DR-1307

AI-AEC-12791 (VOL. II)

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 II**

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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
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CONCEPTUAL SYSTEM DESIGN DESCRIPTIONS
VOLUME II

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DOCUMENT TYPE
System Design Description

PROGRAM TITLE

ANL 1000 Mwe Follow-on Study

KEY NOUNS

Heat Transfer Segment

DOCUMENT TITLE

Heat Transfer Segment Conceptual System Design Description - 1000 Mwe LMFR Follow-on Study

ORIGINAL
ISSUE DATE February 19, 1968

GO NO.

S/A NO.

PAGE 1 OF

7692

13120

REL. DATE

ABSTRACT

The requirements and description for the heat transfer segment for the 1000 Mwe IMFBR Follow-on Study are presented in this document.

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HEAT TRANSFER SEGMENTSYSTEM DESIGN DESCRIPTION1. INTRODUCTION1.1 Functions/Functional Requirements

The main heat transfer system removes the heat generated in the reactor core and transfers it to the steam generator for the production of steam to be used for the generation of electrical power. The main heat transfer system also provides for the removal of decay heat from the reactor following a shutdown for normal operational reasons or for abnormal emergency reasons. The main heat transfer system will provide for the isolation of radioactive sodium within shielded containment areas, and it will be designed to maintain this isolation in the event of a Design Basis Accident (DBA).

This SDD covers the reactor coolant systems, intermediate coolant systems; steam generators, steam and feedwater piping, and coolant fluids on the segment level. The interface areas with other systems are indicated by dotted process lines on Figure 1. Solid lines indicate portions of this SDD. Steam and feedwater piping is part of this SDD segment, but requirements and description of this piping is deferred to SDD-096-40-000, "Energy Conversion System."

1.2 Summary Description

The main heat transfer system consists of three circuits with two loops in each circuit. The two loops are the radioactive primary loop and the non-radioactive secondary loop. The primary loop removes heat from the reactor core by the forced circulation of sodium and transfers the heat to the secondary sodium in an intermediate heat exchanger (IHx).

A free surface pump in each circuit provides the circulation of the primary sodium. The heat is removed from the intermediate heat exchanger and transferred to the steam generator by the forced circulation of sodium with a free surface pump in each secondary loop.

The primary loops are installed in vaults within the reactor building. These vaults provide shielding and containment in the event of radioactive spill. The secondary loops are installed out of doors except for the secondary side of the intermediate heat exchanger and associated secondary loop piping within the primary equipment vaults.

1.3 Requirements

1.3.1 Performance

1.3.1.1 The heat transfer system shall transfer heat from the reactor core to the steam generator to permit the generation of the net full power of 1000 Mwe. This condition is the rated power condition.

1.3.1.2 The system shall be designed to deliver 6,854,000 pounds of steam per hour at 2415 psia and 1000°F at the turbine throttle and to reheat 6,401,900 pounds of steam per hour from 630°F and 573 psia to 1000°F and 515 psia. These design flows are nominally 100% of the rated flows required for the tandem compound turbine selected for the plant.

1.3.1.3 The system shall respond automatically to changes in power demands between 25% of rated load and maximum steam flow conditions. The load following capability shall be at least 25%/min. for emergency load changes and 2%/min. for normal load changes.

1.3.2. Environmental Requirements

The Main Heat Transfer System is designed to contain high temperature sodium. The primary loops and the reactor contain radioactive sodium. To minimize the possibility of a sodium fire in the event of a system leak, the entire primary system is installed in shielded vaults with nitrogen atmospheres.

During normal reactor operation, the primary loops contain sodium that has an equilibrium sodium 24 specific activity of 0.046 curies per cubic centimeter. The primary loops can be expected to be exposed to a neutron flux of 1×10^6 n/cm²/sec or less during normal reactor operation.

The secondary loops do not contain radioactive sodium, and, therefore, do not require shielding. A secondary sodium fire, consequently, is not attended with the same consequences as a fire in the primary system. Most of the secondary sodium systems can be an outdoor installation, as a result. The systems, however, will be designed to withstand the rigors of all weather conditions that may occur at the installation site. Fire retardant support structure for the steam generators will be provided.

1.3.2.1 Steady state conditions are as follows:

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	<u>Rated</u>
Primary Loop	
Reactor Output, Mwt (coolant)	2400
Reactor Inlet Temp., °F	780
Reactor Outlet Temp., °F	1140
Loop Sodium Flow Rate, lb/hr	25.2×10^6
Loop Pressure Drop, psi	150
Loop Sodium Charge, lbs.	$\sim 1.3 \times 10^6$
Cover gas pressure, psia	15
Secondary Loop	
IHX Inlet Temp., °F	670
IHX Outlet Temp., °F	1070
Loop Sodium Flow Rate, lb/hr	22.5×10^6
Loop Pressure Drop, psi	90
Loop Sodium Charge, lb	$\sim 1.6 \times 10^6$
Surge Tank Pressure, psia	115

1.3.2.2 Transient conditions are given in the tables in Section 6 (these tables are estimates and are provided for guidance only.)

1.3.2.3 Chemical Environment

Vault Atmosphere	Nitrogen
Oxygen Content	2% Max.
System Coolant	Sodium
Oxygen Content	10 ppm Max.
Hydrogen Content	0.002 ppm Max.
Carbon Content	20 ppm Max.

1.3.3 Reliability/Availability

The Main Heat Transfer System shall be designed for a 30 year plant life. The mature plant design energy availability objective is 87%. The unavailability allocation for the heat transfer segment is estimated to be 3.7%.

1.3.4 Maintenance

Maintenance shall be accomplished to the greatest possible extent while the plant is in operation. In general, redundancy shall be considered in order to assure continued plant availability. This is particularly true for those components that require plant shutdowns for maintenance action.

All equipment in the shielded cells shall be maintainable after draining the loop sodium, even if contaminated by fission products from failed fuel elements as described in 1.3.7.17. Maintenance by remote means may be necessary, e.g., remote maintenance on an IHX.

Other examples of maintenance include the following:

- Repair or replace spring hangers at 10 year intervals.
- Inspect instrumentation at 1 year intervals.
- Replace pump wearing parts at required intervals.
- Replace valve wearing parts at required intervals.
- Checking running vents for obstructions.

1.3.5 Operation

The design requirement is to permit operation of the plant in a load following mode between 25% of rated power and maximum steam flow conditions. The plant rated power (100 percent steam flow at 2400/1000/1000) shall correspond to the guaranteed gross output of the turbine-generator at 1.5 in. Hg absolute back pressure and zero percent makeup rate.

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1.3.6 Safety

The design of the system shall include due consideration to all factors that pertain to the safety of the system. Considerations that affect safety are represented by, but not limited to, the following:

- 1.3.6.1 Design Based Accident over-pressure of 200 psig.
- 1.3.6.2 Seismic loads of Class 1A earthquake as defined in SDD-096-90-000.
- 1.3.6.3 Prevention of uncovering the reactor core or outlet nozzle in the event of a rupture in the primary loop.
- 1.3.6.4 Prevention of leaks into the systems (such as oil, water, air) and prevention of leaks from the system (such as sodium and cover gases).
- 1.3.6.5 Prevention of loss of integrity of IHX in the event of a tube rupture in the steam generator.
- 1.3.6.6 Provisions for safely handling a sodium/water reaction.
- 1.3.6.7 Assure leakage from least radioactive system to radioactive system if a leak occurs.
- 1.3.6.8 Provisions for cooling of the reactor fuel under all predictable normal and abnormal operating conditions.
- 1.3.6.9 Prevention of large gas bubbles being swept through the core.
- 1.3.6.10 Continuous monitoring of reactor coolant fluid purity.
- 1.3.6.11 The reactor shall be shut down upon outage of more than one circuit.

1.3.7 Design

1.3.7.1 Applicable Documents

ASME	Section I, III & VIII, Unfired Pressure Vessel Code (current issue).
ASA B31.1	American Standard Code for Pressure Piping (current issue).
ASA C1	National Electric Code (1965)

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1.3.7.2 The system shall consist of three heat removal circuits. Each circuit to be of equal capacity and consist of a primary loop and a secondary loop. Each circuit is to be designed to be as independent as possible from the operation of the other circuits.

1.3.7.3 Each heat removal circuit shall be designed with natural convection capability such that any one circuit can remove all afterglow heat immediately after shutdown. The system shall be designed for 15 incidents of this type.

1.3.7.4 The main sodium pump motors shall be supplied by at least two independent sources of power.

1.3.7.5 Primary loop sodium shall be in the shell of the IHX. Secondary loop sodium, then, shall be in the IHX tubes.

1.3.7.6 The three secondary loops shall have steam generators designed and installed such that a failure in any one steam generator shall not affect the integrity of the steam generators in the adjacent loops.

1.3.7.7 The block valves installed in the primary loops shall not permit more than 2×10^{-3} cu. ft. per hour to leak from the reactor side of the valve with a differential pressure of 50 psi.

1.3.7.8 The primary loops shall be completely drainable.

1.3.7.9 The primary heat transfer loops shall be designed to provide for prevention of loss of coolant from the reactor core. The loss of coolant from one loop or loops shall not prevent coolant circulation in the remaining loops or loop, including a leak due to a guillotine type of failure in any one piece of primary sodium piping.

1.3.7.10 The design temperature and pressure for the primary coolant system shall be 1160°F hot leg*/800°F cold leg and 200 psig.

1.3.7.11 The design temperature and pressure for the secondary coolant system shall be 1090°F hot leg/700°F cold leg and 200 psig.

1.3.7.12 The Main Heat Transfer System shall be designed for a 30-year plant life.

1.3.7.13 The pressure of the secondary loops during operation shall be no less than 5 psi higher in the IHX than the pressure of the primary loops in the IHX. This requirement is designed to assure that any leakage will be from the non-radioactive secondary loop into the radioactive primary loop in the event of a leak in the IHX tubes.

*The design temperature contains margin over the maximum operating temperature of 1140°F. This is for a safety uncertainty factor, development flexibility, and stretch capability.

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1.3.7.14 It shall be possible to add sodium to the secondary loops to adjust minor level deficiencies without the need of depressurizing the loops.

1.3.7.15 The Main Heat Transfer System shall be subjected to a hot circulation test at a temperature of (to be determined).

1.3.7.16 The number of modules per bank of steam generators shall be commensurate with economics, H_2 detection capability, and water/steam leak damage isolation considerations.

1.3.7.17 The Heat Transfer System shall be designed for a sudden burst of fuel pins in the fuel element amounting to about 1% of the pins in the core, or continuous operation with 1% failed pins, whichever is worse. It shall be assumed that 0.01% of the solid fission products from these pins leach into the sodium coolant stream and distribute uniformly throughout the system.

1.3.7.18 Provisions shall be made to allow a continuous overflow from the reactor vessel to maintain a constant level in the vessel and to remove a sodium stream for degasification.

1.3.7.19 Design margins shall be applied to the design of the Main Heat Transfer System as follows:

a) Thermal capacity	The design thermal capacity shall correspond to the thermal requirements of the selected turbine at its rated capability. No margins above this shall be applied.
b) Pressure	Max. operating plus 10% or 15 psi, whichever is larger.
c) Temperature	Max. operating plus 1.5%.
d) Piping Corrosion	+ 0.030 in.

1.3.7.20 Running vents shall be provided to prevent gas accumulation in the upper region below the tube sheet of the Intermediate Heat Exchangers (IHX's). These vents shall discharge into the reactor overflow tank. The vents shall be orificed to permit a 10 gpm flow with the pressure drop existing at full loop flow.

1.3.7.21 The Main Heat Transfer System and appropriate subsystems shall be capable of containing the DBA without rupture into the secondary sodium loops. The DBA will result in a pressure of 200 psig at or below 1160°F, in the system external to the reactor shield cavity.

1.3.7.22 The IHX shall be designed to prevent loss of integrity in the event of a tube rupture in the steam generator.

1.3.7.23 In general, the materials that are in contact with sodium at reactor operating temperatures shall be Type 304 stainless steel. However, other materials such as carbon steel and 2 1/4 Cr-1 Mo may be used where compatibility, adequacy, and economics permit.

1.3.8 Interfaces

The main heat transfer primary loops interface with the reactor structure and vessel. The piping reactions and moments on the reactor vessel nozzles are shown in Figure 3 in the Appendix. The primary loops also interface with the sodium supply and treatment system, with the inert gas system, the electrical preheat system, the plant control system, the plant protective system, electrical system, R/A vent system, and site and structures.

The secondary loops interface with the inert gas system, the sodium supply and treatment system, the electrical system, the turbine generator system, the feedwater supply and treatment system, the plant control system, plant protective system, and site and structures.

2.0 DETAILED DESCRIPTION

The Main Heat Transfer System includes three circuits with two loops in each circuit. The system is shown in Figure 1 in the Appendix. The primary loops circulate coolant through the reactor core and are radioactive. The secondary loops transfer heat from the primary loops to the steam generation system and are not radioactive.

Studies of the number of heat transfer circuits indicate that, over the range of two to four, a two-circuit system is the most economic. A three-circuit system, however, was selected for the reference design because of the large size extrapolation from the demonstration plant which would be required for a two-circuit system. As experience is gained in the fabrication of large components and the operation of LMFBR's, the capacity of heat transfer circuits will likely be increased.

Each primary loop includes reactor inlet and outlet block valves, a pump, the shell side of an intermediate heat exchanger, a butterfly throttling valve for regulating loop flow and a check valve located in the reactor inlet piping. The pump in each loop is connected to the reactor vessel by a 44-inch pipe. The intermediate heat exchanger is connected to the pump by a 40-inch pump discharge pipe. The block valves and butterfly valves in the primary loops are remote operated. Each primary loop is provided with a flowmeter in the cold leg and other instrumentation necessary to the safe economical operation of the system. Drains and vents are provided at the appropriate points in the loops to allow adequate filling and draining. The vents are provided with either a vapor trap

or a freeze trap as required. A running vent from the upper plenum of each IHX to the reactor overflow tank is provided to prevent gas accumulation. The pipe size running from the IHX to the reactor is 38-inch.

Each primary loop supplies sodium to the reactor vessel low-pressure region. An 8-inch line with a 6-inch valve for flow regulation connects each primary loop between the reactor vessel and the reactor inlet block valve to the storage area.

The reactor vessel is provided with an overflow line located at the normal sodium level. Sodium flows continuously through the overflow line to the reactor overflow tank. This flow maintains the desired liquid level in the reactor and also allows for gas solution control.

A degasification system is provided as part of the Main Heat Transfer System for the removal of gas to prevent formation of gas bubbles in the sodium coolant. It consists of a reactor overflow tank, a degasifier/surge tank, two full capacity degasifier return pumps, and associated piping and instrumentation. Sodium from three IHX running vents and the reactor overflow line is collected in the overflow tank. The sodium flows from the overflow tank to the degasifier tank where it is mixed with cold leg sodium taken from each of the three reactor inlet lines. Any helium evolved from the sodium is vented and the sodium returned to each of the reactor inlet lines upstream of the block valves.

Each secondary loop is comprised of a surge tank, a pump, the shell side of one intermediate heat exchanger, the shell side of the steam generating equipment, a block valve at the steam generating equipment inlet and outlet, and a butterfly throttling valve at the reheater outlet for regulating the flow split between the superheater and reheater modules. The block valve at the inlet to the steam generating equipment is also a throttling valve used to regulate loop flow. These valves are remotely operated. The secondary sodium pressure is maintained higher than the primary sodium system pressure in the IHX through the inert gas pressure in the secondary sodium expansion tank.

The outlet piping from the IHX to the pump is 36-inch. The discharge from the pump to the superheater/reheater is a 36-inch pipe which splits into a 28-inch pipe to the superheater and a 26-inch pipe to the reheater. Between the superheater/reheater and the evaporator, the piping is 36-inch. The evaporator is connected to the intermediate heat exchanger by a 34-inch pipe. Each secondary loop is provided with a flowmeter in the cold leg and other necessary operational instrumentation including hydrogen detection instrumentation. Vents with appropriate vapor traps or freeze traps and drains are provided to allow adequate filling and draining.

The primary loops are located in the primary equipment cells within the Reactor Building. The secondary loops and steam generation equipment are located outside of, and adjacent to, the Reactor Building.

2.1 Component Description

2.1.1 Main Heat Transfer Pumps

Each primary loop contains one full capacity free surface vertical centrifugal pump capable of circulating 1/3 of the reactor coolant flow requirements when the plant is delivering power at the rated steam flow. Each pump consists of a pump barrel and bowl, impeller and shaft, diffuser, barrel shield plugs, variable speed electric drive and pump control, instrumentation, and cooling systems. Each pump can deliver 62,800 gpm against a 150 psi head. The pump is driven by a combination 7000 HP, 514 rpm wound rotor motor and a 2300 HP D.C. motor, both directly coupled to the pump. The impeller is about 71 inches diameter, the bowl is about 110 inches diameter, and the pump barrel is about 20 feet long. The space above the free surface is vented back to the reactor. The free surface level is maintained by means of overflow piping connected to the reactor upper plenum.

Each secondary loop contains one full capacity free surface vertical centrifugal pump similar to the primary loop pumps. Each pump is of a capacity that permits the circulation of 1/3 the total secondary flow for the 100% power case as for the primary pumps. Each pump can deliver 55,600 gpm against a 90 psi head. Each pump is driven by combination 4000 HP, 900 rpm wound rotor motor and a 1300 HP DC motor, both directly coupled to the pump.

The secondary loop pumps consist of the same components as the primary loop pumps except that no shield plugs are provided. Further, the secondary loop pumps are installed in the secondary loop surge tanks. This arrangement obviates the balance pipe requirement. The pump barrels are about 14 feet long, the impeller is about 30 inches diameter, and the pump bowl is about 55 inches diameter.

2.1.2 Reactor Overflow Tank

The reactor overflow tank is a vertical tank 2 feet 6 inches ID by 15 feet overall height. The tank has semi-elliptical heads and is constructed of 304 SS. There are three separate dip pipes penetrating the top head of the tank and extending down to a point near the bottom of the tank that corresponds to the elevation of the weir at the reactor piping outlet. The IHX running vent lines are connected to these dip pipes. The dip pipe arrangement assures immersion of their

open ends under the same circumstances that the reactor outlet lines are immersed.

The tank is fitted with a nozzle on the upper side for connection to the reactor overflow line. Two nozzles are provided on the top head for connection to the cover gas system and to allow installation of level instrumentation. A nozzle on the side at the desired operating liquid level provides an overflow line connection. The overflow line is connected to the degasifier/surge tank described below.

2.1.3 Degasifier/Surge Tank

The degasifier/surge tank is a horizontal tank 9 feet ID by 36 feet long with semi-elliptical heads. The tank has a capacity of 2100 cu. ft. and is constructed of 304 SS. The tank is provided with a dip pipe which is connected to the line returning the sodium back to the primary loops. A nozzle is also provided on the tank top for connection to the overflow pipe from the reactor overflow tank. On the side shell there is a nozzle for connection of the degasifier return pump barrel sodium overflow lines.

2.1.4 Degasifier Return Pumps

There are two full capacity degasifier return pumps with one in standby. The pumps are free surface centrifugal pumps with a design similar to the main heat transfer pumps. Each pump has a 10 inch impeller driven by a 400 rpm 75 HP electric motor. The pumps deliver 100 gpm against a head of 355 feet. The pump barrel sodium overflow is piped back to the degasifier/surge tank.

2.1.5 Intermediate Heat Exchangers

Each circuit contains one intermediate heat exchanger. The heat exchangers are vertical tube and shell units of the hockey stick design to accommodate tube-to-shell differential expansion. The units are of all welded 304 SS construction.

Each heat exchanger contains 3400 tubes, 1-inch O.D. by 0.058 inch wall. The tubes are arranged for single pass. The tubes average about 54 feet long. The overall exchanger dimensions are 60 feet long and 12 feet O.D.

2.1.6 Steam Generator

Each heat transfer circuit contains one steam generator bank. The modular concept is used for the steam generator. This concept utilizes sufficient hockey stick units connected in parallel to provide the required heat transfer area. Each bank, therefore, consists of 6 evaporator modules, 4 superheater modules, and 3 reheater modules. The modules contain 380 tubes

each. The shell and tubes of the evaporator are made of 2 1/2% chrome 1% moly steel, those of the superheater and reheater are made of 321 H SS.

The steam generator components are arranged vertically with the superheater, reheater and evaporator at the same level. Steam/water is in the tubes of each of the types of modules. All units are of the once through type.

To protect the steam generator shells from high overpressures, each module is provided with one rupture disc. The evaporator modules are equipped with one 8-inch disc each, and the superheater and reheater modules are each equipped with a single 4-inch disc. The relief lines from the modules are manifolded into one central 14-inch discharge line that blows into an 8000 cubic foot pressure relief tank.

2.1.7 Secondary Surge Tank

The secondary surge tank is a vertical 304 SS tank. The tank is 14 feet O.D. by 12 feet overall height. The heads are semi-elliptical. The tank is constructed such that the secondary loop pump barrel and bowl are a part of it with appropriate provisions made for differential thermal expansion between the tank and pump assemblies. The tank capacity is adequate for sodium volume changes occurring during power operation.

2.1.8 Valves

Each primary loop contains four valves. A 44-inch gate valve on the reactor outlet and a 38-inch gate valve on the reactor inlet are used for loop isolation. At the reactor inlet there is also a 38-inch balanced swing check valve. Just upstream of the check valve is the 38-inch butterfly control valve.

The secondary loops each contain three valves. At the inlet side of the superheaters/reheaters there is a 36-inch butterfly control valve. In the superheater outlet manifold is a 28-inch butterfly valve. At the outlet of the evaporators there is a 34-inch block valve.

The check valves are of all welded construction. The gate and butterfly valves use the gas cooled freeze seal principle to prevent leaks. The gate and butterfly valves are provided with valve operators. The valve operators for the primary loop valves are located in an accessible area in the Reactor Building with shafting to transmit the operator power to the valves.

All drain lines on the main heat transfer circuits are provided with double block valves or the equivalent.

2.1.9 Piping

All piping in the primary loops is type 304 stainless steel with all welded construction. The secondary loops are constructed of type 304 SS except in the steam generator bank. Welds are made using a heliarc root pass with an inert gas backing and a shielded arc filler rod. Adequate expansion loops are designed into the piping to minimize stresses from thermal expansions.

Drain lines are not smaller than 2 inches on the loops to limit unintentional blockage due to sodium solidification. Drain lines are additionally made as small as practicable to limit leakage rates in the event of their rupture.

3. PRINCIPLES OF OPERATION

3.1 Startup

3.1.1 Purging

Before admitting the sodium coolant into the various portions of the system, purging with helium must be carried out until the maximum oxygen content is reduced to 1% or less by volume. Appropriate connections are provided to accomplish this operation.

3.1.2 Preheat

It is necessary to preheat all sodium containing components to above the melting point of sodium, 208°F, to prevent plugging and thermal shocks when adding sodium. Heat is supplied by resistance heaters located within a heating oven surrounding the equipment. The preheat temperature is controlled by an on-off controller with a manual set-point. Thermocouples are attached to the equipment in suitable locations to indicate the overall preheat temperatures and to indicate when the desired preheat temperature is attained. The time to preheat will vary from 8 hours to 24 hours depending on pipe size, metal mass etc., being heated and the number of incremental preheat temperatures used. The desired preheat temperature is 350°F.

After reactor power operation there normally will be sufficient reactor decay heat to obviate the use of the preheat on components containing circulating sodium. When the loop is shut down or the system temperature drops below the preheat setpoint, the preheaters are put into operation to maintain a temperature of about 350°F. Some vent and vapor lines must be maintained hot at all times by the preheat system.

3.1.3 System Fill

Sodium at 350°F is pumped from the sodium treatment and supply system into the preheated main heat transfer system. The secondary loops are filled first. Sodium is filled in the secondary loops to the six feet level in the surge tanks at 350°F.

The primary loops are then filled with sodium. The reactor is filled first to the normal operating level at 350°F. Sodium is then admitted to the loops and filled to a level above the pump bearings. Sodium is slowly pumped into the system until the freeze traps, at the high point of the system, seal.

3.1.4 Approach to Load Following Operation

With the Startup, Shutdown, and Decay Heat Removal System (SDD-096-64-100) in service, main heat transfer temperatures will be allowed to increase at rates up to 50°F per hour as reactor power is increased. Sodium flow will be at least 25% of full flow. When steam temperature reaches approximately 485°F, and reactor power is at least 3%, the turbine-generator may be rolled and synchronized. When reactor power reaches 25% the Startup, Shutdown, and Decay Heat Removal System may be removed from service and the normal operation phase entered.

3.2 Normal Operation

3.2.1 Load Following

The normal operation of the main heat transfer system is fully automatic. The system has a load following capability of at least 25% per minute under emergency demands, and 2% per minute under normal demands. Reactor power and sodium flow rates are varied according to steam demands. Normally the primary and secondary loop throttling valves are wide open and flow control is accomplished by varying the speed of the pumps.

3.2.2 Overflow and Gas De-entrainment

As the primary loops and the reactor heat up, the sodium coolant expands. The excess volume overflows from the reactor vessel to the reactor overflow tank. After reaching operating temperatures, an overflow rate of about 10 gpm is maintained. This overflow stream goes into the reactor overflow tank where it combines with the streams from the IHX's. Running vents connected to the IHX upper shell area remove entrained gas that might otherwise collect there. The vents are orificed to permit only about 10 gpm flow for the maximum primary loop flow conditions. The vents each discharge into the reactor overflow tank through separate dip pipes.

The sodium from these two sources flows into the degasifier/surge tank where it mixes with "cold coolant" from the reactor inlet piping. Approximately 40 gpm comes from each primary loop. Thus the potential high gas content coolant from the overflow tank is cooled. The decrease in solubility of helium in sodium at reduced temperatures, will cause the overflow sodium to give up dissolved helium above the saturation temperature existing in the degasifier. The gas separates from the sodium in the degasifier and the sodium is pumped back to the primary loops upstream of the reactor inlet valves.

3.3 Shutdown

A normal shutdown is performed by gradually reducing reactor power and sodium flow to less than 25% of full power. At this point the Startup, Shutdown, and Decay Heat Removal System (SDD-096-64-100) is placed in service, and the turbine-generator load reduced to approximately 3% at which point it is tripped. Reactor power reduction is continued at a sodium flow rate of 25%, with sodium temperatures allowed to decrease at 50°F per 1/2 hour. When delta T across the reactor is less than 100°F, it can be tripped. If it is desired to reduce sodium flow below 25%, the throttle valves are used for flow control.

3.4 Special or Infrequent Operation

3.4.1 One-Circuit Shutdown

Should it be desirable to shut down one of the three circuits for maintenance while the other two remain in operation, it is accomplished by dropping flow in the loop to be shut down by first reducing pump speed until the check valve closes and then closing the block valves and shutting off the pump. The remaining two loops will pick up load until the two loop capacity of about 70% of rated load is reached.

3.4.2 Restoration of an Isolated Circuit To Operation

An isolated loop may be put back into operation if corresponding points in operating and down loops are within 50°F of each other. If this is not the case, the reactor must be shut down to lower the operating loop temperatures to within 50°F of the down loop. The reactor is then restarted and the three loops are brought up together.

3.5 Abnormal or Emergency Conditions

3.5.1 Reactor Trip

In the event of reactor trip, the sodium pump speed is automatically reduced and controlled to maintain the desired thermal condition until thermal convection takes over. The throttle valves then are used to control temperatures. If reactor trip were brought about by some condition that made the normal pump drive motors unavailable, the decay heat would be removed by means of natural circulation. Any one circuit is capable of handling the decay heat load.

3.5.2 Sodium Leak

An increase of radioactivity in a primary heat transfer vault may be an indication of a sodium leak. Verification that such radioactivity is indeed due to sodium can be accomplished by checking sodium levels in the affected loop, and by checking that the activity is due to sodium -24. The action to be taken is to shut down, isolate, and drain the affected loop.

3.5.3 Loss of Pumping Power

If pumping power is lost in a single heat transfer circuit, the check valve in the associated primary loop will close and the loop can then be isolated from the reactor until pumping power is restored.

4. SAFETY PRECAUTIONS

4.1 Hazards

The major hazards associated with operation of the main heat transfer system are:

- A. Loss of pumping power, followed by flow coastdown.
- B. Pipe rupture.
- C. Gas bubble entrainment.
- D. Sodium fires.

4.2 Precautions

A. If pumping power is lost, the protective system should trip the reactor. The operator should be ready to initiate manual trip immediately upon loss of flow if the protective system should fail to take proper action.

B. The system is designed to maintain reactor cooling even if a "guillotine" pipe rupture should occur. Additional precautions which should be taken are:

- 1) Pipe movement due to operation at temperature should be checked to verify that it is as predicted.
- 2) The frequency and magnitude of main heat transfer system thermal transients should be minimized.
- 3) Sodium purity should be maintained to minimize possibility of deleterious corrosion.
- 4) Stringent precautions should be taken during plant construction and subsequent maintenance shutdowns to avoid environmental damage (such as stress corrosion) to the primary piping and components.
- 5) To avoid entrapment of gas bubbles in the primary system piping, sodium flow should be increased to 100% prior to reactor startup, to sweep gas out of the system.

6) The reactor overflow and gas de-entrainment system should be kept in operation at all times.

7) If a single loop has been out of service and completely or partially drained, it should be returned to service only with the reactor shut down.

8) All sodium levels in the primary system should be carefully monitored to assure that they remain at their design levels.

9) If there is significant activation of the primary sodium, a low oxygen, nitrogen atmosphere should be maintained in the primary equipment vaults.

10) If it is necessary to cut into the primary piping system, the temperature of the system should be reduced well below the melting point of sodium.

5. MAINTENANCE

Heat transfer system maintenance methods are explained in the subsystem and component SDD's listed below:

SDD-096-31-000
SDD-096-32-000
SDD-096-33-000
SDD-096-31-100
SDD-096-31-220
SDD-096-32-100
SDD-096-32-220
SDD-096-32-300

6. APPENDIX

6.1 Design Data Sheets

6.2 Drawings

<u>No.</u>	<u>Title</u>
Figure 1	P & I Diagram - 1000 MWe Main Heat Transfer System
Figure 2	Heat Balance

6.3 Tables

Table 1	LMFBR System Thermal Transients System Operation
Table 2	Accidental Conditions. One Contingency
Table 3	Accidental Conditions. One Contingency
Table 4	Accidental Conditions. Two Contingency
Table 5	Accidental Conditions. Two Contingency

PREPARED BY: W. C. Burke				APPROVED BY:		DESIGN DATA SHEET		TITLE: 1000 MWe FBR Main Heat Transfer Segment	
LINE NO.	REVISION			ITEM	DESIGN POINT	REMARKS			
	NO.	DATE	REV						
1.				1. Function					
2.				1.1 To transfer thermal energy generated					
3.				in the reactor to steam generation					
4.				facilities.					
5.				1.2 To provide a highly reliable means of					
6.				removing reactor decay heat under any					
7.				credible set of emergency conditions.					
8.				1.3 To permit plant operation between 25%					
9.				and 100% of rated steam flow.					
10.									
11.				2. General Criteria					
12.				2.1 Heat transfer system to be based on two					
13.				loops per circuit concept					
14.				2.2 Three equal capacity circuits shall be					
15.				used.					
16.				2.3 Convection cooling for decay heat					
17.				removal is required					
18.				2.4 All primary loop sodium components					
19.				shall be located in nitrogen filled					
20.				shielded vaults					
21.				2.5 Cold traps are required, but only					
22.				provisions for hot traps are required.					
23.									
24.				3. Primary Loops					
25.				3.1 Number of loops	3				
26.				3.2 Capacity per loop	1/3 of 100%	Based on steam flow			
27.				3.3 Flow through reactor/lb/hr					
28.				Based on 100% power	75.58×10^6				
29.				3.4 Reactor outlet temperature, °F	1140				
30.				3.5 Reactor inlet temperature °F	780				
31.				3.6 Design temperature, °F					
32.				Hot leg	1160				
33.				Cold leg	800				
34.				3.7 Design pressure, psig	200				
35.									
36.									
37.									
38.									

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PREPARED BY: W. C. Burke				APPROVED BY:		DESIGN DATA SHEET		TITLE: 1000 MWe FBR Main Heat Transfer Segment	
LINE NO.	REVISION		NEW REV	ITEM	DESIGN POINT	REMARKS	NUMBER	REVISION LETTER	PAGE
	NO.	DATE							
1.				4. Primary Sodium Pumps					
2.				4.1 Number per loop	1				
3.				4.2 Type	Centrifugal	Vertical free surface			
4.				4.3 Flow rate per pump, gpm	62,800	100% power			
5.				4.4 Brake horsepower, BHP	7,000	100% power			
6.				4.5 Available NPSH, ft.	30				
7.				4.6 Total dynamic head, ft.	432	100% power			
8.				4.7 Pump speed control range	10 - 110%				
9.									
10.				5. Primary Piping Data					
11.				5.1 Reactor to pump, in OD x in. wall	44 x 1-1/16				
12.				5.2 Pump to IHX, in OD x in. wall	40 x 1				
13.				5.3 IHX to reactor, in. OD x in. wall	38 x 3/8				
14.				5.4 Material	304 SS				
15.				5.5 Reactor outlet block valve, size, in.	44	freeze seal gate			
16.				5.6 Reactor inlet block valve size, in.	38	freeze seal gate			
17.				5.7 Throttle valve size, in.	38	Freeze seal butterfly			
18.				5.8 Check valve size, in.	38	swing check			
19.									
20.				6. IHX Data					
21.				6.1 Number per loop	1				
22.				6.2 Type	shell & tube	hockey stick			
23.				6.3 Capacity per unit, 10 ⁸ Btu/hr	27.35	100% power			
24.				6.4 Number tubes per unit	3400				
25.				6.5 Tube size, in. OD x in. wall	1 x 0.058				
26.				6.6 Material of construction	304 SS				
27.									
28.				7. Secondary Loop					
29.				7.1 Number of loops	3				
30.				7.2 Capacity per loop	1/3 of 100	Based on steam flow			
31.				7.3 IHX inlet temperature, °F	670				
32.				7.4 IHX outlet temperature, °F	1070				
33.				7.5 Flow rate per loop, 10 ⁶ lb/hr	22.6	100% power			
34.				7.6 Design temperature, °F	1090				
35.				7.7 Design pressure, psig	200				
36.									
37.									
38.									

PREPARED BY: W. C. Burke				APPROVED BY:	DESIGN DATA SHEET	TITLE: 1000 MWe FBR Main Heat Transfer Segment	
LINE NO.	REVISION			ITEM	DESIGN POINT	REMARKS	
	NO.	DATE	REV				
1.				8. Secondary Sodium Pumps			
2.				8.1 Number per loop	1		
3.				8.2 Type	Centrifugal	vertical free surface	
4.				8.3 Flow rate per pump, gpm	55,600	Based on 100% power	
5.				8.4 Brake horsepower, BHP	4,000	100% power	
6.				8.5 Available NPSH, ft	50		
7.				8.6 Total dynamic head, ft	256	100% power	
8.				8.7 Pump speed control range	10 - 110%		
9.							
10.				9. Secondary Piping Data			
11.				9.1 IHX to supht/rehtr, in. OD x in. wall	36 x 5/8		
12.				9.2 Evaporator to IHX, in. OD x in. wall	34 x 3/8		
13.				9.3 Material	304 SS		
14.				9.4 Evaporator inlet block valve size, in.	36	freeze seal gate	
15.				9.5 Evaporator outlet block valve size, in.	34	freeze seal gate	
16.				9.6 Throttle valve size, in.	36	freeze seal butterfly	
17.				9.7 Suphtr/rhtr splitter valve size, in.	28	freeze seal butterfly	
18.							
19.				10. Steam Generator			
20.				10.1 Type	AI Modular	hockey stick	
21.				10.2 Banks per loop	1		
22.				10.3 Evap. mods. per bank	6	Once through type	
23.				10.4 Superhtr. mods per bank	4		
24.				10.5 Rehtr. mods. per bank	3		
25.				10.6 Suphtr. temp/reht temp/stm press, °F/°F	1005/1005/2470		
26.				psig			
27.				10.7 Suphtr. Stm. flow per bank, 10 ⁶ lb/hr	2.28	100% power	
28.				10.8 Rehtr stm flow per bank, 10 ⁶ lb/hr	2.13	100% power	
29.				10.9 Feedwater inlet temp., °F	478	100% power	
30.				10.10 Materials of construction (evaporator)	2% Cr - 1 Mo		
31.				10.11 Materials of construction (superheater & reheater)	321 H S.S.		
32.							
33.							
34.							
35.							
36.							
37.							
38.							

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PREPARED BY: W. C. Burke				APPROVED BY:		DESIGN DATA SHEET		TITLE: 1000 MWe FBR Main Heat Transfer Segment	
LINE NO.	REVISION			ITEM	DESIGN POINT	REMARKS	NUMBER	REVISION LETTER	PAGE
	NO.	DATE	NEW REV						
1.				11. Reactor Overflow Tank					
2.				11.1 Number of tanks	1				
3.				11.2 Volume, cu ft	74				
4.				11.3 Tank ID, in	30				
5.				11.4 Tank overall height, ft	15				
6.				11.5 Design temp, °F	1100				
7.				11.6 Design press, psig	200				
8.				11.7 Material	304 SS				
9.									
10.				12. Degasifier/Surge Tank					
11.				12.1 Number of tanks	1				
12.				12.2 Volume, cu ft	2100				
13.				12.3 Tank, ID, ft.	9				
14.				12.4 Tank overall length, ft	36				
15.				12.5 Design temp, °F	1100				
16.				12.6 Design press, psig	200				
17.				12.7 Material	304 SS				
18.									
19.				13. Degasifier Return Pump					
20.				13.1 Number of pumps	2				
21.				13.2 Type	Centrifugal	Vertical free surface			
22.				13.3 Design capacity, gpm	100				
23.				13.4 Brake horsepower, BHP	75				
24.				13.5 Available NPSH, ft.	35				
25.				13.6 Total dynamic head, ft.	355				
26.									
27.									
28.									
29.									
30.									
31.									
32.									
33.									
34.									
35.									
36.									
37.									
38.

LEGEND

GENERAL

LINE SIZE
LINE NO.
LINE NO.

VALVES

VALVE NO.

GENERAL

WATER LINE
WATER LINE

PIPE SPECIALTIES

PIPE SPECIALTIES

CONTROL VALVES

CONTROL VALVES

INSTRUMENTATION

INSTRUMENTATION

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FIG. 1

100% FLOW HEAT BALANCE

FIG 2

FIG 2

LMFBR System Thermal Transients

No.	Transient	Cause & Protective Syst. Action or Control Syst. Action	Number in 30 yrs	Power, %		Flow, %		Location *	Temperature, °F			
				Initial	Final	Initial	Final		Initial	Final	ΔT	Transient, °F/sec or °F/min
A	<u>System Operation</u>											
1	Preheat	Preheat. No PSA	3	0	0	0	0	U	70	350	+280	3 hour
2	Rise to Power	Plant Startup (CSA)	3	0	100	50	100	1	350	780	+430	+50/30 min
								2	350	1140	+790	+50/30 min
3a	Normal Load Decrease	Power Reduction -2%/min. (CSA)	9000	100	25	100	25	1	780	690	-90	-2.4/min
								2	1140	1050	-90	-2.4/min
3b	Normal Load Increase	Power Demand +2%/min. (CSA)	9000	25	100	25	100	1	690	780	+90	+2.4/min
								2	1050	1140	+90	+2.4/min
4a	Partial Load Rejection	Power Reduction -25%/min. (CSA)	720	100	70	100	70	1	780	740	-40	-33/min
								2	1140	1100	-40	-33/min
4b	Power Recovery	Loading After Rejection (CSA)	720	70	100	70	100	1	740	780	+40	+2.7/min
								2	1100	1140	+40	+2.7/min
5a	Refueling Shutdown	Power Reduction & Cooling (CSA) -2%/min	30	100	~2	100	~5	1	780	500	-280	-21/30 min
								2	1140	500	-640	-50/30 min
5b	Refueling Startup	Loading After Refueling (CSA)	30	~1	100	~5	100	1	500	780	+280	+21/30 min
								2	500	1140	+640	+50/30 min
6a	Maintenance Shutdown	Power Reduction & Cooling (CSA) -2%/min	30	100	0	100	0	1	780	350	-430	-27/30 min
								2	1140	350	-790	-50/30 min
6b	Startup After Shutdown	Loading After Maintenance	30	0	100	0	100	1	350	780	+430	+27/30 min
								2	350	1140	+790	+50/30 min

* 1 = Reactor Inlet Sodium

2 = Reactor Vessel Outlet Sodium

(PSA) = Protect. Syst. Action

(CSA) = Control. Syst. Action

U = Throughout the System (uniform)

TABLE 1

LMFBR System Thermal Transients

No.	Transient	Cause & Protective Syst. Action or Control Syst. Action	Number in 30 yrs	Power, %		Flow, %		Location #	Temperature, °F			
				Initial	Final	Initial	Final		Initial	Final	ΔT	Transient, °F/sec or °F/min
B	Accidental Conditions											
	One Contingency											
1	Spurious Reactor Trip (normal flow reduction)	Protect. Syst. Mal-function or Operator Error	85	100	< 5	100	< 5	1 2	780 1140	600 600	-180 -540	-18.8 / 30 min 200 / 8 sec, 0 / 10 sec, +140 / 30 sec, -50 / 30 min
2	Single Control Rod Drop (inserts 1# or more worth)	Holding Magnet Current Interrupted. (PSA). Short Negative Period Trip	15	100	< 5	100	< 5	1 2	780 1140	600 600	-180 -540	Same as above
3	Loss of Primary and/or Secondary Flow	Power Loss, Pump Seizure, Valve Closure										
3a.	One Primary Loop	(CSA) One Circuit Shutdown, Low Flow	10-30	100	~67	100	~67	1 2	780 1140	740 1140	-40 0	-40/min, +10/min +90/5 sec, -90/20 sec
3b.	Three Primary Loops	(PSA) . Power / Flow Trip	1	100	< 5	100	< 5	1 2	780 1140	600 600	-180 -540	-50/min +100/1.7 sec, -280/8 sec, 0/10 sec, +170/30 sec
3c	One Secondary Loop	(CSA) On Circuit Shutdown, Low Flow	10-30	100	~67	100	~67	1 2	780 1140	740 1140	-40 0	+120/60 sec, 150/120 sec +100/60 sec, -100/120 sec
3d	Three Secondary Loops	(CSA) . Two Circuit Shutdown, Low Flow	1	100	< 5	100	< 5	1 2	780 1140	600 600	-180 -540	+120/30 sec, -150/60 sec +100/30 sec, -200/8 sec, 0/10 sec, +100/30 sec
4	Cold Leg Temp. Rise (Primary)	During Shutdown and Decay Heat Operation. Flow Unbalance (high prim. flow). Initiated by Sudden Speed Reduction. No (PSA)	30	< 5	< 5	~20	< 5	1 2	650 660	500 800	-150 +140	-150/60 sec +140/60 sec

* 1 = Reactor Inlet Sodium
2 = Reactor Vessel Outlet Sodium

(PSA) = Protect. Syst. Action
(CSA) = Control Syst. Action

TABLE 2

LMFBR System Thermal Transients

No.	Transient	Cause & Protec- tive Syst. Action or Control Syst. Action	Number in 30 yrs	Power, %		Flow, %		Loca- tion *	Temperature, °F			
				Initial	Final	Initial	Final		Initial	Final	ΔT	Transient, °F/sec or °F/min
B	Accident Conditions (cont'd)											
5	Control Rod With- drawal	Equ'mnt Failure or Operator Error										
5a	At Startup, 25% Flow .05 \$/sec	Interlock Failure (PSA). Low Flux Trip	1	1	2	<2	25	1	600	600	0	0
								2	600	600	0	0
5b	Rods at 100% Power. 100% Flow. .05 \$/sec	Control Syst. Failure (PSA). Power Trip	1	100	<5	100	<5	1	780	600	-180	-17.7/30 min
								2	1140	600	-540	+50/4 sec, -200/8 sec, 0/10 sec, +120/30 sec, -50/30 min
6	Loss of Steam Plant Load	a. Fault of Transmission Line	60	100	<5	100	<5	1	780	600	-180	Same as B.1. Spurious Reactor Trip.
		b. Turbine Tripout	5					2	1140	600	-540	
		c. Generator Tripout	5									
		d. Excitation Tripout	3									
		e. Loss of Vacuum	6									
		f. Loss of Feedwater Pressure	1									
		g. Loss of Control Air	1									
7	Period Spike	Small Dispersed Gas Bubbles. 0.1 Void Fraction; 0.5 \$ at 6 \$/sec (PSA). Power Trip	10	100	<5	100	<5	1	780	600	-180	-17.3/30 min
								2	1140	600	-540	+60/0.6 sec, -200/8 sec, 0/10 sec, +120/30 sec, -50/30 min

* 1 = Reactor Inlet Sodium

2 = Reactor Vessel Outlet Sodium

(PSA) = Protect. Syst. Action

(CSA) = Control Syst. Action

TABLE 3

LMFBR System Thermal Transients

No.	Transient	Cause & Protec- tive Syst. Action or Control Syst. Action	Number in 30 yrs.	Power, %		Flow, %		Loca- tion *	Temperature, °F			
				Initial	Final	Initial	Final		Initial	Final	ΔT	Transient, °F/sec or °F/min
C	<u>Accident Conditions</u>											
	<u>Two Contingency</u>											
1	<u>Reactor Trip Without</u> <u>Normal Flow Reduc-</u> <u>tion:</u>	Pump Coupling Fails to Release										
1a	In One Primary Loop	(CSA). Auto Shutdown	1	100	<5	100	<5	1	780	600	-180	-40/min, -13.4/30 min
								2	1140	600	-540	-220/6 sec, 0/10 sec, +200/40 sec, -50/30 min
1b	In Two Primary Loops	(CSA). Auto Shutdown	1	100	<5	100	<5	1	780	600	-180	-80/min, -9.6/30 min
								2	1140	600	-540	-240/6 sec, 0/10 sec, +220/50 sec, -50/30 min
2	Spurious Reactor Trip Without Primary Flow Reduction	Pump Couplings Fail to Release. Pump Motors Run for 5 to 10 secs.	1	100	<5	100	<5	1	780	600	-180	-100/min, -7.6/30 min
								2	1140	600	-540	-270/6 sec, 0/10 sec, +260/60 sec, -50/30 min
3	<u>Loss of Primary and</u> <u>or Secondary Flow</u>	Power Loss, Pump Seizure, Valve Closure										
3a	One Prim. Loop & Fail- ure of Power/Flow Trip	(PSA). High Reactor Outlet Sodium Temp. Trip	1	100	<5	100	<5	1	780	600	-180	-40/min, -13.5/30 min
								2	1140	600	-540	+130/5 sec, -350/6 sec, 0/10 sec, +200/50 sec, -50/30 min
3b	Three Prim. Loops & Fail- ure of Power/Flow Trip	(PSA). Same as 3a.	1	100	<5	100	<5	1	780	600	-180	-100/min, -7.5/30 min
								2	1140	600	-540	+330/4 sec, -440/8 sec, 0/10 sec, +100/60 sec, -50/30 min
3c	Three Sec. Loops & Fail- ure of Power/Flow Trip	(PSA). Same as 3a.	1	100	<5	100	<5	1	780	600	-180	+150/30 sec, -200/60 sec, -12.5/30 min
								2	1140	600	-540	+150/30 sec, -250/8 sec, 0/10 sec, +80/30 sec, -50/30 min
3d	All Prim. & Sec. Loops & Failure of Power/Flow Trip	(PSA). Same as 3a.	1	100	<5	100	<5	1	780	600	-180	Same as 3b above
								2	1140	600	-540	

* 1 = Reactor Inlet Sodium

2 = Reactor Vessel Outlet Sodium

(PSA) = Protect. Syst. Action

(CSA) = Control. Syst. Action

TABLE 4

LMFBR System Thermal Transients

No.	Transient	Cause & Protective Syst. Action or Control Syst. Action	Number in 30yrs	Power, %		Flow, %		Location *	Temperature, °F			
				Initial	Final	Initial	Final		Initial	Final	ΔT	Transient, °F/sec or °F/min
C	Accident Conditions (cont'd)											
	Two Contingency											
4	Control Rod Withdrawal.	Equ'mnt Failure or Operator Error										
4a	Cold Startup, 25% Flow ≤ .05 #/sec & Failure of Low Flux Trip	(PSA). Power/Flow Ratio Trip	1	1	2	< 2	25	1	600	600	0	0
								2	600	600	0	+ 70/2.2 sec, - 70/60 sec
4b	One Rod . 100% Power and Flow . .005 #/sec & Failure of Power Limit Trip	(PSA). High Reactor Outlet Temp. Trip	1	100	< 5	100	< 5	1	780	600	-180	-17.3/30 min
								2	1140	600	-540	+ 80/50 sec, -260/8 sec, 0/10 sec, +160/30 sec, -50/30 min
4c	All Rods . 100% Power and flow . .05 #/sec & Failure of Power/Flow Trip	(PSA). High Reactor Outlet Temp. Trip	1	100	< 5	100	< 5	1	780	600	-180	-17/30 min
								2	1140	600	-540	+130/8.3 sec, -300/8 sec, 0/10 sec, +160/30 sec, -50/30 min

* 1 = Reactor Inlet Sodium

2 = Reactor Vessel Outlet Sodium

(PSA) = Protect. Syst. Action

(CSA) = Control. Syst. Action

TABLE 5

ATOMICS INTERNATIONAL <small>A Division of North American Rockwell Corporation</small> SUPPORTING DOCUMENT		NUMBER SDD-096-31-000		REV LTR B																																					
PROGRAM TITLE ANL 1000 Mwe Follow-on Study		DOCUMENT TYPE System Design Description																																							
DOCUMENT TITLE Reactor Coolant Conceptual System Design Description - 1000 Mwe LMFBF Follow-on Study		KEY NOUNS SDD - Reactor Coolant System																																							
ABSTRACT The requirements and description for the reactor coolant system for the 1000 Mwe LMFBF Follow-on Study are presented.		ORIGINAL ISSUE DATE 6-28-68 <table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 33%;">GO NO. 7692</td> <td style="width: 33%;">S/A NO. 13500</td> <td style="width: 33%;">PAGE 1 OF TOTAL PAGES 10</td> </tr> <tr> <td colspan="3">REL. DATE</td> </tr> </table>				GO NO. 7692	S/A NO. 13500	PAGE 1 OF TOTAL PAGES 10	REL. DATE																																
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REACTOR COOLANT SYSTEM CSDD**1. INTRODUCTION**

The reactor coolant system is a high temperature liquid metal heat transfer system. Included in this system are the piping and equipment from and including the primary (shell) side of the IHX's to the inlet and outlet of the reactor vessel.

1.1 Functions/Functional Requirements

The purposes of the reactor coolant system are as follows:

1.1.1 To remove heat generated in the reactor core and transfer it to the intermediate heat transfer system through the IHX's.

1.1.2 To transfer heat at the temperatures and rates required by plant power generation demands.

1.1.3 To provide cooling to the reactor core under abnormal as well as normal conditions of operation.

1.1.4 To allow safe operation of a high temperature radioactive liquid metal system.

1.2 Summary Description

The reactor coolant system consists of three parallel loops of equal capacity. A free surface pump in each loop provides the forced circulation of sodium coolant. Flow is out of the reactor vessel upper plenum to the pump, to the inlet nozzle of the primary side of the IHX's, and then back to the inlet nozzle at the reactor vessel lower plenum. Valves are installed in each loop for isolation, for flow control, and for flow reversal prevention.

The primary loops are installed in vaults within the reactor building to provide shielding and containment.

1.3 Requirements**1.3.1 Performance**

1.3.1.1 The reactor coolant loops shall provide for the circulation of coolant through the reactor core at the conditions demanded by the reactor core sufficient for the generation of the net full rated power of 1000 MWe by the plant.

1.3.1.2 The reactor coolant loops shall automatically provide the flow conditions required to meet power demands between 25% rated load and maximum steam flow conditions. The load following capacity shall be at least 25%/min. for emergency local changes and 2%/min. for normal load demands.

1.3.2 Environmental

The reactor coolant system is required to operate with high temperature sodium coolant that is radioactive. To minimize the possibility of a sodium fire and to provide containment in the event of a system leak, the reactor heat transfer system is installed in shielded vaults with nitrogen atmospheres.

1.3.2.1 Steady state conditions for the reactor coolant system are as given in SDD-096-30-000, paragraph 1.3.2.1.

1.3.2.2 Transient conditions for the reactor coolant system are as given in SDD-096-30-000, paragraph 1.3.2.2.

1.3.2.3 The chemical environment that the reactor coolant system is subject to is as given in SDD-096-30-000, paragraph 1.3.2.3.

1.3.2.4 The radiation environment acting upon the system will be as stated in SDD-096-30-000, paragraph 1.3.2.

1.3.3 Operational

It is a requirement of the reactor coolant system to permit operation of the plant in a load following mode between 25% of rated power and max. steam flow conditions. All operational adjustments to the system shall be through manual remote or automatic means, as appropriate.

1.3.4 Safety

The design of the reactor coolant system shall give due consideration to factors pertaining to safety. Some of these factors are as follows:

1.3.4.1 Design basis accident over-pressure of 200 psi.

1.3.4.2 Seismic loads of Class 1A earthquake as defined in SDD-096-10-000.

1.3.4.3 Prevention of uncovering the reactor core or outlet nozzle in the event of a rupture in the primary loops.

1.3.4.4 Prevention of leaks into the system (such as oil, water, air) and prevention of leaks from the system (such as sodium and cover gases).

1.3.4.5 Prevention of large gas bubbles being swept through the core.

1.3.4.6 Continuous monitoring of reactor coolant fluid purity.

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

1.3.5 Design

1.3.5.1 The design temperature and pressure for the reactor coolant system shall be 1160°F hot leg/800°F cold leg and 200 psig, respectively.

1.3.5.2 The system shall be designed for a 30 year plant life.

1.3.5.3 Running vents shall be provided to prevent gas accumulation in the upper region below the tubesheet of the IHX's. The vents shall be orificed to permit a 10 gpm flow when the loop is experiencing full flow conditions.

1.3.5.4 Type 304 stainless steel shall be the general material of construction of the system. Specialized use material such as valve trim material, bearing material, etc., shall be compatible with the system and its contents in all respects.

1.3.5.5 Applicable documents shall be as given in SDD-096-30-000, paragraph 1.3.7.1.

1.3.5.6 The reactor coolant system shall consist of three equal capacity heat removal loops operating in parallel. Each loop shall be designed to operate as independently as possible from the operation of the other loops.

1.3.5.7 Each reactor coolant loop shall be designed with natural convection capability to meet the requirements of SDD-096-30-000, paragraph 1.3.7.3.

1.3.5.8 The reactor coolant loops shall be designed to prevent loss of coolant circulation through the reactor core upon rupture of one or two loops.

1.3.5.9 The reactor coolant loops shall be capable of operation with 1% of the fuel pins in the core being ruptured. It shall be assumed that 0.01% of the solid fission products from these pins leach into the sodium coolant stream and distribute uniformly throughout the system.

1.3.5.10 All equipment and piping shall be covered with thermal insulation.

2. Detailed Description

The Reactor Coolant System includes the three primary loops of the Main Heat Transfer System only. Each primary loop is of equal capacity operating in parallel. Each loop includes the reactor inlet and outlet block valves, a free surface pump, the shell side of an intermediate heat exchanger, a butterfly throttling valve, and a check valve located in the reactor inlet piping. The piping from the reactor vessel outlet to the pump suction nozzle is 44 inch pipe. From the pump discharge to the IHX, the connecting pipe is 40 inches. From the IHX to the reactor vessel inlet, the connecting pipe is 38 inches. A flowmeter is installed in the cold leg of each loop. Other instrumentation is provided as necessary for safe economical operation.

Drains and vents are provided at appropriate points in each loop for adequate filling and draining. A running vent from the upper plenum of each IHX is provided to remove any gas that may collect there. These vent lines are orificed for 10 gpm flows at full primary loop flow.

Each loop of the reactor coolant system is located in a primary equipment cell within the reactor building. An atmosphere of nitrogen is provided in the vaults to prevent sodium fires in the event of leakage. The vaults provide containment and shielding for the radioactive loops.

2.1 Component Description

2.1.1 Main Heat Transfer Pumps

Each loop of the reactor coolant system is equipped with one free surface centrifugal pump capable of circulating 1/3 of the reactor coolant flow requirement when the plant is delivering power at the rated steam flow. Each pump consists of a pump barrel and bowl, impeller and shaft, diffuser, barrel shield plugs, variable speed electric drive and pump control, and monitoring instrumentation. Each pump can deliver 62,800 gpm against a 150 psi head. The pump is driven by a combination 7000 HP, 514 rpm wound rotor AC motor and a 2300 HP DC motor, both directly coupled to the pump. The space above the free surface is vented back to the reactor cover gas system. The free surface level is maintained by means of overflow piping connected to the reactor upper plenum.

2.1.2 Valves

The primary loops each contain four valves. A 44-inch double wedge gate valve on the reactor outlet and a 38-inch double wedge gate valve on the reactor inlet are used for loop isolation. A 38-inch balanced swing check valve is installed in the reactor inlet line just upstream of the gate valve. A 38-inch butterfly valve is used for throttling on convection flow control on the reactor inlet. The double wedge gate valves are provided with a bleed connection between the wedge for leakage control.

The check valves are of all welded construction. The gate and butterfly valves use the gas cooled stem freeze seal principle to prevent leaks. The gate and butterfly valves are provided with valve operators. The valve operators are located in an accessible area in the reactor building with connecting shafting to transmit the operator power to the valves located in the equipment cells.

2.1.3 Piping

All piping in the primary loops is type 304 stainless steel with all welded construction. Welds are made using a heliarc root pass with an inert gas backing and a shielded arc filler rod. Adequate expansion loops are designed into the piping to minimize stresses from thermal expansion.

2.1.4 Intermediate Heat Exchangers

Each reactor coolant system loop contains one IHX. Reactor coolant system loop flow is through the shell. The IHX's are vertical tube and shell units of the hockey stick design to accommodate differential thermal expansion between the tubes and shell. The units are of all welded 304 SS construction.

3. Principles of Operation

Covered in SDD-096-30-000, "Heat Transfer Segment System Design Description".

4. Safety Precautions

4.1 Hazards

The hazards associated with operation of the reactor coolant system include the following:

- a) Coolant leakage from system to heat transfer equipment vault.
- b) Introduction of inert gas into the system, followed by transport of the gas through the system and then through the reactor.

4.2 Precautions

Operational precautions which should be taken to minimize the probability of occurrence and/or the consequences of the aforementioned hazards are:

A. Coolant Leakage

1) A nitrogen atmosphere should be maintained in the primary heat transfer vaults at all times when the system contains significant radioactivity. In case of a coolant leak, the nitrogen atmosphere will prevent a sodium fire.

2) Caution should be exercised during preheating, filling, transient operation, and shutdown to minimize the frequency and magnitude of system thermal transients.

3) Preheating of the system after a shutdown for maintenance should be accomplished progressively, proceeding from a free sodium surface to other parts of the system, to allow a path for expansion of possibly solidified sodium as it is heated and liquified.

B. Gas Introduction and Transport

1) To avoid entrapment of gas bubbles in the primary system piping, sodium flow should be increased to 100%, prior to reactor startup, to sweep gas out of the system.

2) The reactor overflow and gas de-entrainment system should be kept in operation at all times.

3) If a single loop has been out of service and completely or partially drained, it should be returned to service only with the reactor shut down.

4) All sodium levels in the reactor coolant system should be carefully monitored to assure that they remain at their design levels.

5. Maintenance

5.1 Normal/Routine Maintenance

Routine maintenance on the intermediate heat transfer system involves pump drive lubrication, pump bearing and seal check, checking of valves for leakage and proper operation, checking system for leaks, electrical systems testing, and instrumentation checkout. Maintenance is scheduled during refueling or steam turbine maintenance periods. Routine checks may be made during plant operation, and a primary heat transfer loop may be shut down for maintenance while the plant operates on remaining loops.

5.2 Unusual Maintenance

Unusual maintenance tasks which can be anticipated include:

- 1) Primary coolant pump removal/repair/replacement (See SDD-096-32-100).
- 2) Large Sodium valve removal/repair/replacement (See SDD-096-31-220).
- 3) Repair of sodium leaks in piping or equipment. This maintenance activity requires the following major steps;
 - a) Draining the system
 - b) Removal of insulation and electrical heaters
 - c) Cutting out the leaking section, while maintaining an inert atmosphere in the system
 - d) Welding in a new section
 - e) Replacement of heaters and insulation
 - f) Preheat and refill of the system
 - g) Cold trapping to remove contaminants from the sodium

6. Appendices

6.1 Drawing

Figure 1, "Reactor Coolant System Schematic"

FIGURE 1
REACTOR COOLANT SYSTEM
SCHEMATIC

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

NUMBER

SDD-096-31-100

REV

LTR B

DOCUMENT TYPE

System Design Description

PROGRAM TITLE

ANL 1000 Mwe Follow-on Study

KEY NOUNS

SDD-Reactor Na
Pumps and Drives

DOCUMENT TITLE

Pumps and Drives (Reactor Na)
Conceptual System Design Description - 1000
MWe IMFBR Follow-on StudyORIGINAL
ISSUE DATE

8/5/68

GO NO.

7692

S/A NO.

13140

PAGE 1 OF

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REL. DATE

ABSTRACT

The requirements and description for the reactor sodium pumps and drives for 1000 MWe IMFBR Follow-on Study are presented.

PREPARED BY

DATE

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W. C. Burke

731-153

APPROVALS

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1.0 INTRODUCTION

The reactor sodium pumps and drives are installed in the reactor coolant system. This SDD describes these pumps and drives.

1.1 Function/Functional Requirements

The function of the reactor sodium pumps and drives is to circulate sodium coolant for the transfer of heat from the reactor core to the IHX.

1.2 Summary Description

The reactor sodium pumps and drives consist of a vertical free surface centrifugal pump, and a variable speed electric motor drive. There is a single pump in each of the three reactor coolant loops. Since the pumps will handle radioactive sodium, the pumps are provided with a shield plug installed in the pump upper barrel.

1.3 Requirements

1.3.1 Performance

1.3.1.1 The reactor sodium pumps shall circulate sodium coolant as required for the generation of the net full power of 1000 MWe, which is the plant rated power condition.

1.3.1.2 The reactor sodium pumps shall have the capability to provide variable flow rates to meet power demands between 25% of plant rated power and maximum steam flow conditions.

1.3.2 Environmental

1.3.2.1 The reactor coolant pumps shall circulate hot radioactive sodium. The specific activity of the sodium will be 0.046 curies per cubic centimeter.

1.3.2.2 The reactor coolant pumps will be subjected to the steady state conditions as listed in paragraph 1.3.2.1 of SDD-096-30-000.

1.3.2.3 The reactor sodium pumps will be subjected to the transient conditions given in paragraph 1.3.2.2 of SDD-096-30-000.

1.3.2.4 The reactor sodium pumps will be exposed externally to nitrogen gas at 150 F except for the drives which will be in air ranging in temperature from 60 F to 100 F.

1.3.2.5 The reactor sodium pumps will be exposed to a neutron flux that will not exceed 1×10^6 n/cm²/sec.

1.3.2.6 The reactor sodium pumps will use helium cover gas over the sodium free surface in the pumps. The cover gas will be radioactive with a normal activity of 5.6×10^{-3} c/cc of Xe¹³³ or less.

1.3.2.7 The reactor sodium pumps will circulate sodium having impurities as listed in paragraph 1.3.2.3 of SDD-096-30-000.

1.3.3 Operational

1.3.3.1 The pumps shall be capable of satisfactory operation over a speed range of at least 10 to 1.

1.3.3.2 Each pump shall be provided with level instrumentation which will measure and indicate level in the pump barrels. Full instrumentation for the seal oil system is required, including level, pressure, temperature, and flow.

1.3.4 Safety

1.3.4.1 The shaft seal shall be made at the upper end of the pumps and be removable without removal of the shaft, or inner rotating assembly; and shaft seal removal shall not require pump shield plug removal.

1.3.4.2 The seal, thrust bearing, and lubricant system shall be arranged so that a failure of the seal will not permit lubricant to enter the sodium in the pump.

1.3.4.3 The seal coolant and lubricating system shall be an integral part of the pump assembly.

1.3.4.4 For personnel safety, the reactor coolant pumps shall be provided with a shield plug. The pump barrel shield plug shall be a closed container removable with the inner pump column, impeller, and rotating assembly without disturbing the pump barrel.

1.3.4.5 The reactor sodium pumps shall be designed to contain the maximum shock loadings from the DBA.

1.3.4.6 The reactor sodium pump seal leak limitation shall be 18.5 curies Xe¹³³ per day for a pump cover gas concentration of one curie per cc.

1.3.4.7 The pumps shall be designed for seismic loads for Class 1A components as defined in SDD-096-90-000.

1.3.5 Design

1.3.5.1 Applicable Documents

ASME Boiler and Pressure Vessel Code,
Section III, Class "A" Nuclear Vessels

ASME Boiler and Pressure Vessel Code,
Section VIII, Unfired Pressure Vessels

ASAS American Standard Code for Pressure Piping,
Nuclear Power Piping, B 31.7

NEMA Standards

ASA National Electrical Code

1.3.5.2 The reactor sodium pumps shall be designed for the full rated capacity of 62,500 gpm and a TDH of 432 feet. The maximum efficiency shall correspond to this operating condition.

1.3.5.3 The pumps shall be designed to meet the following conditions:

Fluid pumped	Sodium
Design pressure, psig	200
Design temperature, F	1160
Fluid Operating temperature, F	1140
Fluid density @1140°F lb/ft ³	50.0
NPSH available, ft.	30

1.3.5.4 Each pump shall be designed for a service life of 30 years.

1.3.5.5 Each pump shall be of the vertical free surface type.

1.3.5.6 Each pump shall be driven by a combination wound rotor motor and a DC motor, both directly coupled to the pump. This combination shall provide the required speed range of 10 to 1.

1.3.5.7 The motor, pump, coupling, and pump rotating parts shall be statically and dynamically balanced such that there are no critical speeds below 115% of rated speed.

1.3.5.8 The pumps shall be equipped with pressurized hydrostatic radial bearings, cooled and lubricated with the flowing sodium stream. The bearings shall be sized for continuous operation at all pump speeds from 10% to 110% of rated speed, and for short periods (less than 30 minutes) at speeds from zero to 10% of rated speed.

1.3.5.9 Pressure containing parts of the pumps shall be fabricated in accordance with the applicable portions of the ASME Boiler and Pressure Vessel Code, Section III, Class "A" Nuclear Vessels. A code stamp on each pump is required.

1.3.5.10 Type 304 stainless steel shall be used in the fabrication of the pump barrels, casings, shafts, impellers, and other components exposed to the sodium or inert gas environment. This requirement does not apply to the choice of hydrostatic bearing materials which shall be stellite or equal.

1.3.5.11 The primary pump barrel shield plug shall be designed to support a full load of steel shot without excessive deflection. Provisions for filling the plug with shot shall be provided. A four foot depth of steel shot is required for shielding purposes.

1.3.5.12 Halogenated hydrocarbons shall not be used as lubricants for the pump seal and thrust bearing. Lubricants used shall be as compatible as possible with sodium and its vapors to avoid explosions and system damage through contamination if the lubricants should get into the sodium coolant stream.

2.0 DETAILED DESCRIPTION

The primary pump is a free surface, vertical single suction centrifugal pump capable of circulating 1/3 of the reactor coolant flow requirements when the plant is delivering power at 100% of steam flow.

Each pump consists of a pump inner and outer barrel and bowl, impeller and shaft, diffuser, barrel shield plug, variable speed electric drive, control instrumentation and cooling system.

The pump outer barrel connects to the bowl at the bottom and rises to floor level in the reactor building. The outer barrel forms the pump housing that has the concrete floor and vault shield cast around it. The outer barrel has a bolt flange at floor level for connection to the inner barrel and forms part of the housing seal. The sodium level is maintained in the barrel by an overflow which is connected to the reactor vessel by a 8" pipe. The pump cover gas is vented to the reactor vessel cover gas by a 2" vent line, thus maintaining the same cover gas pressure in the pump as that over the reactor vessel.

The pump bowl connects to the outer barrel, the 44" suction pipe from the reactor vessel, and the 40" discharge line to the IHX. The approximate size of the bowl is 110 inches in diameter. The suction connection is located at the base of the bowl and the discharge is located on the side.

The pump inner barrel contains the lower sodium radial hydrostatic bearing housing, thermal barrier assembly, and radiation shield. The inner barrel, instrumentation thimbles, and all internal components are removable for maintenance. The only items that are not removable are the bowl and outer barrel. The radiation shield is penetrated by two instrumentation thimbles. The primary source of radiation which establishes the shielding requirements is the sodium.

The impeller is of the single suction radial type. The impeller diameter is about 71 inches. The pump shaft is connected to the impeller and electromagnetic coupling. The shaft and impeller are supported by the upper thrust bearing and aligned by the lower sodium bearing.

The diffuser is supported by the bowl and is located at the circumference of the impeller. It should be noted that the number of vanes in the diffuser should be different (preferably odd) to the number of vanes in the impeller to reduce the pump noise generation problem.

The distance from the upper housing seal (floor level) to the center line of the discharge is 22 ft. This is the approximate length of the shaft.

The pump is driven by a combination 7000 HP, 514 rpm wound rotor motor and a 2300 HP DC motor, both directly coupled to the pump.

The bearing and seal are mounted above the radiation shield and both are cooled by kerosene. The seal is a double mechanical seal. A drip deflector and catch basin located below the seal is used to prevent oil from entering the sodium in event of seal failure.

Instruments used to measure pump operation will be mounted as part of the pump and include thermocouples, level probes, and shaft tachometer. The thermocouples are chromel-constantan and the level probes are two-point alarm with a three-coil element at each level alarm point. The level probes are housed in the two thimbles which penetrate the shield. Each thimble will house a two-point level probe. Thermo-

couples on the outside of the barrel, bowl and upper cooling systems are not considered part of the pump. Thermocouples mounted on the pump coupling, motor, seal and bearing are part of the pump.

3.0 PRINCIPLES OF OPERATION

The reactor sodium pumps are components of the main heat transfer system. Operation of the system is covered in SDD-096-30-000.

4.0 SAFETY PRECAUTIONS

4.1 Hazards

Hazards associated with the reactor sodium pumps are:

- a) Freezing or plugging of the overflow line.
- b) Plugging of the vent line.
- c) Seal failure which permits cover gas outleakage or possible air inleakage.
- d) Seal oil system failure that leaks oil into the sodium.
- e) Loss of bearing and seal coolant that will lead to failure of these items.
- f) Frozen pump shaft.
- g) Loss of pump power.
- h) Thermal gradients in the pump barrel that could cause shaft and barrel misalignment.

4.2 Precautions

Precautions which should be taken to minimize the occurrence or consequences of the above hazards are:

- a) Overflow and vent lines are provided with heaters that are actuated on low temperature. Periodic checks of these temperatures should be made.
- b) Double mechanical seals are used. The seals should be replaced each year. They should be periodically checked for leakage of cover gas.
- c) A seal oil reservoir large enough to hold the entire system oil volume is provided in the pump. The reservoir should be periodically checked for the presence of oil.

d) Sodium level within the pump casing should be monitored and not allowed to exceed the upper operating limit.

e) Alternate power sources are available to the pump motors. In case of loss of electrical power, these alternate sources should be placed in service.

f) The pump barrels are provided with electric heaters and thermal insulation with appropriate temperature monitors and controls. Temperature distribution on the pump casing should be maintained as constant as possible to avoid case distortion.

5.0 MAINTENANCE

5.1 Normal/Routine Maintenance

Maintenance on the pumps is concerned primarily with lubrication and seals. Seals should be changed yearly. Pump seal change requires that the pump be shut down and requires 4 to 8 hours to complete.

The seals and the thrust bearings require oil for lubrication and cooling. This oil should be changed once each year. The motor drives use grease that requires replacement each six months.

5.2 Unusual Maintenance

The pumps are designed to be removable from their casings if required for repair of those portions of the pump below floor level. This can be accomplished as follows:

- a) Remove drive motor and couplings.
- b) Drain associated reactor coolant loop and allow it to cool to ambient temperature.
- c) Isolate pump cover gas from reactor cover gas.
- d) Purge pump cover gas with clean Argon.
- e) Allow sufficient time for decay of radioactive sodium-24 (nominally 10 days).
- f) Unbolt pump flange.
- g) Place plastic bag over pump and seal to pump casing flange.
- h) Establish Argon atmosphere in plastic bag.
- i) Lift pump into plastic bag.
- j) Tie plastic bag off below pump and seal pump casing.
- k) Perform necessary pump repairs and inspections and replace.

FREE-SURFACE SODIUM PUMP

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SUPPORTING DOCUMENT

NUMBER

SDD-096-31-220

REV
LTR B

DOCUMENT TYPE

System Design Description

PROGRAM TITLE

ANL 1000 Mwe Follow-on Study

KEY NOUNS

Valves (Reactor Sodium)

DOCUMENT TITLE

Valves (Reactor Sodium) Conceptual
System Design Description - 1000 Mwe LMFBF
Follow-on Study

ORIGINAL

ISSUE DATE July 1, 1968

GO NO.

7692

S/A NO.

13140

PAGE 1 OF

TOTAL PAGES 12

REL. DATE

ABSTRACT

The requirements and brief description for the
reactor sodium valves for the 1000 Mwe LMFBF
Follow-on Study are presented

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VALVES (REACTOR SODIUM) CONCEPTUAL SYSTEM DESIGN DESCRIPTION**1.0 INTRODUCTION**

The Reactor Coolant System is a high temperature liquid metal heat transfer system consisting of heat transfer and fluid transport equipment. This equipment is connected together with piping that contains valves for control of the contained fluid. This SDD is concerned only with the sodium valves installed in the loops of the Reactor Coolant System. Requirements for other sodium valves in the plant are found in the Intermediate Sodium Valves SDD and in the Auxiliary Systems Segment SDD. The design of the reactor coolant sodium valves must accommodate various systems and components of other plant segments, whose detailed requirements appear in other SDD's. The major items which must be accommodated, but are not included in this SDD, are as follows:

- a. Reactor sodium piping
- b. Thermal insulation
- c. Pipe supports
- d. Preheat system

1.1 Functions/Functional Requirements

The functions of the valves in the Reactor Coolant System are as follows:

- 1.1.1 To provide for isolation of a loop from the rest of the system.
- 1.1.2 To provide control of flow rate.
- 1.1.3 To provide for prevention of flow reversal in the loops.

1.2 Summary Description

The valves required to perform the above functions consist of three types: Gate valves for loop isolation; butterfly valves for flow control; and check valves for flow reversal prevention. Valve stems use the sodium freeze seal design to control stem leakage. The check valves are of all welded construction of the swing disc type. Gate valves and butterfly valves are power actuated.

1.3 Requirements

1.3.1 Performance Requirements

1.3.1.1 The valves shall permit the circulation of coolant in the reactor coolant loops as required for the generation of the net full rated power of 1000 MWe by the plant.

1.3.1.2 The gate valves shall provide for the isolation of the reactor coolant loops with sufficient tightness to limit leakage to a level low enough to allow radiolytic decay, during continued reactor operation, of the isolated loop contents and to permit maintenance on the loop upon reactor shutdown.

1.3.1.3 The butterfly valves shall provide for the control of coolant flow to match temperature and power as required for the various modes of reactor system operation for decay heat removal.

1.3.1.4 The check valves shall provide for the prevention of flow reversal in a loop upon the loss of pumping power in the loop.

1.3.2 Environment Requirements

The reactor coolant system in which the valves are installed is required to operate with high temperature radioactive sodium coolant. To minimize the possibility of a sodium fire and to provide containment in the event of a system leak, the reactor heat transfer system is installed in shielded vaults with nitrogen atmospheres.

1.3.2.1 Steady state conditions acting upon the valves are as given in SDD-096-30-000, paragraph 1.3.2.1.

1.3.2.2 Transient conditions affecting the valves are as given in SDD-096-30-000, paragraph 1.3.2.2.

1.3.2.3 The chemical environment to which the valves are subject are given in SDD-096-30-000, paragraph 1.3.2.3.

1.3.2.4 The radiation environment acting upon the valves will be as stated in SDD-096-30-000, paragraph 1.3.2.

1.3.3 Operational Requirements

It is required that the reactor coolant system, of which the valves are a part, permit operation of the plant in a load following mode between 25% of rated power and maximum steam flow conditions. In addition, the valves must provide flow control at low flow rates when the reactor system is operated in a decay heat removal mode. All valve operational adjustments must be by remote means.

1.3.4 Safety Requirements

Valve design shall give due consideration to factors pertaining to safety. Some of these factors are as follows:

- 1.3.4.1 Design basis accident overpressure of 200 psi.
- 1.3.4.2 Naturally occurring phenomena such as earthquakes.
- 1.3.4.3 Prevention of sodium leakage from the valves.
- 1.3.4.4 Inspection and testing during service lifetime.

1.3.5 Design Requirements

1.3.5.1 Applicable Documents

ASME Boiler & Pressure Vessel Code, Section III - Class "A"
Nuclear Vessels
USA Standards: Code for Pressure Piping, USAS B31.7 - Nuclear
Power Piping

USA Standards: Steel Pipe Flanges and Flanged Fittings, USA B16.5

- 1.3.5.2 The valves shall be designed for a 30 year plant life.
- 1.3.5.3 The structural and pressure resistive portions of the valves shall be designed in accordance with the requirements for a Class A vessel in the ASME Boiler and Pressure Vessel Code, Section III - Nuclear Vessels, together with related code case interpretations for design temperatures in excess of 800°F.
- 1.3.5.4 The design of the valves shall conform to the requirements for Class I piping of the USA Standard "Code for Pressure Piping - USAS B31.7 - Nuclear Power Piping" with regards to flexibility of the reactor coolant piping system.

1.3.5.5 The dimensional design of any flanges or welding ends shall conform to the applicable requirements of USA B16.5.

1.3.5.6 Design Loadings:

a) Operating Loads:

(1) Steady state normal operation

Internal pressures:

Hot Leg (Reactor discharge to IHX): 200 psig @ 1160°F

Cold Leg (IHX to Reactor Inlet): 200 psig @ 800°F

(2) Transient Loads:

Design pressure of 200 psig

Refer to para. 1.3.2.2 of SDD-096-30-000 for temperature transients

(3) External Pressure Loading: (External pressure may act either independent of or simultaneous with internal pressure)

Hot Leg: 20 psig @ 1160°F

Cold Leg: 20 psig @ 800°F

(4) Live Loads:

Weight of contained fluid. For design purposes assume contained fluid is water at 70°F

Reactions from pipe supports and attached equipment such as piping, flowmeters and interconnecting auxiliary piping

(5) Dead Loads:

Weight of valve, thermal insulation and electrical pre-heaters.

b) Seismic Loads:

The reactor sodium valves are Class IA components as defined in SDD-096-10-000.

1.3.5.7 Determination of Forces and Stresses

Forces and stresses resulting from operating and seismic loads shall be determined utilizing techniques and analysis methods approved by the ASME Boiler and Pressure Vessel Code, Section III.

1.3.5.8 Allowable Stresses

Combined stresses resulting from the loads previously defined, including Design Earthquake loads, shall not exceed the allowable stresses for the material used as permitted by ASME Section III. The valves shall also be capable of withstanding the loads and forces resulting from the Maximum Hypothetical Earthquake when combined with all other simultaneously occurring loads and forces, without gross failure or loss of capability to achieve a safe shutdown of the plant.

1.3.5.9 The valves shall be fabricated from type 304 stainless steel. Trim wearing parts shall utilize suitable materials such as Stellite No. 6. Thickness of all 304 stainless steel components which may come in contact with sodium shall include an additional 0.030 inches of material as a design margin against any possible corrosion.

1.3.5.10 The additional data shown in Table 1 shall also apply to the design of the valves.

ATOMICS INTERNATIONAL

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NO. SDD-096-31-220DATE 7-1-68PAGE 8 OF 12**TABLE 1 Additional Valve Design Data**

Item	Hot Leg Blocking Valve	Cold Leg Blocking Valve	Main Throttling Valve	Bypass Throttling Valve	Check Valve
<u>Type</u>	<u>Split Wedge Gate</u>	<u>Split Wedge Gate</u>	<u>Butterfly</u>	<u>Butterfly</u>	<u>Swing Disc</u>
No. Req'd - Per Loop	1	1	1	1	1
Location	Reactor Outlet	Reactor Inlet	Reactor Inlet	Reactor Inlet Bypass	Reactor Inlet between blocking valves and throttling valve
Operating Temperature	1140°F	780°F	780°F	780°F	780°F
Nominal Operat- ing Pressure	1 psig	120 psig	120 psig	120 psig	120 psig
Flow Rate	25.2x 10 ⁶ lb/hr	25.2x 10 ⁶ lb/hr	25.2x 10 ⁶ lb/hr	2x10 ⁶ lb/hr	25.2x 10 ⁶ lb/hr
Valve Size	44 in.	38 in.	38 in.	6 in.	38 in.
Internal (Seat) Leakage	2x10 ⁻³ ft ³ / hr max. @ 50 psi ΔP**	2x10 ⁻³ ft ³ / hr max. @ 50 psi ΔP**	N.A.	N.A.	N.A.
Pressure Drop at Rated Flow	0.5 psi max.	0.5 psi max.	5 psi max.	5 psi max.	1 psi max.
Flow Control Range	N.A.	N.A.	0% to 25% of rated flow*	10-100%	N.A.
Actuator	Pneumatic or Motor	Pneumatic or Motor	Pneumatic or Motor	Pneumatic or Motor	N.A.
Closure Speed	1 to 2 in./sec	1 to 2 in./sec	15%/sec	15%/sec	N.A.
Stem Seal	Gas Cooled Freeze Seal	Gas Cooled Freeze Seal	Gas Cooled Freeze Seal	Gas Cooled Freeze Seal	N.A.

*Above 25% of rated loop flow, the variable speed pump drives control loop flow rates.

**Zero leakage across valve.

1.3.5.11 The blocking valves and throttling valves shall be power actuated by either pneumatic cylinders or electric motors. Actuating mechanisms and equipment shall be remotely located in areas accessible for maintenance during reactor operation. Linkage between the remotely located actuating mechanism and the valve bodies which penetrate the inner containment boundary as defined by SDD-096-90-000 shall be provided with containment seals. The containment seals shall be capable of accommodating the thermal movement of the valve body and shall satisfy the requirements of paragraph 1.3.7.5.10 of SDD-096-90-000.

1.3.5.12 The butterfly throttle valve power actuators shall be capable of positioning the valves proportional to an instrument signal.

1.3.5.13 The check valves are to be flow actuated and shall be designed to prevent the formation of, or the accumulation of, sodium oxides at the disc hinges. Pressure drop across the valve during periods of low flow without pumping power available shall be such that the natural convection capability specified in SDD-096-30-000 is satisfied.

1.3.5.14 The gate valve bodies shall be provided with a 2 in. pipe connection to drain the volume between the split wedges.

1.3.5.15 The 38-inch butterfly throttle valves shall be provided with physical stops at the 10° open position such that the valve may be opened further but not closed beyond the stop.

1.3.5.16 The butterfly throttling valves shall accommodate full rated flow at the 60° open position with a maximum pressure drop of 5 psig.

1.3.5.17 The check valve shall be of all welded construction.

1.3.5.18 The 44-inch valve body ends shall be prepared to match 44-inch O.D. 1-1/16 inch wall pipe with a standard bevel of $37-1/2^\circ \pm 2-1/2^\circ$ and 1/16 inch, $\pm 1/32$ inch land. The 38-inch valve body ends shall be prepared similarly except that the pipe size shall be 38-inch O.D. with 3/8-inch wall. The 6-inch butterfly body ends shall be prepared similarly for 6-inch schedule 20 pipe.

2.0 DETAILED DESCRIPTION

The primary reactor coolant loops each contain a 44-inch gate valve, a 38-inch gate valve, a 38-inch butterfly valve, and a 38-inch swing check valve. In addition, each of three 8 inch coolant supply lines, one line from the cold leg of each primary loop, for the reactor vessel low pressure plenum contains a 6 inch butterfly valve.

The gate valves utilize a "flexible wedge" disc. The shutoff capability of the valve is enhanced by this type disc which compensates for valve seat distortion. The seating surfaces of the valve disc contact the valve body seat and accomplish nominal closure of the valve. Continued downward movement with increased stem thrust results in deflection of the valve disc such that the disc seats conform with the valve body seats for tight shutoff. The flexible wedge disc configuration also tends to prevent seizing of the valve disc in its seat.

The gate valves have forced convection cooled freeze seal stems with spring loaded packing glands at their upper end for backup. The valve body is designed for through drainage of sodium, and a 2 inch drain connection is provided to drain the volume between the discs when they are in the closed position. All wear surfaces such as on the discs, seats, and stem guides are Stellite surfaced. The valves are provided with remote pneumatic or motor driven actuators.

Since tight shutoff of the main 42 inch butterfly throttling valves is to be prevented, they are of conventional design and a physical stop is provided to limit closure to no less than the 10° open position. The 6 inch valves are not so equipped. The butterfly valves are of the through-shaft design with the outboard bearing internal to the valve body. Bearing wear areas are Stellite surfaced. Valve stems are provided with forced convection cooled freeze seals backed up by a spring loaded packing gland. Pneumatic or motor driven positioning remote actuators are provided for valve operation.

The check valves are of the swing check design. The valve disc swings through an angle which virtually clears the full pipe area. In the full open position, a small portion of the disc is impinged by the flowing stream in order to stabilize the disc. A negative seat angle (1°) is provided to facilitate drainage of the system in which it is installed. The valve is installed in a horizontal pipe run with the valve disc, at rest, hanging vertically. A knife edge hinge structure permits easy disc movement even with adverse conditions such as oxide contamination or misalignment and during periods of low flow such as under convective cooling. Wear areas are surfaced with Stellite. The valve is of all welded construction.

3.0 PRINCIPLES OF OPERATION

The reactor sodium valves are components of the main heat transfer system, operation of which is covered in SDD-096-30-000.

The gate blocking valves are normally kept in the full open position. They are closed only when it is desired to isolate a heat transfer loop from the reactor.

The butterfly throttling valves are normally full open. Upon reactor trip, they partially close to control free convection flow in the heat transfer system.

Check valves are normally open, closing in a single heat transfer loop when flow in that loop drops to a low value compared with the other two loops. (For example, if power is lost to one primary pump.)

4.0 SAFETY PRECAUTIONS

4.1 Hazards

Possible hazards which could be generated by the valves are:

- a) Inadvertent closing of one or more gate or butterfly valves.
- b) Check valve sticking closed.
- c) Leakage across gate valve seat when it is closed.
- d) Stem freeze seal leakage (gate and butterfly valves).

4.2 Precautions

The following precautions should be taken to minimize the above hazards.

- a) Electrical interlocks and administrative controls should be utilized to prevent closure of the gate valves except when desired for single loop isolation.

b) Sodium purity (particularly in regard to sodium oxide) should be maintained and monitored to minimize collections of oxide which might cause check valve sticking or stem freeze seal binding the valve stem.

c) When the gate valves are closed, the space between the gates should be drained and monitored for presence of sodium.

d) Reliable (preferably redundant) cooling system for the valve stem freeze seals should be provided. The backup packing for the stem freeze seal should be of a material which will resist deterioration by high temperature sodium.

5.0 MAINTENANCE

The valves in the primary loops are accessible only upon reactor shutdown and after a suitable decay period. In-place maintenance will be limited to some stem seal parts and actuator linkages only during periods of reactor shutdown. Any major repairs or replacement of components can be accomplished by removal of the valve from the loop. Valve actuating mechanisms will be accessible for maintenance at all times.

6.0 APPENDICES

6.1 References

SDD-096-10-000, "Overall System SDD"

SDD-096-30-000, "Heat Transfer System SDD"

SDD-096-31-000, "Reactor Coolant SDD"

SDD-096-90-000, "Land, Structures and Conventional System Segment SDD"

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INTERMEDIATE HEAT TRANSFER SYSTEM CSDD

1.0 INTRODUCTION

The intermediate heat transfer system is a high temperature liquid metal heat transfer system. Included in this system are the piping and equipment between the IHXs and the steam generator, the secondary (tube) side of the IHXs and the shell side of the steam generator.

1.1 Functions/Functional Requirements

The purposes of the intermediate heat transfer system are as follows:

- 1.1.1 To remove heat from the IHXs and transfer it to the steam generators.
- 1.1.2 To transfer heat at the temperatures and rates required by plant power generation demands.
- 1.1.3 To provide cooling to the reactor coolant system under abnormal as well as normal conditions of operation.
- 1.1.4 To provide for safe operation of a high temperature liquid metal system.

1.2 Summary Description

The intermediate heat transfer system consists of three independent parallel loops of equal capacity. Forced circulation of the sodium coolant is provided by a single free surface pump in each of the loops. Each of the pumps is installed in the surge tank for its loop. The tank/pump is located at the system high point. Flow from the pump is to the steam generator, from which flow is to the tube side of the IHXs, and finally back to the pump inlet. Valves are provided in each loop for flow control and steam generator isolation.

1.3 Requirements1.3.1 Performance

- 1.3.1.1 The intermediate heat transfer system shall transfer heat from the IHXs to the steam generators at a rate and temperature to satisfy the generation of the net full power of 1000 Mwe by the plant.

1.3.1.2 The intermediate heat transfer system shall respond automatically to changes in power demands between 25% rated load and maximum steam flow conditions. The load following capability shall be at least 25%/minute for emergency demands and 2%/minute for normal demands.

1.3.2 Environmental

The intermediate heat transfer system does not contain radioactive sodium, and therefore, does not require shielding. A sodium fire in this system, consequently, is not attended with the same hazards as a fire in the reactor coolant system. Most of the intermediate heat transfer system can be an outdoor installation as a result. The system, however, will be designed to withstand the rigors of all weather conditions that may occur at the installation site.

1.3.2.1 Steady state conditions for the intermediate heat transfer system are as given in SDD-096-30-000, paragraph 1.3.2.1.

1.3.2.2 Transient conditions for the intermediate heat transfer system are as given in SDD-096-30-000, paragraph 1.3.2.2.

1.3.2.3 The chemical environment that the intermediate heat transfer system is subject to is as given in SDD-096-30-000, paragraph 1.3.2.3.

1.3.3 Operational

It is a requirement of the intermediate heat transfer system to permit operation of the plant in a load following mode between 25% of rated power and maximum steam flow conditions. All operational adjustments to the system shall be through manual remote or automatic means, as appropriate.

1.3.4 Safety

The design of the intermediate heat transfer system shall give due consideration to factors pertaining to safety. Some of these factors are as follows:

1.3.4.1 Seismic loads of Class 1A earthquake as defined in SDD-096-10-000.

1.3.4.2 Provisions for handling a sodium/water reaction.

1.3.4.3 Assure leakage from least radioactive to radioactive system if leak occurs.

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

1.3.5 Design

1.3.5.1 The design temperature and pressure for the intermediate heat transfer system (exclusive of IHX's) shall be 1090°F hot leg/700°F cold leg and 200 psig, respectively.

1.3.5.2 The intermediate heat transfer system shall consist of three equal capacity heat removal loops operating in parallel.

1.3.5.3 Each intermediate heat transfer loop shall be designed with natural convection cooling capability to meet the requirements of SDD-096-30-000 paragraph 1.3.7.3.

1.3.5.4 The intermediate heat transfer system loops exterior to the reactor building shall be capable of being drained completely by gravity.

1.3.5.5 It shall be possible to add sodium to the intermediate heat transfer system loops to adjust minor level deficiencies without the need of depressurizing the loops.

1.3.5.6 Type 304 stainless steel shall be the general material of construction of the system. Other materials may be used where compatibility, adequacy, and economics permit.

1.3.5.7 Applicable documents shall be as given in SDD-096-30-000, paragraph 1.3.7.1.

1.3.5.8 The pressure of the intermediate transfer loops during operation shall be no less than 5 psi higher in the IHX than the pressure of the reactor coolant system. This requirement is designed to assure that any leakage will be from the non-radioactive to the radioactive system in the event of tube rupture in the IHX.

1.3.5.9 The intermediate heat transfer system shall be designed for a 30 year plant life.

2.0 DETAILED DESCRIPTION

The Intermediate Heat Transfer System includes the three secondary loops of the Main Heat Transfer System. Each of these loops is of equal capacity. The loops transfer heat from the IHXs to the steam generators. Each loop includes the tube side of an IHX, a surge tank with a free surface pump installed within it, the shell side of the steam generator, two butterfly valves, and a gate valve. An EM flowmeter is installed in the cold leg of each loop. Other necessary operational instrumentation is provided including hydrogen detection instrumentation.

The outlet piping from the IHX to the pump is 36 inches. The discharge line from the pump to the superheater/reheater is a 36-inch pipe that splits into a 28-inch pipe to the superheater and a 26-inch pipe to the reheater. A butterfly valve for flow control is located in the 36-inch pipe downstream of the pump. The second butterfly is located in the 26-inch pipe on the outlet side of the reheater. This valve is used to proportion the flow between the reheater and the superheater. Between the superheater/reheater and the evaporator, the piping is 36 inches. A 34-inch line connects the evaporator outlets with the IHX. This line is provided with a gate valve that in conjunction with the butterfly valve in the 36-inch line provides steam generator isolation in the event of a steam/water leak.

Drain and vent connections are provided appropriately to allow adequate filling/drainage of the loops. The secondary loops are located outside of, and adjacent to, the Reactor Building, except for those portions most intimately associated with the IHXs. These portions are located within the primary equipment vaults.

2.1 Component Description

2.1.1 Pumps

Each secondary loop contains one full capacity free surface centrifugal pump. Each pump can deliver 55,600 gpm against a 90 psi head with an available NPSH of 115 psi. The pump drives are a combination 4000 HP, 900 rpm wound rotor motor and a 1300 HP DC motor, both directly coupled to the pump.

The secondary pumps are installed in the secondary loop surge tanks. This arrangement obviates the balance pipe requirement.

2.1.2 Secondary Surge Tank

Each secondary loop contains a surge tank. The surge tanks are 304 SS vertical tanks, 14 feet OD by 12 feet overall height. The heads are semi-elliptical. The tanks are constructed such that the secondary loop pump barrel and bowl are a part of it with appropriate provisions made for differential thermal expansion between the tank and pump assembly. The tank capacity is adequate for sodium volume changes occurring during power operation.

2.1.3 Valves

The intermediate heat transfer system loops each contain three valves. At the inlet side of the superheaters/reheaters there is a 36-inch butterfly control valve. In the outlet manifold from the reheaters is a 26-inch butterfly valve. At the outlet of the evaporators, there is a 34-inch gate block valve.

The butterfly valves utilize stem freeze seals and have remote pneumatic actuators. The gate valves have double wedges, use stem freeze seal, and remote pneumatic actuators.

2.1.4 Piping

All piping in the intermediate heat transfer system up to the material transition pieces at the steam generator is Type 304 stainless steel with all welded construction. Welds are made using a heliarc root pass with an inert gas backing and a shielded arc filler rod. Adequate expansion loops are designed into the piping to minimize stresses from thermal expansion.

2.1.5 IXX

The intermediate heat transfer system coolant flows through the tube side of the IXXs. Each loop contains one IXX. Each IXX is a vertical shell and tube hockey stick type design. The units are of all welded 304 SS construction.

2.1.6 Steam Generator

The steam generator is of the modular concept. This concept utilizes sufficient small units connected in parallel to provide the required heat transfer area. Each intermediate heat transfer loop contains one steam generator bank. Each bank is made up of evaporator modules in series with superheater modules. Reheater modules are also included in each bank. Steam/water flows in the module tubes. Sodium flows in the shells. The modules and connecting piping are constructed of 2-1/4 chrome-1 moly material 321 or 304 SS.

3.0 PRINCIPLES OF OPERATION

3.1 Pre-operational Preparation

The intermediate heat transfer system loops must be purged with helium before sodium coolant is admitted. The loops are purged until the oxygen content is reduced to 1% or less by volume. The loops then must be heated to a temperature of about 350°F. This is necessary to prevent plugging and thermal shocks when adding sodium. Heat is supplied by resistance heaters attached to the piping and equipment. The steam generator is heated by means of hot water and/or steam to the 350°F preheat temperature.

When the preheat temperature is attained, sodium is pumped into the loops until the level in the surge tanks is at the 350°F equilibrium temperature level. The surge tank cover gas pressure is adjusted, and circulation of the sodium by the pumps is begun.

3.2 Operation

The intermediate heat transfer system is brought to the 25% plant power level in a semi-automatic manner. Above this power level, the system is placed on automatic control to rise to operating levels. During this rise to power phase of operation, temperatures are increased at a rate of about 100°F per hour.

Normal operation of the intermediate heat transfer system is fully automatic. The system has a load following capability of 2% per minute for normal demands. Normally, the intermediate heat transfer system throttle valves are wide open and flow control is through pump speed control. The throttle valves are used to match sodium flow with decay heat removal requirements after reactor shutdown.

Routine shutdown involves gradually reducing load to about 15% power at which point flow control is assumed by the throttle valves. Flow is reduced as required to maintain desired temperatures. The pumps are used in conjunction with the throttling valves until flow is reduced to about 4% of normal, at which time the pumps are shut down and natural convection takes over.

4.0 SAFETY PRECAUTIONS

4.1 Hazards

The hazards associated with the intermediate heat transfer system involves loss of its function of heat transfer capability. Since each loop of the system is independent of the other two, the loss of heat transfer capability in one loop represents a loss in plant capacity with no serious associated hazard. The loss of two loops would lead to plant shutdown to insure adequate decay heat removal capability from the remaining loop.

Loop failure could involve loss of pumping ability through loss of power or pump damage; loss of ability to circulate coolant due to closed valves; loss of coolant through pipe or equipment rupture; steam generator leaks.

4.2 Precautions

The intermediate heat transfer system is designed to minimize the hazards mentioned above. Alternate power supplies are available to the system. System elevations and arrangement are designed to enhance natural circulation. Valves and valve operation are readily accessible for maintenance and repair. Adequate instrumentation is provided to permit monitoring the condition of the system.

The safe operation of the intermediate heat transfer system involves its operation within its design limits and attention to a proper maintenance program.

5.0 MAINTENANCE

Maintenance on the intermediate heat transfer system involves pump drive lubrication, pump bearing and seal check, checking of valves for leakage and proper operation, checking system for leaks, electrical systems testing, and instrumentation checkout. Maintenance is done on a routine basis, being

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scheduled during refueling or steam turbine maintenance periods. Routine checks may be made during plant operation, and an intermediate heat transfer loop may be shut down for maintenance while plant operates on remaining loops.

6.0 APPENDICES

6.1 Drawing

Figure 1, "Intermediate Heat Transfer System Schematic."

FIGURE 1
INTERMEDIATE HEAT TRANSFER
SCHEMATIC

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SUPPORTING DOCUMENT

NUMBER

SDD-096-32-100

REV
LTR

B

DOCUMENT TYPE

System Design Description

PROGRAM TITLE

ANL 1000 Mwe Follow-on Study

KEY NOUNS

SDD- Intermediate Na
Pumps and Drives

DOCUMENT TITLE

Pumps and Drives (Intermediate Na)
Conceptual System Design Description - 1000 Mwe
LMFBR Follow-on Study.

ORIGINAL

ISSUE DATE 8/5/68

GO NO.

7692

S/A NO.

13140

PAGE 1 OF

TOTAL PAGES 10

REL. DATE

ABSTRACT

The requirements and description for the intermediate sodium pumps and drives for the 1000 Mwe LMFBR Follow-on Study are presented.

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INTERMEDIATE SODIUM PUMPS AND DRIVES SDD

1. INTRODUCTION

The intermediate sodium pumps and drives are installed in the intermediate coolant system. This SDD describes these pumps and drives.

1.1 Functions/Functional Requirements

The function of the intermediate sodium pumps and drives is to circulate sodium coolant for the transfer of heat from the IHX to the steam generator.

1.2 Summary Description

The intermediate sodium pumps and drives consist of vertical free surface centrifugal pumps and variable speed electric motor drives. There is a single pump in each of the three intermediate coolant loops. The intermediate sodium pumps are installed in the intermediate coolant loop surge tanks which are located at the high point of the system.

1.3 Requirements

1.3.1 Performance

1.3.1.1 The intermediate sodium pumps shall circulate sodium coolant as required for the generation of the net full power of 1000 MWe, which is the plant rated power condition.

1.3.1.2 The intermediate sodium pumps shall have the capability to provide variable flow rates to meet power demands between 25% of plant rated power and maximum steam flow conditions.

1.3.2 Environmental

1.3.2.1 The intermediate coolant pumps shall circulate hot sodium.

1.3.2.2 The intermediate coolant pumps will be subjected to the steady state conditions as listed in paragraph 1.3.2.1 of SDD 096-30-000.

1.3.2.3 The intermediate sodium pumps will be subjected to the transient conditions given in paragraph 1.3.2.2 of SDD-096-30-000.

1.3.2.4 The intermediate sodium pumps will be exposed externally to air ranging in temperature from 0F. to 120F.

1.3.2.5 The intermediate sodium pumps will use helium or Argon cover gas over the sodium free surface in the pumps.

1.3.2.6 The intermediate sodium pumps will circulate sodium having impurities as listed in paragraph 1.3.2.3 of SDD-096-30-000.

1.3.3 Operational

1.3.3.1 The pumps shall be capable of satisfactory operation over a speed range of at least 10 to 1.

1.3.3.2 Each pump shall be provided with level instrumentation which will measure and indicate level in the pump barrels. Full instrumentation for the seal oil systems is required, including level, pressure, temperature, and flow.

1.3.4 Safety

1.3.4.1 The shaft seal shall be made at the upper end of the pumps and be removable without removal of the shaft, or inner rotating assembly.

1.3.4.2 The seal, thrust bearing, and lubricant system shall be arranged so that a failure of the seal will not permit lubricant to enter the sodium in the pump.

1.3.4.3 The seal coolant and lubricating system shall be an integral part of the pump assembly.

1.3.4.4 The pumps shall be designed for seismic loads for Class 1A components as defined in SDD-096-90-000.

1.3.5 Design**1.3.5.1 Applicable Documents**

ASME Boiler and Pressure Vessel Code, Section III, Class "C"
 Nuclear Vessels

ASME Boiler and Pressure Vessel Code, Section VIII, Unfired
 Pressure Vessels

ASAS American Standard Code for Pressure Piping, Nuclear
 Power Piping, B31.7

NEMA Standards

ASA National Electrical Code

1.3.5.2 The intermediate sodium pumps shall be designed for the full rated capacity of 55,200 gpm and a TDH of 256 feet. The maximum efficiency shall correspond to this operating condition.

1.3.5.3 The pumps shall be designed to meet the following conditions:

Fluid Pumped	Sodium
Design Press., psig	200
Design Temp. F	1090
Fluid Operating Temp. F	1070
Fluid Density @ 1070F, lb/ft ³	50.6
NPSH Available, ft.	350

1.3.5.4 Each pump shall be designed for a service life of 30 years.

1.3.5.5 Each pump shall be of the vertical free surface type.

1.3.5.6 Each pump shall be driven by a combination wound rotor motor and a DC motor, both directly coupled to the pump. This combination shall provide the required speed range of 10 to 1.

1.3.5.7 The motor, pump drive, couplings and pump rotating parts shall be statically and dynamically balanced such that there are no critical speeds below 115% of rated speed.

1.3.5.8 The pumps shall be equipped with pressurized hydrostatic radial bearings, cooled and lubricated with the flowing sodium stream. The bearings shall be sized for continuous operation at all pump speeds from 10% to 110% of rated speed, and for short periods (less than 30 minutes) at speeds from zero to 10% of rated speed.

1.3.5.9 Pressure containing parts of the pumps shall be designed in accordance with the applicable portions of the ASME Boiler and Pressure Vessel Code, Section III, Class "C" Nuclear Vessels. A code stamp on each pump is required.

1.3.5.10 Type 304 stainless steel shall be used in the fabrication of the pump barrels, casings, shafts, impellers, and other components exposed to the sodium or inert gas environment. This requirement does not apply to the choice of hydrostatic bearing materials which shall be stellite, or equal.

1.3.5.11 Halogenated hydrocarbons shall not be used as lubricants for the pump seal and thrust bearing. Lubricants used shall be as compatible as possible with sodium and its vapors to avoid explosions and system damage through contamination if the lubricants should get into the sodium coolant stream.

1.3.5.12 The intermediate sodium pumps shall be installed within the intermediate coolant loop surge tank.

2.0 DETAILED DESCRIPTION

The secondary pump is a free surface, vertical centrifugal pump, capable of circulating 55,600 GPM of sodium at 1070°F, which is rated design flow.

Each pump consists of an expansion tank, barrel and bowl, impeller and shaft, diffuser, inner support structure, variable speed electric drive, control instrumentation and cooling system.

The pump is housed inside the expansion tank with the suction to the pump at the base of the bowl and expansion tank. The suction pipe, is welded to the base of the expansion tank and the pump suction nozzle forms a slip-fit to the expansion tank inlet. The pump bowl is located in the bottom of the tank and is welded to the barrel at the top and to the discharge pipe at the side. The discharge pipe between the expansion tank wall and pump bowl is provided with an expansion joint to allow movement between the bowl and tank. The barrel extends up through the top of the tank and ends in the top mounting and seal flange. The tank also contains two vertically mounted instrument thimbles for level probes. The barrel mounting flange is supported by the pump drive service floor.

The internal support structure forms the housing seal and provides for mounting the shaft seal. The structure contains thermal insulation and heat barrier plates, hydrostatic sodium bearing mount, the anti-whirl baffle plates, instrument thimbles and diffuser hold-down hardware.

The impeller is 30 inches in diameter and is of the single suction radial type. The shaft is connected to the impeller and electric motor drive. The shaft and impeller are supported by the bearing.

The diffuser is supported by the bowl and is located at the circumference of the impeller. It should be noted that the number of vanes in the diffuser should be different (preferably odd) to the number of vanes in the impeller to reduce the pump noise generation problem.

The pump is driven by a combination 4000 HP 900 rpm wound rotor motor and a 1300 HP DC motor, both directly coupled to the pump.

The thrust bearing and seals are located above the thermal shield, and both are cooled by the kerosene cooling system.

Instruments used to measure pump operation will be mounted as part of the pump and are thermocouples, level probes, and shaft tachometer. The thermocouples are chromel-constantan and the level probes are two-point alarm with a three-coil element at each level alarm point. The level probes are housed in thimbles, two in the expansion tank and two in the pump internal support structure. Each thimble will house a two-point level probe. Thermocouples on the outside of the expansion tank and upper cooling systems are not considered part of the pump. Thermocouples mounted on the pump coupling, motor, seal and bearing are part of the pump.

3.0 PRINCIPLES OF OPERATION

The intermediate sodium pumps are components of the main heat transfer system. Operation of the system is covered in SDD-096-30-000.

4.0 SAFETY PRECAUTIONS

4.1 Hazards

Hazards associated with the intermediate sodium pumps are:

- a) Seal failure which permits cover gas outleakage or possible air inleakage.
- b) Seal oil system failure that leaks oil into the sodium.
- c) Loss of bearing and seal coolant that will lead to failure of these items.
- d) Frozen pump shaft
- e) Loss of pump power
- f) Thermal gradients in the pump barrel that could cause shaft and barrel misalignment.

4.2 Precautions

Precautions which should be taken to minimize the occurrence or consequences of the above hazards are:

- a) Double mechanical seals are used. The seals should be replaced each year. They should be periodically checked for leakage of cover gas.
- b) A seal oil reservoir large enough to hold the entire system oil volume is provided in the pump. The reservoir should be periodically checked for the presence of oil.

- c) Sodium level within the combination pump casing and expansion tank should be monitored and not allowed to exceed the upper operating limit.
- d) Alternate power sources are available to the pump motors. In case of loss of electrical power, these alternate sources should be placed in service.

5.0 MAINTENANCE

5.1 Normal/Routine Maintenance

Maintenance on the pumps is concerned primarily with lubrication and seals. Seals should be changed yearly. Pump seal change requires that the pump be shut down and requires 4 to 8 hours to complete.

The seals and the thrust bearings require oil for lubrication and cooling. This oil should be changed once each year. The motor drives use grease that requires replacement each six months.

5.2 Unusual Maintenance

The pumps are designed to be removable from the expansion tanks if required for repair of those portions of the pump within the tank. This can be accomplished as follows:

- a) Remove drive motor and couplings.
- b) Drain associated intermediate coolant loop and allow it to cool to ambient temperature.
- c) Purge expansion tank cover gas with Argon.
- d) Unbolt pump flange
- e) Place plastic bag over pump and seal to pump casing flange.
- f) Establish Argon atmosphere in plastic bag.
- g) Lift pump into plastic bag.
- h) Tie plastic bag off below pump and seal expansion tank opening.
- i) Perform necessary pump repairs and inspections and replace.

Representative Design of Secondary Sodium Pump

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NUMBER

SDD-096-32-220

REV

LTR B

DOCUMENT TYPE

System Design Description

KEY NOUNS

Valves (Intermediate Sodium)

ORIGINAL

ISSUE DATE 7-1-68

GO NO.

7692

S/A NO.

13140

PAGE 1 OF

11

TOTAL PAGES

REL. DATE

PROGRAM TITLE

ANL 1000 Mwe Follow-on Study

DOCUMENT TITLE

Valves (Intermediate Sodium)

Conceptual System Design Description - 1000 Mwe
LMFBR Follow-on Study

ABSTRACT

The requirements and brief description for the
intermediate sodium valves for the 1000 Mwe
LMFBR Follow-on Study are presented

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VALVES (INTERMEDIATE SODIUM) CONCEPTUAL SYSTEM DESIGN DESCRIPTION**1.0 INTRODUCTION**

The Intermediate Coolant System is a high temperature liquid metal heat transfer system consisting of fluid transport and heat transfer equipment. This equipment is connected together with piping that contains valves for control of the contained fluid. This SDD is concerned with the large sodium valves installed in the loops of the Intermediate Coolant System. Requirements for other sodium valves in the plant are found in the Reactor Sodium Valves SDD and the Auxiliary Systems Segment SDD. The design of the intermediate coolant sodium valves must accommodate various systems and components of other plant segments, whose detailed requirements appear in other SDD's. The major items which must be accommodated, but are not included in this SDD, are as follows:

- a. Intermediate Sodium Piping
- b. Thermal Insulation
- c. Pipe Supports
- d. Preheat System

1.1 Functions/Functional Requirements

The functions of the valves in the Intermediate Coolant System are as follows:

- 1.1.1 To provide for isolation of a steam generator bank from the rest of the loop.
- 1.1.2 To provide control of flow rate.

1.2 Summary Description

The valves required to perform the above functions consist of two types: Gate valves for loop isolation; and butterfly valves for flow control. Valve stems use the sodium freeze seal design to control stem leakage. Gate valves and butterfly valves are power actuated.

1.3 Requirements**1.3.1 Performance Requirements:**

1.3.1.1 The valves shall permit the circulation of coolant in the intermediate coolant loops as required for the generation of the net full rated power of 1000 MWe by the plant.

1.3.1.2 The gate valves shall provide for isolation of a steam generator bank with sufficient tightness to limit leakage of water and water/sodium reaction products into the intermediate loop in the event of a tube rupture in a steam generator bank.

1.3.1.3 The butterfly valves shall provide for the control of coolant flow to match temperature and power as required for the various modes of reactor system operation as described in SDD-096-30-000.

1.3.2 Environmental Requirements:

The intermediate sodium valves are part of the intermediate heat transfer system which is required to operate with high temperature sodium coolant. The valves are located inside the Steam Generator Building.

1.3.2.1 Steady state conditions acting upon the valves are described in paragraph 1.3.2.1 of SDD-096-30-000.

1.3.2.2 Transient conditions acting upon the valves are described in paragraph 1.3.2.2 of SDD-096-30-000.

1.3.2.3 The chemical environment acting on the valves is described in paragraph 1.3.2.3 of SDD-096-30-000.

1.3.2.4 Internal environmental conditions within the Steam Generator Building are described in paragraph 1.3.7.6.3 of SDD-096-90-000.

1.3.3 Operational Requirements:

It is required that the intermediate coolant system, of which the valves are a part, permit operation of the plant in a load following mode between 25% of rated power and maximum steam flow conditions. In addition, the valves must provide flow control at low flow rates when the reactor system is operated in a decay heat removal mode. All valve operational adjustments must be by remote means.

1.3.4 Safety Requirements:

Valve design shall give due consideration to factors pertaining to safety. Some of these are as follows:

1.3.4.1 Overpressures up to 300 psig due to the tube rupture in a steam generator module.

1.3.4.2 Naturally occurring phenomena such as earthquakes.

1.3.4.3 Prevention of leakage from the valves.

1.3.4.4 Inspection and testing during service lifetime.

1.3.5 Design Requirements:

1.3.5.1 Applicable Documents

ASME Boiler and Pressure Vessel Code, Section III Class "C" - Nuclear Vessels
USA Standard: Code for Pressure Piping, USAS B31.7 - Nuclear Power Piping

USA Standards: Steel Pipe Flanges and Flanged Fittings, USAS B16.5

1.3.5.2 The valves shall be designed for a 30 year plant life.

1.3.5.3 The structural and pressure resistive portions of the valve shall be designed in accordance with the requirements for a Class C vessel in the ASME Boiler and Pressure Vessel Code, Section III - Nuclear Vessels, together with related code case interpretations for design temperatures in excess of 800°F.

1.3.5.4 The design of the valves shall conform to the requirements for Class I piping of the USA Standard, "Code the Pressure Piping, USAS B31.7 - Nuclear Power Piping", with regards to flexibility of the intermediate coolant piping system.

1.3.5.5 The dimensional design of any flanges or welding ends shall conform to the requirements of USAS B16.5.

1.3.5.6 Design Loadings:

a) Operating Loads

(1) Steady State Normal Operation

Internal Pressures: Hot Leg (IHX discharge to steam generator evaporator module inlet header): 200 psig @ 1090°F;

Cold Leg (Steam Generator evaporator module discharge header to IHX inlet): 200 psig @ 700°F.

(2) Transient Loads:

Internal design pressure of 200 psig.

Refer to paragraph 1.3.2.2 of SDD-096-30-000 for temperature transients.

(3) Sodium-Water Reaction Loads:

Internal pressure of 300 psig at design temperatures of 1090°F for hot leg and 700°F for cold leg. Duration of internal pressure of 1/5 second.

(4) Live Loads:

Weight of contained fluid. For design purposes assume contained fluid is water at 70°F. Reactions from pipe supports and attached equipment such as piping, flow meters and interconnecting auxiliary piping.

(5) Dead Loads:

Weight of valve, thermal insulation and electrical pre-heaters.

b) Seismic Loads:

The intermediate sodium valves are Class IB components as defined in SDD-096-10-000.

1.3.5.7 Determination of Forces and Stresses

Forces and stresses resulting from operating and seismic loads shall be determined utilizing techniques and analysis methods approved by the ASME Boiler and Pressure Vessel Code, Section III.

1.3.5.8 Allowable Stresses

Combined stresses resulting from the loads previously defined, including Design Earthquake loads, shall not exceed the allowable stresses for the material used as permitted by the ASME Boiler and Pressure Vessel Code, Section III. The piping shall also be capable of withstanding the loads and forces resulting from the Maximum Hypothetical Earthquake when combined with all other simultaneously occurring loads and forces, without gross failure or loss of capability to achieve a safe shutdown of the plant.

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1.3.5.9 The valves shall be fabricated from 304 stainless steel. Trim wearing parts shall utilize suitable materials such as Stellite No. 6. Thickness of all 304 stainless steel components which may come in contact with sodium shall include an additional 0.030 inches of material as a design margin against any possible corrosion.

1.3.5.10 The additional data shown in Table 1 shall also apply to the design of the valves.

ATOMICS INTERNATIONAL*A Division of North American Rockwell Corporation*NO. SDD-096-32-220
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PAGE 8 OF 11TABLE 1 Additional Valve Design Data

Item	Blocking Valve	Loop Main Throttling Valve	Reheat Module Throttling Valve
Type	Split Wedge Gate	Butterfly	Butterfly
Number Required per Loop	1	1	1
Location	In Cold Leg between Evaporator Discharge Header & IHX	In Hot Leg between IHX and steam generator inlet headers	In Hot Leg between reheat discharge header and evaporator inlet header
Operating Temperature	670°F	1070°F	905°F
Nominal Operating Pressure	145 psig	190 psig	190 psig
Flow Rate	22.6×10^6 lb/hr	22.6×10^6 lb/hr	13.5×10^6 lb/hr
Valve Size	34 in.	36 in.	28 in.
Internal (Seat) Leakage	2×10^{-3} ft ³ /hr max. @ 50 psi AP**	N.A.	N.A.
Pressure Drop at Rated Flow	0.5 psi max.	5 psi max.	5 psi max.
Flow Control Range	N.A.	0 to 25% of rated flow*	20% - 110%
Actuator	Pneumatic or Motor	Pneumatic or Motor	Pneumatic or Motor
Closure Speed	1 to 2 in/sec	15°/sec	15°/sec
Stem Seal	Gas Cooled Freeze Seal	Gas Cooled Freeze Seal	Gas Cooled Freeze Seal

*Above 25% of rated loop flow, the variable speed pump drives control loop flow rates.

**Leakage across valve should be zero.

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1.3.5.11 The valves shall be power actuated by either pneumatic cylinders or electric motors. Actuating mechanisms and equipment shall be located in areas accessible for maintenance during reactor operation.

1.3.5.12 The butterfly throttle valves power actuators shall be capable of positioning the valves proportional to an instrument signal.

1.3.5.13 The gate valve bodies shall be provided with a 2 in. pipe connection to drain the volume between the split wedges.

1.3.5.14 The 36 in. butterfly main throttling valves shall be provided with a position alarm at the 10° open position such that the valve may be opened further but not closed beyond the point with giving an alarm.

1.3.5.15 The butterfly throttling valves shall accommodate full rated flow at the 60° open position with a maximum pressure drop of 5 psig.

1.3.5.16 The 36 in. valve body ends shall be prepared to match 36 in. O.D. with 5/8 in. wall pipe with a standard bevel of 37-1/2° - 2-1/2° and 1/16 inch $\pm$ 1/32 inch land. The 34 in. valve body ends shall be prepared similarly except that the pipe size shall be 34 in. O.D. with 3/8 in. in wall. The 28 in. butterfly body ends shall be prepared similarly for 28 in. O.D. with 3/8 in. wall pipe.

2.0 DETAILED DESCRIPTION

The intermediate coolant loops each contain a 34 inch gate valve, a 36 inch butterfly valve, and a 28 inch butterfly valve.

The gate valves utilize a "flexible wedge" disc. The shutoff capability of the valve is enhanced by this type disc which compensates for valve seat distortion. The seating surfaces of the valve disc contact the valve body seat and accomplish nominal closure of the valve. Continued downward movement with increased stem thrust results in deflection of the valve disc such that the disc seats conform with the valve body seats for tight shutoff. The flexible wedge disc configuration also tends to prevent seizing of the valve disc in its seat.

The gate valves have forced convection cooled freeze seal stems with spring loaded packing glands at their upper end for backup. The valve body is designed for drainage of sodium, and a 2-inch drain connection is provided to drain the volume between the discs when they are in the closed position. All wear surfaces such as the discs, seats, and stem guides are stellite surfaced. The valves are provided with pneumatic or motor actuators.

Since tight shutoff is not required of the butterfly valves, they are of the conventional design. A physical stop is provided, in fact, to limit closure to no less than the 10° open position on the 36-inch valve. The 28-inch valves are not so equipped. The valves are of the through-shaft design with the outboard bearing internal to the valve body. Bearing wear areas are stellite surfaced. Valve stems are provided with forced convection cooled freeze seals backed up by a spring loaded packing gland. Pneumatic or motor driven positioning actuators are provided for valve operation.

3.0 PRINCIPLES OF OPERATION

The intermediate sodium valves are components of the main heat transfer system, operation of which is covered in SDD-096-30-000.

The gate blocking valve is normally kept in the full open position. It is closed only when it is desired to isolate the steam generator from the remainder of the heat transfer loop.

The loop main throttling valve is normally full open. Upon reactor trip, it partially closes to control free convection flow in the heat transfer system.

The reheat module throttling valve is normally partially open to control sodium flow to the reheater.

4.0 SAFETY PRECAUTIONS

4.1 Hazards

Possible hazards which could be generated by the valves are:

- a) Inadvertent closing of a gate or butterfly valve.
- b) Leakage across gate valve seat when it is closed.
- c) Stem freeze seal leakage.

4.2 Precautions

The following precautions should be taken to minimize the above hazards.

a) Electrical interlocks and administrative controls should be utilized to prevent closure of the gate valve except when desired for steam generator isolation.

b) Sodium purity (particularly in regard to sodium oxide) should be maintained and monitored to minimize collections of oxide which might cause valve sticking or stem freeze seal binding the valve stem.

c) When the gate valves are closed, the space between the gates should be drained and monitored for presence of sodium.

d) Reliable (preferably redundant) cooling systems for the valve stem freeze seals should be provided. The backup packing for the stem freeze seal should be of a material which will resist deterioration by high temperature sodium.

5.0 MAINTENANCE

The valves and actuators in the secondary loops are accessible at all times although maintenance would normally be done only during loop shutdown. In-place maintenance will normally be limited to some stem seal parts and actuator linkage items. Any major repairs or replacement of components can be accomplished by removal of the valve from the loop.

6.0 APPENDICES

6.1 References

SDD-096-10-000, "Overall System SDD"

SDD-096-30-000, "Heat Transfer System SDD"

SDD-096-32-000, "Intermediate Coolant SDD"

SDD-096-33-000, "Steam Generation SDD"

SDD-096-90-000, "Land, Structures and Conventional Systems
Segment SDD"

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SUPPORTING DOCUMENT

NUMBER

SDD - 096-32-300

REV
LTR B

DOCUMENT TYPE

System Design Description

PROGRAM TITLE

ANL 1000 Mwe Follow-on Study

KEY NOUNS

CSDD- Intermediate Heat
Exchanger

DOCUMENT TITLE

Intermediate Heat Exchanger Conceptual
System Design Description - 1000 Mwe LMFB
Follow-on Study.

ORIGINAL

ISSUE DATE 8/8/68

GO NO.

7692

S/A NO.

13140

PAGE 1 OF

TOTAL PAGES 9

REL. DATE

ABSTRACT

The requirements and description for the
intermediate heat exchanger for the 1000 Mwe
LMFBR Follow-on Study are presented.

PREPARED BY

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1.0 INTRODUCTION

The intermediate heat exchanger (IHX) is the common link or interface between the reactor coolant system and the intermediate coolant system. There are three IHX's in the plant, one for each of the main heat transfer circuits. This CSDD describes these IHX's.

1.1 Functions/Functional Requirements

The IHX is required to transfer heat from the reactor coolant system to the intermediate coolant system under all the conditions that the plant is expected to operate.

1.2 Summary Description

The IHX's are vertical, fixed tube-sheet, shell and tube heat exchangers of the hockey stick concept. Reactor coolant system sodium (primary sodium) flows in the shell and intermediate coolant sodium (secondary sodium) flows in the tubes. The IHX's are of all welded 304 SS construction. The IHX's are installed in the primary equipment vaults.

1.3 Requirements

1.3.1 Performance

1.3.1.1 The IHX's shall transfer heat from the reactor coolant system to the intermediate coolant system as required for the generation of the net full power of 1000 Mwe, which is the plant rated power condition.

1.3.1.2 The IHX's shall have the capability to transfer heat under the plant operating conditions to meet power demands between 25% of plant rated power and maximum steam flow conditions.

1.3.1.3 The IHX's shall transfer heat under conditions prevailing during natural convection decay heat removal.

1.3.2 Environmental

1.3.2.1 The IHX's will be installed in shielded equipment vaults which contain a nitrogen atmosphere at about 150F and 14.7 psia.

1.3.2.2 High temperature sodium with a sodium 24 specific activity of 0.046 curies per cubic centimeter will be circulated through the shell of the IHX.

1.3.2.3 High temperature non-radioactive sodium will be circulated through the tubes of the IHX.

1.3.2.4 The IHX's will be exposed to a neutron flux that will not exceed 1×10^6 n/cm²/sec.

1.3.2.5 The IHX's will be subject to the transient conditions given in paragraph 1.3.2.2 of SDD-096-30-000.

1.3.2.6 The IHX's will be subjected to the steady state conditions as listed in paragraph 1.3.2.1 of SDD-096-30-000.

1.3.2.7 The sodium flowing through the IHX on both the tube and shell side will have impurities as listed in paragraph 1.3.2.3 of SDD-096-30-000.

1.3.3 Operational

1.3.3.1 Primary sodium flow shall be downward in the shell side of the IHX in order to enhance natural convection flow in the reactor coolant system.

1.3.3.2 Running vents on the shell side at the top tube sheet shall be provided to eliminate the possibility of gas collection.

1.3.3.3 The IHX's shall be designed to eliminate holdup pockets and to provide appropriate features to enhance complete drainage of the units on both the tube and shell sides.

1.3.3.4 Thermocouple wells or provisions as required to properly control preheat, including monitoring and controlling temperatures near the center of the tube bundle, shall be provided. Further, the outside surface of the IHX shall have provisions for thermocouples. These provisions consist of machine threaded studs, nuts and washers. Location and number of thermocouples to be determined in later stages of plant design.

1.3.4 Safety

1.3.4.1 The IHX's shall be designed to contain the overpressure excursion resulting from a nuclear accident.

1.3.4.2 The IHX's shall be designed for seismic loads for Class IA components as defined in SDD-096-90-000.

1.3.5 Design

1.3.5.1 Applicable Documents

ASME Boiler and Pressure Vessel Code, Section III, Class A
Nuclear Vessels

TEMA Standards

USAS American Standard Code for Pressure Piping,
Nuclear Power Piping, B 31.7

1.3.5.2 Tubes shall be seamless minimum wall and shall be examined 100% ultrasonically.

1.3.5.3 All stainless steel plate shall be ultrasonically tested.

1.3.5.4 Design and construction shall be all welded including tube-to-tubesheet joints. Tubes shall be properly rolled into tubesheet holes prior to welding.

1.3.5.5 The design shall be such that the tubesheets shall be accessible for tube plugging during shutdown, with a minimum of cutting or rewelding.

1.3.5.6 The IHX's shall be of the vertical hockey stick design to accommodate the differential expansion between the shell and the tubes resulting from possible temperature differentials as great as 300F between these parts.

1.3.5.7 The IHX's shall be designed for a service life of 30 years.

1.3.5.8 Tubes shall be sized and supported such that the natural vibration frequency at any point on the tube shall be at least 50% above any predictable vibration frequency induced by flow at any load from no load to 110% of rated power flow conditions.

1.3.5.9 Flow distribution baffles shall be provided at the inlet and outlet to the shell to distribute flow uniformly for optimal heat transfer and to help in minimizing tube vibration due to flow impingement.

1.3.5.10 The IHX's shall be supported from a point near the upper end of the units such that most of the thermal growth is downward.

1.3.5.11 The IHX's shall be constructed from Type 304 stainless steel.

1.3.5.12 The design temperature and pressure for the IHX's shall be as follows:

	Shell	Tubes
Design temp., F	1155	1155
Design pressure, psig	200	300

2.0 DETAILED DESCRIPTION

The IHX is a vertical tube and shell heat exchanger with fixed tubesheets. The IHX is of the hockey stick configuration with a 90° bend at the lower end to provide a short horizontal leg. This design accommodates the differential thermal expansion between the tubes and the shell. A skirt type support is provided near the top of the exchanger that permits downward expansion. The horizontal leg is supported by constant support spring hangers, and a guide is provided at the bottom on the vertical shell centerline to maintain verticality.

The tube bundle consists of 3400 tubes, 1-in OD x 0.058-in. minimum wall on a 1-1/2-in. square pitch. The tubes are seamless Type 304 stainless steel rolled and welded or suitably bonded to the tubesheets in a code approved manner.

Orifice-type baffles 1/2-in thick are provided on 18-in. centers. A shroud encircles the tube bundle at the shell inlet and outlet nozzles. Each shroud has ellipsoidal holes in it. These shrouds improve the sodium flow distribution through the shell side of the tube bundle.

The shell is about 60 ft in overall height. The horizontal leg overall dimension is about 29-ft. The shell OD is

105-in. A 38-in. nozzle is provided for the shell inlet and another one for the shell outlet. The upper head is provided with a 38-in. nozzle. Located on the lower head is a 36-in. nozzle. The shell is fabricated from Type 304 stainless steel.

Primary sodium flows downward in the shell, and secondary sodium flows upward in the tubes.

A vent nozzle is located just under the top tubesheet to bleed any gas that might accumulate there during operation of the unit.

3.0 PRINCIPLES OF OPERATION

The intermediate heat exchanger is a component of the main heat transfer system, operation of which is covered in SDD-096-30-000.

4.0 SAFETY PRECAUTIONS

4.1 Hazards

Hazards associated with the IHX involve the possible release of high temperature radioactive sodium from leaks in the tubes and shell and the possibility of vent lines plugging with frozen sodium or sodium oxide.

4.2 Precautions

To minimize the hazards mentioned above, the following precautions are to be taken:

a) Prior to plant operation, a vibration check on the installed units as a function of sodium flow is conducted to verify that tube vibration is negligible.

b) During preheating of the unit, temperature differentials between the tubes and shell are to be minimized.

c) Thermal shocks should not be imposed on the unit.

d) Sodium purity is to be carefully maintained and monitored to minimize corrosion, and to prevent plugging of tubes or vent lines.

e) During manufacturing, shipment, and installation, measures are to be taken to minimize contamination of the unit with corrosion-inducing materials or atmospheres.

f) Vent lines should be kept hot at all times during heat transfer system operation.

5.0 MAINTENANCE5.1 Normal/Routine Maintenance

The IHX's, inaccessible during plant operation, are maintainable following reactor shutdown. The maintenance procedure is normally limited to inspection for leaks and to check out and repair of electrical heaters or instrumentation attached to the IHX's.

5.2 Unusual Maintenance

Tube leakage, as a result of weld failure, vibration, corrosion, or tube defects, must be located and plugged. The major steps in this plugging activity are:

- a) Isolate affected heat transfer circuit from the reactor and allow time for decay of sodium-24 (nominally 10 days).
- b) Enter heat transfer vault and cut open the manways on each end of the unit (local shielding may be necessary for this and subsequent steps, depending on the amount of fission product and corrosion product deposition in the unit.)
- c) Locate leaking tube by pressurizing shell with helium and searching with a helium leak detector for the leak at each tubesheet.
- d) Plug leaking tube (specific plugging method to be developed).
- e) Inspect integrity of plug.
- f) Reweld manways.
- g) Inspect integrity of rewelded manways.
- h) Preheat and refill unit.

6.0 APPENDICES6.1 Drawing

"1000 Mwe FBR IHX"

1000 Mwe FBR IHX

3-26-69 UNCL

7692-1232

ATOMICS INTERNATIONAL
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SUPPORTING DOCUMENT

NUMBER

SDD-096-33-000

REV
LTR B

DOCUMENT TYPE

System Design Description

PROGRAM TITLE

ANL - 1000 Mwe Follow-on Study

KEY NOUNS

SDD - Steam Generator

DOCUMENT TITLE

Steam Generation - Conceptual System Design
Description - 1000 Mwe LMFBF Follow-on Study

ORIGINAL

ISSUE DATE August 27, 1968

GO NO.

7692

S/A NO.

13500

PAGE 1 OF

16

TOTAL PAGES

REL. DATE

ABSTRACT

The requirements for and the description of the steam generator for the 1000 Mwe LMFBF Follow-on Study are contained in this SDD

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1.0 INTRODUCTION**1.1 Functional Requirements**

The prime function of the steam generator complex is to convert thermal energy contained in the circulating secondary sodium to superheated steam for the purpose of driving a steam turbine for the ultimate conversion to electrical energy. Secondly, but of no less importance, the steam generator provides a means for the removal of decay heat from the reactor following a shutdown.

1.2 Summary Description

The reference design of the steam generator concept provides one bank of steam generators for each of three secondary sodium loops. Each bank of steam generators consist of an array of steam generator modules arranged vertically and supported in a steel frame structure. Each bank consists of three superheater modules mounted beside eight evaporator modules with three reheat modules placed in parallel with the superheater modules. A relief system is provided to relieve overpressure of the sodium in the event of a tube failure. Rupture discs which relieve into the relief system are provided on each module. Remote activated valves are provided at the sodium inlet and outlet of the modular steam generator banks and a flow control valve is provided in the outlet manifold of the superheater modules. Hydrogen detectors monitor the sodium outlet line from each module and the sodium headers in each bank. Main steam desuperheating is accomplished by means of desuperheating stations at the exit of the evaporators. Reheat steam desuperheating is accomplished by means of desuperheating stations in the cold reheat lines.

This SDD defines only the steam generator consisting of evaporator, superheater, and reheater modules assembled in banks. Valves, relief discs, relief systems, and hydrogen detectors are covered elsewhere and constitute interfaces with the steam generator.

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1.3 Requirements1.3.1 Performance

1.3.1.1 The steam generator shall transfer heat from the Intermediate Sodium Coolant System to the water and steam within the steam generator as required for the generation of the net full rated power condition.

1.3.2.2 The steam generator shall have the capability to generate steam under the plant operating conditions to meet power demands up to the plant rated power and steam flow conditions.

1.3.1.3 The steam generator shall provide heat removal capability under conditions prevailing during natural convection decay heat removal in the main heat transfer system.

1.3.2 Environment

1.3.2.1 The steam generator will be housed in a building ventilated with ambient air and may experience temperatures ranging from 0°F to 120°F and 90% maximum relative humidity.

1.3.2.2 The steam generators will have high temperature sodium in the shells of the modules and high pressure steam or water in the tubes.

1.3.2.3 The steam generator evaporator and superheater module will be designed for the transient conditions given in Table I. (These tables are estimates and are provided for guidance only.) Transients for the reheater will be (to be determined).

1.3.2.3 The steam generator will utilize water with the following water chemistry:

Total dissolved solids	0.05 ppm max
Silica, SiO ₂	0.01 ppm max
Iron	0.01 ppm max
Copper	0.002 ppm max
Dissolved oxygen	0.007 ppm max
Free caustic	
pH	(To be determined)

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- 1.3.2.4 The steam generator will be exposed to sodium with impurities as given in SDD-096-30-000, paragraph 1.3.2.3.
- 1.3.2.5 The steam generator shall be designed for seismic loads for class 1A components as defined in SDD-096-90-000.

1.3.3 Operational

1.3.3.1 The steam/water side of the steam generator modules shall be maintained at least 5 psi higher than the sodium side under all operating conditions in order to control the direction of leakage should a leak occur.

1.3.3.2 Temperature and pressure instrumentation shall be provided on the inlet and outlet steam and sodium manifolds for the evaporator superheater, and reheater modules.

1.3.3.3 A hydrogen detector system shall be provided for each set of each group of modules in each bank. The hydrogen detector system shall utilize a selector valve arrangement such that each individual module sodium flow may be monitored for hydrogen content by selective valving.

1.3.3.4 Each rupture disc assembly shall be provided with a leak detection system to monitor for sodium leakage.

1.3.3.5 For turbine protection, the criterion shall be 0% liquid droplet carryover in the steam at the superheater outlet between 25% and 110% of rated power during the expected steady state and transient operating conditions.

1.3.4 Safety

1.3.4.1 Provisions shall be made for pressure relief and reaction products handling which are adequate for the ultimate failure of seven tubes in a module. Direct release of sodium or reaction products to the atmosphere shall not be permitted.

1.3.5 Design

1.3.5.1 Applicable Documents

ASME	Boiler and Pressure Vessel Code Section III, Nuclear Vessels Class "C"
ASME	Boiler and Pressure Vessel Code Case Interpretations, Case 1331-3, Nuclear Vessels in High Temperature.

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1.3.5.2 The steam generator modules of each type shall be of the hockey stick design.

1.3.5.3 The design life of the steam generator shall be 30 years.

1.3.5.4 The design and operating temperatures and pressures of the superheater and evaporator modules shall be as follows:

	Steam			Sodium	
	S.H.Out	S.H.In/Evap.Out	Evap.In	Evap	Sup
Operating Temp. F	1005	715	480	905	1070
Operating Press. psia	2490	2735	2780	246	246
Design Temp. F	1085	920	685	920	1085
Design Press. psia	2670	2970	3020	300	300

1.3.5.5 The design and operating temperatures and pressures of the reheater modules shall be as follows:

	Steam Side	Sodium Side
Operating Temp. F	1005	1070
Operating Press. psia	555	240
Design Temp. F	1085	1085
Design Press. psia	600	300

1.3.5.6 There shall be 6 evaporator modules, 4 superheater modules, and 3 reheater modules per steam generator bank, of which there are three banks in the plant.

1.3.5.7 The thermal duty of each of the modules shall be as follows:

Evaporator	26.8×10^7 BTU/hr
Superheater	16.7×10^7 BTU/hr
Reheater	14.9×10^7 BTU/hr

1.3.5.8 Tubes shall be butt-welded to the tubesheets at each end.

1.3.5.9 Tubes shall be supported by spacers which minimize cross-flow, induced vibration, and fretting.

1.3.5.10 Tube to tubesheet welds shall be radiographically inspectable.

1.3.5.11 Provisions shall be made in the evaporator module type for insertion of flow orifices in the tube inlets. Such orifices would be used in the evaporator modules for boiling stability if required.

1.3.5.12 Surface area provisions shall be based on a 95% probability of meeting 100% of rated power conditions. Allowance for tube plugging shall be made. A maximum of 4 non-adjacent tubes per module may be plugged during fabrication, which represents approximately 1% loss in heat transfer capacity.

1.3.5.13 Sodium side corrosion allowance shall be 0.004 inches.

1.3.5.14 Steam side corrosion and scaling allowance shall be 0.010 inch in the heat transfer tubing and 0.065 inch elsewhere.

1.3.5.15 The water side fouling factor shall be taken as 0.0003 based on the tube OD.

1.3.5.16 The materials of construction shall be 2-1/4 Cr-1 Mo steel for the evaporator modules and 321 H for the superheater and reheater modules.

1.3.5.17 Rupture disc installation on the modules shall be arranged for ease of inspection and replacement.

1.3.5.18 The sodium portions of the modules shall be designed to drain as completely as possible.

2.0 DETAILED DESCRIPTION

The steam generator modules consist of tubes, tubesheets, shells, tube supports, inlet and outlet flow baffles and thermal insulation. The modules are of the hockey stick configuration as shown in Fig. 1 with the bent portion at the top. This configuration provides for tube and shell differential thermal expansion.

The tubes are butt-welded to nipples that are machined into the tubesheets to make a crevice free joint that is radiographically inspectable. The tubes are supported by spacers which embrace the full tube bundle to minimize crossflow, induced vibration, and fretting. The shell is made from straight pipe sections with piping tees or crosses for the sodium inlet, outlet, rupture disc mounting for overpressure relief and fabrication inspection port. A piping elbow provides the bend to make the hockey stick.

Each module contains 380 tubes. The superheater and

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in material, length and evaporator modules are similar except that flow orifices may be installed in the evaporator tube inlets. The reheater modules utilize a tube size that is different than the superheater/evaporator tube size.

The evaporator modules are made of $2\frac{1}{4}$ chrome - 1 moly material.

There are three banks of steam generators in the plant, there being one bank per heat transfer circuit. Each bank consists of 6 evaporators, 4 superheaters, and 3 reheaters. The modules in each bank are arranged in a linear array with the evaporator, superheater, and reheater modules side by side. The modules are supported in a steel structure that is enclosed by metal siding.

Water and/or steam flows in the tubes of the modules with sodium in the shell. Rupture discs for overpressure relief in the event of a tube failure are installed on the module shells.

3.0 PRINCIPLES OF OPERATION

3.1 Startup

The steam generator is preheated with either steam or water to about 350°F. Steam is used when the initial temperature is below 50°F and until the temperature is raised to about 150°F. Pressurized water is used when the initial temperature is above 50°F or after the temperature has been raised to 150°F by steam. Water is circulated through the evaporator and steam is circulated through the superheater and reheater until 350°F is reached. The water-steam mixture is then drained and the modules are dried and held at 350°F by admitting dry steam. Sodium is admitted to the steam generator, and sodium circulation is begun.

Feedwater at 350°F is admitted to the evaporator modules, and upon exiting is flashed to the flash tank. Steam from the flash tank is admitted to the superheater and reheater modules. The steam from the superheater and reheater modules is dumped to the main condensers.

The steam generator is operated in this manner while the sodium flow and temperature are increased in the heat transfer loops

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as reactor power is increased. With increasing sodium flow and temperature, feedwater temperature and pressure are increased until the desired steam conditions exiting the evaporator modules are reached. The turbine is now rolled with this steam, and the superheater and reheater modules are placed in their normal mode of operation. The startup phase of operation is continued until the turbine is loaded to about 10 to 15% power.

3.2 Normal Operation

Operation of the steam generator from about 15% to design power is fully automatic and constitutes normal operation. In this mode of operation feedwater is admitted at the bottom of the evaporator modules, rises, flashes to steam, is slightly superheated, and then exits.

The steam is then passed through the superheater. From the superheater the steam flows to the high pressure (HP) turbine. Steam exiting from the HP turbine flows through the reheater. Steam exiting from the reheater flows to the intermediate pressure (IP) turbine, then the low pressure (LP) turbine, and finally to the condenser.

3.3 Shutdown

Upon reduction of reactor power in preparation for shutdown, steam production is continued in the steam generator. The steam flow to the turbine is continued until the generator is tripped from area load frequency control. At this point, the steam dump valves on the superheaters and reheaters open. The exhausted steam is reduced in pressure, desuperheated, and then admitted to the main condenser. The sodium flow through the reheater is reduced to zero, and the reheater is isolated. Heat from the sodium loops is then removed by means of the evaporator and superheater modules only.

3.4 Emergency

3.4.1 Steam Dumping

Heat rejection by the steam generator upon reactor or turbine trip is handled as described in Section 3.3 above. Any steam

flow above the normal steam dump capacity, however, is exhausted to the atmosphere. If for any reason the main condensers are not available for steam dumping, the steam is exhausted to the atmosphere.

3.4.2 Tube Rupture

Detection of a tube rupture by rupture disc failure of a steam generator initiates isolation of the steam generator bank from its secondary loop by closing the sodium loop valves. Feedwater supply is stopped, the bank is isolated from the steam system, and the water and steam in the bank are dumped. The primary and secondary loop sodium pumps in the affected circuit are shut down. Overpressure in the shell due to steam pressure and the water/sodium reaction is relieved by means of the rupture discs on each of the modules. Nitrogen gas at 200 psig is admitted to the bank on the water side to prevent sodium from entering the steam side of the system.

The above action is automatic, and it is designed to reduce damage to the steam generator to a minimum and leave the steam generator in a suitable condition to allow recovery operations.

4.0 SAFETY PRECAUTIONS

4.1 Hazards

The hazards of the steam generator are those associated with high pressure water/steam and high temperature liquid metal containing equipment. Specific hazards are:

- a) Tube leakage, permitting steam to enter the secondary sodium system and react with the sodium.
- b) Shell leakage, permitting secondary sodium to leak out of the system and ignite in the steam generator building.
- c) Steam leakage
- d) Steam overpressure

4.2 Precautions

The following precautions should be taken to minimize the above hazards.

- a) The presence of hydrogen in the secondary sodium system should be monitored and the affected loop shut down immediately if significant quantities of hydrogen are detected.

- b) Frequent surveillance of the equipment in the steam generator building should be performed to detect possible sodium and steam leakage.
- c) Pressure relief valves should be checked yearly for pop pressure and adequacy of reseal.

5.0

MAINTENANCE

5.1

Normal/Routine Maintenance

Maintenance on the steam generator will consist of the following items:

- a) Inspect spring hangers at one year intervals and repair/replace at ten year intervals or as required.
- b) Inspect/repair insulation at one year intervals.
- c) Inspect instrumentation at one year intervals and repair/replace as required.
- d) Inspect rupture discs yearly and change as required.
- e) Repair or replace relief valves as required during shutdown.
- f) Periodic cleaning of water side deposits in evaporators.

5.2

Unusual Maintenance

Tube leakage will require replacement of a module. This can be done as follows:

- a) Drain affected secondary loop
- b) Cut out damaged module
- c) Replace with spare, if spare is available, or
- d) Cap off pipe stubs where module has been removed and return loop to part-load service while module is being repaired.
- e) Restore loop to full-load service after module has been replaced.

6.0

APPENDICES

6.1

Drawing

Steam Generator Building Equipment Arrangement

NORTH AMERICAN ROCKWELL CORPORATION

CODE IDENT. NO. 02974

NUMBER	REVISION LETTER	PAGE

TABLE 1 - TRANSIENT CONDITIONS

STEAM GENERATOR OPERATING CONDITIONS	NOMINAL TEMPERATURE RANGE °F (ERROR BAND; + 15 F				PRESSURE RANGE, PSIA				CHANGING VARIABLE (INPUT TO ANALYSIS)	MAXIMUM RATE OF CHANGE	NUMBER OF OCCURRENCES IN 30 YEARS	CAUSE
	SODIUM		STEAM		SODIUM		STEAM					
	Inlet	Outlet	Inlet	Outlet	Inlet	Outlet	Inlet	Outlet				
<u>I. TRANSIENTS</u>												
<u>A. Normal</u>												
1. Load swing	1070 to 1030 to 1070	670 to 630 to 670	470 to 353 to 470	NC	204 to 217 to 204	189 to 214 to 189	2780 to 2450 to 2730	2470 to 2410 to 2470	Power - from 100% to 25% to 100%	1/Minute	9000	Normal
2. Initial Preheat	70 to 350	70 to 350	70 to 350	70 to 350	NC	NC	15 to 615	15 to 615	Temperature	100 F/Hour	3	Normal
3. Rise to Temp.	350 to 1070	350 to 670	350 to 470	350 to 1005	15 to 204	15 to 189	615 to 2780	615 to 2470	Sodium Temperature	At Inlet > 900F 100 F/30 Min. At Outlet < 900F 100 F/Hr	3	Normal
4. Refueling Shut-down	1070 to 500 to 1070	670 to 500 to 670	470 to 350 to 470	1005 to 500 to 1005	204 to 80	189 to 110	2780 to 615 to 2780	2470 to 615 to 2470	Sodium Temperature	At Inlet > 900F 100 F/30 Min. At Outlet < 900F 100 F/Hr	30	Normal
5. <u>Dry Maintenance Shutdown</u>												
a. Drop from Temp.	1070 to 350	670 to 350	470 to 350	1005 to 350	204 to 15	119 to 15	2785 to 615	2470 to 615	Sodium Temperature	At Inlet > 900F 100 F/30 Min. At Outlet < 900F 100 F/Hr	30	Normal Maintenance Requir- ing Steam Generator Drainage
b. Shutdown & Drain	350 to 70	350 to 70	350 to 70	350 to 70	NC	NC	615 to 15	615 to 15	Sodium Temperature	100 F/Hr		
c. Initial Preheat	70 to 350	70 to 350	70 to 350	70 to 350	NC	NC	15 to 615	15 to 615	Temperature	3 F/Hour		Normal
d. Rise to Temp.	350 to 1070	350 to 670	350 to 470	350 to 1005	15 to 204	15 to 189	615 to 2780	615 to 2470	Sodium Temperature	At Inlet > 900F 100 F/30 Min. At Outlet < 900F 100 F/Hr.		Normal

(NC) Indicates No Change

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TABLE I - TRANSIENT CONDITIONS (CONT)

STEAM GENERATOR OPERATING CONDITIONS	NOMINAL TEMPERATURE RANGE °F (ERROR BAND + 15 F				PRESSURE RANGE, PSIA				CHANGING VARIABLE (INPUT TO ANALYSIS)	MAXIMUM RATE OF CHANGE	NUMBER OF OCCURRENCES IN 30 YEARS	CAUSE
	SODIUM		STEAM		SODIUM		STEAM					
	Inlet	Outlet	Inlet	Outlet	Inlet	Outlet	Inlet	Outlet				
<u>I. TRANSIENTS (CONT)</u>												
<u>A. Normal (Cont)</u>												
6. Wet Maintenance Shutdown											30	
a. Drop from Temp.	1070 to 350	670 to 350	470 to 350	1005 to 350	204 to 15	189 to 15	2780 to 615	2470 to 615	Sodium Temperature	At Inlet > 900 100 F/30 Min. At Outlet 100 F/hr < 900		Normal maintenance requir- ing Steam Gen. Drainage
b. Rise to Temp.	350 to 1070	350 to 670	350 to 470	350 to 1005	15 to 204	15 to 189	615 to 2780	615 to 2470	Sodium Temperature	At Inlet > 900 100 F/30 Min. At Outlet 100 F/hr < 900		Normal
7. Power Setback	1070 to 1055 to 1070	670 to 655 to 670	470 to 423 to 470	NC	204 to 210 to 204	189 to 202 to 189	2780 to 2595 to 2780	2470 to 2450 to 2470	Power, 100% to 70% 70% to 100%	25%/Min. 2%/Min.	720	Normal
<u>B. Accidents</u>												
1. Reactor Trip												
a. Normal	NC	670 to 660	470 to 375	NC	NC	NC	NC	NC	All Temperature	40 F/10 sec	200	Equipment malfunction
b. Convection flow control mal- function	1070 to 1170	670 to 630	470 to 375	NC	NC	NC	NC	NC			5	Equipment malfunction
2. Turbine Trip	NC	NC	470 to 350	NC	NC	NC	NC	NC	Power, 100% to 25%	25%/Min	80	Equipment malfunction

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PAGE

TABLE I - TRANSIENT CONDITIONS (CONT)

STEAM GENERATOR OPERATING CONDITIONS	NOMINAL TEMPERATURE RANGE °F (ERROR BAND ± 15 F				PRESSURE RANGE, PSIA				CHANGING VARIABLE (INPUT TO ANALYSIS)	MAXIMUM RATE OF CHANGE	NUMBER OF OCCURRENCES IN 30 YEARS	CAUSE
	SODIUM		STEAM		SODIUM		STEAM					
	Inlet	Outlet	Inlet	Outlet	Inlet	Outlet	Inlet	Outlet				
B. ACCIDENTS (CONT)												
3. Flow Mismatch at Constant Power												
a. Primary												
1) Loss of Flow (one pump)	1070 to 630	670 to 630	NC	1005 to 630	NC	NC	NC	NC	Sodium Temperature	180 F/10 sec	1	Butterfly valve closes
2) Flow Ramp Up (one pump)	1070 to 1140	NC	NC	NC	NC	NC	NC	NC	Sodium Temperature	30 F/10 sec	1	Single pump ramp up to 100% flow at 25% power
b. Secondary												
1) Loss of Flow (superheater temperatures)	NC	905 to 560	715 Stm to 470 Water	1005 Stm to 605 Water	204 to 133	178 to 163	NC	2470 to 2800	All Temperatures	Sodium Outlet 57 F/10 sec Steam 50 F/10 sec at outlet 40 F/10 sec at inlet	1	Butterfly valve closes
2) Flow Ramp up	1015 to 820	NC	NC	1005 to 820	217 to 204	214 to 189	NC	NC	Sodium Temperature	80 F/10 sec	1	Ramp up of all secondary pumps to 100% flow at 25% power
c. Feedwater												
1) Loss of flow (Evaporator Temperatures)	905 to 1070	670 to 860	470 to 1070	715 to 860	NC	NC	NC	NC	All temperatures	80 F/10 sec	2	Spurious signal causing instantaneous feedwater valve closure or pump seizure
2) Ramp up	NC	NC	NC	1005 Stm to 590 H ₂ O	NC	NC	2450 to 2780	2410 to 2470	Steam-side Temperature	50 F/10 sec	1	Ramp up of both feedwater pump to 100% flow at 25% power

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TABLE I - TRANSIENT CONDITIONS (CONT)

TABLE 1 - TRANSIENT CONDITIONS (CONT.)										NUMBER		REVISION LETTER						PAGE	
STEAM GENERATOR OPERATING CONDITIONS	NOMINAL TEMPERATURE RANGE OF (ERROR BAND + 15 F)				PRESSURE RANGE, PSIA				CHANGING VARIABLE	MAXIMUM RATE OF CHANGE	NUMBER OF OCCURRENCES IN 30 YEARS	CAUSE							
	SODIUM		STEAM		SODIUM		STEAM												
	Inlet	Outlet	Inlet	Outlet	Inlet	Outlet	Inlet	Outlet											
3. (Continued)																			
4. All Flow																			
1. Loss of Flow	NC	670 to 630	470 to 375	1005 to 1070	NC	NC	NC	NC	All temps.	40F/10sec	1	Spurious Reactor or Turbine Trip Signal. Reactor Trips.							
2. Rampup	1015 to 1070	630 to 670	353 to 470	NC	217 to 204	214 to 189	2450 to 2780		All Flow from 25% to 100%	25%/min	1	Spurious Reactor or Turbine Power Increase Signal							
4. Power Mismatch at Constant Flow																			
a. Control Rod Bank Insertion	1070 to 1060	670 to 660	NC	1005 to 715	NC	NC			Power from 100% to 80%	25%/min.	1	Spurious Reactor Overtemperature Signal.							
b. Control Rod Bank Withdrawal	NC	670 to 640	470 to 375	1005 to 1070	NC	NC	NC	NC	All temps.	40F/10sec	1	Spurious Reactor Under Temperature Signal at 25% Power. Protective System Shuts Reactor Down at About 35% Power.							
5. Flow Distribution at Constant Power																			
a. Sodium Control Valve Failure																			
1. Open	905 to 880**	NC	NC	NC	NC	214 to 220	NC	NC	Sodium Flowrate to Superheater 100% to 80%	Step Change	2	Spurious Signal							
2. Closed	905 to 950**	670 to 743	NC	NC	217 to 265	214 to 168	NC	NC	Sodium Flowrate to Superheater 100% to 200%.	Step Change	2	Spurious Signal							
b. Attenuator Valve Failure																			
1. Closed	NC	NC	NC	NC	NC	NC	NC	NC	N/A	N/A	5	Valve Sticking Shut.							
2. Open	NC	NC	NC	1005 to 770	NC	NC	NC	NC	Steam temp.	Step Change	5	Valve Opens Wide at 25% load.							
6. One Loop Shutdown*	NC	NC	NC	NC	NC	NC	NC	NC	All Flow	Step Change 100% to 0%.	50	Equipment Failure with Normal Protective Action For One Loop.							
II. OTHER CONDITIONS																			
A. Sodium Water Reaction In Module	NC	NC	NC	NC	217 to 2780	214 to 2780	NC	NC	Sodium Pressure	Step Increase	1	Tube Failure							

* For All Practical Purposes Temperatures And Pressures Do Not Change

** These temperatures are for the superheater outlet

FIGURE I

FBR STEAM GENERATOR MODULE

	No. Modules	No. of Tubes	L	Tube Size	Tube Mat'l	Shell Thickness	Shell I.D.	Tube Bundle, Dia.	Nozzle Size
Evaporator	18	380	58'7"	5/8"OD 0.095 min.Wall	2 1/4 Chr. 1 Moly	1-1/8"	31"	27 1/2"	24
Superheater	12	380	44'10"	"	321 H	"	"	"	"
Reheater	9	380	41' 3"	1.0"O.D. 0.058 min.Wall	"	3/4"	38"	34 1/2"	"

Steam Generator Design Data

ATOMICS INTERNATIONAL
A Division of North American Rockwell Corporation

SUPPORTING DOCUMENT

NUMBER

SDD-096-40-000

REV

LTR A

DOCUMENT TYPE

System Design Description

KEY NOUNS

Energy Conversion System

PROGRAM TITLE

ANL - 1000 Mwe Follow-on Study

DOCUMENT TITLE

Energy Conversion Segment Conceptual System
Design Description - 1000 Mwe IMFBR Follow-on
Study

ORIGINAL
ISSUE DATE February 23, 1968

GO NO.

7692

S/A NO.

13120

PAGE 1 OF

TOTAL PAGES 35

REL. DATE

ABSTRACT

The conceptual requirements and design of the
energy conversion system for the 1000 Mwe IMFBR
Plant are contained in this document.

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1. INTRODUCTION

The energy conversion system is designed to transform a portion of the available thermal energy in the steam generated in the steam generators into electrical energy. This transformation is obtained by means of a turbine-generator unit operating in a reheat regenerative cycle.

The energy conversion system contains the necessary equipment and controls used for normal operation of the steam cycle in the load range. The steam generator feedwater treatment, condensate demineralization, and the condenser heat rejection functions are also included in the energy conversion system.

The conceptual system description covers the following main subsystems and components.

- 1) Turbine generator,
- 2) Boiler feed pump turbine drives,
- 3) Main condenser,
- 4) Auxiliary condensers,
- 5) Condensate and feedwater,
- 6) Feedwater treatment,
- 7) Main steam and reheat steam,
- 8) Extraction steam,
- 9) Condenser circulating water.

1.1 Functions/Functional Requirements

The steam-water cycle is selected and designed to obtain a high net plant efficiency consistent with technological, operational and economic constraints. The following are the functional requirements of the energy conversion system.

- 1) Safe and reliable operation during startup, in the load range from 25% to 100% of rated power, during sudden load rejections, and during normal and emergency shutdowns.
- 2) High availability in order to meet an overall mature plant energy availability of 87%.
- 3) Low capital and operating costs to result in an economically competitive, investor owned, power station.
- 4) Base load operation with load following capability in response to load dispatching signals, or plant main control room.
- 5) An operating plant life of 30 years.

1.2 Summary Description

The reference plant net electrical output at the generator terminals is 1000 MW. The turbine generator is a tandem compound unit operating in a reheat regenerative cycle with seven feedwater heaters, as shown in the heat balance Figure 6.2.1. The following flow diagrams included in Appendix 6.2 show the main features of the turbine generator plant.

Main Steam and Reheat Steam
Condensate and Feedwater

Figure 6.2.2
Figure 6.2.3

The main turbine is a 3600 rpm, tandem compound unit with 33.5-in. last stage buckets, designed for initial steam conditions of 2400 psig, 1000°F, with reheat to 1000°F at 500 psig. The rated capability of the turbine at 1.5-in. Hg. condenser pressure, and 0 percent makeup is 1054 MW, under normal operating conditions. The generator capability at rated hydrogen pressure and 0.85 power factor is 1,240 MVA. Steam from seven extraction points is used for heating the feedwater to a final temperature of 478°F, at 100% turbine flow.

Steam from the intermediate pressure turbine exhaust is supplied at 190 psig, 765°F, to two turbines which drive two half-capacity boiler feed pumps. Each turbine exhausts to its respective condenser operating at 2 in. Hg. High pressure steam at 2400 psig, 1000°F, is used during hot starts, low load operating and overload conditions. Cold startups are obtained using steam from two auxiliary fossil fuel steam generators.

The normal path of condensate and feedwater flow from the single shell condenser to the steam generators is as follows: Condenser hotwell, condensate pumps, condensate coolers, condensate prefilters, condensate mixed bed demineralizers, generator hydrogen coolers, generator stator coolers, turbine gland steam condensers, low pressure feedwater heaters, deaerating feedwater heaters, boiler feed pumps, and high pressure feedwater heaters.

The plant power generation, thermal requirements and efficiency, established in the heat balance diagram, Figure 6.2.1, are given in Table 1.1.

1.3 Requirements

1.3.1 Performance Requirements

1.3.1.1 The energy conversion system shall be designed to generate a net electrical output of 1000 Mwe at 0.85 pf, 60 cps, 3-phase and 22 kv. This power level is the rated condition, exclusive of the plant auxiliary power.

1.3.1.2 The energy conversion shall take place in a turbine-generator unit operating in a reheat, regenerative, steam cycle as shown in Figure 6.2.1.

1.3.1.3 The 100% rated steam flow shall be 6,854,100 lb/hr, at 2400 psig and 1000°F at the turbine inlet. One reheat point shall be used at 500 psig and 1000°F with a corresponding steam flow of 6,401,930 lb/hr.

TABLE 1.1

ENERGY CONVERSION SYSTEM DESIGN SUMMARY

100% STEAM FLOW

Gross Generator Output (0.85 pf), Mwe	1040.0
Plant Auxiliary Power, Mwe	38.3
Plant Net Electrical Output, Mwe	1001.7
Plant Net Heat Rate, Btu/kw-hr	8174.6
Plant Net Thermal Efficiency, %	41.7
Evaporators Thermal Power, Mwt	1417.6
Superheaters Thermal Power, Mwt	587.6
Reheaters Thermal Power, Mwt	394.8
Turbine Steam Conditions, psia/°F/°F	2415/1000°/1000°
Steam Flow, million lb/hr	6.854
Condenser Pressure, in. Hg	1.5

1.3.1.4 The condenser pressure at the 100% rated steam flow condition shall be 1.5 in. Hg.

1.3.1.5 The final feedwater temperature shall be 478°F at the 100% rated steam flow condition.

1.3.1.6 Two steam turbines shall be used to drive the two half-capacity steam generator feedwater pumps. The turbines require a total of 252,500 lb/hr of steam at 190 psig and 765°F, which is extracted from the main turbine at the 100% rated flow condition. The turbines exhaust at 2 in. Hg.

1.3.1.7 The net plant heat rate at the conditions given in the above paragraphs shall be 8,174.6 BTU/kw-hr which represents a net plant efficiency of 41.7%. These values are based on a primary loop ΔT of 360°F and a ΔP of 150 psi, and a secondary loop ΔT of 400°F and a ΔP of 90 psi.

1.3.1.8 The condenser circulating water shall be obtained from a fresh water supply at an average yearly temperature of 55°F.

1.3.1.9 Water treatment of the feedwater makeup shall be provided with a capacity to demineralize approximately 3% of the rated turbine throttle flow.

1.3.1.10 Water treatment of the condensate water shall be provided with a capacity to polish and demineralize 100% of condenser flow, up to the 100% turbine rated capability.

1.3.1.11 The system shall provide for programmed control with manual overrides during the startup, synchronization and initial loading sequence up to approximately 30% of rated steam flow. Load following capability shall be provided from approximately 25% to 110% rated steam flow. A shutdown programmed sequence shall also be provided.

1.3.1.12 The system shall permit normal changes (approximately 25% rated) at rates up to $\pm 2\%$ per minute, or $\pm 10\%$ step changes, without exceeding the turbine inlet and reheat pressure and temperature limitations. The step load changes are assumed to be step changes in turbine control valve opening such that when equilibrium conditions are restored, the new load will correspond to a $\pm 10\%$ change.

1.3.1.13 The pressure at the turbine stop valves shall be controlled so that it does not exceed 105% of rated pressure. During abnormal conditions, the pressure may exceed rated pressure briefly by as much as 30%; but the aggregate duration of such brief swings, beyond 105% of rated pressure, shall not exceed 12 hr per 12 month operating period.

1.3.1.14 The steam temperature at each main line to the turbine stop valves shall average not more than rated temperature over any 12 month operating period. In maintaining this 12 month average, the temperature shall not exceed rated temperature by more than 15°F during normal operation. During abnormal conditions, the main steam temperature shall not be in excess of rated temperature by more than 25°F for not more than 400 hr per 12 month operating period. The main steam temperature shall not exceed 50°F above rated temperature, for swings of 15 min. duration or less, aggregating not more than 80 hr per 12 month period.

In maintaining the required temperatures, differences between the main steam lines, feeding a common valve casing or turbine shell, shall not exceed 20°F. During abnormal conditions, the temperature differences shall not exceed 75°F for periods aggregating not more than 400 hr per 12 month operating period.

1.3.1.15 The pressure, at the high pressure exhaust connection to the cold reheat lines, shall not be greater than 25% above that which exists when the high pressure turbine is passing design flow with rated steam and operating conditions. Suitable relief valves shall be provided if necessary.

1.3.1.16 The steam temperature at each turbine reheat admission shall average not more than rated reheat temperature over any 12 month operating period. In maintaining this 12 month average, the reheat temperature shall not exceed the rated value by more than 15°F during normal operation. During abnormal conditions, the reheat temperature shall not exceed the rated value by more than 25°F for not more than 400 hr per 12 month operating period. The reheat temperature shall not exceed 50°F above the rated value for swings of 15 min. duration or less, aggregating not more than 80 hr per 12 month operating period.

In maintaining the required temperatures, differences between the reheat steam lines, feeding a common valve casing or turbine shell, shall not exceed 20°F. During abnormal conditions, the temperature differences shall not exceed 75°F for periods aggregating not more than 400 hr per 12 month operating period.

1.3.1.17 The system controls shall be designed to maintain essentially a constant inlet pressure and temperature and reheat temperature, within the limitations stated in the foregoing paragraphs, in the range from 25% to 110% rated steam flow.

1.3.1.18 The system, in conjunction with the Startup and Shutdown System, shall permit a 25% step load reduction, by means of an automatically controlled steam dump to the condenser, without steam release to the atmosphere. The same steam dump shall be used for removal of residual heat released in the reactor during shutdown.

1.3.1.19 Combined action of the controlled steam dump to the condenser and to the atmosphere provided by the Startup and Shutdown System, shall permit 30% step load reductions without reactor trip. A turbine trip shall result in steam generator safety valve operation and steam release to the atmosphere depending on the load carried prior to the tripout.

1.3.1.20 The system shall be designed to meet the thermal and flow requirements shown in the 100% steam flow heat balance diagram, Figure 6.2.1. Margins to achieve these requirements shall be assigned according to the equipment manufacturers design practices. Margins to obtain stretch capability, or to maintain the rated capability under special operating conditions shall not be provided.

1.3.1.21 The system part load performance shall be established by means of heat balances calculated at the turbine control valve points, which occur at approximately 75%, 40% and 25% of rated steam flow.

1.3.1.22 The maximum turbine exhaust hood temperature shall not exceed 175°F. The system shall provide for water sprays to limit this temperature during emergency conditions.

1.3.1.23 The system shall provide for approximately one minute storage in the condenser hot well and four minutes storage in the deaerator storage tanks, to supply the feedwater requirements at 100% rated steam flow.

1.3.2 Environmental Requirements

1.3.2.1 The energy conversion system equipment shall be located indoors, within the Turbine-Generator Building; with the exception of the condenser circulating water intake equipment, water, chemical and oil storage tanks, deaerators, and the hydrogen, nitrogen and chlorine bottles.

1.3.2.2 The condensate and feedwater equipment shall be subject to feedwater of the highest quality. The following provisions shall be used to meet this requirement.

- 1) Full flow condensate filtering, polishing and demineralization,
- 2) Feedwater makeup demineralization,

- 3) Feedwater heater train of carbon steel and 100% tube welding to minimize copper pickup and tube leakage problems,
- 4) Nitrogen blanketing to prevent feedwater heater corrosion during outage periods,
- 5) Chemical injection to control the condensate and feedwater pH, and
- 6) Feedwater deaeration.

1.3.2.3 Various subsystems of the energy conversion system shall contain the following fluids:

- 1) Lubricating oil,
- 2) Nitrogen gas for equipment protection,
- 3) CO₂ gas for generator hydrogen system purging and fire protection,
- 4) Compressed air for control and station service,
- 5) Hydrogen for generator cooling,
- 6) Raw cooling and service water,
- 7) Sulfuric acid for cation resin regeneration,
- 8) Sodium hydroxide for anion resin regeneration, and
- 9) Hydrazine and ammonia for control of the feedwater oxygen content and pH level.

1.3.2.4 The system shall be subject to the rated steady state operating pressures and temperatures shown in the heat balance diagram, Figure 6.2.1. The corresponding feedwater pressure, at these operating conditions, shall be 3250 psig at the inlet to the feedwater control valves.

1.3.2.5 The system shall be subject to transient pressures, temperatures and flows produced by the following normal and emergency operating conditions.

- 1) Startup preheat from 50°F to 350°F,
- 2) Startup preheat from 350°F to 600°F and pressurization,
- 3) Starting and loading a cold turbine ($< 250^{\circ}\text{F}$),
- 4) Starting and loading a warm turbine ($\approx 400^{\circ}\text{F}$),
- 5) Starting and loading a hot turbine ($\approx 700^{\circ}\text{F}$),
- 6) Load changes at rates up to $\pm 6\%$ per minute,
- 7) Step changes in turbine control valve opening resulting in $\pm 10\%$ load changes,
- 8) Step load reductions of $\pm 25\%$ load demand,
- 9) Step load reductions of $\pm 30\%$ load demand,
- 10) General normal shutdown with load reduction of 2% rated load per minute and turbine tripout at approximately 15% rated load (steam temperature nearly constant),

- 11) General normal shutdown with cooling for short time turbine maintenance. The load reduction rates prior to tripout shall be obtained according to the desired decrease in steam temperature as follows:

$$\text{Rate} = 1.7/(1 + \Delta T/150), \text{ \%/min.},$$

- 12) Shutdown required in shortest period of time. Obtained by reducing load from 100% to 30% in 20 min., with negligible change in turbine inlet temperature, and tripout from the minimum load,
- 13) Emergency turbine generator tripout from 100% rated load,
- 14) Spurious reactor trip due to equipment malfunction or operator error,
- 15) Single control rod insertion,
- 16) Reactor trip without reduction of sodium flow,
- 17) Loss of primary and/or secondary sodium flow,
- a) One primary loop of three,
 - b) Two primary loops of three,
 - c) Two secondary loops of three (primary flow continues),
 - d) Primary and secondary loops,
- 18) Control rod withdrawal,
- a) One rod, initially at rated power,
 - b) One rod, hot restart,
 - c) One rod, cold restart,
 - d) Bank of rods, initially at full power,
- 19) Tripout of feedwater pump turbine drive.

1.3.3 Reliability/Availability Requirements

1.3.3.1 The energy conversion system shall be designed for a 30 year plant life.

1.3.3.2 The system shall have an availability of 95.2% in order to meet an overall mature plant energy availability of 87%.

1.3.4 Maintenance Requirements

1.3.4.1 Maintenance shall be accomplished to the greatest possible extent while the plant is in operation.

1.3.4.2 Redundancy shall be considered in order to meet the required system availability, with due economic justification. Scheduled outages shall be evaluated at 2 mills/kw-hr; unscheduled outages shall be evaluated at 2 mills/kw-hr.

1.3.4.3 The system unavailability, due to shutdown for scheduled maintenance, shall be not greater than 3.8%. Aggregated unscheduled shutdowns are estimated to produce an unavailability not greater than 1.0%.

1.3.4.4 Maintenance facilities, and the arrangement and design of the system components, shall be such as to minimize downtime so as not to exceed the above unavailability.

1.3.5 Operational Requirements

1.3.5.1 The energy conversion system shall provide for local and remote operation. Remote operation shall be from a central control room for both the nuclear and the steam-electric generation plants.

1.3.5.2 Operation of the system shall be according to the performance requirements in Section 1.3.1.

1.3.5.3 The system shall permit partial load operation, during periods of forced or planned outage of certain auxiliary equipment.

1.3.5.4 The system shall operate at an expected load factor of 90% of rated capability after the "break-in" period. This represents an energy output of 7.88×10^6 Mw-hr per year; with an overall plant availability of 87% the energy output per year shall be 6.85×10^6 Mw-hr.

1.3.6 Safety Requirements

1.3.6.1 The design of the system shall include the necessary features to provide safe performance during all the normal and emergency operating conditions listed in Section 1.3.1, during the 30 year expected plant life. Safe performance shall consist of the prevention of injury to operating and maintenance personnel, the prevention of damage to the system components, and the prevention of harmful environment pollution.

1.3.6.2 The safety features shall include due consideration, but not be limited, by the following design practices.

- 1) Analytical investigations to determine the maximum pressures, temperatures, flows and loadings experienced during steady state and transient plant operation. The system and component design conditions, and maximum number of cycles during the plant life, shall be selected accordingly.
- 2) Selection of materials, design factors of safety, welding procedures, heat treatment and testing procedures according to the applicable industrial Codes and Standards.

- 3) Determination of the pressure relief and dump system flow capacities, and setpoints, to prevent exceeding the component design conditions, thus ensuring the structural integrity of the system.
- 4) Determination of the ground conditions and seismic loads at the plant site, for design of the equipment and supports.

1.3.6.3 The energy conversion system shall provide for audio and visual alarms to indicate that an operating design parameter is being exceeded, and for the manual and/or automatic means to correct or isolate the fault and for the initiation of load reduction or shutdown.

1.3.6.4 The system shall provide for periodic on-line testing of those components which are used as protective devices, without affecting plant availability.

1.3.6.5 Provisions shall be made during plant construction, for regular inspection at point of manufacture, or at the plant site, of the architectural, electrical, mechanical and structural equipment and materials; in order to ensure their conformity with the applicable industrial Codes, Standards, and local regulations.

1.3.7 Design Requirements

1.3.7.1 Applicable Codes and Standards

ASME	Boiler and Pressure Vessel Code Section II - Material Specifications Section VIII - Unfired Pressure Vessels Section IX - Welding Qualifications Power Test Codes
USAS B.31.1 B.16.5	USA Standards Institute, Code for Pressure Piping
ASA C1	National Electric Code
NEMA	National Electrical Manufacturers Association.
AWS	American Welding Society
ISA	Instrument Society of America
IEEE	Institute of Electrical and Electronic Engineers

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NFPA	National Fire Protection Association
HEI	Heat Exchange Institute Standards
EET	Edison Electric Institute
HI	Hydraulic Institute
NSC	National Safety Council

1.3.7.2 The structural and material design requirements of the system shall be according to the performance, environmental, availability, maintenance, operational and safety requirements described in the previous sections, and in agreement with the applicable Codes, Standards and local regulations.

1.3.7.3 In the design of the main equipment foundations, the recommendations of the manufacturer shall be followed for loading, deflection limitations, and allowable unit stresses. The manufacturer's recommendations shall also be followed for the thrust and moment limitations at the pipe connections.

1.3.7.4 The system piping design shall be based on engineering and economic considerations: some of which are as follows:

- 1) The ASME boiler and pressure vessel code and the USA Standards Institute code for pressure piping B31.1.0 shall be followed for material, wall thickness and general requirements.
- 2) The overall system shall be designed and the equipment located with the piping layout as one of the major considerations.
- 3) Composite piping layouts shall be made during the early design phase.
- 4) Stress analysis of the main piping systems shall be made using electronic computers to ensure design within the allowable limits, and to obtain minimum piping usage.
- 5) Studies shall be made to establish the most economical fluid velocities, thus establishing the most economical pipe diameters.
- 6) Where possible, butt-welded joints shall be used in preference to flanged fittings, and socket-welded fittings in preference to screwed fittings.

1.3.7.5 Alternate design features, such as bypasses and the use of spares, shall be used whenever justified by engineering, operational and economic considerations.

1.3.7.6 The system shall provide feedwater of the highest quality to the once-through steam generators and feedwater equipment. The environmental requirements in Section 1.3.2.2. and the system design, shall be expected to result in the following feedwater quality.

- 1) Maximum iron concentration, 0.01 ppm,
- 2) Maximum copper concentration, 0.002 ppm,
- 3) Maximum water conductance, 0.5 micro-mho,
- 4) Maximum dissolved solids, 0.05 ppm,
- 5) Maximum dissolved oxygen, 0.007 ppm, and
- 6) Maximum silica content, 0.01 ppm.

1.3.8 Interfaces

1.3.8.1 The energy conversion system shall interface with the following main plant systems.

- 1) Heat transfer system at steam generators,
- 2) Load control system,
- 3) Protective system,
- 4) Auxiliary systems, steam generator isolation and recovery,
- 5) Site, structures and services, and
- 6) Electrical main and auxiliary power system.
- 7) Startup, Shutdown and Decay Heat Removal System

2.0 DESCRIPTION OF SYSTEM

A summary description of the energy conversion system has been given in Section 1.2. The following is a more detailed, but by no means complete, description of the system and its main components. The following heat balance and flow diagrams, included in Appendix 6.2, show the main features of the energy conversion system.

100% Flow Heat Balance	Figure 6.2.1
Main Steam and Reheat Steam	Figure 6.2.2
Condensate and Feedwater	Figure 6.2.3

2.1 Turbine-Generator Unit

The unit is rated 1240 Mva at the generator terminals, with inlet steam conditions of 2400 psig, 1000°F, reheat to 1000°F and exhaust to 1.5 in. condenser pressure. It is of the tandem compound type operating at 3600 rpm.

The 3600 rpm turbine consists of a double-flow, high-pressure element and a double-flow intermediate pressure (reheat) section. Steam is admitted to the high-pressure section through four stop and four control valves. Exhaust steam from this section at 575 psia, 630°F is extracted for feedwater heating in the No. 7 heater; the rest is reheated in the sodium-to-steam re-heater and returned to the intermediate-pressure element through four combination stop-intercept valves at 515 psia, 1000°F. Extraction steam for the No. 6 heater is taken from the intermediate pressure casing. Exhaust steam from this turbine is used for feedwater heating in the No. 5 heater, and for operating the two boiler feed pump turbines. Cross-over pipes carry the rest of the exhaust steam to the low-pressure sections.

The low-pressure turbine consists of three identical double-flow sections which exhaust downward through their respective connections into the single shell condenser. Extraction steam for the deaerator and the Nos. 1, 2 and 3 feedwater heaters is taken from the low-pressure casings. The last-stage buckets are 33.5 in. long and provide a total exhaust annulus area of 396.6 sq. ft. The exhaust moisture does not exceed 10% under the rated normal operating conditions. Each low-pressure exhaust casing is provided with water spray nozzles, and control valves, to prevent exceeding the maximum recommended exhaust hood temperatures during startups, shutdowns and abnormal operating conditions.

The 3600 rpm generator is rated 1240 Mva at 0.85 pf, 60 psig hydrogen pressure, 3-phase, 60 cps, 22 kv. The generator is designed for a short circuit ratio of 0.58.

The generator is totally enclosed to operate with hydrogen gas as the principal cooling medium. This system including fans and coolers is contained within the generator. The stator windings, which have hollow copper conductors, are cooled by purified, de-ionized water which circulates within the coils. Heat exchangers, located outside the generators, are used to reject the cooling system heat to the condensate water.

DC power for excitation is supplied by a static exciter system.

Design data for the turbine generator are given in Table 6.1.1.

2.1.1 Control System

The turbine-generator is provided with an electrohydraulic control system using electronic computing devices and a fire resistant hydraulic system to actuate the steam valves. The control system performs the following functions.

- 1) Speed control,
- 2) Full-arc to partial-arc transfer,
- 3) Load control,
- 4) Initial pressure limit,
- 5) Stop valve flow control,
- 6) Control valve flow control,
- 7) Intercept valve flow control,
- 8) Turbine bypass valve control, and
- 9) Overspeed limit.

2.1.2 Auxiliary Systems

The turbine-generator is provided with the following auxiliary systems.

- 1) Bearing lubrication oil,
- 2) Control fluid,
- 3) High and low speed shaft turning gear,
- 4) Turbine steam seal and leakoff,
- 5) Turbine steam seal regulator,
- 6) Generator hydrogen cooling and sealing, and
- 7) Generator conductor cooling water treatment and control.

2.2 Boiler Feed Pump Drives

Two steam turbines are used to drive the two half-capacity boiler feed pumps. The turbines are provided with dual inlets, which admit low or high-pressure steam, according to the load requirements. At plant loads exceeding approximately 60% of rated, the turbines receive low-pressure superheated steam from the intermediate pressure turbine exhaust. High-pressure steam at 2400 psig and 1000°F is used below 40% of rated load, hot startups, and overload conditions. Dual steam admission occurs, approximately, between 40 and 60% of rated load. Startup may also be obtained using steam from the auxiliary steam generators. Each turbine develops 17,650 shaft horsepower at 5500 rpm, with 190 psig, 765°F inlet steam and 2 in. Hg condenser pressure; steam conditions which correspond to the 100% main turbine flow point. Under these operating conditions the required steam flow per turbine is 126,250 lb/hr.

Each unit is provided with turning gear, control system, lubricating and control oil systems for both the turbine and the pump, and monitoring equipment. The sealing steam and drain systems are connected and controlled from the main turbine gland seal and exhauster system.

2.3 Main Condenser

The main condenser is a horizontal, single shell unit with a single circulating water pass and divided epoxy coated steel water boxes. The unit is solidly mounted on foundations, with its axis parallel to the turbine

generator shaft. Expansion movement between the condenser and the low-pressure turbine casings is provided by belt-type expansion joints. The No. 1 low-pressure feedwater heaters are mounted in the two condenser steam hoods. The deaerating-type condenser hotwell provides sufficient capacity to allow approximately one minute of operation at rated load.

The condenser design provides for receiving the condensate from the boiler feed pump turbines, the low-pressure heater drains, emergency drains, condensate pump recirculation, condensate expansion and demineralized water supply. It also provides for receiving the steam from the superheater bypass; main steam turbine bypass, and reheat steam turbine bypass; up to approximately 30% of rated flow, during plant startup, turbine trip and sudden load rejections.

A vacuum system is provided for evacuation of the water side of the condenser shell, in order to reduce the startup total dynamic head requirements of the circulating water pumps, prior to establishing full flow syphon. The design for the condenser and its auxiliaries is based on a fresh cooling water supply and the materials have been selected accordingly.

Design data for the main condenser are given in Table 6.1.2.

2.3.1 Circulating Water Pumps

Two motor driven, vertical, mixed flow pumps are used to circulate 310,000 gpm of 55°F cooling water through the condenser. The pumps are designed to produce a total dynamic head of 25 ft. at 270 rpm and 87% efficiency. The pump casings are of fabricated steel construction, the rotors have chrome steel shafts and bronze impellers. The motors are rated 1250 shaft horsepower each, at 270 rpm and 4160 volts. Weather protected, NEMA Type II enclosures are used.

2.3.2 Hotwell Pumps

Four motor driven, vertical, centrifugal pumps are used to circulate the condensate from the hotwell to the deaerating heater. One pump is used as a spare. Each pump delivers 3640 gpm of 92°F water, at 800 ft. total dynamic head, with an efficiency of 85% at 1180 rpm. The pump casings are of cast iron, the rotors have stainless steel shafts and bronze impellers.

The motors are rated 900 shaft horsepower each, at 1180 rpm and 4160 volts. Drip-proof enclosures and Class B insulation is used.

2.3.3 Air Removal

Air and noncondensable vapors are removed from the condenser by three positive displacement pumps, one used as a spare. Each pump has a holding capacity of 15 scfm at 1 in. Hg, and is driven by a 125 shaft horsepower motor operating at 575 rpm and 460 volts. The pumps are of cast iron construction with stainless steel shaft.

2.4 Auxiliary Condensers

Two auxiliary condensers are provided to receive the exhaust from the two boiler feed pump turbine drives. Each condenser is a horizontal unit, with double circulating water passes, and divided epoxy coated steel water boxes.

Each unit contains 1870 tubes with a total surface of 8,060 sq. ft.; the tubes are 7/8 in. diameter, 18 BWG, with an effective length of 20 ft. The condenser materials are the same as those used in the main unit. This design permits operation at a condenser pressure of 2 in. Hg, with a flow of 9,200 gpm of 55°F circulating water, and at the conditions existing at the 100% main turbine flow point.

The air removal and cooling water systems are common with the main condenser. Condensate from the auxiliary condensers is pumped to the main condenser hotwell by the auxiliary condensate pumps.

2.4.1 Auxiliary Hotwell Pumps

Three motor driven, vertical, centrifugal pumps are used to circulate the condensate from the two hotwells to the main condenser hotwell. One pump is used as a spare. Each pump delivers 260 gpm of 101°F water, at 50 ft. total dynamic head, with an efficiency of 70% at 1750 rpm. The pump casings are of cast iron, the rotors have stainless steel shafts and bronze impellers.

The motors are rated 5 shaft horsepower each, at 1180 rpm and 480 volts. Drip-proof enclosures and Class B insulation is used.

2.5 Condensate and Feedwater

The thermal cycle is based on seven stages of feedwater heating; three low-pressure stages and three high-pressure stages are located before and after the deaerating heater, respectively. The condensate and feedwater stream is divided into two parallel trains of heaters as shown in Figure 6.2.3. Bypasses and interconnecting headers are provided to permit operation during startup and heater outage conditions. The high-pressure heater drains are cascaded to the deaerator, the low-pressure drains are cascaded to the main condenser.

Full flow filtration and demineralization of the condensate is used to provide feedwater of the highest quality to the once-through steam generators. Heater tube leakage is minimized by using a fully welded design. Copper and zinc pickup problems are mitigated by the use of ferrous materials in the feedwater heating equipment.

Two half-capacity turbine driven pumps are used to circulate the feedwater from the deaerating heaters to the three steam generators. The two pumps are paralleled through suction and discharge headers to minimize unbalance between the heater trains. The turbine drives permit flexibility during startup and normal operation. Feedwater flow control to the steam generators may be accomplished in the normal load range by a combination of individual valve control and pump speed regulation.

Design data for the condensate and feedwater system are given in Tables 6.1.3, 6.1.4, and 6.1.5.

2.5.1 Closed Feedwater Heaters

The six low-pressure heaters are arranged in two parallel trains of three; the same arrangement is used for the six high-pressure units. All heaters, except the deaerator, are closed U-tube, two-pass, removable shell units sized for one-half capacity each. The heaters are made with carbon steel shells and tubing, with 100% tube welding to the tube sheets. A nitrogen blanketing system has been provided to prevent heater corrosion during outage periods.

The heaters are numbered in the direction of feedwater flow. The lowest pressure heaters, No. 1, are located in the condenser steam hoods. The total pressure drop through the three high-pressure heaters, at 100% flow, is approximately 40 psi; that through the three low-pressure heaters is approximately 55 psi. The tube side velocity at the design flow is approximately 8 fps. The heaters are designed with 10°F drain coolers.

The Design data for the closed feedwater heaters are given in Tables 6.1.4 and 6.1.5.

2.5.2 Deaerating Feedwater Heaters

The two deaerating heaters consist of horizontal shells for heating and deaerating purposes mounted upon horizontal storage tanks. The storage tanks have a capacity to supply approximately four minutes of feedwater at rated load. Four downcomer pipes link the heater and storage tanks. The deaerators are located on top of the steam generator building.

The deaerator heaters may be pegged from the No. 6 heater at low loads or from the flash tank during startups and shutdowns.

Design data for the deaerating heaters are given in Table 6.1.6.

2.5.3 Boiler Feed Pumps

The two, half-capacity, turbine-driven pumps are barrel type five-stage units. Each has a capacity of 3.427×10^6 lb/hr (7350 gpm) at a total dynamic head of 8540 ft. when operating at 5500 rpm. The net positive suction head is approximately 125 ft. with 300°F water from the deaerator storage tanks. The pump efficiency at the preceding operating conditions is 84%.

The pump casings are of carbon steel, with chrome steel inner casing, impeller and shaft. Each pump is supplied with adequate lubricating oil from the steam turbine oil system. An automatic recirculation control system is used to protect the pumps under startup, low-load or no-flow conditions.

2.6 Water Treating Plant

The water treating facilities consist of two distinct demineralizing plants. One plant is for feedwater makeup to the main turbine-steam generator cycle, and is designed for approximately 3% of the rated turbine throttle flow. The second plant performs the 100% condensate flow polishing function.

As shown in Figure 6.2.3, the makeup system consists of three strings of single-bed cation exchangers, followed by three strings of single-bed anion exchangers, with a final polish in three parallel mixed-bed units. Capacity of this makeup system can be obtained with the parallel operation of two of the three strings, while the third is in the regeneration cycle. The required sulfuric acid, sodium hydroxide and storage tanks are provided.

The condensate polishing system consists of a parallel bank of prefilters followed by the mixed bed demineralizers and a parallel bank of after strainers.

Design data for the water treating plant are given in Table 6.1.7.

2.7 Main Steam and Reheat Steam

Figure 6.2.2 shows the flow diagram of the main steam and reheat steam systems. These systems contain the piping which interconnects the steam generators, the main turbine and the boiler feed pump drives. Pipe sizes were determined using steam velocities ranging from 170 to 220 ft/sec at the 100% steam flow condition; thickness was determined according to the ASA Code for Pressure Piping.

Table 6.1.8 shows the design conditions and materials used in the main steam and reheat systems.

2.8 Extraction Steam

Steam from seven locations in the turbine is piped to the shells of the two parallel strings of feedwater heaters. Exhaust steam from the high-pressure section is extracted for feedwater heating in the No. 7 heaters. Extraction steam for the No. 6 heaters is taken from the intermediate pressure casing. Exhaust steam from this section is used for feedwater heating in the No. 5 heaters. Extraction steam for the deaerators and the Nos. 1, 2 and 3 feedwater heaters is taken from the low-pressure casings.

Reverse flow valves are provided in all but the lowest pressure extraction lines, to prevent turbine overspeed from backflow of flashed steam from the heaters after a turbine trip. These valves are closed automatically upon a signal from the turbine trip circuit. The lowest pressure extraction lines are located entirely in the two condenser steam hoods and do not contain nonreturn valves.

The 100% heat balance in Figure 6.2.1 shows the extraction flows and steam conditions. The extraction lines are sized to give steam velocities ranging from 150 to 200 ft/sec in order to limit the line pressure drops to the range from 5 to 9% of the turbine shell pressure. The material used in ASTM A106 Grade B, except in the No. 7 extraction where ASTM A335 Grade P11 is used.

2.9 Condenser Circulating Water

The condenser circulating water system is designed for a fresh water flow of 310,000 gpm. Two circulating water pumps are used as described in Section 2.3.1. It is intended that the circulating water screen house and intake structure be entirely of reinforced concrete. The screens, circulating water pumps, fire pumps and wash pumps are for wet pit installation.

3.0 OPERATION

Coordinated operation of the reactor, coolant pumps, intermediate heat exchangers, steam generators and of the energy conversion system during startup, normal operation in the load range, and during normal and emergency shutdowns remain to be investigated. Steady state and transient investigations are required at rated and partial loads.

4.0 SAFETY PRECAUTIONS

4.1 The control and protective systems of the energy conversion system are closely integrated with those of the reactor and heat transfer segments, to prevent exceeding the structural limitations of the components during normal and emergency operation.

4.2 The system and its components are designed according to the applicable industrial codes and standards, and consistent with practices used in the Electric Utility Industry.

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4.3 Additional design provisions for safety may result, after completion of the steady state and transient investigations recommended in Section 3.

5.0 MAINTENANCE

5.1 The arrangement and design of the system is intended to minimize the time required for maintenance.

5.2 The equipment design features are intended to provide for maximum service and optimum maintenance time and cost.

5.3 The facilities provided at the plant site are expected to meet the majority of the system maintenance requirements. Routine maintenance is expected to be performed by plant personnel. Overhauls and other major maintenance is expected to be performed by supplemental crews.

5.4 The required maintenance procedures are outlined in the component and subsystem descriptions, and are performed according to the manufacturer's recommendations.

6.0 APPENDIX

6.1 Tables

- 6.1.1 Turbine generator design data.
- 6.1.2 Main condenser design data.
- 6.1.3 Condensate and feedwater design data.
- 6.1.4 Low pressure feedwater heaters design data.
- 6.1.5 High pressure feedwater heaters design data.
- 6.1.6 Deaerating feedwater heaters design data.
- 6.1.7 Water treating plant design data.
- 6.1.8 Main steam and reheat steam design data.

6.2 Drawings

<u>Title</u>	<u>Drawing No.</u>	<u>Fig. No.</u>
100% Flow Heat Balance	N0960009	6.2.1
Main Steam and Reheat Steam	N0960010	6.2.2
Condensate and Feedwater	N0960011	6.2.3

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6.3 References

1. Engineering Study for Large Steam-Electric Generating Units,
United Engineers and Constructors Inc., May 1, 1966.

6.0 APPENDIX**6.1 Tables****TABLE 6.1.1****TURBINE-GENERATOR DESIGN DATA****100% STEAM FLOW**

Turbine Type	tc-6 flow, 33.5 LSH*
Turbine Rating, Mw (0% makeup)	1054
Turbine Speed, rpm	3600
Inlet Pressure, psia	2415
Inlet Temperature, °F	1000
Inlet Enthalpy, Btu/lb	1460.4
Inlet Steam Flow, lb/hr	6.854×10^6
Reheat Pressure, psia (IP section inlet)	515
Reheat Temperature, °F (IP section inlet)	1000
Reheat Flow, lb/hr (IP section inlet)	6.402×10^6
Condenser Pressure, in. Hg	1.5
Turbine Exhaust Enthalpy, Btu/lb (UEEP)**	1029.6
Turbine Exhaust Loss, Btu/lb	28.6
High-Pressure Casing Extractions	None
Intermediate Pressure Casing Extractions	2
Low-Pressure Casing Extractions	4
Stop and Control Valves	4
Stop and Intercept Valves	4
Generator Rating Mva	1240
Generator Terminal Voltage, Kv	22
Generator Power Factor	0.85
Generator Hydrogen Pressure, psia	60
Generator Short Circuit Ratio	0.58
Generator Frequency, cps	60
Generator Phases	3
Generator Excitation System	Static
Generator Cooling	H ₂ -Stator Core and Rotor Water - Stator Windings
Final Feedwater Temperature, °F	478
Boiler Feed Pump Drives	Condensing Turbines
Heat Balance Diagram	Figure 6.2.1

- * Last stage bucket.
- ** Used energy end point.

TABLE 6.1.2

MAIN CONDENSER DESIGN DATA

100% STEAM FLOW

Number of Shells	1
Circulating Water Passes	1
Number of Water Boxes	2
Condensing Surface, sq. ft.	483,000
Tube, OD in./wall BWG	7/8/18
Tube Effective Length, ft.	75
Tube Material	Admiralty
Tube Sheet Material	Muntz
Tube Material, Air Cooler Section, Steam Impinge Areas	304 SS
Circulating Water Flow, gpm	291,000
Circulating Water Velocity, ft./sec.	7
Circulating Water Inlet Temperature, °F	55
Tube Cleanliness Factor	0.85
Condenser Pressure, in. Hg	1.5
Hotwell Design	Deaerating Reheating Compartmental
Number of Tubes	29,800

TABLE 6.1.3

CONDENSATE AND FEEDWATER DESIGN DATA

100% FEEDWATER FLOW

Condensate Coolers (low vacuum operation)	2
Condensate Filtration, Demineralization	Full Flow
Generator H ₂ Coolers	8
Generator Stator Winding Coolers	2
Turbine Gland Steam Condenser	1
Low-Pressure Feedwater Heaters (2 trains)	6
Deaerator Heaters (parallel)	2
Deaerator Storage Tanks (≈ 4 min supply)	2
Boiler Feed Pumps (half-capacity)	2
High-Pressure Feedwater Heaters (2 trains)	6
Feedwater Control Valves (2 each evaporator)	6
Condensate Flow, lb/hr	5.429×10^6
Deaerator Pressure, psia	65
Feedwater Pump Discharge Pressure, psig	3450
Total Feedwater Flow, lb/hr	6.854×10^6
Final Feedwater Temperature, °F	478

TABLE 6.1.4

LOW-PRESSURE FEEDWATER HEATERS DESIGN DATA

100% FEEDWATER FLOW

Heater Number	1	2	3
Number of Units	2	2	2
Terminal Difference			
Feedwater Outlet, °F	4	5	3
Drain Cooling, °F	10	10	10
Feedwater Flow, lb/hr (each)	2.715×10^{-6}	2.715×10^6	2.715×10^6
Feedwater Inlet Temperature, °F	93.5	160.7	192.8
Feedwater Inlet Enthalpy, Btu/lb	60.7	128.6	160.8
Feedwater Outlet Temperature, °F	160.7	192.8	241.4
Feedwater Outlet Enthalpy, Btu/lb	128.6	160.8	209.7
Feedwater Pressure, psig	300	300	300
Feedwater Pressure Drop, psi	~ 23.8	~ 19.2	~ 11.1
Extraction Steam Flow, lb/hr (each)	161,600	81,960	126,470
Extraction Steam Enthalpy, Btu/lb	1111.5	1155.5	1220.5
Extraction Steam Pressure, psia	5.3	11.0	27.0
Drain Flow In, lb/hr (each)	208,430	126,470	--
Drain Flow In, Temperature, °F	170.7	202.8	--
Drain Flow In, Enthalpy, Btu/lb	138.6	170.8	--
Heater Drain Flow, lb/hr (each)	370,030	208,430	126,470
Heater Drain Flow, Temperature, °F	103.5	170.7	202.8
Heater Drain Flow, Enthalpy, Btu/lb	71.5	138.6	170.8
Heat Transfer, Btu/hr (each)	182.2×10^6	87.1×10^6	132.8×10^6
Tube Surface, sq. ft. (each)	15,700	12,400	7,105
Tube Material	Carbon Steel	Carbon Steel	Carbon Steel
Shell Diameter, in. (approximate)	55	53	50
Shell Length, ft. (approximate)	53	43	27
Tube Side Velocity, ft./sec.	~ 7.6	~ 7.8	~ 7.8

TABLE 6.1.5
HIGH-PRESSURE FEEDWATER HEATERS DESIGN DATA

100% FEEDWATER FLOW

Heater Number	5	6	7
Number of Units	2	2	2
Terminal Difference			
Feedwater Outlet, °F	0	0	0
Drain Cooling, °F	10	10	10
Feedwater Flow, lb/hr (each)	3.427×10^6	3.427×10^6	3.427×10^6
Feedwater Inlet Temperature, °F	340.7	382.6	423.3
Feedwater Inlet Enthalpy, Btu/lb	280.6	360.6	403.6
Feedwater Outlet Temperature, °F	382.6	423.3	478.0
Feedwater Outlet Enthalpy, Btu/lb	360.6	403.6	463.0
Feedwater Pressure, psia	3450	3450	3450
Feedwater Pressure Drop, psi	~ 15.0	~ 10.7	~ 14.6
Extraction Steam Flow, lb/hr (each)	218,445	124,505	226,085
Extraction Steam Enthalpy, Btu/lb	1408.1	1469.5	1311.9
Extraction Steam Pressure, psia	202	320	555
Drain Flow In, lb/hr (each)	350,590	226,085	-
Drain Flow In, Temperature, °F	392.6	433.3	-
Drain Flow In, Enthalpy, Btu/lb	366.9	411.5	-
Heater Drain Flow, lb/hr (each)	569,035	350,590	226,085
Heater Drain Flow, Temperature, °F	314.7	392.6	433.3
Heater Drain Flow, Enthalpy, Btu/lb	284.8	366.9	411.5
Heat Transfer, Btu/hr	274.2×10^6	147.4×10^6	203.6×10^6
Tube Surface, sq ft (each)	15,850	11,250	15,800
Tube Material	Carbon Steel	Carbon Steel	Carbon Steel
Shell Diameter, in.	49	36	49
Shell Length, ft (approximate)	54	55	53
Tube Side Velocity, ft/sec	~ 7.5	~ 7.5	~ 7.3

TABLE 6.1.6

DEAERATING FEEDWATER HEATER DESIGN DATA

100% FEEDWATER FLOW

Heater Number	4
Number of Units	2
Inlet Feedwater Flow, lb/hr (each)	2.715×10^6
Inlet Feedwater Temperature, °F	241.4
Inlet Feedwater Enthalpy, Btu/lb	209.7
Outlet Feedwater Flow, lb/hr (each)	3.427×10^6
Outlet Feedwater Temperature, °F	298
Outlet Feedwater Enthalpy, Btu/lb	267.5
Extraction Steam Flow, lb/hr (each)	143,205
Extraction Steam Enthalpy, Btu/lb	1294.5
Extraction Steam Pressure, psia	65
Drain Flow In, lb/hr (each)	569,035
Drain Flow In, Temperature, °F	314.7
Drain Flow In, Enthalpy, Btu/lb	284.8
Heat Transfer, Btu/hr (each)	313.8×10^6
Diameter, ft	9.5
Straight Side Length, ft	27
Spray Nozzle Material	Stainless Steel
Tray Material	Stainless Steel
Vent Condenser	Internal
Pressure Range Rating, psig	Vacuum to 100
Maximum Residual Oxygen, cc/liter	0.005
Maximum Residual Carbon Dioxide	0
Storage Tank Diameter, ft	12
Storage Tank Straight Side Length, ft	40
Tank Shell Material	Carbon Steel

TABLE 6.1.7

WATER TREATING PLANT DESIGN DATA

3% MAKE-UP

100% CONDENSATE FLOW

Make-Up Water Demineralizer

Capacity, gpm	450
Number of Strings (cation, anion, mixed)	3
Cation Unit Size, ft (diameter/height)	8/10
Anion Unit Size, ft (diameter/height)	7/10
Mixed Bed Unit Size, ft (diameter/height)	4/12
Shell Design Pressure, psig (diameter/height)	100
Shell Lining Thickness, in. (rubber)	3/16
Degassifier Size, ft (diameter/height)	8/15
Degassifier Packing	Raschig Rings
Degassifier Blowers	2
Acid Storage Tank Capacity, gal	10,000
Caustic Storage Tank Capacity, gal	10,000
Maximum Water Conductance, micromho	0.5
Maximum Silica Content, ppm	0.01

Condensate Polishing Demineralizers

Type	Mixed Bed
Number	16
Capacity, gpm (each)	935
Normal Flow Rate, gpm/ft ²	30
Shell Design Pressure, psig	300
Internal Material (rubber lined)	Stainless Steel
Maximum Total Dissolved Solids, ppm	0.05
Maximum Dissolved Oxygen, ppm	0.007
Maximum Silica, ppm	0.01
Maximum Iron, ppm	0.01
Maximum Copper, ppm	0.002

Condensate Storage Tank (active)

Number	1
Capacity, gal	800,000
Diameter, ft	57
Height, ft	42
Material (epoxy coated)	Steel
Location	Ground Level

Polished Water Storage Tank

Number	1
Capacity, gal	300,000
Diameter, ft	41
Height, ft	30
Material (epoxy coated)	Steel
Location	Ground Level

TABLE 6.1.7 (Continued)

Elevated Demineralized Water Storage Tank

Number	1
Capacity, gal	115,000
Diameter, ft	28
Height, ft	11.5
Material (epoxy coated)	Steel
Location	Tower

TABLE 6.1.8

MAIN STEAM AND REHEAT STEAM PIPING DESIGN DATA

100% STEAM FLOW

<u>Service</u>	<u>°F</u>	<u>Psig</u>	<u>ASTM</u>	<u>Grade</u>
Main Steam	1000	2575	A369	FP22
Hot Reheat	1000	675	A155	1 - 1-1/4 Cr
Cold Reheat	650	675	A155	KC70
Low-Pressure to BFP Turbines	770	250	A106	B
High-Pressure to BFP Turbines	1000	2575	A335	P22

REVISIONS			
NO.	DATE	DESCRIPTION	BY
A		SEE REVISION RECORD

MAIN STREAM
 FEEDWATER
 SODIUM
 P PRESSURE PSIA
 T TEMPERATURE °F
 A ENTALPHY BTU/LB
 LB FLOW LB/HR
 TD TERMINAL DIFFERENCE °F
 DC DESIGN COOLING °F

ATOMIC INTERNATIONAL			
1000 MWe LMFR FOLICAN STUDY			
HEAT BALANCE DIAGRAM			
DATE	NO. 1000	REV. 1	NO. 1000
NO. 1000	NO. 1000	NO. 1000	NO. 1000

Fig. 6.2.1

IN 1000-30009
 IN 1000-30009
 SDD-096-40-000

REVISION		DATE	APPROVED
A	SEE REVISION RECORD	7-16-82

ATOMICS INTERNATIONAL
 A DIVISION OF WESTINGHOUSE ELECTRIC CORP.
 3600 MARKET AVENUE, PITTSBURGH, PA. 15204

1000 MWe LMFBR FOLLOW-ON STUDY
 MAIN STEAM & REHEAT STEAM

DESIGNED BY	DATE	SCALE
ENGINEERED BY	DATE	SCALE
CHECKED BY	DATE	SCALE

PROJECT NO. N09630010

Fig. 6.2.2

Fig. 6.2.3

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

NUMBER

SDD-096-50-000

REV
LTR

B

DOCUMENT TYPE

System Design Description

PROGRAM TITLE

ANL 1000 Mwe Follow-On Study

KEY NOUNS

Control & Instrumentation

DOCUMENT TITLE

Control and Instrument Segment Conceptual
System Design Description - 1000 Mwe LMFBR
Follow-on Study

ORIGINAL

ISSUE DATE July 15, 1968

GO NO.

7692

S/A NO.

13500

PAGE 1 OF

TOTAL PAGES 56

REL. DATE 10-22-68

ABSTRACT

The requirements and the description of the
Systems which make up the control and instrument
segment of the 1000 Mwe LMFBR are contained in
this System Design Description.

PREPARED BY

DATE

DEPT/GRP NO.

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731-151

APPROVALS

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CONTROL AND INSTRUMENT SEGMENT CONCEPTUAL SYSTEM DESIGN DESCRIPTION

1.0 INTRODUCTION

The Plant Control and Instrumentation Segment consists of the following systems:

- a. Plant Control System
- b. Plant Protection System
- c. Core Flux Measurement System
- d. Reactor Temperature Measurement System
- e. Reactor Cover Gas Monitor System
- f. Hydrogen Detector System
- g. Sodium Flow Measurement System
- h. Reactor Sodium Level Measurement System
- i. Delayed Neutron Monitor System
- j. Radiation Monitoring System

1.1 Functions/Functional Requirements

1.1.1 Plant Control System

For purposes of conceptual design, the plant control system comprises two functional subsystems and all the panels and other ancillaries to implement their operation with that of the plant. These two subsystems are called:

The Heat Transfer Control System
The Reactor Control System

The primary function of these two subsystems and ancillaries is to assist the plant operator in controlling the required supply of steam to the turbine-generator and in maintaining the steam supply system within safe operating limits during startup, normal operation, and normal and emergency shutdown. The important requirements and limits that define operation of the system are further described in 1.3 below. However, the functional requirement that the system is to assist the operator means that information display is of major importance. It also means that provision must be made so that the operator can select varying degrees of automation (liberal use of hand control and auto-transfer stations).

It is a design requirement that the plant shall be operable with one circuit shut down. Under suitable temperature conditions it is assumed that a shutdown circuit may be restarted. Provision should be made so that the operator can isolate one circuit without requiring plant shutdown to do so.

Considerable assistance to the operator will be required during startup and shutdown. As a minimum, a centralized control room, and a status display of items related to startup and shutdown conveniently arranged in the control room are required. Considerable instrumentation will be supplied by others with the turbine-generator. Integration of these with the plant control system will possibly be made by even a third party (the A&E or the customer). However, other than the possibility of especially terminating or transducing some instrument signals it is not clear how this would affect conceptual design. The foregoing is summarized as follows:

A central control room with a startup panel is required.

The extent and kind of automation for startup and shutdown will not be defined during conceptual design.

Since the steam generators are code vessels they will have a series of safety valves. However, these are difficult to control in reseating. Hence, it is assumed that power relief valves will be furnished and their controls incorporated in the plant control system. It is further assumed that the startup system will incorporate a throttle bypass of at least 15% (and preferably 25%) which also can be used for over-pressure control (and nominally would precede actuation of the power reliefs which would in turn precede operation of the safeties). In any event, the maximum pressure transient on the system appears to be full load rejection. As a design objective the plant control system should act, under turbine trip, to minimize plant pressure and temperature transients, and minimize time to recover load (assuming such load recovery is feasible). This is literally interpreted to mean that turbine trip should not cause reactor trip.

Desuperheating of main or reheat steam gives rise to performance penalties and should thus be avoided. Hence, it is a design objective that the plant control system should minimize desuperheating of the main or reheat steam. (It is assumed that reheat steam flow is not regulated and will nominally be a function of main steam flow).

Since sodium is an excellent heat transfer medium it is possible to transfer heat very quickly to structures. Hence, the rate-of-change of sodium temperatures must be held to the minimum consistent with load following requirements (and within limits established by safety requirements). A feasible but safe rate limit will be established for all control system setpoints. This requirement for temperature rate-of-change must be met under all operating conditions, including shutdown following a reactor trip.

1.1.2 Plant Protection System

The following 26 requirements have been identified for the plant protection system.

1.1.2.1 The protection system shall not unduly interfere with operator action, and the operator shall be able to initiate all actions of the protection system.

1.1.2.2 False alarm action (whether due to component malfunction or unnecessarily restrictive setpoints) is uneconomic and deleterious to the confidence of plant personnel in the protection system. Operating margins, highly reliable components, and protection against accidental actuation by the operator shall be provided.

1.1.2.3 In protected accident studies used to establish setpoints, no credit shall be taken for mitigating action by the operator or control system.

1.1.2.4 Startup and operating interlocks shall be provided so that important components cannot be operated (manually or by the control system) unless it is safe to do so.

1.1.2.5 There shall be two kinds of reactor poison rods. They are called shim-safety (control) rods and safety rods.

1.1.2.6 The shim-safety rods shall be used in normal control, and be capable of mitigating or terminating all credible accidents.

1.1.2.7 The safety rods shall be capable of overriding anomalies from normal operation. These specifically include errors in fuel loading and removal from the core of the most reactive poison rod.

1.1.2.8 Position of all rods shall be monitored, and the reactor shall not be allowed to operate unless safety rods have been withdrawn from the core.

1.1.2.9 There shall be two levels of protection for the reactor. In order of increasing severity they are called inhibit-alarm and reactor trip.

1.1.2.10 Inhibit-alarm is an action that prevents rod withdrawal, and alerts the operator that operating limits have been exceeded. It is a design objective that inhibit-alarm shall precede reactor trip.

1.1.2.11 Reactor trip is an action that rapidly inserts all rods into the core in order to prevent extensive fuel centerline melting. It shall cause inhibit-alarm and drive the safety rods into the core (rod drives actuated "in" even if rods are already in the core), and persist until the reactor is in its most shutdown condition.

1.1.2.12 Reliable and redundant instrumentation shall be provided, sufficient to monitor plant status at all times, including shutdown and fuel handling operations. This shall specifically include monitoring for criticality, high flux, pressure and temperature, and low flow and sodium level.

1.1.2.13 Measurements used for reactor trip shall be derived from at least three separate sensors. Trip from measured variables shall be in two of three coincidence.

1.1.2.14 Every Class I protected accident* requiring reactor trip shall use at least two kinds of signals in protecting against the incident. Specifically, there shall be a primary (first acting) action and at least one backup action derived from different measurement principles. (For example, if primary protection is from measured high flux, the backup could be from measured high temperature.) The primary action shall prevent economic damage; the backup action shall prevent extensive fuel clad melting.

1.1.2.15 No single instrument failure shall prevent proper action of the protection system. This specifically includes short or open failures of power or signal cable, switches or contacts, power supplies, or instrument modules. (A single failure includes also the consequences of such. For example, the failure of a relay coil is a single failure, and it may disrupt a dozen circuits depending on contacts.)

* Class I protected accidents are those accidents that can reasonably be postulated to occur once or more during the life of the plant.

1.1.2.16 It is a design objective that all instruments be continuously self-monitored. For those instruments or devices which are not self-monitored, a schedule for testing shall be specified.

1.1.2.17 It is a design objective that any instrument be testable or replaceable during operation. Removal or test of any instrument shall be annunciated and shall automatically place the circuit in action condition. (For example, if in a 2 of 3 coincidence, that function would then become one of two for the remaining two instruments).

1.1.2.18 Operation with one heat transfer circuit inoperative is permitted, and provision shall be made to automatically shut down one circuit for conditions shown in Figure 6.1.1.

1.1.2.19 Operation with two circuits inoperative is not allowed, and shall cause reactor trip.

1.1.2.20 With respect to electrical power, the protective system shall be divided into three distinct sections. This specifically includes control rod drives and other devices related to control rods. It is a design objective that power or signal interconnection between these sections be minimized.

1.1.2.21 Pressure relief will be provided by components normally not considered part of the plant protection system. This shall include provision for a throttle bypass (steam dump) of adequate size to safely start up and shut down the plant and to remove decay heat.

1.1.2.22 Power reliefs and safety valves shall be provided. It is a design objective that power reliefs and steam dump be sized so that safety valves will not normally operate on turbine trips from full load.

1.1.2.23 A system for pressure relief and containment of reaction products in the event of sodium water reaction in the steam generator shall be provided.

1.1.2.24 It is a design objective that connections between the control and protection systems shall be minimized. However, there are certain off-normal actions required by the protection system under reactor trip. These actions shall persist until the conditions causing them are cleared. Return of the control system to normal operation shall be prevented until the operator takes positive action. This specifically includes the reactor flux control system.

1.1.2.25 Special attention shall be paid to equipments whose maloperation or malfunction could possibly negate proper action by the protection system. This specifically includes the fuel handling machine, building gantry, position of the top shield, position of and all electrical connections to the rods or drives.

1.1.3 Core Flux Measurements

Nuclear instrumentation plays a major role in both control and protection of the plant. Hence, reliability requirements form an important part of the system design. Redundancy, i.e., three instruments of each type, are shown in the nuclear diagram, Section 6.0, Figure 6.1.3. In order to avoid blind startups it is necessary to be able to measure shutdown flux. Along with capability to measure to at least 150% of power, this leads to a requirement for about ten decades of coverage. In the design shown, two types of sensors, source range and power range, provide this coverage, and they overlap about a decade.

There are two requirements that have been identified that require measurement of flux within the reactor. The first of these is a performance (or mapping) requirement, and the second is a safety (or shutdown or loading) requirement. In addition to the requirements of economics and protection, both are believed required in order to obtain an operating license.

Core startup may require measurement of neutron flux at a number of points within the core, and possibly at several energy levels to verify physics calculations. (Spatial variations cannot be determined from ex-core instruments.)

The second requirement has to do with safety during initial loading experiments, and during subsequent refueling operations. In the first instance, control rods will be operated. Measurements of flux level and period are required for protection of the core (by causing reactor trips). In the case of refueling operation, rods will be inserted in the core (and the plant will be in trip condition). Interlocks on the fuel handling machine may be included to prevent insertion of a new fuel element into a nearly critical core. Since this second requirement has to do with the safety system, redundancy (at least triplication) in sensors is required.

1.1.4 Reactor Temperature Measurements

The requirement for reactor temperature measurements stems from the necessity to control plant operations, to enhance protection of the reactor system, and to monitor or verify design of the reactor structure.

1.1.4.1 These have been functionally interpreted as requirements for measurement of the following temperatures:

- Sodium out of fuel channels
- Sodium out of control rod channels
- Sodium out of selected blanket element channels
- Grip plate structure
- Top shield plugs
- Reactor vessel
- Foundation and insulation on reactor vessel

Note: The first three items are related to plant control and protection. The last four are concerned with preheating control and monitoring or design verification.

1.1.4.2 There are other temperatures in the reactor which might be measured. However, no requirements have been established. Specifically, the following measurements are not required.

1.1.4.2.1 Fuel or fuel cladding temperature (or pressure).

1.1.4.2.2 Control rod temperature (or any other measurement on the poison column).

1.1.5 Reactor Cover Gas Monitor

1.1.5.1 Monitoring of the cover gas serves as part of the cleanup control system in order to keep the activity level below the limit (to be specified) established in the plant Technical Specification. It also serves as a pre-alarm for the R/A Vent System. Finally, it functions as a failed element detection system. It should be noted that this system does not provide for failed element location.

1.1.5.2 Provision should be made for operation with vented fuel elements.

1.1.6 Hydrogen Detectors

The function of the hydrogen detectors on the steam generator modules is twofold; first, to indicate as quickly as possible the presence of a leak in a steam generator; and secondly, to identify the module in which the leak has occurred. The first requirement is to enable the plant operator to shut down a steam generator to minimize the extent of damage in the heat transfer system, and the second is to reduce the unavailability of the plant.

1.1.7 Sodium Flowmeters

The functional requirements are to provide reliable and reasonably accurate, within system tolerances, signals representing primary and secondary sodium flow for control and protection of the plant. Information on sodium flow will also be used to check plant performance. The transducer or sensor used to detect sodium flow will be of the type that assures the integrity of the sodium containment and provides maintenance free operation for the life of the plant. All environmental conditions in the vaults and interface requirements with instrument systems can be met by the EM flow-meter.

1.1.8 Reactor Sodium Level Measurements

A major safety problem is the loss of cooling for the core. This could be caused by a major rupture of the vessel or primary piping. Sodium level measurement appears to be one of the earliest possible means of detecting and automatically protecting against the consequences of such an event. Additionally, safety considerations require that the bottom of the top shield be immersed in sodium (also reducing accumulation of frost and amount of dissolved gas). On a slower time scale, level provides backup to leak and smoke detectors.

Level signals are used in fill and drain, and in reactor overflow and dissolved gas systems.

1.1.9 Delayed Neutron Monitoring

1.1.9.1 Fuel cladding is the first and most important containment barrier for fission products. Hence, it is important to know when cladding has failed. Continued operation with a major rupture or numerous small failures may lead to accumulation of fission products in primary system equipments (preventing access normally possible after Na 24 decay). For these reasons a system capable of detecting small amounts of fission products in the primary sodium is required. This system is in parallel with the cover gas monitor which also will detect some cladding failures. (However, the delayed neutron monitor has a much shorter time response, and will detect fuel and other solids). Present concept of operation of this system is that it is a guide for the operator and would not actuate any automatic shutdown. This system will be used only for monitoring of bulk sodium. A system for location of failed elements (FELS) is specifically excluded.

1.1.10 Radiation Monitoring System

The Radiation Monitoring System assists in protecting the general public and plant personnel against damaging radiation. Effluent release and direct or inhalation dose must be limited to amounts specified in 10 CFR 20. (For reference purposes, important limits from 10 CFR 20 and 100 are included in 1.3 below). However, certain of radiation protection functions are properly included under health physics, and are not included as part of the RD&M System (hand held and survey type equipments are typical examples of these).

Additional requirements of the system comprise criticality monitoring in fuel storage or handling areas, and assisting in the determination of other equipment failures or breaches of containment.

When effluent limits are exceeded, controls must be provided to prevent further release. Sufficient holdup should be provided so that minor incidents - to be expected during operation - should not unduly compromise operation of the plant. However, credit will be taken both for dilution factors and the fact that daily and yearly averaging is allowed for in some releases.

1.2 Summary Description

1.2.1 Plant Control System

The plant control system is organized into two major subsystems called the Heat Transfer Control System and the Reactor Control System. In general terms, the Heat Transfer Control System is in cascade with the turbine-generator in a direct energy balance and the Reactor Control System is in cascade with the Heat Transfer Control System through the use of total primary sodium flow as a main index for the Flux Controller. When a load increase is demanded by the machine, simultaneous increases are demanded (through PLD) in feedwater flow, secondary sodium flow, and primary sodium flow. When primary sodium flow increases there will be an increase in demand flux which will be met by pulling control rods in the reactor. Hence, with increases in load there is a rapid increase in system flows and in reactor flux. Thereafter, a number of minor trim controls occur to maintain steam pressure and temperatures to achieve a constant 2400/1000/1000 over the normal load following range (nominal 25% of rated to the valve wide open (V.W.O.) condition). These include desuperheating of main steam with feedwater between the evaporator and superheater, attemperation of reheat steam with cold reheat steam; changes in feedwater pressure, and change in reactor outlet and inlet temperature; change in the split in superheater-reheater sodium flow; and change in secondary cold leg temperature.

1.2.2 Plant Protection System

This system provides two levels of protection. The first of these is called "inhibit-alarm", and prevents the withdrawal of the control rods. Audio-visual alarms alert the operator to this condition and what initiating event (first-out annunciators) caused the action. The function of this action is to aid in plant protection, but it also acts to minimize the number of "reactor trips". The final action of the system is called "reactor trip". This action causes all rods (both safety and shim-safety rods) to drop into the core and the drive motors to run in. In order to avoid temperature shocks, the turbine must also be tripped when the reactor is tripped. The converse is not true, since the reactor can be run back to within the capacity of the dump system when the turbine is tripped.

1.2.3 Core Flux Measurement System

The schematic diagram of the nuclear instrumentation given in Section 6.0, Figure 6.1.3, shows the core nuclear detectors. The three ex-core source range channels will probably use BF_3 or fission chambers. There are six ex-core power range channels, three for plant control and three for plant protection. The in-core system consists of a neutron detection chamber which can be inserted in a special fuel element for axial flux distribution measurements; and three additional neutron detectors located in thimbles close to the top of the axial blanket, to provide trip signals to the safety system during critical experiments.

1.2.4 Reactor Temperature Measurement System

The system includes the thermocouples which sense the temperatures of the sodium exiting from each fuel element, and the grid plate thermocouples.

1.2.5 Reactor Cover Gas Monitor

This system consists of a multichannel analyzer and scaler, fed by a gamma detector mounted in an auxiliary gas circulating loop off the reactor cover gas plenum.

1.2.6 Hydrogen Detection System

This system consists of the nickel diffusion type hydrogen detectors mounted at the sodium outlet of the steam generators.

1.2.7 Sodium Flow Measurement System

This system consists of electromagnetic flowmeters used to measure primary and secondary sodium flow.

1.2.8 Reactor Sodium Level Measurement System

This system consists of the induction type sodium level alarm and indicating devices used for measuring sodium level in the reactor vessel.

1.2.9 Delayed Neutron Monitoring System

This system consists of multiple shielded BF_3 chambers which sense delayed neutrons in a sidestream of sodium taken from the reactor vessel outlet.

1.2.10 Radiation Monitoring System

This system of vent gas and liquid waste activity monitoring and control subsystems, primary equipment cell activity monitoring devices, secondary loop activity monitors, area radiation monitors, air particulate monitors, portal monitors, criticality monitors, and fuel handling machine radiation monitors.

1.3 Requirements

1.3.1 Performance Requirements

1.3.1.1 Plant Control Performance Requirements

1.3.1.1.1 Inputs to the plant control consist of signals from process sensors in the reactor, heat transfer components and the turbine-generator.

Outputs are to final control elements and operator-display devices. Further details are discussed in section 2.0.

1.3.1.1.2 Main interfaces for the Plant Control System are with the control room, main and reheat steam lines, feedwater pumps and lines, sodium pumps and heat exchangers, reactor, plant annunciators and data logs, Reactor-Scanner-Logger, startup-shutdown and steam dump system, plant protective system nuclear instrumentation system, and plant electrical system.

1.3.1.1.3 Tolerances are further defined in the following.

1.3.1.1.4 Design margins have not yet been established. (However, tentative rate margins are a factor of two. This will apply to the flux controller (and control rod drives) and to pump speed control.)

1.3.1.1.5 Steady-state (full load/part load) and transient requirements, are estimated as follows:

1.3.1.1.5.1. The plant is designed to be a minimum incremental cost station and will be operated at nearly constant (20-24 hours/day) rated load, with nominally only one power swing (nominally to 50% load) each day. Throttle conditions are 2400/1000/1000 (to a TC-6F machine).

1.3.1.1.5.2 Normal load changes will not exceed 2% of rated load per minute.

1.3.1.1.5.3 Response of the system to non-normal demands should be at least 25% minute. Non-normal means emergency pickup or partial load rejection.

1.3.1.1.5.4 Expected load following range is from 25-100% of rated load. Load increase above rated load may be made less than normal rate.

1.3.1.1.5.5 Performance testing of the plant to ASME Power Test Codes will be required.

1.3.1.1.5.6 Provision shall be made for automatic control of steam pressure and temperature, with variable setpoints (range of setpoint adjustment to be specified).

Note: A point in time is called steady-state if, during the preceding and succeeding half minute, steam temperatures (at the main and reheat headers) do not vary more than 7°F, steam pressure by more than 10 psi, and power by more than 1%.

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

1.3.1.1.5.7 Providing feedwater temperatures are not varied by more than 10°F the Plant Control System shall be capable of maintaining the plant in steady state intervals of at least 4 hours.

1.3.1.1.5.8 Steam pressure shall not exceed 130% of rated (1.3 x 2400 psi) at any time. Accumulated time in excess of 105% of rated (1.05 x 2400 psi) shall not exceed 12 hrs/yr.

1.3.1.1.5.9 Main or reheat steam temperature shall not exceed rated +25°F (1025°F) more than 400 hrs/yr; shall not exceed rated +50°F (1050°F) more than 15 minutes at any time nor more than 80 hrs/yr.

1.3.1.1.5.10 It shall be a design objective that no steady state point shall exceed rated steam temperature +100°F (1100°F) under any operating or casualty condition (including reactor or turbine trip).

1.3.1.1.5.11 Under normal load following, steam temperatures shall not vary more than 25°F, and initial pressure not more than 100 psi.

1.3.1.1.5.12 Sodium temperatures, when below 900°F, shall not be changed more than 100°F in ½ hour. Above 900°F, no sodium temperature shall be changed more than 50°F in ½ hour.

1.3.1.1.5.13 Steady state temperatures (part load program) have not been determined. (They will be set by heat transfer characteristics and required steam temperatures.) However, it is a design objective that steady-state sodium temperature be maintained within 15°F of nominal. Transient limits have not been established.

1.3.1.2 Plant Protection Performance Requirements

1.3.1.2.1 Input/outputs are shown in Figure 6.1.1.

1.3.1.2.2 Ranges of operating variables are:

1.3.1.2.2.1 Twelve decades in flux to at least 150% power.

1.3.1.2.2.2 Approximately 600-1200F on sodium temperature.

1.3.1.2.2.3 Normal reactor level minus one foot plus about four feet.

1.3.1.2.2.4 Coolant flows about 10-110% of rated flow.

1.3.1.2.3 Tolerances on design points have not been specified, but will range between 1/2 and 5% of scale.

1.3.1.2.4 Preliminary values of some steady-state requirements are shown in Figure 6.1.1. All values will be established and tabulated during preliminary design.

1.3.1.3 Core Flux Measurements Performance Requirements

1.3.1.3.1 Inputs are from neutron flux within the core with output to Channel 10, and from the region above the axial blanket (from a nominal 3" above the axial blanket up to the top shield) with output to Channels 7, 8, and 9.

1.3.1.3.2 Channel 10 interfaces with fuel elements, core structure, top shield, core cables and connectors, and nuclear instrumentation (possibly also with the plant protective system). Channels 7, 8, and 9 interface with core structure, top shield, fuel handling machine, plant protection system, core cables and connectors, nuclear instrumentation, and plant annunciators.

1.3.1.3.3 Steady state (full load/part load) and transient performance are as follows:

1.3.1.3.3.1 Channel 10 must operate over the range of $10^5 - 5 \times 10^{14}$ nv. Target sensitivity is expected to be 10^{-4} cts/nv for 235 or 239 sensors, 10^{-7} cts/nv for 238 sensors, and 10^{-18} A/nv for enriched boron sensors.

1.3.1.3.3.2 Channels 7, 8, and 9 must operate over the range of about $10^2 - 10^{14}$ nv. Target sensitivity is expected to be .001-1 cts/nv, with the system kept on scale ($10 - 10^6$ cps) by moving the detectors.

1.3.1.3.6 The Channel 10 detector must be capable of axial traverse with a determinate position throughout its traverse.

1.3.1.4 Core Temperature Measurements Performance Requirements

1.3.1.4.1 Input to the thermocouples is temperature and output is a millivolt signal (per chromel-contantan conversion tables) to the plant protective system, the plant control system, the reactor core scanner-logger, and the preheat panel.

1.3.1.4.2 Thermocouples interface with core and vessel structure, top shield, core cables and connectors, control rods, and instrument systems listed in 1.3.1.4.1.

1.3.1.4.3 Inaccuracy of temperature measurement (as read out by core scanner logger) shall be less than 15°F . (T/C bought to 3/4%, see 1.3.5.4.1).

1.3.1.4.4 Steady state (full load/part load) and transient performance requirements are only severe with the channel outlet thermocouples. The temperature at these sensors will cycle each time there is a power change. Nominal rates are less than $100^\circ\text{F}/1/2$ hour. The time constant of these thermocouples should be 3 second or less, i.e., the time to detect 63.3% of a step change in sodium temperature. (This obviously requires high conductivity liquid like NaK to achieve good thermal bond.).

1.3.1.4.5 Nominal leak rate of an individual seal at top of loading face shield is less than 6.5×10^{-6} atm/cc/sec at 1 psi, linear with pressure.

1.3.1.5 Reactor Cover Gas Monitor Performance Requirements

1.3.1.5.1 The system shall detect at least Xe 133 in concentrations of 10^{-4} microcuries/cc in flowing Helium (nominal 0.2-1.0 cfm).

1.3.1.5.2 An internal calibration source is required, along with pump, flowmeter and regulator, particulate filter (to remove 0.5 micron and larger).

1.3.1.5.3 Electronics will cover six decades, with provision to change sensitivity by detector movement. Observed readings will be within 6% of true in steady-state. Transient response requirements will be minimal since process response (order of ten minutes) is controlling.

1.3.1.6 Hydrogen Detector Performance Requirements

1.3.1.6.1 Input

1) Hydrogen gases entrained in flowing sodium in concentrations of 0 to .03 ppm.

2) Vacuum system to maintain a pressure of approximately 5 microns in that part of the system containing the diffusion membrane and Pirani gage.

1.3.1.6.2 Output

1) Electrical signal to Alarm Panel, alarm level to be .03 ppm H_2 in Na in presence of 10-20 ppm oxide.

2) Response time, max. one min.

1.3.1.7 Sodium Flow Measurement Performance Requirements

1.3.1.7.1 Input to Flowmeter

Sodium flow - .2 ft/sec - 20 ft/sec.

The velocity of the conductor (sodium) is the flow rate and can be stated in ft/sec or gal/min or pounds/hr or any convenient unit for flow.

Sodium Temperature - 350°F to 1140°F.

1.3.1.7.2 Output from Flowmeter

The signal generated by the flowmeter will be 0 to 20 MV proportional to 0 to 100% flow.

1.3.1.7.3 Interfaces

The flow meters interface with the Main Heat Transfer System, Service Systems, Plant Control and Protective System, and data collecting.

1.3.1.7.4 Tolerances

1.3.1.7.4.1 The desired absolute accuracy is less than 5% scale at 20 to 110% flow.

1.3.1.7.4.2 The desired repeatability is less than 2% of scale at 20 to 110% flow.

1.3.1.7.5 Steady State and Transient

1.3.1.7.5.1 The response of the meter shall be 15 cps or better (10 ms time constant or less), over complete scale.

1.3.1.7.5.2 A monotone increasing signal is required with 20 MV output for rated flow. From 20-110% of rated flow, the sensitivity shall not change by more than a factor of 2.

1.3.1.8 Reactor Sodium Level Measurement Performance Requirements

1.3.1.8.1 Alarm and continuous level coils shall indicate within 2".

1.3.1.8.2 Dip-stick probes shall indicate within 1/2".

1.3.1.8. Nominal alarm level is 1/2 foot below normal operating level. Trip level is 1 foot below. Target system time constant is less than 100 ms.

1.3.1.9 Delayed Neutron Monitoring Performance Requirements

1.3.1.9.1 Input lines will come from at least two of the reactor outlet pipes to the detector system, which should be located close to the reactor.

1.3.1.9.2 Output flow from the system is back to the reactor overflow tank. Output signal of the electrical system is pulses to scalars and count rate meters and then to the annunciator and plant data system.

1.3.1.9.3 Target sensitivity of detector is at least 10 counts/nv, shielded from Na 24 of equilibrium activity of about 0.05 curies/cc.

1.3.1.9.4 Pump capacity of 10 gpm and 15 psi is required to deliver sodium at about 4 feet per second.

1.3.1.9.5 Overall sensitivity of the system (from a given size leak to detector count rate) has not been determined. (However, considerable leeway exists by varying counting time.) It is a system objective to be able to detect occurrence of fuel-pin failures (at a value to be specified) less than 1%.

1.3.1.10 Radiation Monitoring Performance Requirements

Requirements on the RD&M system derive from 10 CFR 20 and 10 CFR 100.

1.3.1.10.1 Radiation level in routinely occupied areas shall be detectable to 0.01 mr/hr (assumed background of 0.02 mr/hr), and nominally alarm at 0.2 mr/hr or less (0.1 MPL as in (a) above).

1.3.1.10.2 The requirement in 1.3.1.10.1 above applies to secondary sodium detectors mounted on secondary piping.

1.3.1.10.3 Target sensitivities for fixed detectors in normally non-occupied areas are:

10^{-6} microcuries/ml of Krypton 85 in air (1.2.10.1)

10^{-6} microcuries/ml of Xenon 133 in air (1.2.10.1)

10^{-6} microcuries/ml of Strontium 90 in water (1.2.10.2)

0.01 mr/hr 80 Kev - 2.5 Mev gamma (1.2.10.3, 1.2.10.5, 1.2.10.7)

1 mr/hr 80 Kev - 2.5 Mev gamma (1.2.10.5)

1.3.1.10.4 Air particulate monitors shall detect Xenon and Krypton as in 1.3.1.10.3 above and at least 10^{-8} microcuries/ml of Sodium 24 in air. (It seems likely that a separate requirement will be established for Plutonium.)

1.3.1.10.5 Nominal system accuracy requirement is that indicated dose rate shall be within 20% of actual.

1.3.1.10.6 Nominal system time constants will not exceed 25 seconds.

1.3.2 Environmental Requirements

1.3.2.1 Plant Control Environmental Requirements There is considerable variation in environmental requirements depending upon the particular component or subsystem. Hence, only a few general comments are made.

1.3.2.1.1 Sensors and final control elements obviously require careful environmental specifications to assure suitability of service. These will be prepared individually.

1.3.2.1.2 Most instruments will be in the control room or preamp room, a typical clean service and 40°C requirement.

1.3.2.1.3 Units located in the switchgear room will be subject to dirty (non-corrosive) service and 50°C requirement.

1.3.2.1.4 Units located on the loading face or in the high bay will be subject to clean service and 65°C (without forced cooling) requirement.

1.3.2.2 Plant Protection Environmental Requirements

1.3.2.2.1 Temperatures are as follows:

1.3.2.2.1.1 15-40°C in control or preamp room, nominal at 20°C.

1.3.2.2.1.2 10-60°C in cable spreading or switchgear room, nominal at 35°C.

1.3.2.2.1.3 10-80°C in reactor building or on reactor loading face, nominal at 40°C.

1.3.2.2.1.4 Sensor temperature requirements are called out in 1.3.1 above.

1.3.2.2.2 Radiation levels are as follows:

1.3.2.2.2.1 Radiation levels for all parts of the system except sensors and cabling is that of normally occupied areas.

1.3.2.3 Core Flux Measurement Environmental Requirements

1.3.2.3.1 The operating sequence of events (startup, shutdown, load changes, etc.) runs as follows: Following checkout of both systems, the wet critical will be run and eventually power will reach 10% and temperatures near 700°F. After the core has adequately been mapped, Channel 10 will be removed from the system and stored. Channels 7, 8, 9 will be removed and stored or will be positioned in the top shield. In this case they will be called Channels, 11, 12, 13. When the reactor has been shut down, Channels 7, 8, 9 will be reinstalled in the core.

1.3.2.3.2 Expected number of cycles is at least one for Channel 10 and the initial loading plus every refueling cycle for Channels 7, 8, 9.

1.3.2.3.3 The expected range of environmental variables in each operational phase is as follows:

1.3.2.3.3.1 Temperature is not expected to exceed 700°F for all phases of operation of this system.

1.3.2.3.3.2 Pressure on the performance sensor should not exceed 100 psig. The safety sensors, being in thimbles will see only a nominal gas pressure expected not to exceed 15 psig; however, plugs must be designed to accommodate the DBA (possibly up to 200 psi).

1.3.2.3.3.3 The maximum radiation environment on performance sensor will be between 10^8 and 10^9 R/hr, and less than 5×10^{14} nv. For the safety sensor it is less than 10^8 R/hr and 10^{14} nv during the initial operating period. Thereafter, flux will normally be substantially less than 10^6 nv and 10^7 R/hr.

1.3.2.3.3.4 No loads other than pressure and impact head are expected, but no analysis has been made.

1.3.2.3.3.5 Transient conditions are expected to be the same as steady state. A possible exception to this would be in the event that thimbles for the safety sensors were inserted into the core immediately after shutdown. It has been tacitly assumed that design margins on materials of construction would be such no problem with temperature would exist in the non-operating condition.

1.3.2.3.3.6 There are no anticipated problems with chemicals (oxide, hydroxide, etc.) attack that are different than for other core components. Expected construction is with stainless steel sheathing detectors.

1.3.2.3.3.7 Velocity is near nominal zero for the performance sensor (Channel 10) since it will be in a thimble in the fuel element, which thimble must be severely orificed in order to avoid starving fuel pins in the special fuel element. (It is assumed that the thimble is not closed in order to be able to drain sodium.) Velocity is zero for the safety sensor (Channels 7, 8, 9) since they are contained in a gas-chilled thimble.

1.3.2.4 Core Temperature Measurement Environmental Requirements

1.3.2.4.1 The operating sequence of events (startup, shutdown, load changes, etc.) runs as follows: initial operation will be the preheating of the core before wet critical. Nominal temperature will likely not exceed 350-400°F at rates less than 1°F/hr. Wet critical and full loading will likely not exceed 700°F. Normal power operation will nominally be limited to 1160°F at the inlet. Refueling will be nominally at 500°F. A major event is reactor trip which will cause down-shocks in temperature in the order of 200°F/10 seconds, recovering possibly in 30 seconds. Cross-checking for calibration run every time the system is isothermal. During refueling, checks made against newly calibrated standards.

1.3.2.4.2 In addition to the refueling cycle (30 during plant life) there will be about 10,000 daily load cycles, and 10^6 - 10^7 minor (1-5%) power changes. Reactor trip is indeterminate as to frequency, but may occur six times per year.

1.3.2.4.3 The expected range of environmental variables in each operational phase is as follows:

1.3.2.4.3.1 Temperature as shown in 1.3.2.4.1.

1.3.2.4.3.2 Pressure will be greatest (100-150 psi) on certain of the thermocouple located in the grid plate structure. All other locations will see only 0-15 psig. However, system must contain DBA pressure.

1.3.2.4.3.3 Expected maximum radiation environment will occur for fuel channel outlet T/C, and 30 years life is approximately:

2×10^{12} R	Core γ 's
3×10^{20} nvt	fast
1.2×10^{23} nvt	epithermal
2.8×10^{22} nvt	thermal
1×10^{14} R	sodium-24

1.3.2.4.3.4 Vibration analysis has not been made. However, vibration is assumed to be of any consequence only for core exit T/C. These are all in tubes with closed ends - although slotted both to vent and to improve contact with sodium - so that forcing functions will be of lesser magnitude and frequency than in the sodium stream, particularly with stiff support tubes. Vibration testing seems required.

1.3.2.4.3.5 Transients are as defined in 1.3.2.2 above, and will nominally be limited to 100°F/1/2 hour and 15 psi/sec. Additional cycles - of an undefined number, but possibly at the rate of 6/year, will be as described in 1.3.2.4.1 above.

1.3.2.4.3.6 There are no anticipated problems with chemical(oxide, hydroxide, etc.) attack, provided seals do not leak and cause local buildup of oxides. (Minor leaks should not cause in-leakage in any event, as the reactor will be maintained at slightly positive pressure.)

1.3.2.5 Reactor Cover Gas Monitor Environmental Requirements

1.3.2.5.1 The system will operate continuously and is relatively unaffected by plant cycles.

1.3.2.5.2 Temperature of the gas plenum is not expected to exceed 1170°F at any time, and nominally will not change more than 100°F/hr. Gas temperature at the detector will be less than 200°F.

1.3.2.5.3 Pressure of the gas plenum is normally inches of water. However, design pressure must be consistent with the DBA.

1.3.2.5.4 There appear to be no problems associated with loads, vibration, chemical compatibility, piping reactions, or weather conditions.

1.3.2.5.5 Gas velocity is less than 2 fps.

1.3.2.6 Hydrogen Detector Environmental Requirements The hydrogen detection system will interface with the Heat Transfer Segment, Steam Generator, Control & Instrument Segment and the Site Services and Structures Segment.

1.3.2.6.1 Tolerances

$\pm .01$ ppm H_2 in Na.

1.3.2.6. Steady State Continuous monitoring of hydrogen background in the secondary sodium system and alarm on hydrogen concentration of .03 ppm H_2 in sodium.

Transient - Alarm on rapid rise in hydrogen concentration.

1.3.2.6. Operating Sequence of Events

Sodium side temperature changes 30-yr life.

I. Initial Preheat

Three (3) cycles 70°F-350°F @ 3°F/hr.

II. Rise to Power

Three (3) cycles 350-1070°F @ 7.3°F/hr with maximum ramps of 100°F/30 min.

III. Load Swings

Nine thousand cycles $\pm$ 100°F/30 min.

IV. Refueling Shutdown

Thirty (30) cycles 1070-500°F @ -100°F/30 min. maximum.

V. Maintenance Shutdown

Thirty (30) cycles 1070-350°F @ -100°F/30 min. maximum, and -50°F/30 min. maximum to ambient.

VI. Casualty Conditions

1. Sodium flow continues feedwater flow stopped - 1 cycle.
2. Earthquake Force

Horizontal	.33g
Vertical	.2g
3. Withstand overpressure of secondary sodium system from tube failures in the modules. Maximum pressure to be determined.

VII. Static Loads

1. Sodium pressure - 100 psig
2. Static head and dead weight.

VIII. Fluctuating Loads

1. Casualty conditions
2. Vibration
3. Thermal movement of piping
4. Earthquake

IX. Chemical Environment

Sodium Side - Normal	10-20 ppm oxide
Normal	20 ppm carbon
Casualty	NaOH
	H ₂
External	Air

1.3.2.7 Sodium Flow Measurement Environmental Requirements

The magnetic circuit will be supported by an independent hanger to remove the major portion of the load from the pipe. With this type of mounting the environmental requirements for the magnet are that of the vault atmosphere and a small amount of heat transferred from the pipe through the mounting stabilization brackets to the magnet circuit assembly.

1.3.2.7.1 Sodium Temperature Changes 30-Year Life

I. Initial Preheat

3 cycles 70°F-350°F @ 3°F/hr

II. Rise to Power

3 cycles 350°F Operating Temp. at 7.3°F/hr with maximum ramps of 100°F/30 min. (max.).

III. Load Swings

9000 cycles (cold leg) $\pm$ 50°F/30 min. ramp.

IV. Refueling Shutdown

60 cycles - 100°F/30 min. max. from 915°F to 550°F.

V. Maintenance Shutdown

30 cycles - 100°F/30 min. max. from 915°F to 350°F and -50°F/30 min. max. to ambient.

VI. Casualty Conditions

1. Reactor trip, turbine trip, sodium flow continues
feedwater flow stopped

5 cycles

2. Feedwater flow lost but sodium flow continues,
delayed reactor trip

5 cycles

3. Earthquake horizontal force .33g horizontal,
.2g vertical

VII. Static Loads

a) Normal Operating Pressure

b) Static Head and Dead Weight

Fluctuating Loads

a) Casualty Conditions

b) Vibration (pump induced)

c) Thermal Movement of Piping

d) Magnet Mounted on Independent Hanger (Min. Pipe Load).

VIII. Chemical Environment

Sodium Side	Normal	10-20 ppm oxide
	Normal	20 ppm carbon
	Casualty	xxx NaOH to be determined

IX. Velocity

Sodium Side .2 ft/sec to 20 ft/sec.

X. Radiation

The meter shall be able to absorb 10^{14} ergs/gm from sodium without changes in signal or failure rate.

1.3.2.8 Reactor Sodium Level Measurements Environmental Requirements

1.3.2.8.1 Nominal integrated dose over a 30-year life near the top of the axial blanket will be approximately as follows

Fast Flux	1.9×10^{15} nvt
Epithermal Flux	3×10^{20} nvt
Thermal Flux	6.3×10^{18} nvt
Na 24 Gamma	9.5×10^{13} rad
Core Gammas	2.5×10^{12} rad

1.3.2.8.2 Nominal sodium temperature will be 1140F.

1.3.2.8.3 Coils and connectors will be sealed in thimbles containing an inert gas (helium suggested).

1.3.2.8.4 Pressures are nominally a few psi. However, shield plugs and thimble seals must be designed for the DBA.

1.3.2.9 Delayed Neutron Monitoring Environmental Requirements

1.3.2.9.1 Saturated activity of Na 24 will be 0.046 curies/cc.

1.3.2.9.2 Nominal sodium temperature is 1140F at full power conditions (the sodium will be insulated from the cooled detectors).

1.3.2.9.3 Environment for BF_3 chambers is yet to be determined. Sensitivity and life decrease with increasing temperature, but there is a design compromise to avoid forced cooling if possible; such compromise has not been made.

1.3.2.9.4 Detector will operate continuously, but it is likely to be ineffective when power drops much below 50%. Hence, variations in sodium temperature will be minimal.

1.3.2.9.5 Maximum sodium velocity will be less than 5 fps. Vibration or structural loads are not considered a problem.

1.3.2.9.6 Nominal pressure is only a few psi. However, system must be designed to handle the DBA.

1.3.2.10 Radiation Monitoring Environmental Requirements

1.3.2.10.1 Environment for most instruments is that of the control room.

1.3.2.10.2 Water detectors must be suitable for 100°F, 200 psi operation (this assumes mechanical pumps.)

1.3.2.10.3 GM tubes for secondary sodium monitors must be suitable for 300°C operation.

1.3.2.10.4 Cell monitors must be suitable for 350°F operation, in fields up to 1000 R/hr.

1.3.2.10.5 Other detectors will be specified for 150°F operation (with 100-110 expected).

1.3.2.10.6 All monitors penetrating containment barriers must be designed to handle the DBA pressures.

1.3.3 Operational Requirements

1.3.3.1 Plant Control Operational Requirements

1.3.3.1.1 Insofar as manual operation is feasible, provision should be made for manual operation of any controller. (It is not considered feasible to operate manually under changing load or temperature. However, under steady load it is believed feasible. Selected hand control stations may require remote (from controller) location.

1.3.3.1.2 Operation of the plant is assumed to be with a single unit operator. This is, one operator will operate both the turbine-generator and reactor steam supply. This requires careful attention to and integration of the plant control, data analysis and presentation, and annunciator systems, liberal use of deviation alarms, and auto transfer of critical functions.

1.3.3.1.3 Considerable assistance must be given to the operator during startup and shutdown. These details have not been worked out; however, it is assumed that largely automatic (although of permissive - operator choice - type) operation will be required, including a special startup board.

1.3.3.1.4 Trim of individual rods or of a gang of rods (not necessarily those selected for auto control) is required. The three modes of operation are then, individual, manual gang, and auto gang. The auto gang further must have provision for operation by the flux controller or by the operator. This is, the auto gang has both an auto position and a manual position (required for controller alignment and auto transfer, and emergency condition).

1.3.3.1.5 Deviation alarms (deviation between setpoint and measured) are required for certain critical controllers.

1.3.3.1.6 Auto transfer to manual on deviation alarm is required for selected units (to be specified).

1.3.3.1.7 Limited rates of change (nominal 50%/min and 100F°/hr) on set-points will be required for critical controllers.

1.3.3.1.8 Provision shall be made to adjust circuit flows in varying ratios to account for differences in heat transfer characteristics between circuits. (This is not required to be a front-of-board adjustment.)

1.3.3.1.9 Provisions shall be made to operate any circuit at constant load while varying load in the other circuits (within the capability of that circuit to meet steam temperatures requirements).

1.3.3.2 Plant Protection Operational Requirements

1.3.3.2.1 A reactor start key is required. It shall be in the "on" position before any rods can be pulled.

1.3.3.2.2 A system check according to a written operating procedure is required following reactor trip. This shall include functional check of the setback and reactor trip busses including modules, holding magnets, and control rods and drives.

1.3.3.2.3 At least two source range or two power range nuclear instruments shall be on-scale, and at least two primary pumps shall indicate flow above 10% before any rod can be pulled.

1.3.3.2.4 All safety rods must be above a minimum value (to be specified) before any shim-safety (control) rod can be withdrawn. (Normal practice for power operation will be to withdraw all safety rods to their upper limit.) Power operation above a specified value (presently set at 5% of rated power) shall not be allowed if any safety rod is in the core.

1.3.3.2.5 Provision shall be made to exercise control rod drives through the use of a keylock switch when it is in the "on" position. Such position shall cause a reactor trip.

1.3.3.2.6 Reduction of sodium pumps to a prescribed speed, actuation of the convection control system, and start of emergency feedwater or reset of feedwater pump pressure control shall occur automatically on reactor trip. (Operation of steam dump or power reliefs is by pressure control.) Provisions shall be made to reset these to other values after suitable time-delay.

1.3.3.2.7 Signals for initiation for containment isolation have not been established. (Probable candidates are high building pressure or high building leak rate, fire protection system, and radiation monitoring system.) If reactor trip is an initiating condition, a keylock bypass will be required.

1.3.3.2.8 Automatic lockout of the flux controller shall be possible when the "rod drive exercise" keylock (1.3.3.2.5) is in the "on" position.

1.3.3.2.9 See 1.1.2.4

1.3.3.2.10 See 1.1.2.6

1.3.3.2.11 See 1.1.2.7

1.3.3.2.12 See 1.1.2.8

1.3.3.2.13 See 1.1.2.13

1.3.3.2.14 See 1.1.2.19

1.3.3.2.15 See 1.1.2.20

1.3.3.2.16 See 1.1.2.22

1.3.3.2.17 See 1.1.2.23

1.3.3.2.18 See 1.1.2.24

1.3.3.2.19 See 1.1.2.25

1.3.3.2.20 Provision shall be made to operate the reactor trip bus from at least one place besides the control room. (Tentative locations are the switchgear room, the preamp room, and the reactor loading face.)

1.3.3.2.21 After modification to the protection system, a complete continuity check is required in addition to functional checks prescribed in 1.3.3.2.2.

1.3.3.2.22 During the initial critical experiments, special circuits may be added under approved administrative procedures. They must be removed before normal plant operation. Such installation and removal is considered a system modification.

1.3.3.3 Core Flux Measurements Operational Requirements

Operation of the mapping system is largely nonautomatic in that location of fuel bundle and position of the sensor is at the discretion of the test engineer, and range switching is anticipated. The safety system part, once detectors have been positioned, is automatic in the following sense: operation of the fuel handling machine depends upon satisfaction of interlocks derived from high and low limits on flux measured by safety sensors.

1.3.3.4 Core Temperature Measurements Operational Requirements

It should be possible to make checks during shutdown or at any other time the system is isothermal, and procedures will call for this to be done.

A design objective is to minimize the number of times that connections must be broken. To this end, an operational requirement to avoid breaking any T/C connections during fuel handling, except as required for repair or replacement, makes very good sense. Considering the requirement for top shield rotation (and perhaps also raising the guide tubes in-core), this is not a trivial design problem.

1.3.3.5 Reactor Cover Gas Monitor Operational Requirements

The operation of the monitor will require some expertise, and there are some problems with calibration and setpoints. However, once the procedures have been worked out they can be reduced to operational documents.

1.3.3.6 Hydrogen Detector Operational Requirements

To be furnished later.

1.3.3.7 Sodium Flow Measurement Operational Requirements

To be furnished later.

1.3.3.8 Reactor Sodium Level Measurements Operational Requirements

1.3.3.8.1 Continuous monitoring of sodium level is required during fill and drain operations, and during refueling.

1.3.3.8.2 Provision for calibration during shutdown or refueling is required

1.3.3.8.3 It is assumed that cross-calibration during power operation will be the usual procedure. However, it would be desirable to be able to perform

on-line calibration. If this should become a requirement, a drive system for the dip-stick would likely become required (present design assumes manual operation).

1.3.3.9 Delayed Neutron Monitoring Operational Requirements

System calibration and the physical interpretation of signals is difficult. Tests are being performed in TREAT to determine typical signatures of fuel failures. Further tests have been run in EBR-II and data will become available as operation continues. However, with the establishment of a stable background (either constant or constantly increasing), fuel failures will be evidenced by step changes. Calibration (against a similar-to-expected energy distribution) and establishment of a real background are the primary operational requirements. It should be noted that this system is complementary to the cover gas monitor, and some modes of fuel failure will be seen on both of them.

1.3.3.10 Radiation Monitoring Operational Requirements

1.3.3.10.1 There are numerous control devices in the system. A status display shall be provided for the operator. Present intent is not to supply a full graphic panel, but either a mimic bus or CRT display.

1.3.3.10.2 Provision shall be made for adequate environmental sampling (instrument access plugs where appropriate) of normally unoccupied areas prior to entry.

1.3.4 Safety Requirements

1.3.4.1 Plant Control Safety Requirements

The major safety criterion established for the plant control system is that malfunction of the control system should not simultaneously cause failure, malfunction, or prohibit the operation of plant protective devices. However, this is specifically not construed to mean that there should be no connection to or common devices with the protective system. (Specific examples of such include flowmeter and flux signals, control rods, and many others. Common signals and functions can be shared provided there is no interfering effect.)

1.3.4.2 Plant Protection Safety Requirements

These are already stated in sections 1.1.2.1, 1.1.2.2, 1.1.2.3, 1.1.2.4, 1.1.2.16, and 1.1.2.26.

1.3.4.3 Core Flux Measurements Safety Requirements

Safety is of considerable importance since the control and protection of the core during wet critical testing is only by this system. This implies good absolute calibration and careful attention to detail in installation and operation. An additional hazard is the possibility of shearing off instruments or cables which have been inserted into the core but not removed before shield rotation. This appears unlikely by reason of administrative procedure, but is possible.

1.3.4.4 Core Temperature Measurements Safety Requirements

There appear to be three problems related to safety. First is improper seals, connections, and feed-throughs that lead to excessive failure rates and resulting loss of protection. This has been attacked by redundancy, attention to design, and hopefully by attention to detail in the field. Second is the problem of exposure during maintenance, which must be handled by good design and administrative procedures. The third is the possible consequence of broken T/C's being swept around through the system. The latter could apply to any T/C, but most seem to be adequately protected. Possibly only the T/C on the lower grid plate (approximately 60 feet long) are at all likely (not very) candidates. In any event, safety considerations require that design minimize this possibility (and the consequences thereof should be examined).

1.3.4.5 Reactor Cover Gas Monitor Safety Requirements

These are typical of radioactive systems with respect to leakage and exposure. This is aggravated by periodic filter replacement, and possibility of sodium contamination. Other safety requirements relate to electric motors and detector high voltage.

1.3.4.6 Hydrogen Detector Safety Requirements

Personnel safety only.

1.3.4.7 Sodium Flow Measurement Safety Requirements

The inherent safety characteristics of electromagnetic flowmeters, because they do not penetrate the sodium containment, is greater than all other types and should not affect the overall plant safety.

1.3.4.8 Reactor Sodium Level Measurements Safety Requirements

The only apparent safety requirements concern proper sealing against the DBA and radiation protection during operation or maintenance.

1.3.4.9 Delayed Neutron Monitor Safety Requirements

1.3.4.9.1 Principal safety requirements are related to maintenance of the integrity of the primary system. Design should reflect the fact that penetration of the reactor containment is made.

1.3.4.9.2 Operational safety requires shielding, consideration of the consequences of leaks involving radioactive sodium into the high bay, and standard precautions for electric power and detector high voltage.

1.3.5 Design Requirements

1.3.5.1 Plant Control Design Requirements

1.3.5.1.1 At least the following documents pertain to design.

NEMA Standards

National Electrical Code

TID-20893

(Instrument Modules)

ISA-RPI-1, 20

(Recommended Practices)

USASI-N1, N42

(Instrument Design)

SDD-096-10-000

(Overall SDD)

SDD-096-22-000

(Reactivity Controls)

1.3.5.1.2 It is a design objective that solid state units of the smallest size and voltage acceptable to the plant operator shall be specified.

1.3.5.1.3 There shall be at least three isolated instrument power supply electrical buses, nominally one for each heat transfer circuit. Standby sources (through switching) shall be furnished. For subsystems not part of a heat transfer circuit (flux controller or plant load demand computer), dual-feed power buses are required.

1.3.5.1.4 The largest possible use of modular design in instrument chassis shall be made. Modules will be furnished with plugs for power feed.

1.3.5.1.5 Every instrument shall be designed, constructed, and installed so that it can conveniently be replaced without affecting other units.

1.3.5.1.6 Test points shall be furnished for checking inputs, outputs, power and critical adjustments. The latter shall be readily identified, and with lock-in-place features.

1.3.5.1.7 All terminal strips and test points shall be identified in a permanent manner and be accessible for test and maintenance.

1.3.5.1.8 The largest possible use shall be made of digital indicators and lighted push-button control stations.

1.3.5.1.9 Important parts of the flux control system shall be redundant (so that single failures shall not cause plant malfunction). The design shall incorporate majority type logic for use with at least three nuclear channels. (Figure 6.1.4 is an acceptable mechanization of this requirement.)

1.3.5.2 Plant Protection Design Requirements

1.3.5.2.1 At least the following documents are applicable to the Plant Protection System.

10 CFR 50	(Licensing Criteria)
TID-20893	(Instrument Modules)
USASI-N1, N42	(Instrument Design)
ISA-RP-1, 20	(Recommended Practices)
SDD-096-10-000	(Overall SDD)
SDD-096-22-000	(Reactivity Controls)

1.3.5.2.2 It is a design objective that solid state units of the smallest size and voltage acceptable to the plant operator shall be specified.

1.3.5.2.3 The largest possible use of modular design in instrument chassis shall be made. Modules will be designed for front of board mounting with service loops (of a length to be specified) and mechanically lockable connectors.

1.3.5.2.4 The system shall be sectionalized into at least three physically separated subsystems each with individual electrical feed.

1.3.5.2.5 Modules shall be furnished with a keylock to fasten the module to the sectionalized board.

1.3.5.2.6 Test points shall be furnished for checking inputs, outputs, power and critical adjustments. The latter shall be readily identified, and with lock-in-place features.

1.3.5.2.7 It is a design objective that compensation for variations in ambient temperature over the ranges specified in 1.3.1.2 and 1.3.2.2 above shall be performed automatically or be not required.

1.3.5.2.8 Alternate methods of reactor trip called out above shall be furnished in color-coded locked boxes with remote annunciator contacts.

1.3.5.2.9 Provision for self-monitoring shall be provided.

1.3.5.2.10 All terminal strips and test points shall be identified in a permanent manner and be accessible for test and maintenance.

1.3.5.2.11 It is a design objective that signal wiring for the protection system be of different color from any other wiring in the plant. (Hallam practice was solid red.) Low level wiring shall be low noise of interweaved design with two wires per signal, pulled in conduit, and spatially separated from power conductors.

1.3.5.2.12 Manual switches for trip and setback shall be furnished with protective covers.

1.3.5.2.13 Annunciators for the protection system shall be specially color coded of lock-in "first out" type, wired in parallel with lighted setback and reactor trip plaques.

1.3.5.2.14 Outputs from protection system sensors that are used in other circuits shall be isolated by high input impedance amplifiers.

1.3.5.2.15 It is a design objective that making and breaking of protection system connections be minimized (particularly significant for signals from core sensors).

1.3.5.2.16 No instrument plugs or other devices to bypass proper functioning of any instrument or subsystem will be allowed in system design.

1.3.5.2.17 Loss of power or signal to any module shall place it in an action state.

1.3.5.2.18 Status of power and logic shall be shown on the face of each module.

1.3.5.3 Core Flux Measurements Design Requirements

1.3.5.3.1 At least the following documents are applicable to this system:

10 CFR 20	(Radiation Protection)
USASI Std. N2.6	(Administrative Procedures)
USASI Std. N42	(Nuclear Insts.)
SDD-096-10-000	(Overall System)
SDD-096-20-000	(Reactor Segment)
SDD-096-81-000	(Subassembly Handling)

1.3.5.3.2 The system interfaces with fuel elements, core structure, top shield, fuel handling machine, core cables and connectors, preamp and control room, plant annunciators and data system, nuclear instrumentation system, protective system, and electrical system.

1.3.5.3.3 Target dimension for in-core detectors (Channel 10) is nominal $3/8$ " O.D. It is assumed that they will be inserted in a hole in a fuel element, which hole is formed by the removal of seven pins, which pins are replaced by a thimble whose I.D. is about $3/4$ ", which thimble is properly orificed so as to achieve equivalent (to other channels) flow and pressure drop. (Testing required.)

1.3.5.3.4 Materials have not been specified; however, detectors will be contained in stainless steel tubes.

1.3.5.3.5 A spares policy has not been defined. Prudence, however would indicate at least one spare sensor of each type since sensor cost is small compared with schedule delays (especially so since detectors are special small-lot fabrication and will have long delivery times).

1.3.5.3.6 Provision for axial movement (up and down) is required for both types of detectors.

1.3.5.3.7 Provision for position location determination is required.

1.3.5.4 Core Temperature Measurements Design Requirements

1.3.5.4.1 At least the following documents apply to specifications related to this SDD.

NBS Circular 561	(Reference Tables)
ISA-RP-1.3	(T/C and extension wire error)
SDD-096-10-000	(Overall System)

1.3.5.4.2 Design interfaces are reactor core and vessel structure, top shield, core cables and connectors, preamp room, main control board, plant control system, plant protective system, core scanner-logger, cable and raceway schedules, and the instrument index.

1.3.5.4.3 Structural requirements have not yet been defined. However, it appears likely that T/C, and guide tubes must withstand velocities of 30 feet per second. Seals must withstand the DBA along with the top shield, and this may run as high as 200 psi.

1.3.5.4.4 Materials required are inconel sheathed, MgO packed, chromel-constantan T/C, nominal $1/8$ inch O.D. Consideration is being given to $1/16$ inch O.D. assemblies, which should also be tested in conjunction with the $1/8$ inch assemblies.

1.3.5.5 Reactor Cover Gas Monitor Design Requirements

1.3.5.5.1 At least the following documents pertain to system design.

10 CFR 20 (Radiation Protection)

NEMA Standards

National Electrical Code

USASI Std. N2.6 (Admin. Procedures)

SDD-096-10-000 (Overall System)

SDD-096-20-000 (Reactor Segment)

SDD-096-60-000 (Auxiliary Systems)

1.3.5.5.2 The system interfaces with the cover gas cleanup system, helium supply system, R/A vent system, preamp room, main control room, RD&M system, plant data and plant annunciator systems, and electrical system.

1.3.5.5.3 Entire unit (detector, sample chamber, pump, etc.) to be enclosed in a stainless steel box, and mounted on a steel base plate for bolting to concrete floor. Box to be flanged with 200 psi 1/2 inch (inlet and outlet) pipe connections. Electrical connections to be made through hermetic seal connections, both for power (motor and detector HV) and signal.

1.3.5.6 Hydrogen Detector Design Requirements

Applicable Codes - ASA Code for Pressure Piping

1.3.5.7 Sodium Flow Measurement Design Requirements

1.3.5.7.1 Applicable Codes

ASA Code for Pressure Piping

1.3.5.7.2 Package Limits

4' x 4' x 6' (largest)

1.3.5.7.3 Structural Criteria

The structural requirements will consider the steady-state and abnormal modes of operation of the sodium heat transfer system with respect to electrode attachment and supporting structure. External loads from pipe movement and magnet weight shall be considered.

1.3.5.7.4 Materials

Magnetic circuit to be constructed from alnico V and 1010 stl. The throat is the sodium pipe and as such as 304 SS.

1.3.5.8 Reactor Sodium Level Measurements Design Requirements

1.3.5.8.1 At least the following documents apply to system design.

SDD-096-10-000	(Overall System)
SDD-096-20-000	(Reactor Segment)
SDD-096-30-000	(Heat Transfer Segment)
SDD-096-50-000	(Control & Instr. Segment)

1.3.5.8.2 The system interfaces with the reactor vessel, top shield, core cables and connectors, preamp and control room, reactor overflow tank and level control system, plant protective system, plant annunciators and data system, and electrical system.

1.3.5.9 Delayed Neutron Monitoring

1.3.5.9.1 At least the following documents pertain to system design:

10 CFR 20	(Radiation Protection)
NEMA Standards	
National Electrical Code	
USASI Std. N2.6	(Admin. Procedures)
SDD-096-10-000	(Overall System)
SDD-096-20-000	(Reactor)
SDD-096-30-000	(Heat Transfer)

1.3.5.9.2 Principal interfaces are with reactor vessel, top shield, reactor building, reactor overflow tank, preamp and control room, plant annunciators, plant data systems, electrical system, and possibly H&V system (if detector forced cooling is required).

1.3.5.9.3 Materials of construction include stainless steel for piping and cylinder (5-10 liters), lead for shielding, polyethylene and graphite for moderator, steel and copper for EM pump, and insulating material.

1.3.5.9.4 Stainless steel isolating valves are required. Design should include provision for remote handling of structure (including detectors, pump, shielding and moderator) between shutoff valves.

1.3.5.9.5 Provision should be made for easy replacement of detectors (wired in parallel). Design should include provision for calibration and check of detectors in place.

1.3.5.9.6 Electrical connections to system shall be made through plugs.

1.3.5.10 Radiation Monitoring Design Requirements

1.3.5.10.1 At least the following documents pertain to system design:

10 CFR 20	(Radiation Protection)
10 CFR 100	(Siting Criteria)
NEMA Standards	
National Electrical Code	
USASI Std. N2.6	(Admin. Procedures)
SDD-096-10-000	(Overall System)
SDD-096-30-000	(Heat Transfer Segment)
SDD-096-60-000	(Auxiliary Systems Segment)
SDD-096-80-000	(Reactor Support Systems Segment)
SDD-096-90-000	(Land Structures & Conventional Systems Segment)

1.3.5.10.2 Principal interfaces are with R/A Vent System, R/A Liquid Waste System, H&V System, Main and Local Control Boards, and Electrical System (120/208).

1.3.5.10.3 At least the stack or pre-stack monitor should be redundant.

1.3.5.10.4 Except as noted, detectors will be furnished with internal calibration sources.

1.3.5.10.5 Radiation alarms (lights, horns, and annunciator drops) shall be provided locally for all instruments in occupied areas (in addition to those provided in the control room).

1.3.5.10.6 Detector location and evacuation alarms shall be coordinated with plant communications system design.

1.3.5.10.7 Provision shall be made for manual operation of system controls with suitable interlocks.

2.0 DETAILED DESCRIPTION

2.1 Plant Control System

The system is a direct energy balance scheme wherein sodium and feedwater flows are nominally set proportional to power, and vary as power demand varies. There are minor changes in setpoints based on pressure and temperature. A schematic diagram of the system is shown in Figure 6.1.2. Referring to the schematic, in the lower right hand corner there is a large box marked "plant load computation". The inputs to this are from generator demand and from a steam pressure controller. The latter signal can be both additive to the generator demand signal in case steam pressure is too low or subtractive if pressure is too high. (The generator demand represents most of the signal and has the dominant time constant; the pressure trim signal is nominally zero in steady state.) The sum of these two signals is called plant load demand (similar to the master index in a fossil-fired boiler), and establishes setpoints for all of the Heat Transfer Control System and parts of the Reactor Control System. (These are alternative ways of describing the Plant Control System: Set-Point Programming, or Supervisory Control.) Consistent with the use of hand-control stations across all controllers, it is possible for the plant operator to set PLD manually, which he would nominally do only for startup or under circumstances where the turbine would be operated reactor-following.

The important points to be noted in this summary description are that setpoints are calculated for each of the local control subloops, and that the operator can set any of these manually through individual hand control stations. Whether a separate division-of-load computer is required depends upon the scheme of mechanization chosen. Computers may take a variety of forms between special purpose analog equipment and general-purpose digital computer, and should not be given too narrow an interpretation.

The Heat Transfer Control System then supplies secondary sodium flow according to PLD; primary sodium flow to PLD plus a trim on reactor inlet temperature; feedwater flow to PLD plus a trim on secondary cold leg temperature (and secondary flow during transients); main and reheat steam and reheater exit sodium temperatures are controlled according to PLD, as is feedwater pressure. Or, if the operator desires, the final elements (valves or pumps) may be set manually.

The Reactor Control System is shown on the left-hand side of Figure 6.1.2. The control elements consist of absorber rods inserted into the core to reduce power and pulled out to increase it. They are positioned by a contactor (on-off) controller in response to a flux feedback. This is called the flux controller, and shown attached to the control rod drives.

The main index for the flux controller is total primary sodium flow, which represents power. There is actually a flow-delta T computation performed, but the delta T is set by PLD. During normal operation delta T is nearly constant, but in any case merely represents a scaling factor on measured flow. In addition to the main index there is a trim signal based on fuel channel outlet sodium temperature for short term and on bulk sodium temperature for long term control.

Two additional subsystems are not shown on the system diagram. These are the Reactor Level Control system and the Rod Cooling System. The latter is required in order to properly match temperatures in control rod channels, fuel storage areas, and the reactor vessel. Sodium supply for cooling comes from three pipes each of which is teed from the main inlet lines. Each line contains a control valve operated from a common controller sensing an average of control rod coolant temperatures.

The reactor Level Control maintains a constant level in the reactor by means of an overflow tank and pump-back system.

2.2 Plant Protection System

The essential facts of the protection system are briefly shown in Figure 6.1.1, which is an input/output matrix of system performance. Primary protection of the core and the public is given by three levels of operation of control rods, in response to signals (flux, flow, temperature, level) from the reactor. The first level of protection is called the "inhibit-alarm". This simultaneously gives audible and visible alarm, and prevents withdrawal of control rods. The second level is called "setback". Under setback the shim-safety rods are driven into the core at a high speed (possibly as much as 50¢/second) until the condition causing the setback is cleared. The third and last level is called "reactor trip". When this is activated all holding magnets are released and both the safety and shim-safety rods drop into the core. Simultaneously all rod-drives act to drive the rods into the core, and the convection control system is actuated.

The turbine is tripped whenever the reactor is, but the converse is not true. The expected method of operation on turbine trip would be to reduce the plant load demand signal in the control system (which reduction takes place automatically by reason of reduced load) on a maximum schedule dumping steam to the capacity of the steam dump, with excess steam blown off through the power relief system (automatic under the action of high pressure).

Another action of the system, intended to protect the heat transfer system and turbine initially, is the automatic - or operator - shutdown of one circuit (a circuit comprising the primary and secondary sodium loop, and the feedwater - steam loop making up a connected portion of the heat transfer system). Conditions causing this action are major mismatches

in flow which would lead to temperature shocks in the heat transfer system - and fairly soon thereafter in the reactor - if not quickly corrected. The one-circuit shutdown is also caused by activation of the sodium-water-reaction relief system in this case the turbine is also tripped for the short time required to isolate the offending circuit from the turbine. The other two circuits would operate as in normal turbine trip.

The last action shown is the "plant power limit alarm", which is a special display unit that alerts the operator that he cannot increase the power of turbine-generator, or that he should take action to reduce power before the initial pressure regulator does. The display consists of five indicators, four of which show totals of feedwater flow, secondary and primary sodium flow, and reactor flux. The fifth comprises plant power and floating alarms established by plant power. The unit serves a second function in the case of turbine-following operation, which may be occasionally required for on-line equipment maintenance.

2.3 Core Flux Measurements

The measurement of flux in the reactor during initial and subsequent loading operations consists of two parts: one for performance and one for safety.

2.3.1 The performance part (further indentified as Channel 10) requires a special fuel element in which a neutron sensor is inserted from the top and removed during power operation or when the element is moved to other locations in the core (enough to adequately map the core).

2.3.2 The safety part consists of at least three detectors (further identified as Channels 7, 8, 9) located in thimbles. Detectors will be located close to the top of the axial blanket, but without interfering with rotation of the top shield. Interlocks for the fuel handling machine will be provided (period and level). Trip signals for the safety system will be provided during critical experiments.

2.4 Reactor Temperature Measurements

Fuel element outlet temperatures will be measured by Type E Chromel-Constantan thermocouples, Inconel sheathed, MgO packed type, and will be inserted into closed thimbles called thermocouple guide tubes. These are welded into the top shield structure, and closed at the top with a compression seal which will allow T/C replacement. The guide tubes will be part of the control rod shield plug and dummy control shield plug assemblies, and will be filled with a liquid to enhance response. Each channel will be provided with one guide tube containing three thermocouples, which will be 1/8 inch OD.

All fuel element outlets will be rapidly scanned by the core scanner-logger (estimated 200 points per second) in order to detect over-heating. A selected few will be used by the protective system to shut down the plant if a gross temperature excursion occurs. Other selected elements will be used by the reactor control system.

The other thermocouples will be of the same type, and will be installed permanently. The grid plate thermocouples will be routed within the core vessel to the selected temperature measuring points. These thermocouples will have limited life due to the radiation exposure they will receive; expected life will be less than one year of full power operation. If after this time additional information is needed, temperature sensors can be inserted into the grid plate area by means of dummy control rod shield plugs.

2.5 Reactor Cover Gas Monitor

This system consists of an auxiliary loop off the cover gas plenum, a vapor trap, a particulate filter, detector assembly and a second vapor trap. The detector output will feed to a multichannel analyzer and scaler in parallel with a count rate meter, and then to the plant data system, annunciator system, cover gas cleanup control system, and R/A vent system.

2.6 Hydrogen Detectors

The reference design for the hydrogen detector employs a nickel diffusion membrane in contact with the sodium on one side and a maintained vacuum on the other. The vacuum is maintained at a constant pressure by a pressure regulating valve and vacuum pump. A Pirani gage located in the constant pressure area is used to indicate the presence of hydrogen by utilizing the thermo conductivity properties of gases. An alarm would be actuated when the hydrogen exceeds some level above background. Background is caused by hydrogen from various sources, (e.g. water disassociation in the steam generators) other than the water-sodium reaction due to leaks.

2.7 Sodium Flowmeters

It is anticipated that electromagnetic flowmeters will be used to measure primary and secondary sodium flow. Remote flux measurements will be made, if required, by using a flip coil.

2.8 Reactor Sodium Level Measurements

Sodium level measurements will be made by two plug assemblies, each containing four thimbles. They will protrude into the sodium well below the minimum 350F level, but enough above the top of the core that rotation of the top shield can be accomplished. One of the thimbles in each plug will normally be used for a dip-stick probe, and will require an off-set in the shielding region to reduce direct streaming. It is desired to be able to probe down to the lower grid plate.

Alarm coils and continuous coils will be provided. The dip stick probes will be the tri-coil assemblies.

The system includes the following:

2.8.1 Four channels each to be capable of annunciating a high and low level condition and providing a low-low level shutdown signal to the plant protective system.

2.8.2 Two continuous channels to provide level information to the operator and to the reactor level control system.

2.8.3 Portable dip-stick probes for use in calibrating the above channels and for probing the sodium level during fill and draining of the reactor vessel.

2.8.4 Plugs and thimbles for the above.

2.9 Delayed Neutron Monitoring

Connections are made to the main reactor outlet sodium pipes (at least two). The sodium is pumped through the delayed neutron monitor located in a trench at the operating level, and thence to the reactor over-flow tank.

The detector system comprises multiple BF_3 chambers (similar to 20th Century 31EB70G, 1" detectors for 250 C operation but operated under 300 F) in parallel, polyethylene moderated, with lead and graphite shielding to reduce background.

2.10 Radiation Monitoring System

Items in this system are briefly discussed below.

2.10.1 Principal effluents are gaseous and are vented through the stack, since the building is under slight negative pressure. The vent gas system consists of detectors, compressors, and diverting valves. The latter divert the gas streams to holdup tanks on high radiation in the stack or inlet lines to the stack, or on low flow. The stack contains an isokinetic nozzle for flow measurements.

2.10.2 The liquid waste system monitors the liquid waste from drains. The latter is either released to a leaching field or is concentrated and shipped out by truck.

2.10.3 Primary cell monitors determine the activity concentration of the nitrogen atmosphere in tank areas, pipe chases, IHX cells, cold trap cells, and pump cells.

2.10.4 Secondary loop monitors measure activity of sodium in the secondary system. The most probable source of this would be a combination of an IHX leak and higher primary pressure than secondary.

2.10.5 Area monitors are permanently mounted ionization chambers, and cover both normally non-R/A areas (control and preamp room, high-bay, change room), and normally low active areas, (local boards, maintenance cell, decon room, control rod drives).

2.10.6 Air particulate monitors are portable self-contained units with pumps and moving filter paper systems. They will monitor at least the high bay, control room, change room, and fuel handling and storage area.

2.10.7 Criticality monitors are installed in the fuel storage and handling areas.

2.10.8 Portal monitors (hand and foot counters) are installed at reactor building exits, at man-lift near the liquid waste system, fuel storage and handling exits, and change room.

2.10.9 Monitors are installed on the fuel handling machines.

3.0 PRINCIPLES OF OPERATION

3.1 Plant Control System

3.1.1 The Plant Load Computer is a setpoint generator for the nuclear steam supply system. It translates the requirements of the turbine generator (as determined by measured load and turbine throttle pressure) into setpoints for all the control subsystems of the plant control system.

3.1.2 The Feedwater Pressure Control System maintains the feedwater pressure required to achieve a satisfactory differential pressure (approximately 50 psi) across the valves in the Feedwater Control System. Pressure is scheduled with load and maintained by varying speed of the hydraulically coupled shaft-driven boiler-feed pump.

3.1.3 The Feedwater Control System maintains desired flow in a particular circuit by throttling with regulating valves. Valve control is required because the circuits operate with different heat transfer and load characteristics. The system is three-element, with the third element made up from a temperature computer to maintain secondary cold leg temperature at its programmed value, with an additional signal from secondary sodium flow (a filtered rate or "kicker" which adds a feedforward signal on flow change).

3.1.4 The Reheat Temperature Control System maintains temperature by attemperation with cold reheat steam through control valves.

3.1.5 The Main Steam Temperature Control System maintains temperature out of the superheater by desuperheating with feedwater between the evaporator and superheater. Final control is with valves.

3.1.6 The Reheat Sodium Temperature Control acts to maintain a reasonable mixed mean temperature of sodium coming from the reheater and from the superheater. It is a compromise between mixing temperature and minimizing reheat attemperation. Final control is with valves.

3.1.7 The Secondary Sodium Flow Control maintains desired flow by varying secondary pump speed in response to flow demand set by the plant load computer.

3.1.8 The Primary Flow Control System is quite similar to the secondary except that it additionally acts to maintain the secondary hot leg temperature on a programmed-with-load schedule. This schedule reduces secondary hot leg slightly with reduced load in order to minimize desuperheating of main steam. The temperature schedule is derived from PLD and matched against measured secondary hot leg temperature to determine an incremental flow change (actually an incremental change in demand flow, which added to the demand from PLD makes up the actual demand). This trim control is direct-acting in that a reduction in primary flow will cause -- among other actions -- a reduction in secondary hot leg temperature. Such temperature measurements are derived from thermocouples installed in thermowells in the discharge piping of the IHX on the secondary side.

3.1.9 Convection control is by throttle valves, and is a flow control is cascade with a reactor outlet temperature control. It operates only during plant shutdown and is actuated by the Plant Protection System. It functions to maintain a smooth rate-of-change in the reactor outlet temperature, following reactor trip.

3.1.10 The Reactor Temperature Control acts to maintain the reactor outlet temperature. During normal plant operation (as opposed to convection control) it acts through the flux controller. It is a cascade system whereby the plenum outlet temperature, with a full-load time constant of about 30 seconds, establishes the setpoint for fuel channel outlet temperature.

3.1.11 The Flux Controller maintains desired neturon flux by positioning control rods according to demand. This demand is primarily the steady-state power as set by total primary sodium flow. Actual power is proportional to flow times actual reactor delta T. Since reactor delta T is nearly constant power is nearly proportional to flow. An added change in flux is demanded from the Reactor Temperature Control, increased for low outlet temperature and decreased if too high.

3.1.12 The Rod Cooling Control acts to maintain control rod exit sodium temperature at proper values by regulating throttle valves on lines to the low pressure plenum in the reactor. The system is partially self-regulating in that there is natural circulation between the sodium in control rod channels, the fuel storage areas, and sodium cooling the reactor vessel - all of which are connected to the low pressure plenum. Only partial control of these temperatures is by throttling; unregulated leakage flow from the high pressure plenum to the low pressure plenum will be a considerable contribution and may be dominating.

3.1.13 The Reactor Level Control is a continuous overflow to the reactor overflow tank. Pumpback at scheduled flow-rates at the reactor inlet lines maintains the level and continuous overflow.

3.2 Plant Protection System

There are two items of operational significance in this category, and these are the dT/dt computer and the self-checking circuits. Other parts of the system comprise relays, contact closures, and comparison circuits.

3.2.1 The dT/dt Computer The high temperature circuit consists simply of a comparison, and is not further discussed. The rate circuit is part of the same unit and makes up a filtered rate circuit with input-output relationship of the form

$$e_o/e_i = KTS/(1 + \tau_s)$$

The unusual feature of the circuit is that it is designed for time constants covering 20 minutes to four hours.

The response of this circuit to a step-change at the input is initially a step followed by an exponential decay. The steady-state value is zero (hence zero-centered meters to show both positive and negative changes). If the scaled value of the trip setting is say three volts, then the circuit will always respond when the difference between input and output reaches three volts.

The response to a ramp is initially a ramp with slope zero, but with increasing slope that is proportional to the difference between input and output. The steady-state value is a ramp of the same slope as the input, but delayed in time (following error). Steady-state is reached after about three time constants. However, the trip point is still reached at three volts between input and output. This can thus be achieved with a step-change of some value, X, in temperature or at a value of about 3X/hr sustained for three time constants. (The choice of the time constant depends upon response of the physical system in the plant being protected, which was the reactor vessel and set at 20 minutes in Hallam.) Trip can also occur on higher rates

for shorter times, but must persist long enough to exceed the step-change allowable. On the other hand, step changes of less-than-trip amounts can be made, and provided no further changes are made for a while, the output will reduce to safe values (33% in one time constant and under 5% in three). This is exactly analogous to the stresses in a large plate when a differential temperature is created across it, and is the basis for the circuit design.

3.2.2 Self-Monitoring Circuit

The module acts to make normally passive circuits into active ones (by making dc circuits into ac), and has a scheme for annunciation of failures. It is important to note that this is not just overall input-output (particularly since the "black-box" is a two-state device), but detailed information on component failure.

The circuits and logic used are such, so that loss of any signal or overly high signal (still well below saturation) will cause the module to be in the alarm state, will also indicate on the face of the module (front-of-board so that the operator will see it), and will also annunciate the state of the module (lighted plaque and horn, bell, or chime).

3.3 Core Flux Measurements

Operating principles are those of ion chambers and fission counters, which have been calibrated to give a particular current flow or count rate when in a specified flux field.

3.4 Core Temperature Measurements

Principle of operation involves thermocouple thermometry.

3.5 Reactor Cover Gas Monitor

Essential components are a scintillation counter and multi-channel plus-height analyzer (comparable to TMC's gammascope II, model 102, which is a 100-channel analyzer). Normal operation will be broad-band, or narrow band probably set for Xe-133 or KR-85. Periodically or upon alarm, the total energy range will be scanned.

3.6 Hydrogen Detector

The principles of operation are given in the discussion of paragraph 1.2.6.

3.7 Sodium Flow Measurement

The electromagnetic flowmeter operates on the principle of Faraday's Law of Induction. Hence, when the conductor (liquid sodium) moves through a magnetic field, an electromotive force is induced across the sodium proportional to its velocity.

3.8 Reactor Sodium Measurement

The system is operated with a pair of coils in a bridge. The coils themselves are surrounded by an electromagnetic field caused by current through the coil. When sodium surrounds the coil, the impedance of the coil changes because of the changed field reluctance (similar to shorted secondary transformer). The use of two coils in a balanced bridge allows for convenient means of temperature compensation.

The continuous system actually comprises multiple coils in a stacked array, with one coil fixed as a reference and the other coils introduced in the bridge by switching. All coils above the coil partly covered with sodium (actual contact with the sodium does not occur) will read downscale, coils below will read upscale, and the coil being used will show intermediate readings.

3.9 Delayed Neutron Monitoring

When a physical breach is made in the fuel cladding it becomes possible for fuel and fission products to enter the coolant stream within the fuel bundle and then get mixed into the upper plenum. Some of these will emit delayed neutrons; those with an average half-life of about 22 seconds appear with concentrations of 0.002-0.005 n/fission (depending on isotope). Side streams can be taken from the outlet and pumped in 10-20 seconds to a detector system where they are resident for another 5 seconds. Neutrons given off are then thermalized and counted. The resulting count rate then depends upon the many variables of size of the rupture, dilution through mixing, transport delays and counting efficiency.

In addition to neutrons due to breaches in fuel cladding there is a background of neutron emitters in the coolant, and also a background at the detector location. Hence, careful attention to shielding is required. In tests run in EBR-II, however, a reported signal to noise ratio of 6.7 was achieved, with 17 second response. A test of unclad fuel showed a contribution of 38% to signal, with 13% from tramp uranium, and 49% from Na ²⁴. Periodic sampling of the sodium stream, followed by decay of Na ²⁴ and subsequent analysis for fission products will develop a system calibration. So the system, in principle, and through experiment, appears feasible. What remains to be determined is the relationship between the size of failures or kinds of failures in the core and observed count rate of the system.

3.10 Radiation Monitoring System

Principles of operation involve basic radiation detection devices consisting of G-M tubes, ion chambers, and scintillation counters. These devices measure direct radiation, activity in a gas stream, in a water pipe, or on a fixed or moving filter paper. Readouts are generally on count rate meters or with counting room techniques for very low level emitters.

4.0 SAFETY PRECAUTIONS

4.1 Plant Control System Safety Precautions

These are described in 1.3.4.1 above.

4.2 Plant Protection System Safety Precautions

The system is intrinsically safe in that it comprises only static equipment operating at low voltage. Thus, any safety precaution relates to proper performance of its safety function.

4.2.1 Safety will be enhanced by early recognition that real safety must be developed in hardware and not in specifications or reports about safety.

4.2.2 Safety will be enhanced by early recognition that the safety hardware must be installed and operate in the real environment of an operating power plant and not in development labs.

4.2.3 Safety will be very much enhanced by early feedback between system studies and actual plant hardware, both to establish and to verify adequacy of performance and requirements.

4.3 Core Flux Measurements Safety Precautions

4.3.1 Shielding and administrative procedures are required to avoid exposure during handling and operation of this system.

4.3.2 A careful operational check of the system - which is a major part of the protective system during shutdown - must be made prior to fuel handling operation.

4.3.3 Administrative procedures (or electrical interlocks) should be provided to assure that the in-fuel detectors are not in the core before the top shield is rotated (a good mechanical sweep arm appears adequate).

4.3.4 Caution is required in handling detector high voltage and power cables.

4.4 Core Temperature Measurements Safety Precautions

See paragraph 1.3.4.4.

4.5 Reactor Cover Gas Monitor Safety Precautions

See paragraph 1.3.4.5.

4.6 Hydrogen Detector Safety Precautions

The safety considerations involving the hydrogen detectors are essentially the same as those of the steam generator complex since their prime function is to provide early detection of leaks and minimize the extent of damage through operator action. The main safety considerations are protection of loss of heat sink for the reactor and the risk to operating personnel. The hydrogen detector must stand pressure of steam generator tube rupture.

4.7 Sodium Flow Measurement Safety Precautions

See paragraph 1.3.4.7.

4.8 Reactor Sodium Level Measurements Safety Precaution

See paragraph 1.3.4.8.

4.9 Delayed Neutron Monitor Safety Precaution

See paragraph 1.3.4.9.

4.10 Radiation Monitor Safety Precautions

4.10.1 Since the detectors are installed (in some cases) in liquid or gas lines expected to contain radioactive products, attention to installation detail is required, and the possibility of exposure is high. Where detectors form a part of reactor containment structure this is particularly so.

4.10.2 Entering normally unoccupied areas should not be dependent on (assumed) operation of permanently installed detectors.

4.10.3 Dilution factors and system sensitivity should be carefully checked to assure that allowable releases or release rates are not exceeded.

5.0 MAINTENANCE

The general maintenance policy shall be that all repairs are made on-line.

5.1 Plant Control System

The maintenance policy shall be that all repairs are made on-line. Exceptions to this occur only in final control elements (valves, pumps, and control rods). This is interpreted as a design requirement that all instrument subsystems be furnished with disconnect plugs for power feed, and preferred use of modular concepts.

5.2 Plant Protection System

5.2.1 All maintenance shall be performed on-line. (See also 1.1.2.17.) This is not to be construed that major maintenance cannot be performed during scheduled shutdowns for refueling or turbine overhaul.

5.2.2 See 1.1.2.18

5.3 Core Flux Measurements

Maintenance is only of moderate concern since the units are not expected to be used during power operation. However, the units must be conveniently replaceable during shutdown operation, and this will require some remote handling.

5.4 Core Temperature Measurements

On-line maintenance is a design objective for all instrumentation. It has been suggested as a requirement for channel outlet T/C also, but this partly depends upon seal design requirements and the extent to which a primary reactor containment barrier can be breached during operation. On-line replacement would allow for shorter failure rates. If on-line replacement is not allowed, then the refueling cycle is also a variable.

There is still some question of the maintenance policy as discussed above. However, it is clear that remote handling is required for removal of failed thermocouples. Tooling requirements for insertion of new elements are not so clear, and remain to be worked out.

5.5 Reactor Cover Gas Monitor

Access to equipment is required in order to be able to perform on-line maintenance, primarily circuit testing and filter replacement.

5.6 Hydrogen Detectors

Provisions shall be made for case of repair and in-place calibration of readout instruments.

5.7 Sodium Flow Measurement

It is anticipated that little if any maintenance to the E.M. flowmeters will be required. In general, the supporting structure will be designed to facilitate removal of the magnet.

5.8 Reactor Sodium Level Measurements

On-line maintenance of the system electronics is required. It is assumed that on-line replacement of coil assemblies would normally not be allowed.

Expected maintenance is nominal, consisting of periodic calibration. (Solid state power supplies of adequate rating are expected to cure the Hallam problem.)

5.9 Delayed Neutron Monitor

Provision for on-line maintenance is required. This will require capability to isolate the system (valving), and to replace the detector assembly (in the event of leaks).

5.10 Radiation Monitoring

The stated policy on plant instrument maintenance is that it be performed on-line. Except for sensors located in normally high radiation areas, there appears to be no problem in achieving this requirement.

6.0 APPENDICES

6.1 Drawings

6.2 References

- 6.2.1 Federal Register, Vol. 32, No. 132, Tuesday, July 11, 1967, pp. 10213-10218, "Proposed Rule Making" (Proposed Amendments to 10CFR50).
- 6.2.2 ANL 7110, "Reactor Physics Division Annual Report", page 295
- 6.2.3 ANL 7067, "Cladding Failure Simulation Tests in EBR-II, "Smith and Doe, December 1966
- 6.2.4 NAA-SR-5313 (Rev.), "Development of Temperature Protective Circuits for the HNPF fuel Channel Exit"
- 6.2.5 Smith, Doe, and Porges, "Fuel Element Failure Detection in EBR-2," Trans. ANS 8, 2 (Nov. 1965).

— ACTIONS TAKEN —											
INITIATING EVENTS		STEAM GENERATOR ISOLATION PLANT POWER LOAD REDUCTION AUTOMATIC ONE-CIRCUIT ISOLATION START POWER LIMIT ALARM ACTUATE EMERGENCY PUMPING INHIBIT CONVECTION CONTROL ROD WITHDRAWAL TRIP REACTOR TRIP									
		STEAM GENERATOR ISOLATION	PLANT POWER LOAD REDUCTION	AUTOMATIC ONE-CIRCUIT ISOLATION	START POWER LIMIT ALARM	ACTUATE EMERGENCY PUMPING	INHIBIT CONVECTION CONTROL	ROD WITHDRAWAL TRIP	REACTOR TRIP		
NUCLEAR INSTRS DOWNSCALE											
NEUTRON FLUX > 3%											KEYLOCK CUTOUT ABOUT 1% POWER
NEUTRON FLUX > 5%											
FLUX-FLOW RATIO > 1.20											INCLUDES FLUX > 115%
FLUX-FLOW RATIO > 1.35											INCLUDES FLUX > 120%
SOD. TEMP. STEP CHANGE > 65F											
SOD. TEMP > 1100°F											
SOD. TEMP > 1150°F											
SHORT PERIOD < 30 SECS.											CUTOUT ABOVE 5% AFTER 30 SEC. DELAY
SHORT PERIOD < 20 SECS											
SOD. LEVEL 1/2 FOOT LOW											
SOD. LEVEL 1 FOOT LOW											
ROD DRIVE EXERCISE SW. ON											KEYLOCK
BUILDING CRANE POSITION											MUST CLEAR RODS
FUEL HANDLING MACHINE											MUST BE LOCKED IN OPERATING POSITION
TOP SHIELD HOLD DOWN											
SODIUM-WATER REACTION											
ONE BLOCK-VALVE CLOSED											
ONE PUMP OFF											
ONE-CKT FLOW < 10%											
TWO CIRCUITS ISOLATED											
SAFETY ROD NOT WITHDRAWN											
SHIM-SAFETY ROD DOWN											FLUX ABOVE 27%
SHORT NEGATIVE PERIOD											
TURBINE-GENERATOR FAULT											
FLUX < GENERATED POWER											5-10% LOW
FLOW < GEN POWER ②											
MACHINERY ACCESS DOOR OPEN											KEYLOCK
REACTOR START SW. OFF											
MANUAL											

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PROTECTION SYSTEM FUNCTIONAL DIAGRAM

Figure 6.1.1

Figure 6.1.2

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Figure 6.1.3
Nuclear Instrumentation Schematic

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