reactor multiplication factor will be less than one when the safety rods are withdrawn to their position of least worth. That is, the safety rods are not used to bring the reactor to criticality.

Footnotes

- a) "Representative value." Calculations have not yet been made for the 1000 Mwe ANL study, but the values given are believed to be representative.
- b) "To be determined." Values not yet calculated for 1000 Mwe study. Values from other programs may not be representative.

1.1 Functional Requirements

The functions of the Reactivity Control System (RCS) are:

- 1) To control the reactivity of the reactor core, hence flux and power
 - a) during startup, normal power operation and scheduled shutdown, in conjunction with the Plant Control System, and
 - b) for emergency shutdown, in conjunction with the Plant Protective System.

1.2 Summary Description

1.2.1 System

The control rods are similar to previous designs developed by Atomics International. Each assembly consists of:

- a) a neutron-absorbing "poison" element, or absorber section, which is positioned vertically in the core to change the neutron multiplication rate (reactivity) by varying the number of parasitic absorptions according to the amount of insertion
- b) the absorber guide tube, inside which the absorber section is guided in the core
- c) drive mechanisms (a power unit and a rotary-to-linear motion converter), to adjust the position of the absorber element in the core
- d) a trip magnet and trip latch to release the poison element for emergency shutdown (trip)
- e) a drive extension, connecting the trip magnet and latch, and the drive extension housing

- f) a hydraulic snubber to absorb the kinetic energy of the falling rod at the end of its drop following trip, and
- g) instrumentation to continuously monitor the position of the rod in the core, trip latch condition, drive load, and control rod coolant outlet temperature.

The assembly of absorber element and associated mechanisms can perform both control and shutdown functions.

The system reference design is an array of 15 such control rods, located in the core as shown in Figure 1, and having a total reactivity worth of \$28.49. The rods are divided into two banks. Nine rods, positioned at the boundary of the inner and outer fuel zone of the core and having a total worth of \$11.74 are designated as shim-safeties. The lower ends of these rods, at the top of their travel, are at the core-blanket interface. Six rods, positioned in the fourth ring of the inner fuel zone of the core and having a total worth of \$16.75 are designated as safeties. These rods are withdrawn from the core before reactor startup. During operation, the lower ends of these rods are at the top of the upper blanket. All rods are designed to be capable of driving at both normal control and setback speeds; however, only the shim-safeties will have the motors installed for setback (rapid reduction of power without tripping). All rods will be dropped on reactor trip.

1.2.2 Control Rod

Tantalum is used as the poison material in both absorber sections. The absorber sections are of the multiple-pin type with NaK-bonded stainless-clad Ta pins. They are cooled by thermosyphon-induced sodium flow through the absorber guide tube from the low pressure plenum. Since only the absorber section is dropped during scram, it has at its top a handling head which is gripped by the trip latch during normal operation.

The rod drive is of the electromechanical type using two electrical motors, a stepper and a dc motor mounted on the same shaft. The power unit drives a ball-screw through a gear reducer. A trip magnet is attached to the lower end of the screw through a load cell coupling. The entire drive mechanism including motors, gear box, seals, position sensors, limit switches, and load cell operate in a cool, low-radiation environment and are easily accessible for maintenance or adjustment.

The drive extension is connected to the ball-screw with a bayonet-type coupling and extends through the top shield, cover gas, and sodium pool to the top of the absorber section in the reactor core. It consists of:

- 1) two concentric shafts, an outer drive shaft and an inner trip-latch actuating shaft
- 2) sodium and sodium vapor seals consisting of bellows for sealing the annulus between the trip-latch shaft and main drive tube, and a helium gas purge as well as elastomer seals in cooler areas far removed from the sodium pool
- 3) the trip latch, mounted on the lower end, which grasps the absorber, and
- 4) the housing which establishes alignment between the drive and the absorber guide tube, and inside which the drive tube operates.

The drive tube is surrounded by a double-walled housing

which contains thermal and radiation shielding where it passes through the top shield. Gas pressure in the drive and extension housings is maintained slightly above the static pressure of the reactor cover gas so that if a leak should develop in either the immersed bellows or other seals, entrance of sodium or vapor into the drive extension is minimized and operation may continue until replacement is convenient.

1.2.3 Major Components

The major components of a control-safety rod are:

1. Control Rod Drive
 - a. Actuator
 - (1) Power Units
 - (2) Rotary-to-Linear Motion Converter (ball-screw)
 - b. Trip Magnet
 - c. Instrumentation
 - d. Drive Housing
3. Brake and Clutch
2. Control Drive Extension
 - a. Main Shaft
 - b. Latch Actuating Shaft
 - c. Trip Latch

d. Sodium and Gas Seals

(1) Reciprocating Sodium Seals

(a) Trip Latch Bellows

(b) Gas Purge System & Drive Tube Shaft Seals

(2) Static Gas Seals (O-Rings)

e. Thermal and Radiation Shielding

f. Drive Extension Housing

g. Holddown Tube (may be deleted)

3. Absorber Section

a. Poison Pins

b. Hanger Rod and Handling Head

c. Snubber Assembly

d. Guide Tube with Handling Fitting

1.3 Requirements

1.3.1 Performance Requirements

1.3.1.1 Total reactivity worth of system: \$2849. This is the upper 3 σ limit estimate. See Table 1 for worth requirements.

1.3.1.2 Total number of control-safety rods in system: fifteen (for location of rods in core see Figure 1).

1.3.1.3 Number of rod banks in system: the Reactivity Control System shall consist of two rod banks, a shim-safety bank, and a safety bank.

1.3.1.4 Functions of Rod Banks

1.3.1.4.1 Shim Safety Bank

The shim-safety bank shall be used to cover all normal operating requirements of the reactor. It shall be capable of terminating or mitigating all credible accidents.

1.3.1.4.2 Safety Bank

The safety bank shall be used to cover only those reactivity anomalies outside the normal operating requirements, such as a refueling error or a stuck rod.

- 1.3.1.5 Reactivity Worth of Rod Banks
- 1.3.1.5.1 Shim-Safety Bank: \$11.74. This is the upper 3σ limit.
- 1.3.1.5.2 Safety Bank: \$16.75. This is the upper 3σ limit.
- 1.3.1.6 Maximum Allowable Worth of a Single Rod
- 1.3.1.6.1 Shim-Safety: value to be determined. Limited by 1.3.6.2.
- 1.3.1.6.2 Safety: no limit.
- 1.3.1.7 Maximum allowable rate of insertion of positive reactivity during startup and normal control: value to be determined. Limited by 1.3.6.3.
- 1.3.1.8 System Response During Reactor Trip (Scram): Note Credit will be taken for the contribution of the shim-safety bank only.
- 1.3.1.8.1 Response time (time from interruption of power to scram magnet to initiation of negative insertion): 0.1 second maximum.†
- 1.3.1.8.2 Negative reactivity insertion rate: to be determined.
- 1.3.1.9 System Response During Fast Setback
- A second level of safety action, in addition to reactor trip, shall be used to reduce power in the event reactor operating parameters (flux, period, temperature, etc.) exceed allowable limits. This action, termed fast setback, shall be initiated prior to trip, to minimize the number of severe thermal transients to which the reactor is subjected. Only the shim-safety bank shall be used for fast setback.
- 1.3.1.9.1 Response Time (Time from receipt of signal to reach 95% of nominal insertion rate): 300 milliseconds.†
- 1.3.1.9.2 Negative Reactivity Insertion Rate: to be determined.
- 1.3.1.10 Control Dead-band: $\pm 0.5 \times 10^{-3}\% \Delta k/k$ (0.15¢).†
Based on 0.1% power dead-band at 10% flow.
- 1.3.1.11 Rate of change of Reactivity for Load Following: 0.00255% $\Delta k/k$ -sec (0.77¢/sec) minimum.† Note: For determination of rod drive rate requirement, a 3-rod bank was assumed.

† Preliminary value

- 1.3.1.12 Loss of Worth Due to Burnout
- 1.3.1.12.1 Individual Rod
- Loss of reactivity worth of any individual rod due to burnout during its design life shall not exceed 10% of its original worth.
- 1.3.1.13 Control Rod Requirements
- 1.3.1.13.1 Poison Material Length
- 1.3.1.13.1.1 Shim Safety Rod: 43 inches
- 1.3.1.13.1.2 Safety Rod: 31 inches
- 1.3.1.13.2 Control Drive Travel: 50 inches. Nominal stroke for both rods is 43 inches. Additional travel of 7 inches will be necessary to lift the drive extension and holddown tube above the top of the core to allow shield rotation for refueling. Drive overtravel beyond stroke limits shall be prevented by physical stops.
- 1.3.1.13.3 Absorber Position Limits
- 1.3.1.13.3.1 Shim-Safety Rod: at the uppermost position of the absorber section, the bottom of the poison material is at the core upper blanket interface. At its lowermost position, the bottom of the poison material is at the bottom of the active core.
- 1.3.1.13.3.2 Safety Rod: at its uppermost position of the absorber section, the bottom of the poison material is at the top of the upper blanket. At its lowermost position, the bottom of the poison material is one foot above the bottom of the active core.
- 1.3.1.13.4 Drive Requirements
- 1.3.1.13.4.1 Motor Normal Drive: this slow drive motor, which must be reversible, is used to bring the reactor to criticality and power and to regulate power during normal operation. The max speed shall be 7.5 in./min.*
- 1.3.1.13.4.1.1 Response Time: time to reach 95% maximum speed in or out after power is applied to motor shall be 0.2 sec.* Downward adjustability by a factor of 3 to 1 (to 2.5 inch/min) at site is required. Variation in normal drive speed during operation shall be prevented by

* Representative value

mechanical means. Note Based on meeting requirement of 1.3.1.7 with 9 shim-safeties in their maximum worth position.

1.3.1.13.4.1.3 Minimum Incremental Rod Motion ("Jogging Action"): the maximum motion per step shall be no greater than 0.004[†] inch.

Note Set by control band requirement of 1.3.1.10 and assuming 9-rod regulating bank.

1.3.1.13.4.2 Fast Setback Requirements

1.3.1.13.4.2.1 Setback Drive Response Time

Acceleration: the time to reach nominal speed after power is applied to the motor shall be no greater than 250 ms.[†]

Braking: the time to halt the drive when traveling at maximum speed is 500 ms[†] after power is applied to the brake.

1.3.1.13.4.2.2 Fast Setback Drive Speed

The maximum speed shall be at least 15 ft/min.[†] The nominal speed is 10 ft/min[†] based on meeting or exceeding the requirements of 1.3.1.9.2 when the shim safety rods are in their minimum worth positions.

1.3.1.13.5 Reactor Trip

This mode is used to shut the reactor down should any operating parameter exceed allowable limits. Trip action is achieved by releasing the absorber element from the drive extension and dropping it into the core.

1.3.1.13.5.1 Fail-Safe Feature

When the trip magnet is de-energized, the control-safety rod absorber element shall drop into the core.

1.3.1.13.5.2 Trip Magnet and Trip Latch Response Time

(Time from interruption of magnet power to signal indicating latch release) 0.1 sec maximum.[†]

[†] Representative value

1.3.1.13.5.3 Rod Drop Characteristics

1.3.1.13.5.3.1 Minimum average acceleration before snubbing starts - 0.5g*.

1.3.1.13.5.3.2 Minimum free fall distance from fully withdrawn position (total distance rods must fall before snubbing starts) 3 ft.*

1.3.1.13.5.3.3 Trip Capability versus Rod Position

Rods must be capable of being dropped from any position in the core (including the snubbing region) without hangup.

1.3.1.13.5.4 Drive Interchangeability

The drives and drive extensions for all rods shall be identical and interchangeable.

1.3.1.13.5.5 Seal Design Philosophy

The latch actuating rod seals include double bellows and a gas purge backup. The main drive rod is sealed by a helium gas purge backed up by double shaft seals. The alternate seal design for the main drive rod is a full stroke (see 1.3.1.13.2) bellows and the design philosophy to be followed shall be to allow installation of this bellows at any time with a minimum of design modification of the assembly. Standard design practice for all top shield penetrations including the control-safety rods is that there shall be double seals with leak test capability between radioactive sodium or cover gas and any outside environment to which personnel have access. Since the drive housing may be removed for maintenance during refueling, no credit for drive housing seals may be taken in meeting this requirement.

1.3.1.13.6 Instrumentation

1.3.1.13.6.1 Rod Position Indication

Indication of the position of the absorber section in the core shall be provided at all times when the reactor is operating. Momentary loss of position signal during trip is permissible but must be

* Representative value

re-established before startup of the reactor.

1.3.1.13.6.1.1 Absolute Position Indication (as measured at the drive screw) ± 0.030 inch.*

1.3.1.13.6.1.2 Accuracy of Differential Rod Position Measurements (as measured at the drive screw) ± 0.010 inch.*

1.3.1.13.6.1.3 Limit Switches

Redundant limit switches shall be provided to indicate the following:

Absorber and drive extension full-in

Setback cut-off

Absorber full-out (43 inches withdrawn)

Drive extension overtravel limit (50 inches)

1.3.1.13.6.1.4 Absorber Drop Time Measurements

Provision shall be made to allow measurement of absorber drop time at any time during the life of the plant. Drop time is defined as the time from latch release to impact upon entering the snubber.

Special instrumentation may be arranged near the drive for this measurement. Measurement of drop times for individual absorbers is not required when absorbers are dropped as a group; only measurement of first and last impact times is required.

1.3.1.13.6.2 Coolant Outlet Temperature Measurement

Redundant temperature detectors shall be provided to continuously measure absorber section coolant outlet temperature within $\pm 15^\circ\text{F}$.

1.3.1.13.6.3 Drive Load

Provision shall be made to continuously measure the load on the control rod drive with an accuracy of ± 20 lbs.

1.3.1.13.6.4 Integrity of Drive and Drive Extension Seals

Means shall be provided for continuous monitoring of the integrity of the seals in the rod drive and drive extension and of the flow rate through the gas purge seal system.

*Representative values

1.3.1.13.6.5 Trip Latch Position

Provision shall be made for measurement of trip latch position, i.e., open or closed.

1.3.1.13.7 Design Life

The design life of the equipment shall be as follows:

- 1) Rod drive and instrumentation
30 years, with periodic maintenance
- 2) Drive extension
10 years (design objective 30 years)
- 3) Absorber section and absorber guide tube
8 years (design objective, 10 years)

1.3.1.13.8 Duty Cycle**1.3.1.13.8.1 Average Number of Rod Drops and Double Drive Strokes**

Cycles per year

25 for the first 10 years of plant life
10 for subsequent years

1.3.1.13.8.2 Design Total Number of Rod Drops and Double Drive Strokes

- 1) Rod Drive 500
(30-year design life)
- 2) Drive Extension 250
(10-year design life)
- 3) Absorber Section and Core Housing 250
(10-year design objective)

1.3.1.13.8.3 Regulating Cycles

Note: Defined as a 1/4-inch motion in one direction, followed immediately by a 1/4-inch motion in the opposite direction distributed evenly over upper 1/3 of normal stroke.

1.3.1.13.8.3.1 Per year 4×10^5 cycles**1.3.1.13.8.3.2 Design total (based on use of all 9 shim-safety rods operated as a bank for regulation)****1.3.1.13.8.3.2.1 Absorber Section and Guide Tube
 3.2×10^6** **1.3.1.13.8.3.2.2 Drive Extension
 4×10^6** **1.3.1.13.8.3.2.3 Drive
 1.2×10^7**

1.3.1.13.8.4 Refueling Cycles

Note: To lift holddown tube for fuel handling (50-inch stroke).

1.3.1.13.8.4.1 Drive Extension
40 over 10-year design life

1.3.1.13.8.4.2 Drive
120 over 30-year design life

1.3.1.13.9 Control Drive Design Load

Nominal load, including drive extension and absorber section is 500 lb* (tension) maximum compression load to dislodge a stuck rod is estimated at 750 lb.* Slip clutch overload protection should be set at this latter figure.

1.3.1.13.10 Maximum Cover Gas Leak for 15 Control-Safety Rods**1.3.1.13.10.1 During Normal Operation and Refueling***

The allowable leak rate is set by the amount of Xe^{133} which can be allowed to escape based on limiting high bay activity to 0.1 mpc during refueling. The maximum allowable leak rate allotted to the control rod seals is 2.2×10^{-8} SCF of Xe^{133} /day, based on the assumption of 1% fuel pin failures and a reactor cover gas activity of 2.4×10^{-4} curies of Xe^{133} per cc and a maximum leakage requirement of 29 curies Xe^{133} /day per curies Xe^{133} /cc of helium cover gas. This can also be expressed as 1×10^{-3} ft³ Xe /day per ft³ Xe /ft³ of helium cover gas. For the purposes of testing, the leakage can also be expressed as 29 cc Xe (STP)/day with pure xenon at one atmosphere pressure on the cover gas side of the seal.

1.3.1.13.11 Maximum misalignment

The control-safety rod shall be capable of normal operation (driving or dropping rod) with a minimum misalignment, between drive and absorber section guide tube (displacement of holddown tube-absorber guide tube interface) approximately 1 to 2 inches.*

* Representative value

1.3.2 Environment

1.3.2.1 Operation Sequence of Events

The control rods are exposed to the radiation, thermal, mechanical, and pressure environment produced during plant preheat, startup, operation (including load swings), trip, refueling and maintenance shutdowns and restarts throughout their design life. Since the control-safety rod assembly extends from above the biological shield through the shield, cover-gas space, outlet coolant sodium pool and the reactor core, the environment varies greatly along its length.

Number of Cycles	<u>10 Years</u>	<u>30 Years</u>
Preheat	1	3
Startups	1	3
Load Swings	3000	10,000
Refueling Shutdowns	10	30
Maintenance Shutdowns	10	30

1.3.2.1.1 Normal Operation

The control-safety rod temperature environment at 100% power is given in Tables (to be determined). The radiation environment at 100% power is given in Figures (to be determined).

1.3.2.1.2 Shutdown

During shutdown the reactor sodium will be maintained approximately isothermal at 500°F. Radiation exposure will be negligible compared to normal operation.

1.3.2.2 Thermal Shocks

During reactor trip the control-safety rod absorber section will experience a thermal shock of approximately 400°F in 0.7 sec.* The trip latch will experience a transient from 1140°F of about 100°F in 25 sec,* as the drive extension moves down to re-engage the absorber section.

*Representative value

1.3.2.2.1 Accident Condition Transients

To be determined.

1.3.2.3 Sodium

The sodium environment of the control-safety rod is reactor grade with the following levels of contaminants:

(a) Oxygen	15 ppm max 10 ppm design objective
(b) Carbon	20 ppm max
(c) Hydrogen	0.002 ppm

1.3.2.4 Gas

1.3.2.4.1 Cover Gas

The cover gas is helium, saturated with sodium vapor and containing fission products under the following temperatures and pressures.

Temperature

Operating	1140°F nominal
Shutdown	500°F

Pressure

Normal (both operating and shutdown)	10 inches H ₂ O
Transient (both operating and shutdown)	5 psig for 1 hour*

1.3.2.4.2 Purge Gas

During operation the drive extension will be continuously purged with either fresh reactor grade helium or recycled cover gas helium purified to the composition given below. Either gas will be supplied at ambient temperature and a pressure of (to be determined).

1.3.2.4.3 Pressurization Gas

The control-safety rod drives will be pressurized with reactor grade helium which will be maintained at least (to be determined) psi above the cover gas pressure during operation.

* Representative value

1.3.2.4.4 External Environment (Containment Dome and High Bay)*

Temperature: 50 - 150°F, nominal 100°F

Pressure - Normal: $-1/4$ inch Hg with respect to atmosphere outside building

Test (Containment Dome): To be determined

Humidity: 50% relative

Atmosphere: air or nitrogen

1.3.2.5 Loads

1.3.2.5.1 Power Operation

The control rod shall be designed to withstand the loads expected due to the misalignments given in 1.3.1.13.11.

1.3.2.5.2 Seismic Loads

The control-safety rod shall be designed to withstand, without impairment of function or loss of sealing ability, a horizontal ground acceleration of 0.33 g, plus a vertical acceleration of 0.22 g in combination with all normal loads.

1.3.2.5.3 DBA Loads

The drive shall be designed to prevent any part of the control rod from becoming a missile when subjected to the following DBA loads. A DBA can occur only while the reactor is operating; hence, the drive will always be bolted down and the containment dome will be in place during a DBA. Loss of control-safety rod sealing capability and gross damage to the rods are acceptable. The rods are not required to be operable following a DBA.

1. TNT Rationale (1300 # TNT, 14 ft from blast center)

a) Shockwave Energy Density

$$e = 16000 \text{ in.-lb/in.}^2$$

b) Shockwave Pressure

$$P_s = 16,000 \text{ psi}$$

c) Residual Pressure (assumes that shockwave ruptures vessel as per design)

$$P_r = 20 \text{ psig}$$

* Representative value

2. Na Vapor Expansion (No shock wave)**a) Peak Pressure**

$$P_p = 8,000 \text{ psi}$$

b) Impulse

$$I = 70 \text{ psi-sec}$$

1.3.3 Reliability/Availability Requirements**1.3.3.1 Reliability**

The control rods serve three different primary functions.

The first and most important function is to provide the safe shutdown of the reactor in response to signals from the plant protective system plus fail-safe features in the design and operation of the rods themselves. This fail-safe criterion can be expected to account for some loss in energy availability through spurious trips which will be charged against the plant protective system. The system objective will be a reliability in excess of 0.9999 to perform a shutdown under the range of operating conditions specified in Section 1.3.2. This requirement is interpreted to mean that there shall be a probability in excess of 0.9999 that upon receipt of a trip signal sufficient rods will drop to shutdown the reactor safely. The objective will be achieved through, among other things, a high order of redundancy in that only a small number (3 shim-safeties or 2 safeties) of the 15 rods are required to terminate any credible reactor transient. Provision has been made, in establishing the reactivity of the system, for the possibility of a rod sticking (failing to fall) during reactor trip.

A corollary requirement to that of dropping the rods is that there shall be a probability of less than 0.0001 of withdrawing any rod except on valid command of the control system.

The second function of the control rods is to adjust reactivity during startup. Before the reactor is allowed to startup, the integrity of the control rods must be guaranteed. Therefore, the

availability of power will be influenced by repair activities accumulated since the last startup. Failed redundant items will require replacement which may cause the shutdown period to be extended.

The third function of the system is to regulate the reactivity of the reactor within specified limits during power operation. In this function accumulated operating time and duty cycles are the chief factors in the reliability model. Part failures will be random except where influenced by unexpected events such as changes in the environment outside prescribed limits, e.g., excessive contamination of reactor sodium, jamming moving parts, or flow blockage, causing excessive temperatures. A chief objective in the design will be to enable continued safe operation in the event of failed components, through use of redundancy and backup systems or components. The components involved in the operating mode shall be maintained independent from those involved in the safety or shutdown mode, insofar as possible, so that a fail-as-is design policy can be followed here, i.e., a shim-safety rod which has failed as a shim will still be operable as a safety. The design of the control rods shall minimize spurious scrams and setbacks.

There is no specified reliability goal for these latter two functions (startup and reactivity regulation at power) since they are control rather than protective functions. Their reliability must be high enough to meet the availability requirements for the control rod system.

1.3.3.2 Availability Requirements

Reactor unavailability due to control rod failures (15 rods) shall not exceed 0.107%. The allocation of this unavailability among the sub-systems and components of the control rod is the responsibility of the manufacturer. Verification of attainment of

the availability goal shall be by failure mode and reliability analysis, taking into account development and prototype test data.

1.3.4 Maintenance

Provision shall be made in the design of the control rod and on-site facilities and through planning of required procedures to insure minimum contribution to plant unavailability from control rod failures. Particular attention shall be given to:

- 1) Maintenance of adequate spacing and arrangement of rod drive assemblies on the top shield to allow rapid and efficient adjustment and calibration of the assemblies during scheduled shutdowns for refueling every year.
- 2) Design of sub-systems and components of the assembly to be easily removable, each as an entity; e.g., drive, drive extension, trip latch, absorber guide tube and absorber section, to allow replacement, repair, or adjustment.
- 3) Handling, cleaning, and storage of radioactive and sodium-contaminated assemblies. Use of the building crane, auxiliary handling machine, and both fuel handling machines is assumed. A storage facility in which a drive extension can be stored in an inert gas atmosphere, examined, and adjusted, must be available.
- 4) Provisions in the design of the control-safety rod for flushing and draining of internal parts (e.g., with hot inert gas, steam, cleaning solutions, etc.) to remove sodium.
- 5) Establishment of periodic preventive maintenance inspection procedures to detect incipient failure modes, gross changes in materials properties, impending wearout, accumulation of contaminants in clearances, etc. Preventive maintenance cycle will be one year.
- 6) Bellows which have been in contact with sodium or its vapor, unless subsequently shielded against contact with oxygen (see 3 above), should not be re-used, due to the high probability of subsequent corrosion and failure.

7) In general, if a component requires repair, the entire subassembly will be removed and replaced by a spare to avoid excessive unavailability. The removed subassembly will then be repaired elsewhere while the reactor is operating.

8) Provision of an on-site control-rod test facility, with associated control and instrumentation console, to allow checkout and adjustment of each assembly prior to installation in the reactor.

9) Design so that all parts of the assembly, including the in-core absorber guide tube, shall be replaceable.

10) Design so that the above-shield portion of the drive shall be removable separately from the drive extension to prevent excessive handling equipment length for the latter.

1.3.5 Safety Requirements

1.3.5.1 The reactivity control system must have a shutdown capability sufficient to make and hold the core subcritical from any operating condition, with appropriate margins for contingencies. As a minimum, shutdown margin equivalent to the maximum worth of the most effective rod when fully withdrawn shall be provided.

1.3.5.2 The maximum worth of the most effective shim-safety rod shall be limited such that a continuous rod withdrawal at full power does not cause fuel rod damage in the hot channel (cladding failure).

1.3.5.3 The reactivity control system should be capable of sustaining a control system malfunction (inadvertent withdrawal of all shim-safety rods) without causing a reactivity transient resulting in exceeding core transient design limits.

1.3.5.4 The reactivity control system shall include a secondary or backup reactivity control means independent of the primary method of reactivity shutdown, with capability to shut down the reactor from any operating condition.

- 1.3.5.5 Electrical safety devices shall be employed to prevent simultaneous raising of all control rods at any significant distance above their fully inserted position.
- 1.3.5.6 Each control rod drive mechanism shall be capable of positive engagement and disengagement with its rod by remote control.
- 1.3.5.7 Both shim-safety and safety rods shall have a known and predictable reactivity worth and life residency in the core.
- 1.3.5.8 Both shim-safety and safety rod assemblies shall be designed to withstand "Design Earthquake" conditions at the reactor site without failure or loss of primary containment integrity.
- 1.3.5.9 The design of the control-safety rod shall be such as to prevent inadvertent handling of control rods during fuel handling.
- 1.3.5.10 The reactivity control system shall not introduce instabilities in core reactivity.
- 1.3.5.11 At least two independent systems shall be provided for controlling the shim-safety rods in the core.
- 1.3.5.12 All penetrations subject to failure or deterioration in service, such as resilient seals and expansion bellows, shall permit leak-tightness of a control rod assembly to be demonstrated at any time.
- 1.3.5.13 The shutdown margin in all rods shall be sufficient to hold the reactor core subcritical under all credible reactivity anomalies.
- 1.3.5.14 The design of the control rod shall be such as to prevent the absorber element from being driven from the core by coolant flow-induced forces.
- 1.3.5.15 The control rods shall include, to the extent practicable, provisions to demonstrate operability at any time during the life of the plant. Specifically, it should be possible to test drive and drive extension without connecting to the absorber element when the reactor is shut down.
- 1.3.5.16 The drive shield penetration shall contain appropriate shielding to prevent excessive localized radiation levels. Drives may be removed only when the reactor is not in operation.

1.3.6 Interfaces

1.3.6.1 Electrical Interfaces

The Reactivity Control System interfaces electrically with the Control and Instrumentation Segment (SDD-096-50-000).

1.3.6.1.1 The plant control system shall provide the power for the normal operation of the control-safety rod drives and readout devices for the instrumentation in 1.3.1.13.6. The following types of power are required and will be supplied:

(to be determined)

1.3.6.1.2 The plant protective system shall provide continuous power for the trip magnet. The following types of power are required and will be supplied:

(to be determined)

1.3.6.2 Mechanical

The Reactivity Control System forms a part of the Reactor Segment (SDD-096-20-000), hence interfaces and must be compatible with the following systems, also parts of the Reactor Segment:

- 1) Reactor Vessel (SDD-096-21-000)
- 2) Fuel and Blanket Elements (SDD-096-28-000)
- 3) Top Shield (SDD-096-23-000)

Due to handling requirements, the Reactivity Control System also interfaces and must be compatible with the Fuel Handling System (SDD-096-81-000) which forms a part of the Support System Segment.

1.3.6.2.1 Absorber Guide Tube Assembly

The Guide Tube Assembly interfaces with the fuel elements and grid structure (see Figure 2). As shown there, the guide tube is supported by its nozzle on the lower grid plate with lateral support provided by the upper grid plate. The housing incorporates a snubber just above the upper grid plate, to decelerate the absorber section

following a reactor trip. There is approximately $5/32$ inch radial clearance between the absorber section guide tube and the adjacent fuel element housings permitting the guide tube to be unaffected by fuel element displacement to this extent. The upper end of the guide tube assembly is 3 inches lower than the top of the fuel element housing to ensure that the fuel handling machine can differentiate between them. The absorber section is captured in the guide tube assembly and its handling head is always below the reactor parting plane.

The guide tube/absorber section assembly is handled by the in-vessel fuel handling machine and must be compatible with it with regard to design of handling fitting.

1.3.6.2.2 Drive Extension

The drive extension interfaces with the top shield as shown in Figure 4. The drive extension is suspended from the top shield, passing through the center of a carrier tube. The carrier tubes are fixed in the top shield and extend only to the bottom of the shield. The $1/2$ to $3/4$ inch radial clearance between the drive extension housing and the carrier tube provides freedom for misalignment and thermally-induced displacements.

During reactor operation the drive extension housing mates with and holds down the absorber guide tube. The normal procedure for shutting the reactor down is to lower all the absorber sections all the way into the core. After releasing the absorber section the drive extension is then driven to the top of its travel, finally raising the holddown tube about 7 inches. A slip joint in the drive extension housing accommodates this movement and also compensates for thermal growth during operation.

The control rod assemblies are aligned to be as near vertical as possible during refueling. This results in a maximum misalignment during operation (to be determined), including the maximum displacement resulting from core fuel element thermal bowing

during reactor operation.

During installation and removal from the reactor the drive extension is handled by the Auxiliary Handling Machine (SDD-096-80-000). It must, therefore, be compatible with it regarding overall dimensions, grappling attachments, and inert gas requirements.

1.3.6.2.3 Rod Drive

The control rod drive is mounted above the top shield on the upper end of the instrument carrier tube (see Figure 3). The envelope dimensions of the drive are set by the center-to-center spacing (see Figure 1) and considerations of adjustment and maintenance.

1.3.6.2.4 Thermosyphon Cooling System

The control-safety rod absorbers are one of several components cooled by the thermosyphon cooling system. Others are the reactor vessel wall, thermal shield, and reflector elements. The absorbers are usually the largest heat source and thus influence cooling of other components. To avoid under-cooling the vessel wall or reversing flow, the maximum total sodium flow to all metal safety rods shall be limited to _____* lb/hr and the pressure drop to _____* psi. Coolant will be supplied at 780°F and $17 \pm$ _____* psig.

1.3.6.2.5 Pressure/Purge System Gas Supply

Provision should be made in the design of the cover gas purification system to deliver pressurization and purge gas to each control-safety rod drive of purity of (to be determined) and at a flow rate of up to (to be determined).

* To be determined.

2.0 DETAILED DESCRIPTION

2.1 Subsystems/Components2.1.1 Absorber Section (see Figure 2)

The absorber sections for both the shim-safety and safety rods utilize tantalum metal as the poison material. The tantalum is in the form of small diameter rods, or pins, contained in Type 316 stainless steel cladding tubes. To insure good thermal contact between the tantalum and the cladding, the gap between them is filled with NaK. Specially designed end fittings and tapering of the ends of the tantalum pins serve to lower the heat load and temperature of the end fittings thus minimizing thermal shock stresses on these heavier sections during trip.

The shim-safety rod pins are 43 inches in length whereas the safety rod pins are 31 inches in length. During normal operation the weight of the pins is supported by the cladding through the top end fittings, which are pinned to stainless support rings, in turn, attached to the hanger tube. However, during trip deceleration in the snubber, the inertia load of the pins is transmitted directly through the lower end fittings into the lower support rings, thence into the snubber dash ram.

Rubbing surfaces to guide the absorber section inside the guide tube are provided by guide rings, concentric with and part of the support ring assemblies. Spacing of the pins, to prevent excessive bowing due to unbalanced flow distribution near the guide tube wall, is achieved with sheet-metal spacers located at three places along the length of the shim-safety rod poison section, at two on the safety rod. As indicated previously, the lower end of the absorber section forms the dash-ram for the scram hydraulic shock absorber (snubber). At the top of the hanger rod, a handling head is mounted. This head is gripped by the trip latch (see Section 2.1.4 below) during normal power operation, and released for reactor trip or disengaged for fuel handling operations.

The absorber section hanger contains a "weak link" section (not shown in Figure 2) between the handling head and the upper guide ring. In case a trip latch malfunctions such that it will not delatch

its absorber section, the absorber section will be pulled up against the absorber guide tube upper fitting (backed up by the holddown tube) and the hanger rod broken at the weak link. This will allow transfer of the absorber section and core housing to the in-core storage area and removal of the drive extension in the normal manner with the auxiliary handling machine.

2.1.2 Absorber Section Guide Tube Assembly

The guide tube assembly consists of three components, the handling fitting, guide tube, and snubber dash-pot. The handling fitting, mounted at the top of the housing, serves three functions: (1) it is gripped by the in-vessel handling machine latch for installation or removal of the housing from the reactor core, (2) its upper surface is bevelled to match that on the holddown tube (see Section 2.1.4 below) assuring positive alignment of the absorber section with the drive extension, (3) it serves as a positive mechanical absorber travel limit.

As implied by its name, the guide tube serves to guide the absorber section as it moves through the core. The forcing of trapped sodium from the snubber dash-pot, by the snubber dash-ram attached to the absorber section, dissipates the kinetic energy of the falling absorber section during reactor trip.

2.1.3 Rod Drive (see Figure 3)

The control rod drive is of the electromechanical type, using two electrical motors, a stepper, and a dc motor mounted on the same shaft. The stepper motor drives the rod for startup, shimming, and regulating functions. A gear train increases the output torque to that required which is delivered to a ball-screw rotary-to-linear motion converter. A slip-clutch in the drive train prevents damage to the drive in the event of travel limit switch failure and a "No-Bak" clutch prevents over-hauling of the drive by the load.

The dc motor is used to drive the rod at high speed (~ 15 ft/min) for the fast setback (shim-safeties only) and for fast

rundown of all drive extensions after release of the absorber sections at reactor trip. An over-running one-way clutch between the dc and stepper motors insures that the rods can be driven at the fast speed in the down direction only.

The trip magnet at the lower end of the drive screw holds the latch actuating rod during reactor operation or releases it for reactor trip. The drive is attached to the drive extension through bayonet fittings in the main pull tube and latch actuating rod just below the trip magnet. Both the drive screw and the scram magnet operate inside the sealed drive housing, inside which an inert gas is maintained under a pressure slightly above the static pressure at the lower end of the drive extension. This allows the integrity of the latch actuating rod bellows to be monitored continuously through use of a gas flow measuring device in the inlet line to the drive housing. The drive housing also backs up the drive extension seals.

Instrumentation for the control rod consists of

1) Both digital and analog position sensors to continuously indicate the position of the drive rack (and the absorber section when gripped by the trip latch).

2) Limit Switches (see 1.3.1.13.6.1.3)

3) A load cell, between the drive rack and the trip magnet, to continuously monitor the load on the drive, i.e., drive extension and absorber section weights and friction loads.

4) Gas Pressure and Flow Measuring devices to monitor (a) the integrity of the bellows (main stroke if used and latch actuating rod) and (b) the gas flow through the gas purge seal system.

5) Thermocouples to monitor the temperature of the outlet coolant from the absorber section (see Section 2.1.4 below).

6) An accelerometer attached to the drive housing to sense absorber section impact in the snubber, allowing measurement of rod drop time.*

* Preliminary choice.

7) A shorting-type detector to indicate the presence of sodium inside the drive extension in the event of a bellows failure.

2.1.4 Drive Extension (see Figure 4)

The drive extension performs the following functions:

- 1) It connects the drive, located above the shield, to the absorber section and establishes alignment between the drive and guide-tube absorber section assembly in the reactor core.
- 2) It seals the penetration of the rod through the shield against leakage of sodium, sodium vapor or fission products from the reactor, and against inleakage of oxygen.
- 3) It contains thermal and radiation shielding to prevent excessive heat losses or radiation streaming at the control rod penetrations through the shield.
- 4) It provides positive holddown of the control rod guide tube during reactor operation and of the drive extension shafts during maintenance operations.

The main components of the drive extension are: two concentric shafts, an outer main drive tube and inner trip-latch actuating shaft; seals (two latch actuating shaft bellows, a drive tube packing seal, static O-ring seals, and a gas purge system), the trip latch, holddown tube and housing.

Shafts

The drive extension shafts are connected to the drive with bayonet couplings in similar shafts just below the trip magnet. They extend through the top shield, the reactor cover gas space and sodium pool, to the top of the absorber section in the reactor core. During power operation, the main drive tube supports the weight of the absorber section through the trip latch attached to its lower end. The latch actuating shaft, operated by the trip magnet armature, holds the latch closed to grip the absorber section, or opens it to release the absorber section for trip.

Bellows and shaft packing seal

The primary drive tube seal design is a helium gas purge (see below). If tests show that the gas purge is not sufficient, a backup design consisting of bellows and purge will be used. If used, the drive bellows would be located between the outer housing and the main drive tube and accommodates the full rod stroke of 50 inches. (Normal stroke 43 inches plus 7 inches to lift the holddown tube to allow shield rotation for fuel handling.) A smaller bellows of the same type and material seals between the main drive tube and the latch actuating shaft and accommodates the relative motion between the two required to actuate the scram latch ($\sim 5/8$ inch). The gas purge seal (or main bellows) is backed up by a shaft packing seal at the top of the drive extension.

Gas Pressure/Purge System

Pressure System

Under normal operation gas is maintained inside the drive housing and main drive tube at a pressure slightly above the maximum static sodium head at the bottom of the assembly. The gas is contained by the trip latch bellows and drive tube shaft seal. Should a leak develop in the bellows, it will be indicated by gas flow. The pressure in the drive housing (hence trip latch bellows) is slightly above that in the drive tube purge system to prevent upward diffusion of radioactivity.

Purge System

It is necessary to prevent or minimize the upward diffusion of sodium vapor and radioactive fission products into the drive extension which could (a) interfere with operation by freezing of sodium or formation of frost around the main shaft and (b) create radiological problems at control rod penetrations through the shield. The gas purge system accomplishes this by maintaining a downward flow of purified cover gas between the shaft and housing sufficient to effectively block upward

diffusion and convection. The purge system gas flow path (see Figure 4) is through the clearance annulus between the main drive tube and the housing guide tube, along the length between the packing seal at the upper limit of the wetted drive tube as shown in Figure 4. The bearings are hard-surfaced to prevent galling or excessive wear on the shaft. In addition to providing shaft guidance, the ring serves to increase the purge gas velocity thus increasing the efficiency of the gas in blocking upward convection and diffusion.

If the backup design full stroke bellows is used a blow-out disk or ring will be installed in the housing above the main bellows. Should gas flow through the system indicating a leak in the bellows, this disk is deliberately ruptured to provide a gas flow path around the bellows and a sodium flow path to prevent pumping of sodium up into the drive extension by compression of the bellows. As part of this system, an electrical shorting type indicator is provided to indicate the presence of sodium inside the main bellows.

A ball-check valve in the top of the main drive tube prevents loss of gas pressure inside it when the drive housing is removed for maintenance of the drive or preparatory to removal of the drive extension from the reactor.

O-Ring Gaskets

Four O-ring gaskets (with leak-monitoring lines between each of the two pair) seal between the drive extension housing and the reactor shield penetration.

Trip Latch

The trip latch, attached to the lower end of the main drive tube, serves two functions: disengaging the drive extension from the absorber section for fuel handling; and quick release of the absorber

section for reactor trip. It is of the movable-finger, cam-operated type based on the concept proven reliable in the Fermi reactor. It is latched to grip the absorber section handling head by downward motion of the drive extension in the absorber guide tube so that the latch fingers enclose the head. It is delatched or released by de-energization (either by the Plant Protective System or loss of power) of the trip magnet. When this occurs the weight of the latch-actuating rod, aided by the delatch-assist spring (part of the magnet assembly) causes downward motion of the rod releasing the fingers from their collet and forcing them open by cam action, dropping the absorber section. The latch will be fabricated of adhesion and galling-resistant materials, such as the refractory metals or their alloys (e.g., TZM) or the cobalt-based alloys (e.g., Stellites) as selected by tests in sodium.

Shielding

The part of the drive extension housing which passes through the biological shield contains thermal and radiation shielding equivalent to that in the top shield (SDD-096-23-000) to minimize heat losses and radiation streaming at control rod penetrations.

Holddown Devices

The drive extension housing contains two holddown devices. The holddown tube forms the lower part of the drive extension housing. It is bevelled to match the upper fitting on the absorber element guide tube assembly and is forced against it when the drive extension is driven to its lowermost (full-in) position. The holddown tube maintains alignment between the drive and the in-core assembly and prevents inadvertent lifting of the absorber guide tube (beyond the 7-inch travel of the slip joint) with the absorber section, should binding between the two develop. The holddown tube is raised ~ 7 inches by the drive prior to shield rotation for fuel handling.

The second holddown device is that which prevents upward motion of the drive extension housing itself, when the drive has been removed. It consists of a large snap-ring which fits into a groove in the shield plug penetration tube and bears against the top of the drive extension housing.

Thermocouples

As mentioned in Section 2.1.3, thermocouples are required in each control rod assembly to monitor the outlet temperature of the absorber section coolant. Two sheathed thermocouples (not presently shown in Figure 4) will be located in the coolant stream just above the trip latch. The thermocouple leads will be routed inside the latch rod with seal welds to keep sodium out of the rod.

3.0 PRINCIPLES OF OPERATION

3.1 Startup

Since a neutron absorbing material is used as the controlling element, lowering the rods into the core reduces the reactivity and raising the rods increases reactivity. Startup of the reactor is accomplished as follows: (1) prior to startup, the safety rod bank will be raised to its uppermost position (bottom of the poison section at the top of the upper blanket); (2) the shim-safety rods are raised, either singly or in banks of two or more, to bring the core to criticality and then to power.

3.2 Normal Operation

After reaching power, one or more of the shim-safeties (probably banks of three) are used to continually regulate the core flux and power, either at a constant or changing level as determined by load demand. While performing this function should the regulating rods reach previously-established limits of travel, the remaining shim-safeties are adjusted up or down so as to bring the regulating rod travel back within limits. Individual shim-safeties may also be adjusted up or down to obtain the desired power and temperature distribution in the core.

Heat is generated in the tantalum absorbers during operation of the reactor by (1) gamma rays generated in the Ta by the $n-\gamma$ neutron-absorption reaction and, (2) core-generated gamma rays. Cooling of the absorber pins is achieved by thermosyphon-induced coolant flow through the control channels. The control channels are isolated from the high-pressure plenum and receive coolant from a low-pressure plenum, fed through a bypass valve from the main high-pressure coolant line. With thermosyphon cooling, the pressure drop and resultant flow through the control rod channel is balanced by a fluid density differential between warmer (hence more buoyant) sodium in the rod channel and cooler (hence more dense) sodium in a parallel flow path through

the region surrounding the reactor core. Since flow is closely linked to sodium temperature, hence heat load, this method is self-regulating and results in an essentially constant outlet sodium temperature in spite of the wide variation in absorber heating as its position in the core changes.

3.3 Shutdown and Emergency Operation

The most vital function of the Reactivity Control System, which it performs in conjunction with the Plant Protective System, is that of safely shutting the reactor down from any credible operating condition without damage to the plant or danger to the public. Since with neutron absorbing rods reactivity decreases with insertion into the core, this safety function is achieved by rapidly lowering the rods. For the 1000 Mwe LMFBF two levels of safety action are used. In the first, called fast setback, the shim-safeties are driven downward (at ~ 30 times the normal rate) with the rod drives. Should the plant operating parameter(s) continue to diverge from the acceptable operating range, the second level of safety action (reactor trip) is reached at which time all rods are dropped into the core. Previous safety studies have indicated that shutdown can be achieved from any credible accidental transient with the expected rod release time (~ 0.1 sec) and an effective acceleration of $\sim 1/2$ g.

4.0 SAFETY PRECAUTIONS**4.1 Hazards**

The possible hazards associated with the Reactivity Control System are:

- a) Excessive gas leakage from the reactor due to faulty seals on the control rod drive shaft or housing.
- b) Leakage of one or both of the latch bellows, permitting sodium to enter the space around the latch actuating rod and possibly interfering with trip action.
- c) Trip of a control rod without sufficient sodium in the reactor to permit proper snubbing action.
- d) Failure of rod to actuate due to accumulation of sodium frost around the drive tube.
- e) Damage to the control rod drives during installation and removal of the reactor containment dome, or as a result of interference with the fuel handling equipment.
- f) Damage to the drive extension and/or absorber column as a result of top shield rotation before the drive extensions are delatched from the absorber columns.
- g) Jamming of absorber column due to distortion of the core matrix as a result of material swelling and thermal gradients.
- h) Self-welding of latch materials.

4.2 Precaution

- a) Control rod seals should be periodically checked for leakage and replaced as necessary.
- b) At the next shutdown following failure of either of the latch bellows, the drive extension should be removed and the faulty bellows repaired.
- c) Control rods should never be withdrawn, even for testing purposes, unless sodium level is verified to be above that necessary for snubbing action.

d) Rods should be periodically "exercised" during operation to verify that they are movable. Periodic inspection should be performed during reactor shutdown to determine presence of sodium deposits around the drive shaft.

e) Extreme caution must be exercised while performing all operations in the vicinity of the control rod drives. Licensed operators only should be utilized in these activities.

f) Strict procedural control (in addition to the top shield rotational interlocks) must be exercised, in conjunction with the sweep arm, to assume that all rods are delatched before shield rotation.

g) See precaution d) above.

h) Control rod latches should be exercised at every shutdown and examined periodically for evidence of self-welding.

5.0 MAINTENANCE**5.1 Normal/Routine Maintenance**

The general approach to maintenance of the Reactivity Control System is exemplified by the maintenance requirements, Section 1.3.4. Routine maintenance is expected to consist of periodic examination of the absorber column and drive extension, replacement of drive units with spares on a scheduled basis, and seal replacement, also on a scheduled basis. Performance checks on drive speed and drop time will be made during reactor shutdown and corrective action taken as necessary.

5.2 Unusual Maintenance

No requirement for unusual maintenance is presently foreseen.

6.0 **APPENDICES**

6.1 Tables

- 1) Reactivity Worth Requirements

6.2 Drawings

- 1) NO 9622001, Control Rod-Absorber 1000 Mwe LMFBR
(Conceptual Layout) Attached as Figure 2.
- 2) NO 9622002, Control Rod Drive 1000 Mwe LMFBR
(Conceptual Layout) Attached as Figure 3.
- 3) NO 9622003, Control Rod-Drive Extension,
1000 Mwe LMFBR (Conceptual Layout)
Attached as Figure 4.
- 4) NO 9620023, Reactor Plan Section 1000 Mwe LMFBR
Follow-on Study
Attached as Figure 1.

Table 1
Reactivity Worth Requirements

Primary Shutdown System
(all rods dropped)

Expansion	\$ + .60 ± \$.60
Fuel burnup	3.45 ± 1.72
Doppler	2.56 ± .77
Sodium temperature effect	- .54 ± .32
Ta burnup	1.05 ± .32
Shutdown margin (1% ΔK)	3.00 ± 0.00
Maximum sodium void worth	8.37 ± 5.02
Control margin	.28 ± .14
Tantalum worth uncertainty	0.00 ± 2.82
Sub-total	<u>\$18.77 ± \$6.10</u>
Total uncertainty	6.10
System worth required	<u>\$24.87</u>

Backup Shutdown System
(all rods driven in at setback speed)

Expansion	\$.60 ± \$.60
Fuel burnup	3.45 ± 1.72
Doppler	2.56 ± .77
Sodium temperature effect	- .54 ± .32
Ta burnup	1.05 ± .32
Shutdown margin (1% ΔK)	3.00 ± 0.00
Control margin	.28 ± .14
Ta worth uncertainty	0.00 ± 1.56
Sub-total	<u>\$10.40 ± \$2.56</u>
Total uncertainty	2.56
System worth required	<u>\$12.96</u>

Shim-Safety Bank Worth

Expansion	\$.60 ± \$.60
Fuel burnup	3.45 ± 1.72
Doppler	2.56 ± .77
Sodium temperature effect	- .54 ± .32
Ta burnup	0.82 ± .25
Shutdown margin *	1.00 ± .00
Control margin	.23 ± .14
Ta worth uncertainty	0.00 ± 1.22
Sub-total	<u>\$ 8.17 ± \$2.36</u>
Total uncertainty	2.36
System worth required	<u>\$10.53</u>

Safety Bank Worth = (Primary Shutdown System Worth) - Shim-Safety Bank Worth)

$$\text{Safety Bank Worth} = \$24.87 - \$10.53 = \underline{\underline{\$14.34}}$$

* The shutdown margin is large enough to prevent fuel damage when the largest undetected bubble passes through the core.

CORE LEGEND

- FAST FUEL - 274
- ⊗ BLANKET FUEL - 138
- REFLECTOR - 78
- ⊙ SHIM SAFETY-CONTROL ROD
- ⊗ SAFETY-CONTROL ROD (15 TOTAL)

REACTOR DRAWINGS
NO9620021 REACTOR - VERTICAL SECTION
NO9620022 REACTOR - PLAN VIEW

- Figure 1 -

ATOMIC INTERNATIONAL A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION 3800 110TH AVENUE, CARLSBAD, CALIF.			
REACTOR PLAN SECTION 2000 MW(L) LMFBR FOLLOW-ON STUDY			
DRAWN	CHECKED	ENGINEER	APPROVED
DWG NO		SHEET NO	
NO9620023

REVISIONS			
NO.	DESCRIPTION	DATE	BY
A	SEE REVISION RECORD	1/4	2/3/62

SDD-096-22-000

Page 45

- Figure 3 -

ATOMIC INTERNATIONAL			
CONTROL ROD DRIVE			
1000 MW-LMFB FOLLOW-ON STUDY			
ESTIM. D. 22000	REV. 3	DATE	1/4
DESIGNED	BY	DATE	1/4
DRAWN	BY	DATE	1/4
CHECKED	BY	DATE	1/4
APPROVED	BY	DATE	1/4
N09622002

ATOMICS INTERNATIONAL*A Division of North American Rockwell Corporation***SUPPORTING DOCUMENT**

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SDD-096-23-000

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System Design Description

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Shield-Energy Absorber

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Reactor Shielding Conceptual System Design
Description - 1000 Mwe LMFBF Follow-On StudyORIGINAL
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ABSTRACT

This SDD presents the requirements for, and a brief description of, the following: equipment shielding, top shield and plugs, DBA energy absorbing system, top shield cooling system, and the biological shield cooling system.

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REACTOR SHIELDING SDD

1. INTRODUCTION

The following items are included as part of this design description:

- a. Biological Shield - This includes all plant shielding used to protect people from radiation. All information on this item is presented under the "Land, Structures and Conventional Systems" segment. Refer to SDD-096-90-000.
- b. Equipment Shielding - This includes all plant shielding used to protect equipment from radiation. It includes "special" shielding such as calcium borate and lead as well as concrete.
- c. Top Shield and Top Shield Plugs - This includes the entire top shield and all plugs which are not an integral part of other systems, such as fuel handling or the control rods. It also includes the energy absorbing system used to hold down the shield and plugs during a DBA.
- d. Blast Shield - This includes the shielding used to protect the structures and/or equipment immediately surrounding the reactor vessel from the energy and missiles released during a rupture of the vessel. At the present time, the necessity of, or requirements for, such shielding have not been determined, consequently no further information is presented. When information becomes available it will be incorporated into this SDD.
- e. Shield Cooling Systems - This includes the cooling system for the top shield and the system used to cool only that portion of the biological shield immediately surrounding the reactor (during normal plant operation).

The following are not included as part of this design description:

Cooling systems used to protect portions of the cavity liner from the effects of a DBA.

Cooling systems used to cool the pipe chases around the reactor on the primary equipment vaults.

The reactor containment dome and instrument ring. The process equipment (pumps, heat exchangers) associated with the biological shield cooling equipment.

1.1 Functions/Functional Requirements

1.1.1 System Functions

The general functions of the components covered in this SDD are to protect systems and equipment from the effects of radiation which

would interfere with maintenance functions, to protect operating personnel and the general public from radiation, to cool the reactor shielding so that it can function properly throughout plant life, and to provide blast and missile protection for the inner containment boundary (reactor containment dome).

1.1.2 Component Functions

1.1.2.1 Equipment Shielding

The functions of the equipment shielding are to:

- a. Provide neutron shielding as required to limit equipment and component activation to the point where maintenance will not be impaired.
- b. Provide neutron shielding as required to prevent significant activation of secondary sodium.

1.1.2.2 Top Shield and Plugs

The functions of the top shield and top shield plugs are to:

- a. Provide the vessel top closure to seal the reactor cover gas within the vessel, maintaining the low leakage required to accept failed fuel elements.
- b. Shield the high bay area from the nuclear radiation of the reactor.
- c. Minimize heat loss from the reactor.
- d. Minimize the sodium-cover gas interface area to limit cover gas entrainment by the sodium.
- e. Provide support for the control rod drives, reactor assembly instrumentation, and fuel transfer machines.
- f. Act functionally as a part of the fuel transfer system by positioning the fuel handling machines over individual fuel, blanket or control rod elements in the core or in the in-vessel transfer storage positions.
- g. Provide for distribution of DBA energy, applied to the bottom surface of the shield, into the DBA energy absorbing structure located above the shield.
- h. Provide access to the reactor internals for instrumentation.

1.1.2.3 DBA Energy Absorbing System

The function of the DBA energy absorbing system is to provide shock and blast protection for the inner containment boundary.

1.1.2.4 Top Shield Cooling System

The functions of the top shield cooling system are to:

a. Isolate the high bay arca and temperature limited components located above the top shield from reactor temperatures.

b. Lower the top shield surface temperature to permit contact work during fuel handling and maintenance.

c. Limit thermal gradients across the shield to insure that such gradients do not impair proper rotation of shield plugs or cause excessive seal leakage.

d. Lower shield temperature to insure that seal integrity is maintained.

1.1.2.5 Biological Shield Cooling System

The functions of the biological shield cooling system are to:

a. Limit concrete temperatures to those which will not impair the neutron shielding effectiveness of the concrete.

b. Limit concrete temperatures to prevent loss of structural strength over the plant lifetime.

c. Limit temperatures and temperature fluctuations across the steel cavity liner to prevent liner buckling and loss of integrity during normal plant operation.

1.2 Summary Description

1.2.1 Equipment Shielding

The equipment shielding basically consists of bulk concrete surrounding the reactor vessel, concrete shielding around the heat transfer piping into and out of the vessel, and concrete shielding around the IHX. Supplemental shielding in the form of lead and borated material is used as required to reduce neutron flux to appropriate levels.

The bulk concrete around the reactor functions both as biological and equipment shielding. As equipment shielding, it limits neutron flux levels to minimize activation of process equipment adjacent to the reactor, principally within the sodium drain tank vault. The shielding around the heat transfer piping, basically the walls of the main pipe chases

around the reactor, reduce flux levels to prevent significant activation of the primary heat transfer loops, up to and including the loop block valves. The IHX shielding reduces the neutron flux levels even lower, to prevent activation of the secondary sodium within the reactor building.

1.2.2 Top Shield and Plugs

The top shield is approximately 11 feet thick, consisting of layered steel gamma shielding plates, layered reflective insulation plates, serpentine neutron shielding and then a thick steel structural top plate. It is composed of a small rotating plug within a large rotating plug which fits closely within the 24'-8" diameter of the top end of the vessel. By appropriate rotation of the large and small plugs during reactor shutdown the fuel transfer ports in the shield can be positioned over any element of the reactor assembly or any of the in-vessel storage positions. During reactor operation the plugs are locked in position by means of pins with the control rod ports, all in the small plug, in position over the reactor assembly.

The top shield is shown on Dwg. Nos. NO9620021 and NO9620022.

The Top Shield Assembly consists of the following components or subsystems:

1. Rotating Plugs.
2. Rotating Plug Bearings.
3. Rotating Plug Seals and Sodium Vapor Barriers.
4. Rotating Plug Drive and Indexing Subsystem.
5. Misc. Access Ports.
6. Reactor Instrument Thimbles.
7. Reactor Instrument Support Structure.
8. Shield Cooling System.

1.2.3 DBA Energy Absorbing System

The DBA energy absorbing system is composed of the following major components:

- a. Energy absorber
- b. Support ring
- c. Support columns

The energy absorber is a hold-down structure for the top shield and its plugs which prevent these items from penetrating the reactor containment dome during the energy release associated with a DBA. The support ring is used to transmit the associated shock loading from the absorber to the support columns which, in turn, isolate the loads from the building structure and transmit them to the building foundation. The support ring is also used, during normal operation, to support the reactor and its internals, and to provide shielding. The energy absorbing system and the reactor complex is shown on Dwg. No. NO9620021.

The energy absorber is a ribbed-plate shell structure designed to absorb the kinetic energy of the top shield through uniform deformation of guided crush columns in the event of a DBA. A missile barrier is provided above the control rod drives and other shield penetrations to protect the dome. The absorbed load is transmitted into the support ring and thence into the columns and building foundation by shear lugs which can be inserted and removed for fuel handling operation.

1.2.4 Top Shield Cooling System

The top shield will be cooled by a forced-circulation cooling system using air or an inert gas as the cooling medium. A firm design is yet to be established.

1.2.5 Biological Shield Cooling System

The cooling system for the biological shield surrounding the reactor has not been established at this time. Based on recent studies of similar systems, the most promising design appears to be a forced-circulation system using water as the cooling medium. The water is circulated through pipe coils imbedded in the concrete.

1.3 Requirements

1.3.1 Performance Requirements

1.3.1.1 Design Life

All components and/or systems included in this SDD shall be designed for a 30 year lifetime.

1.3.1.2 Availability

All components of the Reactor Shielding CSDD shall be designed such that the total unavailability shall be .97%. The unavailability for the individual components shall be as follows:

<u>Item</u>	<u>Unavailability (%)</u>
Biological Shielding	< 0.001
Equipment Shielding	-
Top Shield and Plugs (includes extension of refueling outages)	.96
DBA Energy Absorbing System	< 0.001
Blast Shielding	-
Top Shield Cooling System	Included in Top Shield
Biological Shield Cooling System	0.01
<hr/>	
Total Unavailability	.97 %

1.3.1.3 Equipment Shielding

During normal plant operation, the equipment shielding shall limit component activity levels, due to neutron activation to the following:

- a. Equipment in the heat transfer vaults, up to and including the block valves--15 mr/hr
- b. Equipment in the drain tank vaults--1.5 mr/hr
- c. Secondary sodium--0.25 mr/hr

1.3.1.4 Top Shield and Plugs1.3.1.4.1 Operating Floor Radiation Requirements

During operation at 100% power, the neutron and gamma radiation levels at the operating floor elevation shall not exceed:

Gross - 2.0 m rem/hr

Local Streaming - 10.0 m rem/hr.

1.3.1.4.2 Floor Level Temperature

The floor level temperature shall not exceed 135°F.

1.3.1.4.3 Fuel Handling

Rotation - The large and small rotating plugs shall be capable of 360° bi-directional rotation.

Time to Rotate - The maximum time required to maneuver the incore port from any core location to any transfer storage location shall be 2 minutes.

Index Repeatability Tolerance

a. In-core - The in-core port shall be indexable to any core or transfer storage position within a concentricity of 0.25 inches.

b. Ex-core - The excore handling port shall be indexable to the transfer storage positions within a concentricity of 0.25 inches.

1.3.1.4.4 Reactor Alignment

When the shield is pinned in its position for reactor operation the control rod ports shall be located at true positions relative to reactor vessel centerline within .030 inch.

NOTE: The system will be designed to have the control rod ports and instrumentation supported by the shield nominally in line with the elements of the reactor assembly at the 500°F refueling temperature.

1.3.1.5 DBA Energy Absorbing System

1.3.1.5.1 System Performance Requirements

a. Normal Conditions - During normal operation of the plant, the DBA energy absorbing system shall provide structural support for the reactor vessel and top shield assemblies, access ports to the reactor cavity for reactor vessel inspection, and instrument ports for nuclear instrumentation located in the cavity.

b. DBA Conditions - The DBA energy absorbing system shall absorb the maximum energy release associated with a DBA and any other less severe emergency condition. The inner containment barrier shall be protected from any damage which would impair its containment integrity. Provision shall be included to vent excess pressure within the dome into the reactor cavity.

1.3.1.5.2 Component Performance Requirements

1.3.1.5.2.1 Energy Absorber

The energy absorber shall be:

a. Capable of absorbing an energy release equivalent to a 500 psig pressure moving a 24' - 8" diameter plug up to a 4 foot vertical distance.

b. The energy absorber structure shall be removable in conjunction with, or following removal of, the containment dome to allow access to the reactor loading face for fuel handling or maintenance operations.

1.3.1.5.2.2 Support Ring

a. The support ring shall be capable of transmitting to the support columns a load of up to 35×10^6 lb acting axially either upward or downward resulting from the interaction of DBA generated forces on the containment dome, instrument ring, energy absorber or reactor vessel.

b. In addition to the DBA loads, the support ring shall be capable of transmitting to the support columns the dead load associated with the reactor and top shield assemblies under all reactor operating condition as well as all fuel handling loads.

NOTE: Ex-core fuel handling equipment is not normally supported by the top shield, support ring and support columns; however, these structural components must be designed to support these loads in event equipment malfunction requires the fuel handling equipment to rest on the shield face.

c. The support ring shall be provided with a minimum of 12, 24-inch diameter ports evenly distributed around the periphery of the reactor vessel to allow remote inspection of the reactor vessel. Contained within the inspection ports shall be removable pressure relief disc assemblies connecting the containment dome with the reactor vessel cavity.

d. The support ring shall provide sufficient radiation shielding such that during normal reactor operation the radiation level at the top surface of the support ring is no greater than the radiation level at the upper surface of the reactor top shield.

e. The support ring may include a number of ports evenly distributed around the periphery of the reactor vessel for the purpose of admitting nuclear instrumentation to the reactor vessel cavity. Refer to SDD-096-50-000 "Control and Instrumentation Segment" for requirements.

1.3.1.5.2.3 Support Columns

The support columns shall be capable of withstanding the DBA loading of up to 35×10^6 lb acting axially either upward or downward and transmitting this loading into the reactor foundation.

NOTE: The support columns are eventually loaded in tension as a consequence of restraining the top shield during a DBA; however, arrival of a DBA shock wave at the bottom of the vessel prior to reaching the shield could conceivably result in column compressive loading for a short time interval.

In addition to the DBA loads the support columns shall be capable of withstanding the reactor vessel and top shield assembly dead loads under all reactor operating conditions as well as reactor fuel handling loads and transmitting these loads into the reactor foundation.

1.3.1.6 Top Shield Cooling System

a. The top shield cooling system shall maintain the temperature of the top surface of the shield at a maximum of 135°F.

b. The top shield cooling system shall limit the thermal gradient across the diameter of the upper surface to $< 50^\circ\text{F}$

1.3.1.7 Biological Shield Cooling System

The biological shield cooling system shall limit the temperature of the concrete shield to a maximum of 150°F.

1.3.2 Environmental Requirements

1.3.2.1 Equipment Shielding

Normal Conditions

Atmosphere: Nitrogen - Max. oxygen content 2 vol.%

Temperature: 150° F. Max.

Pressure: $-\frac{1}{4}$ " to $-\frac{1}{2}$ " w.g.

Emergency Conditions

Atmosphere: Nitrogen containing radioactive sodium, fission products and gasses and possibly fuel or reactor structure particles or vapors.

Temperature: Approx. 650°F Average Max.

Pressure: 16. psig. peak pressure

1.3.2.2 Top Shield and Plugs**1.3.2.2.1 Operation Sequence of Events**

The top shield is exposed to the radiation, thermal, mechanical and pressure environment produced during plant preheat, startup, operation (including load swings), refueling and maintenance shutdowns and restarts throughout its design life of 30 years.

Number of Cycles (30 years)

Preheat	3
Startup	3
Load Swings	9,000
Refueling Shutdown	30
Maintenance Shutdown	30
Plug Rotation	4,800 element transfers

Approx.

1.3.2.2.2 Na Pool Temperature**a. Steady State**

100%

During fuel transfer
shutdown 500 F

During reactor operation 1140 F

Design 1160 F

b. Transients

Refer to SDD-096-20-000 "Reactor Vessel Assembly

Conceptual System Design Description."

1.3.2.2.3 Reactor Cover Gas**Type**

Helium, saturated with sodium vapor and containing fission products.

Temperature

Operating 1140 F max

Refueling Shutdown 500 F max

Pressure

Operating	0.3 psig
Refueling Shutdown	0.3 psig
Transient (8 hrs max)	5.0 psig

1.3.2.2.4 Reactor Containment AtmosphereType - Nitrogen - 2% vol. max oxygen contentTemperature 150 F maxPressure (-)1/2" W.G. with respect to atmosphere
surrounding reactor building.

NOTE: Leak rate tests will be performed on the reactor containment boundary at 20 psi differential. The maximum differential across the top shield during this test has not been established.

1.3.2.2.5 Radiation Fluxes

The bottom face of the top shield is exposed to the following radiation spectrum during normal, full power, operation:

<u>Neutron Flux</u>	<u>Flux</u> <u>n/cm²-sec</u>	<u>Integrated Dose</u> <u>nvt</u>
Fast (> 1.0 Mev)	TBD*	
Epithermal (27.0 ev-1.0 Mev)	"	
Thermal (< 27.0 ev)	"	
<u>Gamma Dose Rate</u>	<u>Dose Rate</u> <u>m/rem/hr</u>	
Na ²⁴	TBD *	
Capture γ in Na	TBD*	
Core γ rays	TBD*	
Inelastic neutron scattering	TBD*	
Cover gas activity - Operating	TBD*	
Shutdown	TBD*	

1.3.2.2.6 Loadsa. Reactor Operating

The top shield will carry the following approx. equipment weights during reactor operation, i.e., the rotary plugs are stationary.

Control Rods 15 @ 700 lb	10,500 lb
Small Rotating Plug Drive	500
Power & Instr leads & supports	6,500

*To be determined.

b. Reactor Shutdown (Refueling)

The top shield will carry the following approximate equipment weights during refueling, i.e., the fuel handling plugs are performing fuel handling functions.

Control Rods 15 @ 700 lb	10,500 lb
Small Rotating Plug Drive	500
Power and Instr. Leads and Supports	6,500
Floor Valve (In-vessel FHM)	12,000
In-vessel FHM	10,000
Floor Valve (Ex-vessel FHM)	12,000
Lower Portion of Ex-vessel FHM	(To be determined)
Force capability of in-core FHM	±4,000 lb

NOTE: In addition to the loads indicated in a and b, the top shield must also support its own dead weight which, estimated from the conceptual design shown on Dwg. NO9620021, is as follows:

Small Rotating Plug	223,000 lb
Large Rotating Plug	465,000 lb

c. DBA Load

In a DBA a resisting pressure of 500 psi (total force equal to approximately 35×10^6 lb) will be applied against the top face of the shield to resist the DBA impulse loads. The shield will move upward through a 4 ft stroke, being resisted by crush columns of a DBA energy absorbing structure.

1.3.2.3 DBA Energy Absorbing Systema. Normal Conditions

Atmosphere: Nitrogen - Maximum Oxygen Content 2 volume %

Temperature: 135°F Maximum

Pressure: $-\frac{1}{4}$ " to $\frac{1}{2}$ " w.g.

b. Emergency ConditionsPost DBA Environment

NOTE: The transient and quasi-steady state conditions during and subsequent to a DBA depend upon the manner in which the DBA energy is released to the primary containment envelope. It presently appears that reactor vessel rupture leads to higher temperatures and pressures in the

reactor cavity than in the reactor dome, whereas if the vessel remains intact, higher pressures and possibly temperatures will occur in the reactor dome than in the cavity. The conditions cited below are preliminary and subject to revision as the results of more detailed analysis become available.

Atmosphere: Nitrogen and helium containing radioactive sodium, fission products, and gases; and possibly fuel or reactor structure particles or vapors.

Temperature: Transient peak gas temperature in cavity below support ring of 1900°F following vessel rupture. Transient peak gas temperature in containment dome of 960°F with no vessel rupture.

Pressure: A transient pressure of (TBD) will exist within the cavity for a period of less than one second after rupture of the reactor vessel while the cavity pressure is being relieved to the heat transfer vaults; equilibrium pressure in the cavity and vaults is 16.2 psig (10.6 from blast expansion plus 5.6 from fire). The containment dome pressure will reach a maximum of 18 psig by communication through the vessel rupture and blown out top shield seals. In event of no vessel rupture, a transient pressure of 22 psig will exist within the reactor containment dome for a period of less than one second after reactor vessel pressure is released to the dome while the dome pressure is being relieved to the reactor cavity; equilibrium pressure in the dome and cavity is 9 psig, of which 7 psig is due to sodium fire.

1.3.2.4 Top Shield Cooling System

The top shield cooling system will cool the upper surfaces of the top shield. Generally, the portion of the system in, or close to, the top shield will experience the same environmental conditions as the energy absorber. Refer to Paragraph 1.3.2.3.

1.3.2.5 Biological Shield Cooling System

The environmental conditions experienced by the biological shield cooling system will depend on the type of system selected for use. This is yet to be established.

1.3.3 Operational Requirements

1.3.3.1 Equipment Shielding - Not Applicable

1.3.3.2 Top Shield and Plugs

1.3.3.2.1 Rotating Plug Control

The control of plug rotation shall be by remote or local command. The control dead band shall be established at limits that meet the index repeatability requirements of Paragraph 1.3.1.3.3. Visual position indication markings shall be provided in addition to any position indicating system incorporated with the indexing system.

1.3.3.2.2 Temperature Measurement

The shield shall contain sufficient temperature instrumentation to permit monitoring of temperatures in critical areas.

1.3.3.2.3 Other Instrumentation

The design shall include provisions for monitoring the following parameters:

1. Rotating seal inflation pressure
2. Large and small rotating plug drive torques

1.3.3.3 DBA Energy Absorbing System - Not Applicable

1.3.3.4 Top Shield Cooling System - To be determined

1.3.3.5 Biological Shield Cooling System - To be determined

1.3.4. Safety Requirements

1.3.4.1 Equipment Shielding - Not Applicable

1.3.4.2 Top Shield and Plugs

1.3.4.2.1 Structural Deflection

The structural deflections due to mechanical and thermal loads on the top shield shall be limited to values which will preclude problems with alignment of control rods and the fuel handling machines.

1.3.4.2.2 Access Plug Locks

The holddown mechanism for all normally removable access plugs shall incorporate features that do not rely on operator functions in securing the plug to the top shield. A redundant operator installed holddown shall also be supplied.

1.3.4.2.3 Shield Rotation Interlock

A mechanical interlock shall be provided to prevent top shield rotation during reactor operation.

1.3.4.2.4 DBA Energy Transfer

Under the DBA loading of paragraph 1.3.2.2.6 (c) the top shield shall retain its form and structural integrity to the extent required to:

1. Properly distribute the applied DBA energy to the crush columns of the DBA energy absorbing structure.
2. Provide for restriction of flow area through plug annuli into the DBA containment dome.
3. Preclude release of shield components such as access port plugs, plug drives, etc., or passage of core components to the containment dome to prevent penetration of the containment.

1.3.4.2.5 Hydrogenous Material Containment

Any hydrogenous material used in the shield shall be separated from the reactor interior by a high integrity barrier.

1.3.4.2.6 Fuel Element Holddown

The instrument support structure shall be designed to provide backup holddown for the fuel elements by limiting their upward movement. The backup holddown shall be designed to minimize restriction of flow through a raised element and shall not otherwise damage the element.

1.3.4.3 Top Shield and Biological Shield Cooling Systems

The cooling medium used in the top shield cooling system shall be inert gas; hydrogenous materials shall not be used.

Any hydrogenous material used as cooling fluids in the biological shield cooling system shall be separated from the reactor and/or cavity by a high integrity barrier.

1.3.5 Design Requirements

1.3.5.1 Applicable Codes and Standards

Federal

U. S. Code of Federal Regulations - "Protection Against Radiation"
Title 10, Part 20

U. S. Code of Federal Regulations - "Licensing of Production
Title 10, Part 50 and Utilization Facilities,"
Appendix A, "General Design
Criteria for Nuclear Power
Plant Construction Permits."

U. S. Code of Federal Regulations "Reactor Siting Criteria"
Title 10, Part 100

SOCIETY

American Institute of Steel Construction: Specification for
the Design, Fabrication, and Erection of Structural Steel for Buildings.

Research Council on Riveted and Bolted Structural Joints:
Specifications for Structural Joints Using ASTM A325 Bolts.

American Welding Society: Code for Welding in Building
Construction, AWS D1.0.

American Concrete Institute: Building Code Requirements for
Reinforced Concrete, ACI 318.

American Society for Testing Materials: Material Specifications.

National Fire Protection Association: Standards.

USA Standards: Code for Pressure Piping, USA B31.1

American Society of Mechanical Engineers: Boiler and Pressure Vessel Code.

American Nuclear Society: Proposed Standard 7.60, Leakage Rate Testing of Containment Structures.

Atoms International

SDD-096-10-000 Overall System Design Description - 1000 Mwe LMFBF Follow-on Study

1.3.5.2 Equipment Shielding

1.3.5.2.1 Design Criteria

The equipment shielding shall be designed to limit the thermal neutron flux to the following levels:

- 1) At the primary loop block valves
and cavity liner atmos. seals 1×10^6 n/cm²/sec
- 2) Equipment in the drain tank vault 1×10^5 n/cm²/sec
- 3) Areas containing secondary sodium 4×10^2 n/cm²/sec

1.3.5.2.2 Structural Criteria

Where the equipment shielding consists of concrete walls or other structures integral with the reactor building, the structures shall be classified as Class II B structures. They shall be designed in accordance with the criteria and requirements for Class II B structures as defined in SDD-096-90-000 "Land, Structures, and Conventional Systems Segment Conceptual System Design Description."

1.3.5.3 Top Shield and Plugs

1.3.5.3.1 Applicable Code

To the extent that it can be applied, the top shield shall be designed and constructed in accordance with the ASME Code for Nuclear Vessels (Section III Class A) including Case 1331.

1.3.5.3.2 Materials

1. Na Contact

Unless other materials are required for special characteristics, such as wear or adhesion resistance, all material in contact with sodium (liquid or vapor) at temperatures above 500°F shall be Type 304 Stainless Steel.

2. Coolant Contaminants

Materials of the top shield which could contribute to coolant contamination shall be contained within high integrity metal liners which will prevent the possibility of release of such materials to the coolant. Special care shall be taken when any potential contamination could result in significant reactivity, coolant channel plugging and/or dangerous chemical reactions.

3. Serpentine

The maximum permissible temperature to which the serpentine shield material may be subjected is 800°F.

1.3.5.3.3 Penetrations

The following access ports are required in the top shield.

1. In-core FHM Access Port

One port shall be located within the small rotating plug such that it can be indexed over any core element position and any storage position by appropriate orientation of the large and small shield plugs. This port shall be sized to permit removal of a cluster of 7 core elements. A semi-permanent plug (weld sealed) shall contain an inner port for the in-vessel FHM.

2. Ex-vessel FHM Access Port

One port shall be located in the large rotating plug, providing access to a ring of the element storage positions by appropriate indexing of the large rotating plug.

3. Control Rod Ports

Fifteen ports shall be provided in the small rotating plug for the control rods. These ports shall be located directly over the control positions when the reactor system is at nominal refueling temperature (500°F isothermal sodium).

4. Sodium Liquid Level Probe Ports

Two ports shall be provided in the large rotating plug for liquid level measuring instruments.

5. Instrumentation Ports

Penetrations shall be provided in the small rotating plug for core instrumentation.

6. Miscellaneous Access Ports

Three penetrations shall be provided in the small rotating plug for miscellaneous access and/or core sweep arm devices.

1.3.5.3.4 Design Features

1.3.5.3.4.1 Seals

All seals shall have multiple (2 or more) sealing surfaces and the design of the seal installations shall include features for continuous monitoring of seal effectiveness during reactor operation and shutdown.

1.3.5.3.4.2 Insulation

The insulation shall be designed to minimize deterioration of its performance due to accumulation of sodium or sodium oxide deposits.

1.3.5.3.4.3 Penetration Steps

The steps in all shield penetrations shall be sloped to minimize accumulation of sodium or sodium oxide deposits.

1.3.5.3.5 Design Criteria

1.3.5.3.5.1 Lifetime - The Top Shield System

The top shield system shall be designed for an operational life of 30 years. However, easily replaceable components, such as the plug seals and rotating plug drives, can be designed to normal commercial lifetimes.

1.3.5.3.5.2 Design and Analysis (Structural)

a. Operational - The system shall be analyzed for stress and deflection with the intent of the code listed in paragraph 1.3.5.3.1.

b. Seismic - All structural components of this system shall be designed as Class 1A structures as defined by SDD-096-10-000.

c. Operational Tolerances - All possible combinations of operational tolerances shall be evaluated and factored into the design.

d. Uncertainties - The level of uncertainty associated with each analysis method shall be considered and factored into the design,

e. Allowable Stress -

1. Type 304 - 30-year unirradiated. (The internal shielding will be such that the fast neutron dose, at the bottom surface of the shield will not cause deterioration of structural performance of any metallic material).

2. Other materials - Allowable stresses shall be established in accordance with the criteria used in the ASME Code for Nuclear Vessels (Section III) with due consideration given to the design lifetime (approximately 250,000 hours).

1.3.5.3.5.3 Operational Margins

a. Mechanical Loads - The rotating plug drives shall be designed to include a factor of 2 in drive torque over-capacity. The drive capacity calculation shall consider:

1. Maximum frictional loads due to seals
2. Maximum frictional loads due to bearings
3. Load required to accelerate plugs to operating speed
4. Inefficiency of motor, drive, gears, etc.

b. Radiation Shielding - The radiation level for the unlimited access high bay area shall include a safety factor of 2.5. The AEC 10CFR20 requirement is 2.5 m rem/hr.

1.3.5.3.5.4 Gas Entrainment/Collection

1. The bottom surface of the shield shall be submerged in the sodium pool when at operating level, to minimize the cover gas and sodium interface area.

2. The bottom surface of the shield shall be sloped at a 2-5° angle to prevent gas collection beneath the shield.

1.3.5.3.5.5 Na Frosting of Rotating Plug Annuli

The rotating plug annuli shall be protected from Na frost accumulation. Alternate concepts to accomplish this requirement shall be considered.

1.3.5.3.5.6 Condensate Drainage

All horizontal surfaces in the cover gas space shall be sloped or provided with features for run-off of sodium condensate.

1.3.5.3.5.7 Reactor Cover Gas Leakage

The leakage rate of Xe¹³³ through the following top shield seals shall not exceed:

Normal Operation

Rotating Plug Dynamic Seals (To be determined)

Static Plug Seals (To be determined)

Shutdown

Rotating Plug Dynamic Seals (To be determined)

Static Plug Seals (To be determined)

1.3.5.3.5.8 In-leakage

The top shield shall be designed to prevent inleakage of air, water, or other contaminants into the primary system or coolant under all normal and credible abnormal operating conditions.

1.3.5.3.5.9 Seal Redundancy

Redundancy shall be provided at all seals such that a single seal failure will not cause leakage limits (both operational and refueling) to be exceeded.

1.3.5.4 DBA Energy Absorbing System

1.3.5.4.1 Seismic Design

For seismic design purposes the DBA energy absorbing system is a Class I B structure as defined in SDD-096-10-000. Ground acceleration loadings, stresses and other design criteria shall be as defined in SDD-096-90-000 for the reactor building.

1.3.5.4.2 Dead Loads

Structure dead loads shall be determined on the basis of structure configuration, dimensions, and materials.

Estimated weights of some of the major pieces of equipment to be supported are:

Reactor and Reactor Vessel (To be determined)

Vessel Top Shield 700,000 lbs.

Vessel Ring Assembly (To be determined)

In-Vessel FHM w/Floor Valve 25,000 lbs.

In addition, it may be necessary to support the following major pieces of equipment in event of equipment malfunction:

Ex-Vessel FHM (To be determined)

Auxiliary HM (To be determined)

1.3.5.4.3

Containment Design Basis (CDB) Loadsa. Energy AbsorberKinetic Energy Input: 140×10^6 ft-lbMaximum Loading: 35×10^6 lb

Maximum Stroke: 4 feet

Missile Loading: (To be determined)

b. Support RingMaximum Loading: $\pm 35 \times 10^6$ lb (vertical)Max. Differential
Pressure1) reactor cavity
to containment
dome (To be determined)2) containment dome
to reactor cavity 25 psic. Support ColumnsMaximum Loading: $\pm 35 \times 10^6$ lbd. Reactor Containment Dome

Internal Pressure: 25 psig at a wall temp of 650°F

e. Blast Protection Components (To be determined)f. Pressure Relief Disc Assembly

Non-load bearing assembly

1.3.5.5

Top Shield Cooling System

Max. Top Surface Temp. of Shield 135°F

Heat Duty (To be determined)

Cooling Medium (To be determined)

Max. Temp. Diff. Across Dia. at Top Surface 50°F

1.3.5.6

Biological Shield Cooling System

Heat Duty (To be determined)

Cooling Medium (To be determined)

Radiation Fluxes (To be determined)

Max. Concrete Temperature 150°F
(Cavity Surface)

1.3.5.7

Interfaces

Items covered by this SDD interface with other segments of
the plants in the following manner:

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<u>Item</u>	<u>Interface</u>
Equipment Shielding	Land, Structures & Conventional Syst. Segment Reactor Building Structure Heat Transfer Segment Shields around piping and components
Top Shield and Plugs	Reactor Segment Reactor vessel Control rods DBA energy absorber Reactor Support System Segment Fuel handling components Control & Instru. Segment Reactor instrumentation Heat Transfer Segment Reactor sodium Auxiliary Systems Segment Helium system R/A vent system Component cooling system
DBA Energy Absorber	Reactor Segment Top shield Control rods and drives Reactor Support System Segment Fuel handling components Land, Structures & Conventional Sys. Segment Reactor Building Structure Reactor Containment Dome
Top Shield Cooling System	Reactor Segment Top Shield Auxiliary Systems Segment Component cooling system Land, Structures & Conventional Sys. Segment Reactor building structure Reactor containment dome
Biological Shield Cooling System	Auxiliary Systems Segment Component cooling system Land, Structures & Conventional Sys. Segment Reactor cavity Cavity liner

2.0 DESCRIPTION OF SYSTEM**2.1 Equipment Shielding**

The equipment shielding consists primarily of concrete surrounding the reactor and components of the heat transfer system. The general shielding arrangement is shown on the Reactor Building Equipment Arrangement Dwgs., N096960007 and N09696008.

The largest item is the bulk concrete immediately surrounding the reactor vessel. This concrete provides both biological and equipment shielding. At the lower level of the building the shield is used to protect operating personnel, on the south side, and equipment in the drain tank room, on the other side. For personnel protection the shielding is approx. 8 feet thick. For equipment protection, that is, to prevent excessive neutron activation of components, the shield is approx. 6 feet thick. Concrete thicknesses are based on ordinary concrete, at approx. 150 lbs/ cu. ft. At the upper level, the concrete serves to protect the main heat transfer components, and forms large pipe chases for the reactor inlet and outlet piping. The walls of the chases adjacent to the primary pumps are approx. 4 feet thick, based on concrete at 220 lbs/cu. ft. Supplemental shielding within the chases, in the area of the reactor nozzles, is provided to minimize neutron streaming. Borated material is used in these, and other locations, as required, to minimize bulk.

Equipment shielding is provided around the IHX within the heat transfer vaults. This shielding also serves as both biological as well as equipment shielding. During plant shutdown, the shield protects maintenance personnel in the vault from the IHX which could become an activity source due to fission product plate-out on the surface. During operation, the shield serves to further reduce the neutron flux level at the IHX in order to prevent significant activation of the secondary sodium. The shielding consists of approx. 18 inches of ordinary concrete.

2.2 Top Shield and Plugs

2.2.1 Top Shield Assembly

The top shield is the upper closure of the reactor vessel. It provides radiation shielding for the operating floor area, a seal for the reactor cover gas, a heat flow barrier, and mechanical support for the control rod drives. It is a double rotatable plug, and acts as an integral part of the fuel handling system. The lower portion of the shield is submerged in the sodium pool to minimize the cover gas-sodium interface area available for gas entrainment and solution. The sodium level is controlled by an overflow line located on the reactor vessel at this location. The bottom surface is sloped to prevent collection of gas below the shield.

The larger of the two plugs is concentric with the reactor vessel and turns on a thrust bearing approximately 25 ft. in diameter. The smaller plug at 15 ft.-6 in. in diameter is of sufficient size to accommodate all of the control rod drives and to allow for removal of the grid plate or other vessel internals if necessary. It is mounted on a separate thrust bearing and has its axis about 3 ft. from the center of the larger plug. The smaller plug has one penetration near its periphery for the in-vessel fuel handling equipment. This port is concentrically located in a larger seal-welded plug which has been provided to allow removal of several (7) stuck elements, should the need arise. The large plug has one penetration through which the ex-vessel fuel handling machine operates. In addition to the 15 ports for the reactor control rods, additional ports are provided for instrument (thermocouples, in-core multiplication monitors during refueling, core material surveillance thimbles, etc). tubes.

Over each fuel and blanket element, a thimble passes through the shield to an elevation approximately 4 inches above the top of the element for instrumentation. Instrument support structures extend down from the bottom of the small rotating plug to provide lateral positioning of the instrument thimbles. The instrument support structure also limits the upward movement of the core elements in the event hydraulic hold-down fails to keep the element seated.

2.2.2 Shielding

The overall thickness of the top shield is approximately 11 ft. In the current reference design the shielding consists of an approximately 15 inch thick structural top plate and approximately 5 feet of serpentine rock. Below are approximately 2 feet of reflective insulation plates and a 2.5 foot gas space which is vented to the reactor cover gas.

2.2.3 Shell

All components of the shield shell that are exposed to liquid sodium or hot cover gas are constructed of stainless steel. The two large plugs as well as all penetrations through the top shield, are "stepped" to prevent excessive radiation streaming. The serpentine rock is completely encased in the shell structure which is sealed and vented to an external gas handling system.

The reflective insulation, which provides a thermal barrier between the reactor sodium pool and the high bay area, consists of stainless steel sheets evenly spaced over a 24 inch depth. The insulation space and the gas space below are vented to the reactor cover gas atmosphere.

2.2.4**Seals**

Double static seals with a helium purge of the intermediate space are used at all plug penetrations. The seals for the two large plus are elastomer double lip seals with the intermediate space pressurized with helium. Friction is reduced during rotation of the plugs by reducing the pressure. During reactor operation an inflatable static seal located below the bearing isolates the bearing and the lip seal from the reactor atmosphere.

2.2.5**Bearings and Drives**

The rotating plugs are supported on large diameter ball thrust bearings. The bearings are approx. 25 and 16 feet in diameter at the ball pitch for the large and small plugs, respectively. The races and balls, made of 17-7 Ph steel, will require no lubrication.

Each rotating plug has a ring gear which is driven by electric motor thorough an appropriate gear reducer. Positive indexing, to locate the fuel handling ports over the core element and storage positions, is provided by tape control of the drive with a system similar to that used for controlling machine tools. Locking keys will be provided to secure the rotating plus in position during reactor operation. These insure close alignment of the shield over the reactor assembly and prevent inadvertent rotation of the shield plugs.

2.3**DBA Energy Absorbing System****2.3.1****Energy Absorber**

The energy absorber, shown on Dwg. No. N09620021 is composed of:

- a. ribbed-plate shell structure
- b. guided crush columns
- c. tension straps and shear lugs
- d. missile barrier
- e. guide pins

The ribbed-plate shell structure is statically supported by the tension straps. The structure is constructed from high strength welded steel plate. It is designed to transmit the crush column imposed load into the support ring via a ring of tension straps and shear lugs. Adequate clearance is provided between the structure and top shield mounted components to assure ease of removal and reassembly at fuel handling preparations with the assistance of guide pins.

The crush columns, distributed over the surface of the rotating plugs, are provided to absorb the DBA produced energy and distribute it into the ribbed structure. The number and location of the crush columns has not been determined. The columns are designed to crush uniformly through a 4' stroke and thus transmit a constant load into the structure. They are guided to preclude primary column instability failure during loading.

The tension straps transmit the crush column load into the support ring via the shear lugs, these lugs are removed to permit lifting of the structure for fuel handling access. A mechanical-electrical interlock is provided between the lugs and control rods to prevent reactor operation unless all lugs are in position.

Finally, a missile barrier is located over the control rods to prevent damage to the dome in the event of a control rod holddown failure.

2.3.2 Support Ring

The support ring is a 6' deep box-section ring which is designed to transmit the DBA loads from the energy absorbing structure and reactor vessel to the support columns. Also, this ring provides the primary support for the reactor vessel and top shield. The ring contains concrete shielding having a bulk density of about 200 lb/ft³ and a minimum of 12 penetrations for the pressure relief disc assemblies. The pressure loads taken out by the containment dome are transmitted to this ring via the instrumentation ring and the quick disconnect bolts.

2.3.3**Support Columns**

The support columns are designed to transmit the down and up loads imposed on the support ring during a DBA into the building foundation and also provide normal static support for the reactor/top shield system. These columns are isolated from the gallery concrete and welded at both the inner and outer containment boundaries to provide leaktight sealing under DBA pressure and loads.

2.3.4**Pressure Relief Disc Assembly**

The pressure relief disc assembly contains a burst disc set to relieve dome pressures up to 25 psig to the reactor cavity; dome pressures above the leak test pressure are thus limited to very short duration and do not create a radiological leakage hazard. The relief disc assembly also provides biological shielding. It is part of a removable plug that fits within the combined vessel inspection access ports and dome vents shown on Dwg. N09620021.

2.4**Top Shield Cooling System**

The detailed design of the top shield cooling system has not been established. A design presently under consideration involves cooling the upper surface of the shield by circulation of nitrogen through coils attached to the structural top plate. The nitrogen lines will be routed out through the instrument ring to a kerosene cooled heat exchanger within the reactor building. Hose-reel systems will be provided so that circulation of the coolant can be maintained during plug rotation for fuel handling.

Alternate design methods are being considered.

2.5**Biological Shield Cooling System**

The cooling system design which currently appears most promising is a forced circulation system involving pipe coils imbedded in the concrete surrounding the reactor. The piping is imbedded directly behind the steel plate which lines the inner surface of the reactor cavity. Water is used as the cooling medium. A separate cooling system is provided for the bottom plate of the cavity liner. This system is provided to prevent melt-through of the liner in the event of a DBA and reactor vessel rupture. Whether or not these two cooling systems will be combined has not been determined.

3.0 Principles of Operation

Operational aspects of the reactor shielding system are related to the rotation of the top shield, and the cooling systems for the top shield and equipment shielding. The top shield is rotated only during reactor shutdown, for the purposes of fuel handling. Therefore, its operation under these conditions is described in SDD-096-81-000, "Subassembly Handling."

3.1 Startup

Just prior to filling the reactor vessel and heat transfer systems with sodium, the cooling systems for the reactor top shield and for the equipment shielding will be activated. The top shield does not have heaters or other means of preheat. The temperature will rise due to exposure of the underside to reactor cover gas initially and then to sodium when the operating liquid level is reached. The cooling system will be adjusted to maintain the top surface at or below 135°F.

3.2 Normal Operation

During normal operation, the components of this system are passive except for their thermal response to the changes in sodium temperature and gamma heating.

Temperatures in critical areas of the shielding will be monitored and the shield cooling system controlled to maintain shield temperatures within design limits.

3.3 Shutdown

In general, once activated initially, the shielding systems, and in particular their associated cooling systems will not be shut down. If a problem develops requiring complete shutdown of a shield cooling system, it will usually be necessary to shut the reactor down and reduce sodium temperature to a minimum ($\sim 350^{\circ}\text{F}$) to minimize heat load on the affected shielding.

3.4 Special or Infrequent Operation

The most likely infrequent operational mode will be continued operation of the reactor and heat transfer systems with a temporary inter-

ruption of cooling to some portion of the shielding. Because of the large shielding mass generally involved, rates of temperature change will often be low enough to permit interruption of cooling for significant lengths of time (fractions of an hour to several hours). This operational mode requires close attention to shield temperatures, to assure that they do not exceed maximum design values. (They may exceed normal operating limits.)

3.5 Emergency Conditions

Emergency conditions associated with the shielding systems include loss of shield cooling, or a hypothetical explosive core disassembly (DBA).

3.5.1 Loss of Shield Cooling

Action must be taken to restore cooling. If this is unsuccessful, reactor power may have to be reduced (to reduce gamma heating in the affected shield) and/or sodium temperature may have to be reduced (to reduce thermal loads on the affected shield).

3.5.2 Design Basis Accident (DBA)

During a DBA a large impulse force will be exerted against the bottom face of the top shield. The shield will move upward, being resisted by energy absorbing crush columns of the DBA Energy Absorbing System, with a constant force of 35×10^6 lb through a 4 ft stroke. Although the shield and most other reactor components will be functionally destroyed, the shield will retain sufficient integrity to distribute the loading into the crush columns and to provide a high impedance to flow of reactor fluids into the containment dome space above the shield.

4.0 Safety Precautions

4.1 Hazards

The potential hazards associated with the Reactor Container/Top Shield System are:

- a. Excessive misalignment between the reactor core and the top shield, caused by lateral displacement of the vessel due to thermal effects or mechanical loads, failures of support structure, or inadvertent rotation of the top shield, can cause jamming of the reactor control rods and loss of ability to control the reactor.

b. Excessive radiation due to leakage of radioactive gas through the top shield and its seals can severely restrict required operations such as refueling or maintenance in the reactor building.

c. Leakage of radioactive gas during top shield seal maintenance or replacement can endanger operating personnel.

d. Excessive pressure of the reactor cover gas can lift one or both of the rotating plugs.

4.2 Precautions

Precautions which can be taken to minimize the possibility of occurrence of the above hazards or to mitigate their consequences are as follows:

a. Rapid thermal shocks on the reactor should be avoided whenever possible. This should minimize the possibility of misalignment between the reactor core and top shield.

b. Strict administrative control of top shield rotation must be exercised, over and above the interlocks which will be provided to prevent rotation during reactor operation.

c. Top shield seals should be checked for leakage during each scheduled shutdown and replaced as necessary.

d. Functional checks of the reactor cover gas control and relief system should be performed during each scheduled shutdown.

5.0 Maintenance

5.1 Normal or Routine Maintenance

5.1.1 Equipment Shielding

No maintenance is expected to be required for the equipment shielding.

5.1.2 Top Shield and Plugs

5.1.2.1 Rotating Plugs

The rotating plugs will be designed for operation without maintenance or replacement for the life of the plant. Although replacement of one or both of these plugs is possible, no special provisions, such as hoisting capability, are being included in the plant design.

5.1.2.2 Rotating Plug Bearings

The plug bearings will be designed for operation for the life of the plant without maintenance or replacement. Design of the plug bearing and seal installation does, however, allow replacement if required. Removal can be accomplished by removing the inner race to plug bolts while supporting the plug from above and then lowering the plug onto a step. The holddown ring, drive gear, inner race and outer race can then be removed in sequence. This procedure is reversed for replacement.

5.1.2.3 Rotating Plug Seals

The rotating plug seals will require replacement at intervals over the life of the plant (estimated lifetime 5 to 10 years). Replacement of the double lip seal and the static seals associated with this seal require only removal of the holddown ring and the ring gear. Replacement of the vapor barrier will require removal of the bearing, and use of plastic bagging technique or the containment dome as an inert atmosphere cell during the seal exchange.

5.1.2.4 Rotating Plug Drive and Indexing System

The plug rotation drive and indexing system components should be inspected, serviced and calibrated on a periodic schedule. A spare parts inventory should be maintained.

5.1.2.5 Access Plug Seals

The static seals on the access ports will be inspected and serviced each time the plugs are removed from the shield. An inventory of spare seals will be maintained.

5.1.2.6 Instrument Thimbles

The instrument thimbles will not require scheduled maintenance but some failures are anticipated and provisions for replacement are included in the design. The plug at each core position is replaceable as a unit. This will involve removal of the seal weld at the top surface of the shield, extraction of the old and insertion of a new unit using the auxiliary handling machine, and then replacement of the seal weld.

5.1.3 DBA Energy Absorbing System

The DBA Energy Absorber is a steel structure and as such should require minimal maintenance. Scheduled periodic visual inspections should be utilized to assess the condition of exposed surfaces, and any remedial action such as repainting will be scheduled for periods which would not interfere with generation of power.

The relief assemblies located in the support ring are removable. During the plant lifetime these assemblies should be periodically replaced and destructively tested to assure that the design relieving pressures are being maintained.

5.1.4 Top Shield Cooling System

Components such as the gas circulators and heat exchangers will require periodic inspection and maintenance. The system will be divided into partial capacity subsystems so that an individual portion can be isolated and shut down for service during reactor refueling when the shield heat load is approximately one-third of the operating heat load. The hose-reel units will only be used during refueling and are therefore available for inspection and service during reactor operation.

5.1.5 Biological Shield Cooling System

External components such as pumps and heat exchangers will be periodically inspected and maintained. The imbedded piping is not maintainable. Extra pipe coils are provided within the shield to assure the required availability and long service lifetime. The piping arrangement is such that the ends of individual pipe coils are accessible, to permit plugging of fail coils.

5.2 Unusual Maintenance

The major unusual maintenance task associated with the shielding system is removal, repair, and replacement of either of the rotating top shield plugs. This may be required if one of the plugs jams due to mechanical interference or sodium frost buildup. The anticipated sequence is:

- a. Remove containment dome, energy absorbing structure, control rod drives and drive extensions, and all fuel and blanket elements from reactor.
- b. Fabricate a cover plate for the opening to be left by the plug which is removed.
- c. Lower sodium temperature to approximately 300°F and establish an argon cover gas in the reactor. The sodium may be (optionally) drained from the reactor vessel; however, it may be preferable not to do so in order to prevent formation of sodium oxide at undesirable locations, and to utilize the sodium as a shield.
- d. Remove bearing and seal assembly from plug to be removed.
- e. Set up temporary shielding around plug opening to shield against radiation from the lower part of the plug.
- f. Install plug lifting fixture and remove plug. The building crane may be used to lift the small rotating plug. The large rotating plug must be jacked out of place. It may be desirable to remove these plugs into a plastic tent to minimize oxygen contamination of the reactor cover gas.
- g. Cover opening with specially fabricated cover plate.
- h. Perform necessary repairs.
- i. Replace plug.

6.0 Appendices

6.1 Drawings

N09620021 - "Reactor-Vertical Section - 1000 Mwe LMFBR Follow-On Study"

N09620022 - "Reactor-Plan View - 1000 Mwe LMFBR Follow-on Study"

N09696007 - "Reactor Building Equipment Arrangement - Plans - ANL 1000 Mwe Follow-on Study"

N09696008 - "Reactor Building Equipment Arrangement - Sections - ANL 1000 Mwe Follow-on Study"

6.2 References

SDD-096-10-000 - "Overall Conceptual System Design Description - 1000 Mwe LMFBR Follow-on Study"

SDD-096-20-000 - "Reactor Segment Conceptual System Design Description - 1000 Mwe LMFBR Follow-on Study"

SDD-096-30-000 - "Heat Transfer Segment Conceptual System Design Description - 1000 Mwe LMFBR Follow-on Study"

SDD-096-50-000 - "Control and Instrumentation Segment Conceptual System Design Description - 1000 Mwe LMFBR Follow-on Study"

SDD-096-60-000 - "Auxiliary Systems Segment Conceptual System Design Description - 1000 Mwe LMFBR Follow-on Study"

SDD-096-80-000 - "Reactor Support Systems Segment Conceptual System Design Description - 1000 Mwe LMFBR Follow-on Study"

SDD-096-90-000 - "Land, Structures, and Conventional Systems Segment Conceptual System Design Description - 1000 Mwe LMFBR Follow-on Study"

SDD-096-21-000 - "Reactor Vessel Assembly Conceptual System Design Description - 1000 Mwe LMFBR Follow-on Study"

SDD-09623-000
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REFERENCE DRAWINGS
N03620022 REACTOR - PLAN VIEW
N03620023 REACTOR - PLAN SECTION

ATOMICS INTERNATIONAL	
1000 MVA LMFBR FOLLOW-ON STUDY	
N03620021	
REACTOR - VERTICAL SECTION	
1000 MVA LMFBR FOLLOW-ON STUDY	
N03620021	

ATOMICS INTERNATIONAL	
1000 MVA LMFBR FOLLOW-ON STUDY	
N03620021	
REACTOR - VERTICAL SECTION	
1000 MVA LMFBR FOLLOW-ON STUDY	
N03620021	

REVISIONS		DATE	APPROVED
REV	DESCRIPTION

EQUIPMENT

1. REACTOR TOP SHIELD
2. REACTOR VESSEL
3. REACTOR VESSEL LINER
4. CAVITY LINER
5. REACTOR CONTAINMENT DOME
6. SPENT FUEL STORAGE VESSEL
7. FUEL CANISTER ENCAPSULATING CELL
8. REACTOR COMPONENT WASH CELL AND MISC. STORAGE PITS
9. SHIPPING CASE
10. SHIPPING CASE AIR LOCK - 8'-10" CLEAR
11. FUEL HANDLING GANTRY
12. PLUG FOR BELOW-FLOOR MAINTENANCE OF FUEL HANDLING MACHINES.
13. PROCESS EQUIP. WASH CELL - 8' DIA. - SIZED BY PRI. PUMP
14. VIEWING WINDOW
15. COOLING EQUIPMENT FOR SPENT FUEL STORAGE VESSEL
16. RECIRCULATING H.E.V. UNITS
17. PRE-AMP ROOM
18. PRIMARY PUMP DRIVE
19. VALVE OPERATORS
20. ACCESS PLUGS FOR IMX MAINTENANCE
21. COLD TRAP ACCESS PLUGS
22. IMX ENCLOSURE
23. EQUIP. ACCESS HATCH (THROUGH TO BOT. LEVEL)
24. PRIMARY PUMP
25. IMX
26. PRIMARY BLOCK VALVES
27. CHECK VALVE
28. THROTTLE VALVE
29. SECONDARY PIPING, PENETRATION DEALS (DOUBLE BELLOWS)
30. COLD TRAPS
31. H.E.V. AND SERVICE PIPE CHASE
32. PRESSURE RELIEF PANELS FOR VAULT RELIEF TO DRAIN TANK
33. BELOW (DDA RELIEF) - TYPICAL EACH VAULT
34. ATMOSPHERE SEAL - TYPICAL
35. REACTOR OVERFLOW TANK
36. DEGASIFIER/SURGE TANK
37. DEGASIFIER RETURN PUMPS

38. PRIMARY NO. SERVICE PUMPS
39. PRIMARY NO. SERVICE PIPE CHASE
40. PRIMARY NO. DRAIN TANK - 50,000 GAL
41. PRIMARY NO. DRAIN TANK - 80,000 GAL
42. R/A LIQUID WASTE TANKS - DRAINS FROM WASH & ENCAPSUL. CELLS
43. COVER GAS PURIFICATION SYSTEM
44. POST DDA COOLING EQUIP. - USED DURING NORMAL OPERATION - EMERGENCY EQUIPMENT IS OUTSIDE BUILDING
45. PERSONNEL AIR LOCK - 5'-6" x 7' CLEAR - AT GRADE
46. LARGE EQUIPMENT ACCESS OPENING - 32' DIA. - WILL CLEAR REACTOR TOP SHIELD AND IMX

FOR SECTIONS SEE DWG. NO. N09696008

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REACTOR BLDG. EQUIP. ARRANGEMENT - PLANS			
ANL 1000NHL FOLLOW ON STUDY			
DESIGN	DATE	BY	NO. 09696007
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SHEET NO.		OF 1	

SDP-096-73-000

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REVISIONS			
REV	ZONE	DESCRIPTION	DATE APPROVED

SECTION D-D

SECTION E-E

SECTION H-H

SECTION F-F

SECTION G-G

EQUIPMENT

1. REACTOR TOP SHIELD 2. REACTOR VESSEL 3. REACTOR VESSEL LINER 4. CAVITY LINER 5. REACTOR CONTAINMENT DOME 6. SPENT FUEL STORAGE VESSEL 7. FUEL CANISTER ENCAPSULATING CELL 8. SHIPPING CASE 9. SHIPPING CASE AIR LOCK 10. FUEL HANDLING GANTRY 11. PROCESS EQUIP. WASH CELL 12. VIEWING WINDOW 13. COOLING EQUIPMENT FOR SPENT FUEL STORAGE VESSEL 14. RECIRCULATING HEV UNITS 15. PRIMARY PUMP DRIVE 16. VALVE OPERATOR 17. ACCESS PLUG FOR IMX MAINTENANCE 18. IMX ENCLOSURE 19. PRIMARY PUMP 20. IMX 21. PRIMARY BLOCK VALVE 22. CHECK VALVE 23. THROTTLE VALVE 24. SECONDARY DIPPING PENETRATION SEAL 25. COLD TRAP 26. ATMOSPHERE SEAL 27. REACTOR OVERFLOW TANK 28. DEGASIFIER/SURGE TANK 29. DEGASIFIER RETURN PUMPS 30. PRIMARY NO. SERVICE PUMP 31. PRIMARY NO. DRAIN TANK - 55,000 GAL. 32. PRIMARY NO. DRAIN TANK - 30,000 GAL.	41. R/A LIQUID WASTE TANKS (PUMPS - DRAINS FROM WASH & ENCAPSUL. CELLS) 42. COVER GAS PURIFICATION SYSTEM 43. PRIMARY FLOWMETER 44. BUILDING CRANE - 120 TON CAPACITY 45. EX-VESEL FUEL HANDLING MACHINE 46. TOP SHIELD HOLD-DOWN STRUCTURE

FOR PLAN VIEWS SEE -
DWG. NO. N09696007

same as before

ATOMIC INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.
8800 BIRCH AVE., CARLSBAD, CALIF.

REACTOR BLDG. EQUIP. ARRANGEMENT - SECTIONS - AML 1000 MW FOLLOW-ON STUDY			
DRAWN	HAGGELTON	DATE	DEC 68
CHECKED	W. J. J.	BY	E
ENGINEER	W. J. J.	SCALE	N09696008
APPROVED		SHEET NO.	OF 20

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FORM 734-C REV. 3-69

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1.0 INTRODUCTION

Fuel and blanket elements are the building blocks which form the core of the fast breeder reactor. These elements must have the properties which shall result in sustaining a nuclear fission chain reaction upon assembly of the reactor. The elements must provide for production and transfer of the thermal energy released by the fission process to sodium coolant and have sufficient integrity to produce large amounts of thermal energy. In addition to providing a sustained chain reaction, the combination of fuel and blanket elements must promote the type of nuclear processes which ultimately achieve a net gain or breeding of fissionable material. Fast region fuel elements shall produce the bulk of the power; blanket region elements shall serve as a reflector to the core and as a region for fissile fuel breeding. Both solid and gaseous fission products shall be retained within the fuel and blanket element cladding.

The fuel and blanket elements must function compatibly with the physical, safety and regulatory aspects of the events of their life cycle. These events include: high production rate fabrication of components and assembly; pre-irradiation shipment to the plant site; pre-irradiation storage; installation in a core position by means of the Fuel Handling System; power operation within the core; movement to temporary storage in the ex-core spent fuel storage vessel; removal to a shipping cask; transport to a reprocessing facility.

1.1 Functional Requirements

The fuel elements and blanket elements shall:

1. Produce energy by fission process in a neutron energy spectrum which results in a net gain of fissile isotopes.
2. Contain the fuel and all isotopes produced, including fission gases, for the required life.
3. Hold the fuel in the required configuration.
4. In conjunction with reactor internals,
 - a) be capable of assembly into the core;
 - b) prevent wrong placement;
 - c) maintain the required core configuration.

1.2 Summary Description

1.2.1 Fuel Elements

The fuel elements are hexagonally shaped, thick-walled (0.140 in.) wrapper tubes 5.617 in. across flats containing 217 fuel pins, coolant inlet and hold-down fixtures, and upper-end fittings compatible with fuel handling needs. The fuel pin cladding is austenitic Type 316 stainless steel* tubing with a 0.300 in. outside diameter and a 0.0175 in. wall thickness. The cladding envelope constitutes the first radiological boundary and is designed to limit the total failure rate, from all causes, to less than 1% with a reliability of 0.999. The cladding contains a column of (U,Pu) O_2 fuel pellets 42.84 inches long, 12 and 15 inches of depleted UO_2 blanket pellets at the upper and lower ends respectively, and a 38 inch upper region for storage of fission product gases released from the fuel material. The fuel pins are spaced in the hexagonal housing by "wire-wrap" spacers.

The overall length of the fuel element is 180 inches. Structural material is Type 316 stainless steel. The elements are designed to preclude their mislocation in the core. Coolant orifices are provided in each element to regulate core coolant flow.

The maximum power rating of the fuel element is 10.171 Mwt. The elements attain an average discharge burnup of 67.360 Mwd/metric ton.

1.2.2 Blanket Elements

The blanket elements have the same external configuration and dimensions as the fuel elements. The hexagonal wrapper tubes are .100 in. thick. There are two types of elements for operation in the inner and outer rows of the radial blanket; they differ in the number of blanket pins and the diameter of the blanket pins.

* Type 316 is a tentative selection. Both 304 and 316 appear to have the necessary strength. Type 316 has been extensively irradiated, but 304 is more economical.

Each of the inner row blanket elements has 91 blanket pins with 0.50 inch outside diameter. The outer row elements have 61 pins with an 0.617 inch outside diameter. All blanket elements are cooled by forced convection from the high pressure plenum. Because of the steep neutron flux gradient through the blanket elements, all blanket elements are rotated 180° at mid-life. This tends to give the elements a more uniform burnup. The blanket pins contain depleted UO₂ pellets forming a column centered axially on the fuel core. The inner and outer radial blankets are 56.3 in. and 42.8 in. long respectively.

1.3 Requirements

1.3.1 Performance

1.3.1.1 Thermal

Core plus blanket power, Mwt	2400
Reactor inlet sodium temperature, °F	780
Reactor outlet sodium temperature, °F	1140
Core lifetime, years	2
Refueling interval, years	1
Plant factor	0.8

1.3.1.2 Hydraulic

All fuel and blanket elements shall be cooled by forced convection from the high pressure plenum.

Coolant Pressure Drop Across Element (plenum-to-plenum) at Rated Full Flow	87 psi
---	--------

Flow in individual fuel elements will be controlled by fixed orifices, sized to provide for ten core flow zones. Flow for the rated full power conditions are listed below. The tolerance on these flows is $\pm 2\%$.

Zone 1 flow	86.23 lb/sec
Zone 2 flow	(to be determined)
Zone 3 flow	(to be determined)
Zone 4 flow	(to be determined)
Zone 5 flow	(to be determined)
Zone 6 flow	(to be determined)

Zone 7 flow	(to be determined)
Zone 8 flow	84.47 lb/sec
Zone 9 flow	(to be determined)
Zone 10 flow	(to be determined)
Inner row blanket element flow	15.88 lb/sec
Outer row blanket element flow	7.68 lb/sec

1.3.1.3 Lifetime

<u>Element</u>	<u>Residence In-Core (years)</u>	<u>Average Exposure (Mwd/metric ton)</u>
Fuel	2.0	67.360
Inner blanket	5.0	15.010
Outer blanket	7.0	10.340

1.3.1.4 Steady-State Performance

Elements shall be capable of unrestricted operation throughout their specified lifetime at any power up to maximum rated power and at coolant outlet temperatures appropriate to the part-load temperature program specified in SDD-096-20-000.

The design requirements stated herein do not provide for a power stretch-level of operation.

1.3.1.5 Transient Performance

Elements shall be capable of a power change equal to 25% of rated load/min within the range of 25 to 110% of rated power in order to provide an emergency load pickup capability.

Element design, in conjunction with the protective system design, shall assure the capability for undergoing a 20% overpower transient without sufficient fuel center melting to cause economic damage.

Elements shall be of a power change of 2% per minute from 25% to 100% full load at one cycle per day.

Elements shall be capable of withstanding all protected accidents without damage which would impair performance up to 120% of nominal power.

1.3.1.6 Design Margins

Fuel element design margins shall be applied so that the reliability against all modes of failure will equal or exceed that required to satisfy the plant availability allocation (See 1.3.3).

1.3.2 Environmental Requirements

1.3.2.1 Coolant

Elements will be immersed in flowing sodium. Oxygen impurity will be less than 10 ppm. Other established impurity limits are listed in the SDD-096-30-000, Heat Transfer Segment.

1.3.2.2 Temperature

Maximum nominal steady inlet coolant temp. 780°F

Maximum nominal steady outlet coolant temp. 1140°F

The temperature transients expected during the lifetime of the reactor are given in SDD-096-20-000. These data shall be utilized for detailed stress analysis of the element in order to meet the requirements.

1.3.2.3 Pressure

Maximum normal operating pressure of high pressure plenum (core element inlet). 96 psig

External pressure at the top surface of grid. 9 psig

External pressure at element outlet. 5 psig

(See Section 1.3.6 for safety pressures)

Pressure transient (pump ramp) of high pressure plenum, once per day. 100 to 17 psig

1.3.2.4 Radiation

The radiation environment which the fuel and blanket elements will be exposed to is as follows:

TABLE II
RADIATION ENVIRONMENT

<u>Neutron Spectrum</u>	<u>Flux (1 & 2)</u> <u>n/cm²-sec</u>	<u>Exposure</u> <u>years</u>	<u>Fluence⁽¹⁾</u> <u>nvt (.8 p.f.)</u>
<u>Fuel Element</u>			
Greater than 100 kev		2	
Thermal (< 25 ev)		2	
<u>Inner Row Blanket Element</u>			
Fast (> 1.1 Mev)		5	
Epithermal (25 ev to 1.1 Mev)		5	
Thermal (< 25 ev)		5	
<u>Outer Row Blanket Element</u>			
Fast (> 1.1 Mev)		7	
Epithermal (25 ev to 1.1 Mev)		7	
Thermal (< 25 ev)		7	

(1) Values for the 1000 Mwe LMFBR are to be determined.

(2) Maximum flux values for each category. Category maximums may not occur at same location.

1.3.2.5 Matrix Loading

The core tightening device (refer to Reactor Vessel Internals Description, SDD-096-21-300) provided to control reactor assembly configuration against the fuel element bowing tendency will result in compressive forces applied at the housing contact pads of all elements. The magnitude of the peripheral constraint on the reactor assembly will depend upon two things. The first is the extent to which inter-channel mixing mitigates the temperature gradient which develops across the elements. The second is the amount of material "growth" due to the fast neutron irradiation. Maximum compressive forces are to be determined.

1.3.2.6 Mechanical

Vibrational excitation frequencies and magnitudes present in the environment will be determined.

Earthquake design criteria are to be determined.

1.3.3 Reliability/Availability Requirements

The reliability objectives of the fuel and blanket elements originate with the unavailability objective assigned to these elements. This objective states that the potential combination of a, b, and c, below shall not result in an energy availability loss greater than 0.43%.

- a. Unscheduled plant outage caused by the fuel element failures (e.g., rupture of more than 1% of the fuel pins).
- b. Extensions of scheduled outages caused either by inadequate design (e.g., fuel subassembly interference causing increased fuel handling times) or by improper handling.
- c. Plant reactor power derating resulting from inadequate fuel performance (e.g., core stretchouts).

1.3.4 Maintenance Requirements

Fuel elements are consumable items which must operate without maintenance between refueling intervals. Blanket elements must be rotated 180 degrees at their mid-life.

1.3.5 Operational Requirements

The fuel and blanket elements shall be capable of remote handling for installation and removal. The elements shall:

- a. Include a handling pickup at the top.
- b. Be self-locating in the core.
- c. Maintain integrity throughout their required lifetime.
- d. Contain all fission products produced within the reliability specified for the elements.

1.3.6 Safety Requirements

The fuel element safety requirements are as follows:

- a. The maximum allowable fuel pin motion shall not introduce a reactivity effect greater than the doppler reactivity increment.

b. The mechanical design of the fuel element shall minimize the dispersal of fuel pin spacers and element assemblies under conditions of sodium boiling.

c. The tendency for fuel slumping and redistribution under transient conditions shall be minimized.

d. The design shall provide a positive means of identifying an element and a positive means for orifice inspection prior to loading.

e. The fuel element flow exit shall induce mixing to assure prompt accurate T/C response to temperature perturbations, thus contributing to the early detection of system malfunctions.

f. The fuel element housing shall be capable of resisting transient sodium boiling pressure, local sodium-fuel interactions and all credible temperature transients without fuel element to element failure propagation.

1.3.7 Design Requirements

1.3.7.1 Element Types

The different types of fuel elements required and their identifying features are as shown in the following table.

<u>Types</u>	<u>Flow lb/sec</u> <u>See Sec. 1.3.1.2</u>	<u>Enrichment Zone</u> <u>Sec Sec. 1.3.8</u>	<u>Estimated Annual</u> <u>Throughput</u> <u>Elements/Year</u>
Type 1 fuel element	86.23	A	67
Type 2 fuel element	to be determined	A	
Type 3 fuel element	to be determined	A	
Type 4 fuel element	to be determined	A	
Type 5 fuel element	to be determined	A	
Type 6 fuel element	to be determined	A	
Type 7 fuel element	to be determined	A	71
Type 8 fuel element	84.47	B	
Type 9 fuel element	to be determined	B	
Type 10 fuel element	to be determined	B	13
Inner row blanket	15.88	C	
Outer row blanket	7.68	C	10

The axial and radial blanket material shall be depleted UO_2 pellets. Other requirements to be determined.

1.3.7.2 Dimensional

Fixed mechanical dimensions other than those designated as interface requirements (Section 1.3.8) are as follows:

Core height	42.84 inches
Axial blanket (upper)	12 inches
Axial blanket (lower)	15 inches
Radial blanket (inner row)	56.34 inches
Radial blanket (outer row)	42.84 inches
Lattice pitch	5.667 inches
Wrapper tube distance across flats	5.617 inches

1.3.7.3 Fuel

The fuel shall be mixed oxide, $(\text{Pu}, \text{U})\text{O}_{1.97}$. The fuel shall be solid pellet with a fabricated density of 88.3% theoretical density (corresponding to a smeared density of 85.0% theoretical density based on a mixture density of 11.06 grams/cc). The fuel elements shall have two types of enrichments.

1.3.7.4 Fission Gas Plenum

The fission gas plenum shall be above the fuel and the fission gas shall be contained within the fuel pin.

1.3.7.5 Installation

The elements shall be designed to mechanically preclude mislocating elements of different types within the core (Section 1.3.7.1 and 1.3.8.b).

1.3.7.6 Element Holddown

All elements shall provide for a hydraulic holddown against the upward coolant flow of section 1.3.1.2.

1.3.7.8 Hydraulic

a. Coolant mixing directly above the fueled core is desired to minimize fuel element bowing (Section 1.3.2.5).

b. The fuel element outlet coolant shall be mixed so that the coolant temperature sensed by the outlet temperature T/C shall not differ from the true mean effluent temperature by more than 15⁰F (See Section 1.3.5).

1.3.7.9 Structural Design

a. Cladding Stress

Allowable cladding stress limits shall be established to provide a 1000 hour margin on stress rupture life for the most severe combination of fuel pin gas pressure and cladding temperature; with due consideration for irradiation effects on material properties.

b. Housing Stress

Allowable housing stress limits shall be established to provide:

1. A 0.9999 probability that no two adjacent elements will so distort by creep or plastic deformation so that they will contact each other with sufficient contact pressure as to significantly increase the force required to withdraw an element.
2. A 0.9999 probability that the housing will not crack or rupture.

1.3.8 Interface Requirements

The fuel and blanket elements interface with the following systems:

- SDD-096-20-000 Reactor Segment
- SDD-096-21-000 Reactor Vessel Assembly
- SDD-096-21-300 Vessel Internals
- SDD-096-81-000 Subassembly Handling
- SDD-096-22-000 Reactivity Controls

The interface requirements are as follows:

- a. External configuration of all elements shall be as defined on Drawings N09624006 and N09624007.
- b. Diameter and tolerances of the elements and the grid plate holes at the upper and lower grid plate shall be as defined on the drawing. The diameter at the lower end of the element is to be determined.*
- c. The fuel element shall seat on the lower grid plate.
- d. Spacer pads on the hexagonal housing shall be located above the core section and shall have the dimensions and tolerances as shown on the drawing. Forces applied on the pads will be as specified in Section 1.3.2.5.
- e. Weight of the fuel element shall be approximately 750 lbs.
- f. The elements shall be capable of withstanding shipping and handling forces of 2.3g in any direction.
- g. The elements shall be capable of withstanding a 4000 lb. applied axial load during insertion or removal from the reactor lattice.
- h. The elements shall orient themselves within the core.

2.0 DETAILED DESCRIPTION

2.1 Fuel Elements (Drawing N09624006)

The fuel elements consist of 217 wire wrapped fuel pin assemblies, lower and upper end fitting assemblies, fuel pin support bars and a wrapper tube.

The fuel pins are located in a triangular lattice with a pitch of 0.3592 in. The pins are spaced by .0552 inch diameter wire wraps wound on a 12 inch pitch. Fuel elements will differ from each other depending on their radial location in the core. The differences arise from the ten orificing zones and two fuel enrichment zones. The fuel element is designed mechanically to preclude mislocation of a fuel element in the core. The weight of the element is approximately 750 lbs.

* These dimensions shall vary to preclude mislocating elements within the core.

2.1.1 Housing

The housing material is Type 316 stainless steel. It is a hexagon which is 5.337 inches across the inside flats. It is 118 inches long with a wall thickness of 0.140 inch. Holes are provided in the housing for attachment to the end fittings. The six external corners of the housing have spacer pads located 10-3/4 inches above core center line to space the elements from each other. The distance across pads flats is 5.667 inches.

2.1.2 Spacer

A helically-wrapped wire spacer is installed on each fuel pin. These are 0.0552 inch diameter stainless steel wires wrapped on a 12 inch pitch and welded to the end fittings. On the outside quarter turn of the pins next to the housing the wire height is reduced to 0.0247 inch.

2.1.3 Support Bars

The fuel pins are supported axially within the housing by support bars which pass through the slot in the pin lower end fitting and are in turn secured to an 0.062 in. thick support sleeve. Each element has 17 bars 0.093 in. thick by 1.250 in. wide.

2.1.4 Fuel Pin

The fuel pin consists of the cladding, core fuel, blanket fuel spring retainer, and end fittings.

The cladding is Type 316 stainless steel.* It is a 0.300 in. OD tube, has an 0.0175 in. wall and is approximately 109 inches long. The pin has a 12 - inch upper and a 15 inch lower blanket and a 42.84 inch core region. An upper gas reservoir (plenum) of 37.6 inches is provided. A retainer spring assembly positions the fuel in the fuel pin to preclude movement of the fuel during handling but permits axial thermal and burnup expansion.

The Type 316 stainless steel cladding is seamless tubing and has the following chemical composition.

* Type 316 is a tentative selection. Both 304 and 316 appear to have the necessary strength. Type 316 has been extensively irradiated, but 304 is much cheaper.

<u>Element</u>	<u>Percent by Weight</u>
Carbon	0.04 - 0.06
Manganese	2.00 Max.
Phosphorus	0.03 Max.
Sulfur	0.015 Max.
Silicon	1.0 Max.
Chromium	16.0 - 18.0
Nickel	12.0 - 14.0
Molybdenum	2.0 - 3.0
Copper	0.50 Max.
Cobalt	0.10 Max.
Boron	0.0005 Max.
Nitrogen	.05 Max.
Iron	Balance

The mechanical properties at room temperature are as follows:

Tensile strength	80,000 psi min.
Yield strength (0.2%)	45,000 to 60,000 psi
% elongation	45 min.

The fuel consists of ceramic pellets of substoichiometric (U,Pu)O₂ manufactured by cold pressing and sintering. The diameter of the cylinder is 0.262 inches and the length to diameter (L/D) ratio is 1/1 to 2/1. The oxygen to metal (O/M) ratio is 1.96 ± 0.02. The fabricated density of the fuel pellet is 88.3% of the theoretical density. The total impurity content of the pellets shall not exceed 4000 ppm. Individual impurity upper limits are:

<u>Element</u>	<u>ppm by Weight</u>
Al	200
B	5
C	250
Ca	100
Cd	5
Cl + Br + I	50
Co	8
Cr	200
Cu	85
F	25
Fe	500
Mg	50
Mn	50
Mo	150
N	200
Na	100
Nb	200
Ni	200
Pb	20
Si	200
Ti	40
V	25
W	200
Zr	200
Dy + Eu + Gd + Sm	5
H (exclusive of amt of water)	15

The average enrichments at mid-cycle equilibrium conditions are 10.5 and 13.14 atom %* for the inner and outer zones respectively. Initial enrichments are to be determined.

* Fissile atoms/U + Pu

2.1.5 Miscellaneous Structure

The upper fitting is Type 316 stainless steel. It provides a grapple to engage with the fuel handling machine. A guide way is machined into the fitting so subsequent fuel elements will align themselves as they are installed.

The lower fitting is Type 316 stainless steel. It engages the upper and lower core grid plates and seats on the lower grid. Holes are provided in the fitting between the grid plates to permit flow through the element. These holes vary in number to orifice each row in the core. The lower guide diameter and the bored hole in the lower grid plate vary for each row in the core to permit a "go" "no-go" for locating the elements.

2.1.6 Dimensions

Dimensional data for the fuel elements are listed in Section 6.1.1.

2.2 Blanket Elements (Drawing N09624007)

The blanket elements have a basic design configuration identical to that of the fuel element as described in Para. 2.1 except as follows:

1. The blanket material is depleted UO_2 contained within pin assemblies that are 0.500-in. OD and 0.617-in. OD in the inner and outer rows respectively.

2. Inner row elements have 91 pins and the outer row elements have 61 pins.

3. The radial blanket axial boundaries are at the axial blanket midpoints on the inner row elements and at the core boundaries on the outer elements.

2.2.1 Dimensions

Dimensional data for the radial blanket elements are listed in Section 6.1.2.

3.0 PRINCIPLES OF OPERATION

(See SDD-096-20-000)

4.0 SAFETY PRECAUTIONS**4.1 Criticality**

The fuel and blanket elements contain fissile isotopes and, under certain conditions, may constitute a critical mass. Criticality safeguards criteria and procedures are required to assure safe handling in compliance with applicable Federal and local regulations for handling, shipment and storage of the elements.

4.2 Radiation Hazard and Toxicity

The fuel material is radioactive and contains plutonium which is toxic when ingested. After irradiation in the core, both the fuel and blanket elements will be highly radioactive and will contain significant quantities of gaseous fission products. Safety criteria and procedures are required to assure safe handling in compliance with applicable Federal and local regulations for handling, shipment and storage and reprocessing of the elements.

5.0 MAINTENANCE

The fuel elements are consumable items which must operate without maintenance between refueling intervals. Blanket elements must be rotated 180 degrees at their mid-life.

6.0 APPENDICES**6.1 Design and Operating Data****6.1.1 Fuel Elements****Design Dimensions (70°F)**

1. Number of pins per element.	217
2. Fuel pin pitch, inch	0.3592
3. Fuel pin diameter, inch	0.3000
4. Wire-wrap wire diameter, inch	0.0552
5. Quarter wrap wire on outer pins diameter, inch	0.0247
6. Housing ID across flats, inch	5.337

7. Average assembly clearance (pin-to-pin and pin-to-wall), inch	0.0040
8. Housing wall thickness, inch	0.140
9. Housing OD across flats, inch	5.617
10. Fuel element cell size across flats, inch	5.667
11. Fuel clad thickness, inch	0.0175
12. Fuel pin clad ID inch	0.2650
13. Core length, inch	42.84
14. Lower axial blanket length	15.0
15. Upper axial blanket length	12.0
16. Gas plenum length, inch	37.6
17. Overall fuel pin length (w/o end fittings) inch	107.4
18. Mass of fuel per unit length, kgF/ft	0.102

Operating Data

1. Fuel pin pitch (hot), inch	0.3633
2. Fuel pin diameter (hot, clean), inch	0.3030
3. Wire diameter (hot) (12-inch pitch), inch	0.0557
4. Quarter wrap wire on outer pins diameter (hot), inch	0.0250
5. Housing ID across flats (hot), inch	5.387
6. Housing wall thickness (hot), inch	0.141
7. Fuel element cell size across flats (hot), inch	5.719
8. Fuel clad thickness (hot), inch	0.01767
9. Fuel pin diameter (hot, midcycle, average) inch	0.3053
10. Flow area (hot, midcycle, average), inch ²	8.743
11. Total sodium area (hot, midcycle, average) inch ²	9.234
12. Housing steel area (hot, average), inch ²	2.700
13. Clad and wire area (hot), inch ²	3.964
14. Total steel area (hot), inch ²	6.664
15. Fuel pin ID (hot, midcycle, average), inch	0.2700
16. Fuel area (hot, midcycle, average), inch ²	12.424
17. Total cell area (hot), inch ²	28.322

18. Fuel element cell volume fractions, fuel	0.4387
steel	0.2353
sodium	0.3260
19. Core length (hot), inch	43.26
20. Lower axial blanket length (hot), inch	15.11
21. Upper axial blanket length (hot), inch	12.12
22. Gas plenum length (hot), inch	38.00
23. Overall fuel pin length (hot), inch	108.49
24. Number of fuel elements	274
25. Average linear pins power, kw/ft	10.147
26. Mass per unit length of fuel pin, kgF/ft(hot)	0.1010
27. Max linear pin power (ith row element)kw/ft(hot)	16.03
28. Power splits, core	0.9062
axial blanket	0.0456
radial blanket	0.0482
29. Radial blanket outlet temp degradation, °F	11.5
30. Core mix-mean temp rise, °F	371.5
31. Peaking factors, ϕ_a (axial)	1.23
ϕ_r (radial)	1.22
ϕ_c (8th row)(circumferential)	1.053
ϕ_e (8th row)(elemental)	1.075
32. Row 8 max element power, kwt	9.963
33. Row 8 max flow rate, lb/sec	84.47
34. Central element power, kwt	10.171
35. Central element flow rate, lb/sec	86.23
36. Row 8 max power element average velocity, ft/sec	26.92
37. Central element average velocity, ft/sec	27.49
38. Row 8 Reynolds number	97.760
39. Central element Reynolds number	99.800
40. Row 8 friction factor	0.02205
41. Central element friction factor	0.02198
42. Row 8 bundle pressure loss, psi	67.07
43. Central element bundle pressure loss, psi	69.72

44. Row 8 element inlet loss, psi	7.53
45. Central element inlet loss, psi	7.85
46. Row 8 element outlet loss, psi	4.04
47. Central element outlet loss, psi	4.21
48. Element outlet sodium mixer and flow meter loss, psi	5.00
49. Row 8 element total pressure loss, psi	83.6
50. Central element total pressure loss, psi	86.8
51. Row 8 housing wall ΔP at top of core, psi	40.0
52. Central housing wall ΔP at top of core, psi	41.4
53. Sodium properties at 965°F, density, lb/ft ³	51.67
specific heat, Btu/lb-°F	0.3010
viscosity, lb/ft-hr	0.5667
thermal cond., Btu/hr-ft-°F	38.24
54. Equivalent diameter of bundle (hot), inch	0.1327
55. Heat transfer coefficient(row 8), Btu/hr-ft ² -°F	23,000
56. Peclet Number (row 8)	457
Nusselt Number (row 8)	7.22
57. Film ΔT at hot-spot (row 8), °F	13.9
58. Clad ΔT at hot-spot (row 8), °F	37.2
59. (Avg pwr of pins in 8th row elem)/(Avg pwr of fuel pins in the core)	1.135
60. Coolant temp rise of max pwr pin, row 8, °F	399.4*
61. Location of clad max temp, x/L**	0.5
62. Coolant temp around max pin pwr at x/L-0.5, °F	1170.3
63. Max clad outer surface temp. °F	1184.2
64. Max clad inside surface temp. °F	1221.4
65. Hot channel max inner pin clad temp. °F	1265.6
66. Hot spot max inner pin clad temp, °F(x/L-0.46)	1282

* Based on no mixing

** x - 0 is core

6.1.2 Radial Blanket Elements**6.1.2.1 Design Dimensions (70°F)**

	<u>Inner Row</u>	<u>Outer Row</u>
1. Pins per element	91	61
2. Pin OD, in.	0.5000	0.6170
3. Cladding thickness, in.	0.0175	0.0175
4. Wire wrap diameter, in.	0.0530	0.0570
5. Fuel pin pitch, in.	0.5547	0.6758
6. Housing ID across flats, in.	5.417	5.417
7. Housing wall thickness, in.	0.100	0.100
8. Housing OD across flats, in.	5.617	5.617
9. Cell size C/L - C/L	5.667	5.667
10. Fuel length, in.	56.34	42.84
11. Gas plenum length, in.	10	10
12. Overall fuel pin length, in. (w/o end fittings)	66.34	52.84
13. Number of Elements	66	72

6.1.2.2 Dimensional Data

Item	<u>Inner</u>	<u>Row</u>	<u>Outer</u>	<u>Row</u>
	<u>Cold</u>	<u>Hot</u>	<u>Cold</u>	<u>Hot</u>
1. Fuel length, in.	56.34	56.89	42.84	43.26
2. Fuel pin OD, in.	0.5000	0.5047	0.6170	0.6228
3. Cladding thickness, in.	0.0175	0.01766	0.0175	0.01766
4. Wire-wrap wire diameter, in.	0.0530	0.05349	0.0570	0.05753
5. Assembly clearance per pin, in.	0.00166	0.00199	0.00176	0.00175
6. Fuel pin pitch, in.	0.5547	0.5602	0.6758	0.6821
7. Pitch/diameter ratio	1.109	1.110	1.095	1.095
8. Housing ID across flats, in.	5.417	5.467	5.417	5.467
9. Housing wall thickness, in.	0.100	0.10093	0.100	0.10093
10. Housing OD across flats, in.	5.617	5.669	5.617	5.669

Item	Inner Cold	Row Hot	Outer Cold	Row Hot
11. Sodium gap between elements, in.	--	0.050	--	0.050
12. Cell size, distance between C/L, in.	5.667	5.719	5.667	5.719
13. Flow area (inside), in. ²	--	7.474	--	7.142
14. Equivalent (hydraulic diameter, in.	--	0.1675	--	0.1913
15. Sodium gap area, in. ²	--	0.4931	--	0.4931
16. Housing area, in. ²	--	1.947	--	1.947
17. Cladding area, in. ²	--	2.459	--	2.048
18. Wire-wrap wire area, in. ²	--	0.2045	--	0.1586
19. Fuel area (clad ID), in. ²	--	15.748	--	16.537
20. Volume fractions, fuel	--	0.556	--	0.584
sodium	--	0.281	--	0.269
steel	--	0.163	--	0.147

6.1.2.3 Thermal-Hydraulic Performance Data*

	Inner Row	Outer Row
1. Total power, Mwt	74.2	41.6
2. Average element power, Mwt	1.121	0.578
3. Maximum end-of-life element power, Mwt	1.766	0.910
4. Maximum fuel pin power (end-of-life), kw	27.17	18.65
5. Maximum linear pin power (end-of-life), kw/ft	8.296	6.363
6. Average coolant temperature rise of element at end-of-life, °F	350	373
7. Maximum nominal coolant temperature rise at end-of-life (50% mixing), °F	420	420
8. Maximum nominal coolant outlet temperature from hot channel at end-of-life, °F	1200	1200
9. Mix-mean coolant outlet temperature from element at end-of-life, °F	1130	1153
10. Mix-mean coolant outlet temperature from all elements, °F	1002	1016

* Midcycle equilibrium condition values given unless otherwise indicated.

	<u>Inner Row</u>	<u>Outer Row</u>
11. Element flow rate, lb/sec	15.88	7.68
12. Element bulk velocity, ft/sec	5.94	3.01
13. Reynolds number	27,600	16,000
14. Peclet number	122	70.6
15. Nusselt number	4.27	3.43
16. Heat Transfer coefficient, Btu/hr-ft ² -°F	11,600	8,150
17. Sodium film resistance, ft-hr-°F/Btu x 10 ³	0.602	0.775
18. Cladding resistnace, ft-hr-°F/Btu x 10 ³	0.855	0.688
19. Location of clad hot spot, x/L	0.981	1.00
20. Maximum nominal inside cladding temperature at end-of-life, °F	1206	1216
21. Assumed clad hot-channel factor	1.15	1.15
22. Maximum hot-channel clad inside temperature °F	1270	1281
23. Element pressure drop, psi	3.7	0.76
24. Degradation of reactor outlet temperature by blanket, °F		11.1

6.2**Drawings**

- a. N09624006 - Fuel Element Assembly - 1000 Mwe LMFBR
Follow-On Study
- b. N09624007 - Blanket Element Assembly of 1000 Mwe
LMFBR Follow-On Study

REVISIONS			
REV	DATE	DESCRIPTION	APPROVED
A		SEE REVISION RECORD	
		RD 52222

WEIGHTS	
INNER ROW DEPLETED U ASSEMBLY	144 KGS. 562 LBS.
OUTER ROW DEPLETED U ASSEMBLY	115 KGS. 491 LBS.

ATOMIC INTERNATIONAL
A DIVISION OF WESTINGHOUSE ELECTRIC CO.
PITTSBURGH, PENNSYLVANIA, U.S.A.

BLANKET ELEMENTS ASSEMBLY
1000 MW LMFBR FOLLOW-ON STUDY

NO 52222