1000-MWe LMFBR FOLLOW-ON STUDY
— Task II and III Final Report —
Volume 3
Conceptual System Design Descriptions
(Sections 8 - 14)

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PREFACE

Babcock & Wilcox (B&W) is participating in the AEC's 1000-MWe Fast Breeder Reactor Follow-On Study Program under Argonne National Laboratory (ANL) Contract No. 31-109-38-1998. As stated by ANL,

"The over-all objective of the current 1000 MWe LMFBR Follow-On Study Program is to provide the AEC with information needed to establish research and development programs which will lead to the achievement of economic and safe LMFBR power plants by a viable nuclear industry. This information is to be developed through studies of 1000 MWe LMFBR power plant reference conceptual designs favored by the nuclear industry for further development and eventual sale to the electric utilities in the period of 1975 to 1985."

The B&W Follow-On Study consists of four tasks:

Task I — The development of four basic plant concepts based on the established ground rules and the estimated state of technology in 1980, and the selection of a preliminary reference design that incorporates the best features of each concept.

Task II — Detailed engineering work on the preliminary reference design, and, in parallel, the performance of in-depth engineering trade-off studies.

Task III — Parametric studies to identify and evaluate basic problem areas and to provide a basis for the selection of an optimized reference design.

Task IV — Preparation of a document outlining the research, development, and component testing required for the successful design and construction of the reference 1000-MWe LMFBR design.

This five-volume report describes the work performed under Tasks II and III. Volumes 2 and 3 contain the 14 individual conceptual system design descriptions (CCDDs) for the reference 1000-MWe LMFBR central

station power plant. The seven CSDDs presented in Volume 3 are as follows: (1) reactor auxiliary, (2) intermediate coolant, (3) steam generator, (4) energy conversion, (5) instrumentation and control, (6) electrical, and (7) safety systems. The titles and a brief description of the contents of Volumes 1, 2, 4, and 5 are listed below.

Volume 1 — Summary Description and Cost Estimate

Cost data and a description of the overall plant and the reference operating conditions and parameters.

Volume 2 — Conceptual System Design Descriptions (1 - 7)

Descriptions of land, structures, and conventional systems and the reactor and associated systems, such as shielding, control, and supports.

Volume 4 — Trade-Off Studies

A compilation of the various trade-off studies that were carried out to evaluate alternate choices for the conceptual design.

Volume 5 — Parametric Studies

Descriptions of the parametric studies carried out under Task III.

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Section 8

REACTOR AUXILIARY SYSTEMS

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A3117E 1000-MWe LMFBR - Reference Concept Decay Heat
Removal System, Schematic Diagram

8. REACTOR AUXILIARY SYSTEMS

8.1. General

The auxiliary systems perform essential support functions in B&W's 1000-MWe LMFBR reference plant. The auxiliaries include the decay heat removal system, the auxiliary cooling system, the radioactive waste disposal system, and the preheat system.

The decay heat removal system is primarily concerned with the maintenance of prescribed reactor temperatures during normal periods of operation when the plant output is reduced to zero; however, special provisions are included in the design to handle emergencies and so that the system can be used for preheating the reactor system during startup.

Many areas in a power plant require a positive means of preventing waste heat from creating an intolerable condition for delicate equipment (such as the plant computer and control instrumentation in the computer room), temperature sensitive structural materials, and manned access. The auxiliary cooling system removes waste heat from such areas and rejects it to the service water system.

Radioactive solid, liquid, and gaseous wastes produced in a reactor must be processed and disposed of in a manner that prevents an accidental release to the environment. Section 8.4 describes the system adopted to handle wastes from the 1000-MWe LMFBR reference design plant.

All systems handling or receiving liquid sodium must be preheated and maintained at a temperature in excess of 208 F to prevent the solidification of sodium. Preheating needs can be divided into two areas: system restart and plant commissioning. Section 8.5 is primarily devoted to describing commissioning preheating although system or loop restart needs are discussed whenever an interface exists.

8.2. Decay Heat Removal System

8.2.1. Introduction

8.2.1.1. System Functions

The primary function of the decay heat removal

system is to remove fission product decay heat from the reactor after (1) an anticipated reactor shutdown, (2) an unanticipated reactor shutdown, and (3) any accident up to and including a design basis accident (DBA). In addition, the system will be capable of manipulating the reactor coolant temperatures during plant shutdown; i.e., add or remove heat as required.

8.2.1.2. Summary Design Description

The decay heat removal system consists of two similar closed loops which are separate and independent. Each loop has a decay heat removal capability of 171×10^6 Btu/hr* and incorporates the necessary components, instrumentation, and controls required for operation while performing any of the designated decay heat removal functions. The relationships between the decay heat removal components and the reactor vessel system, reactor coolant system, and the reactor shielding system are shown in the overall schematic diagram A31117E.** The locations of the Na-NaK heat exchangers are shown in B&W drawing A31077E; the reactor vessel, pit liner, and biological shielding are shown in drawing A31078E.** The major components of each loop are the electromagnetic pumps, storage tank and NaK-air heat exchanger, one two-pass NaK-air heat exchanger, three 50% -capacity electromagnetic pumps, and biological shield cooling coils.

The connecting inlet and outlet piping incorporated into the loop for supplying the NaK flow to the Na-NaK heat exchanger is doubly contained within the reactor building and is equipped with electrical trace heating; the inlet and outlet isolation valves are located outside and adjacent to the reactor building. These valves can isolate the Na-NaK heat exchanger from the loop during a post-DBA decay heat removal operation or during the malfunctioning of a heat exchanger. The remainder of the loop piping outside of the reactor building is insulated with 4 inches of standard insulation and equipped with the required trace electrical heating.

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* See Appendix 8-A for a discussion of the sizing philosophy for the decay heat removal system.

** B&W drawing A31117E is placed at the end of this section, and drawings A31077E and A31078E are included in Volume 1 of this report.

The NaK-air heat exchanger, together with the propane heater and the four 100,000 scfm, propeller-type fans, is located on the top of the reactor building. An NaK surge tank equipped with inert cover gas, inlet and outlet argon lines, and pressure relief valves is located at the top of the two-pass NaK-air heat exchanger. The surge tank is sized to accommodate the change in volume of the NaK heat transport medium within the loop due to thermal expansion during loop operation. A single NaK storage tank, which is isolated from the system during operation, is used for initial filling of the loops.

Major design data for the decay heat removal system are summarized as follows:

Number of loops	2
Thermal duty per loop, Btu/hr	
Normal	171×10^6
Emergency	171×10^6
Post-DBA (biological shield coils)	85.5×10^6
NaK flow rate per loop, lb/hr	
Normal	1.75×10^6
Emergency	1.75×10^6
Post-DBA (biological shield coil)	8.75×10^5
NaK inventory, total, lb	25,452
Material of construction	304 SS
Design temperature, F	900
Design pressure, psig	300
Number of cold traps per loop	1
Number of Na-NaK heat exchangers per loop	1
Number of NaK-air heat exchangers per loop	1
Number of EM pumps per loop (50% capacity)	3
Number of surge tanks per loop	1
Number of fans per loop	4

8.2.1.3. System Design Requirements

Performance

Both decay heat removal loops will be capable of rejecting to the atmosphere 171×10^6 Btu/hr of heat generated by the

reactor core and the activated reactor coolant during normal and emergency shutdown conditions.

During post-DBA decay heat removal operations, each decay heat removal loop will be capable of transferring and rejecting heat at a rate of 85.5×10^6 Btu/hr, utilizing cooling coils embedded in the biological shields.

Any one of the three 50% capacity electromagnetic pumps in a decay heat removal loop shall be capable of providing a NaK flow of 8.75×10^5 lb/hr to the circumferential biological shield cooling coils and the remaining closed loop configuration during a post-DBA decay heat removal operation.

Each of the four electric motor driven propeller-type fans shall provide 100,000 scfm of air flow to the NaK-air heat exchanger.

Environment

The Na-to-NaK heat exchangers will be designed to operate satisfactorily with reactor-grade sodium at temperatures up to 900 F.

Reliability/Availability

Two heat removal paths will be provided for emergency core cooling. These paths must be capable of fulfilling their functions in any situation that compromises the normal path of heat removal.

Maintenance

The system will be designed so that the components, including heat exchangers, pumps, fans, valves and interconnecting piping, can be removed and replaced.

Operations

Facilities for periodic testing of the delivery capability of the heat removal mechanisms will be provided as close to the core as is practicable. Provisions shall be included for testing, under conditions as close as practical to design condition, the full operational sequence that would bring the emergency heat removal mechanisms into operation, including the transfer to alternate power sources.

Safety

Means to safely relieve excess pressure in the decay heat removal system will be provided. All valves, motors, pumps, NaK-air heat exchangers, and the major portion of the interconnecting piping will be located outside of the reactor building in isolated and protected areas. All NaK piping in the reactor building will be doubly contained.

Isolation valves will be located outside and adjacent to the reactor building in order to isolate an Na-NaK heat exchanger from the loop in the event that it has been damaged. In the event of tube leaks in the Na-to-NaK heat exchangers, the leakage will be in the direction of the sodium.

The circumferential biological shield cooling coils will be located in the concrete at the interface of the biological shielding and the reactor pit liner, which protects the coils. In addition, the system will have instrumentation to indicate to the operator whether a malfunction has occurred within the decay heat removal system.

Design Standards

All system components that are designed to function in the reactor coolant environment or to contain liquid NaK will be constructed of 304 stainless steel.

The decay heat removal system will be designed and fabricated in accordance with the requirements of Section III, Nuclear Vessels, ASME Boiler Codes, and supplemented by the special requirements of creep, liquid-metal thermal shock, NaK corrosion, and mass transport. Standards not covered specifically by these codes will comply with accepted state-of-the-art material, fabrication, assembly, and inspection procedures used in the design and construction of nuclear vessels. The system will be designed in accordance with the earthquake criteria specified in Volume 1, section 2, Plant System Design Description. Other site effects, such as winds, ice, and floods, will receive adequate consideration in the design.

Purity requirements for the argon cover gas required in the surge tanks and storage tank will be as analyzed for commercial-grade liquid argon:

<u>Impurity, ppm</u>	<u>Maximum concentration</u>
O ₂	5
N ₂	20
Others	5

(Dew point, F -76)

(See Appendix 8-B for a discussion of NaK purity requirements.)

Interface

The heat removal mechanisms and the core will be designed to prevent damage to the fuel and cladding that would interfere with the emergency core cooling function. Since the removal of decay heat from the core by natural convection has not been investigated, it will be conservatively assumed that low-flow operation of the reactor coolant pumps will be mandatory.

8.2.2. Detailed System Description

The decay heat removal system contains two separate and similar decay heat removal loops which can be operated independently. Each loop incorporates one Na-NaK heat exchanger, one NaK-air heat exchanger, three electromagnetic pumps, a circumferential biological shield cooling coil, a surge tank, four fans, adjustable air flow dampers, a propane-fired heater, a complete control system, and the required interconnecting piping and valves necessary to complete the closed loop configuration.

The system performs normal decay heat removal during periods of intentionally initiated reactor shutdown. This function is accomplished by circulating the liquid NaK heat transport medium through a single-pass Na-NaK heat exchanger submerged in the primary reactor coolant, and through a two-pass NaK-air heat exchanger on the roof of the steam generator building. While this function is being performed, the liquid NaK heat transport medium is prevented from circulating through the circumferential biological shield cooling coils by the biological shield cooling coil inlet valve. This valve is normally closed and interlocked with the Na-NaK heat exchanger inlet and outlet piping isolation valves. Three 50%-capacity electromagnetic pumps provide an

NaK flow of 1.75×10^6 lb/hr within each closed loop configuration. Decay heat transferred to the NaK in the Na-NaK heat exchanger is transported through insulated piping to the NaK-air heat exchanger where the NaK rejects its heat to the air. The NaK then returns through insulated piping to the suction side of the pumps.

Emergency decay heat removal is also performed by this system during periods of unanticipated reactor shutdown initiated by the temperature sensing instrumentation associated with the protection system. This instrumentation automatically starts the decay heat removal system when it senses that the cold leg of the reactor coolant system has attained a temperature of 850 F.

This system also removes decay heat after a DBA. If both of the Na-to-NaK heat exchangers are damaged, then heat is removed by isolating the Na-NaK heat exchangers and using the electromagnetic pump to circulate the NaK through the circumferential biological shield cooling coils. These cooling coils, which are located behind the reactor pit liner in the biological shield, can remove the 85.5×10^6 Btu/hr of decay heat. Upon leaving the cooling coils, the NaK transports its heat through insulated bypass piping to the NaK-air heat exchanger where the heat is rejected for return through insulated piping on the suction side of the pumps.

In addition, the system is capable of supplying heat to or removing heat from the reactor coolant system in order to raise or lower the temperature of the entire reactor coolant system. Each loop in the system can perform this function independently during either reactor operation or reactor shutdown. Heat is added to the reactor coolant system by the actuation of propane heaters firing directly on the NaK coils in the NaK-air heat exchanger. The hot NaK is then transported to the NaK-to-sodium heat exchangers where it deposits the heat. The propane heaters can be actuated when the sodium temperature drops to 300 F.

8.2.2.1. Components

Sodium-to-NaK Heat Exchanger

The stainless steel sodium-to-NaK heat exchanger has a design operating temperature of 900 F and pressure of

300 psig. It provides a decay heat removal capability of 171×10^6 Btu/hr during normal and emergency decay heat removal operations. The heat exchanger receives an inlet flow of liquid NaK of 1.75×10^6 lb/hr at a temperature of 300 F. The cold NaK circulates through the heat exchanger and leaves at 700 F.

Thermal communication between the liquid NaK heat transport medium and the primary reactor coolant occurs in the single-pass shell-and-tube heat exchangers which are suspended from the reactor vessel cover structure and submerged in the primary coolant. Each heat exchanger has 453 straight, 1-inch OD tubes of 25-foot length, giving an effective heat transfer surface area of 2965 ft².

The liquid NaK heat transport medium enters a plenum above the charge face floor and is divided into 4-inch-diameter downcomers which merge into a single 12-inch-diameter downcomer passing through the center of the tube bundle. An elliptically shaped head, welded to the lower tube sheet, forms the lower distribution manifold for the tube bundle. The upper tube sheet forms the bottom of the upper collection manifold which routes the hot NaK into two risers of 4-inch ID. These risers terminate in an annular manifold plenum above the charge face floor, and the hot NaK exits from the heat exchanger into outlet piping. The downcomers are anchored to the shield plug can and carry the weight of the tube bundle and all other structures extending down into the reactor vessel. The risers are not anchored to the shield plug and are permitted to expand freely in a vertical-upward direction for stress relieving of the materials.

Special precautions must be taken to ensure that radiation dosages are reduced to acceptable levels in the areas directly adjacent to and above the portions of the sodium-NaK exchangers extending above the operating floor level. Gamma and neutron streaming through the inlet and outlet liquid NaK pipework is the primary consideration in this regard. The 10-foot-thick, stepped stainless steel can, which is the basic load-carrying structure in the heat exchanger is filled with boron steel shot in the space around the stepped inlet and outlet piping. Surrounding the top of the heat exchanger is a 4-inch-thick layer of steel. The inner layer is a 1-inch thickness of 1.5 wt % boron steel, and the outer 3-inch layer is stainless steel. This steel

cap also extends the double containment piping philosophy for the liquid NaK heat transport medium up to the reactor vessel, and acts as a lifting hanger for handling the heat exchanger.

The heat exchanger shroud surrounding the tube bundle is 48 inches in diameter, 1 inch thick, and 23 feet long. The shroud channels a naturally circulating flow of 1.9×10^6 lb/hr of hot primary sodium down through the tube bundle. The hot sodium is cooled by the liquid NaK heat transport medium inside the heat exchanger tubes as it flows down through the shell side, and is permitted to escape through the open bottom of the shroud back into the reactor pool.

Important engineering data for the Na-to-NaK heat exchangers are summarized as follows:

Number of tubes	453
Tube length, ft	25
Tube size, in.	
OD	1.00
Wall	0.049
ΔP , shell side, psi	0.2
Central downcomer ID, in.	12
Material of construction	304 SS
Total heat transfer area, ft ²	3000
Design temperature, F	900
Design pressure, psig	300
Thermal duty, Btu/hr	171×10^6

NaK-Air Heat Exchanger

The NaK-air heat exchanger is a forced air, two-pass, finned-tube unit. The finned-tube array is constructed of 0.875-inch OD tubing with 12 fins per inch. The array presents a total heat transfer area of 31,250 ft² and is contained within a structure 13 feet high, 30 feet long, and 12 feet deep. The finned tubes and the distribution manifold headers are in contact with the liquid NaK and form an integral part of the decay heat removal loop. The stainless steel

tubes and distribution manifold headers have a design operating temperature of 900 F and pressure of 300 psig.

The heat exchanger receives an inlet flow of 1.75×10^6 lb/hr of liquid NaK at a temperature of 700 F. Ambient air, supplied at a maximum rate of 100,000 cfm by four 60-inch-diameter fans, cools the NaK flow to 300 F. Each fan is powered by a 200-hp electric motor. Air enters the heat exchanger at a design ambient temperature of 90 F and exits at a temperature of 490 F. Motor-operated variable inlet vanes are used to control the air flow and hence the rate of heat rejection. Propane heaters located between the damper assembly and the tubes are capable of introducing large amounts of heat into the reactor coolant during periods when the core afterglow heat is not sufficient to maintain the prescribed reactor coolant temperature. Propane heaters, rather than electrical heaters, were selected because they are more economical and have proven to be very reliable heat sources. These burners are designed for a wide range of thermal duty.

Design data for the NaK-to-air heat exchangers are summarized as follows:

Tubing size, in.		
OD		0.875
Wall		0.062
Number of cooling fins per in. of tube length		12
Cooling fin diameter, in.		1.875
Approximate NaK ΔP , psi		2
Total air flow rate, scfm		400,000
Material of construction		304 SS
Total heat transfer area, ft ²		31,250
Design temperature, F		900
Design pressure, psig		300
Tube array dimensions, ft		
Height		13
Length		30
Depth		12
Number of fans		4
Flow rate per fan, scfm		100,000
Fan propeller diameter, in.		60
Fan motor horsepower		200

Electromagnetic Pumps

The three 50%-capacity electromagnetic pumps are located in the decay heat removal loop outside of the reactor building in an accessible and protected area. During loop operation the pumps receive 300 F liquid NaK from the NaK-to-air heat exchanger return line and discharge into the common Na-NaK heat exchanger inlet line. Flow can also be routed to the circumferential biological shield cooling coils inlet line. The pumps provide a liquid NaK flow of 1.75×10^6 lb/hr to the closed loop configuration during normal and emergency decay heat removal operations and 8.5×10^5 lb/hr during post-DBA decay heat removal operations. These pumps are arranged in the loop in parallel and provide a redundant pump. Any two of the three pumps can meet the total loop flow requirements during normal and emergency decay heat removal operations, and any one of the three pumps can meet the total loop flow requirements during post-DBA decay heat removal operations. Valves are located at the discharge and suction ends of each pump to permit isolation of a pump or a leak source.

Each pump is an a-c, flat-linear-induction electromagnetic pump requiring 480 volts of electric power for operation. All of the pump components exposed to the liquid NaK heat transport medium will be constructed of 304 stainless steel and be capable of operating at a design temperature of 900 F and pressure of 300 psig. The pumps have a capacity of 2000 gpm at 35 psig and an operating efficiency of 35 to 40%. The polyphase multipole windings in the magnetic core and the stators are cooled by forced air convection. Transition pieces are attached to each end of the rectangular pump duct. These transition pieces provide connections to the 4-inch-diameter inlet and outlet nozzles and contain the expansion joints which accommodate the changes in the length of the pump duct caused by thermal expansion.

Pertinent engineering data on the electromagnetic pumps are summarized as follows:

Pump capacity, gpm	2000
NaK inlet temperature, F	300
Pump inlet pressure, ft of NaK	≈ 150
Material of construction (pump duct & fittings)	304 SS
Type	Flat linear induction
Operating efficiency, %	35 - 40
Total developed head, psig	35
Electrical input, volts	480
Horsepower	115
Inlet nozzle diameter, in.	4
Outlet nozzle diameter, in.	4

NaK Surge Tank

The surge tank accommodates the increase in NaK volume due to thermal expansion and permits a controlled overflow of the system during the initial filling operation. The controlled overflow feature ensures the complete filling of the system in the event that gas pockets were formed during the initial filling operations. The pressure of the decay heat removal system is maintained above the primary system pressure by a static head of NaK to avoid the radioactive contamination of the system in the event of a leak in an Na-NaK heat exchanger tube.

The 304 stainless steel surge tank has a design pressure of 300 psig and temperature of 900 F. The tank has a 42-inch inside diameter and a 66-inch-long cylindrical section. The top and bottom heads are elliptical. Location and mounting structures for the tank are incorporated into the NaK-air heat exchanger supporting structures.

Tank design data are summarized as follows:

Volume, ft ³	53
Inside diameter, in.	42
Height of inside heads, in.	72
Design temperature, F	900
Design pressure, psig	300
Material of construction	304 SS
Operating temperature, F	304
Operating pressure, atm	1
Operating liquid level, in.	4.1
Safety valve setting, psig	75

Circumferential Biological Shield Cooling Coils

The circumferential biological shield cooling coils accomplish post-DBA decay heat removal in the event that the heat-exchange equipment within the reactor vessel is destroyed. These coils, which are located behind the reactor pit liner and in the biological shield, can remove 85.5×10^6 Btu/hr of decay heat. To ensure the required thermal communication between the cooling coils and the decay heat source, the argon gas annulus between the reactor vessel and the reactor pit liner must be flooded with sodium. If the reactor vessel does not fail during a DBA* so that the annulus is filled with reactor sodium, then action will be taken to dump the contents of an intermediate loop into this void. The flooded annulus reduces the resistance and permits the decay heat to be conducted into the circumferential biological shield cooling coils.

The cooling coils are made of 1.5-inch-ID stainless steel pipe and are designed to function at temperatures up to 900 F and pressures to 300 psig. The coils are positioned on 6-inch centers and circumscribe the lower cylindrical section of the reactor pit liner. Two of the three 50%-capacity electromagnetic pumps located

- - - - -
*It would be impossible to reason that a vessel of this design could not fail; therefore, the post-DBA decay heat removal system has been designed for this contingency.

outside of the reactor building can force 8.5×10^5 lb/hr of liquid NaK through these coils.

Valves, Pipework, and Trace Heating

The valves in this system are constructed of 304 stainless steel and are designed for a temperature of 900 F and a pressure of 300 psig. These valves which direct and isolate flow to or from the major loop components, are located outside of the reactor building in protected areas. All valves are installed within the closed loop configuration by full penetration butt-welding to the adjacent components. During loop operation all shutoff valves are in the fully open or closed position and are equipped with electrically controlled operators actuated by the protection system. In addition, the valves can be manually operated to ensure operability under all conditions.

The interconnecting piping in the decay heat removal loop has a design temperature of 900 F and pressure of 300 psig; it is constructed of 6-inch stainless steel, schedule 40 pipe. Concentric piping is used inside the reactor building to double contain the NaK. The annulus between the pipes will be filled with argon gas. All piping runs outside the reactor building will be insulated with 4 inches of standard insulation. The piping is fabricated into the closed loop configuration by full penetration butt-welding. Long-radius fittings or pipe bends are used in conjunction with offsets and loops to provide the flexibility required to accommodate thermal expansion of the loop. In addition, the piping will be supported by spring hangers, sway braces, or flexible snubbers to minimize piping material stresses and loop vibration.

Trace electrical heating equipment is provided to preheat the piping and major loop components before the initial NaK fill or to maintain the NaK in the liquid state during periods of standby. Trace heating is not intended to provide a means of maneuvering primary system temperature; propane heaters are included for this function. These electrical resistance heaters are attached directly to the heated surfaces and are designed to ensure a minimum NaK temperature of 120 F. Redundant heaters will be installed on equipment and piping located in areas where contact maintenance is difficult to perform.

8.2.2.2. Decay Heat Removal System Support Service

Inert Gas Supply

The inert cover gas required for use in the surge tanks within the decay heat removal system and the liquid NaK storage tank is available from the plant argon manifold at a pressure of 40 psig.

Electrical Supply

The electrical power supply required for the operation of the electromagnetic pumps, fan motors, and all other electrically operated components associated with the decay heat removal system's operations and functions originates from the temporarily interruptible power supply.

NaK Storage Tank

The liquid NaK storage tank has a storage capacity equal to 135% of the total volume of liquid NaK required for filling the entire decay heat removal system. This storage volume provides for a 10% excess volume of liquid NaK to remain in the storage tank after the decay heat removal system filling operations are completed, and a 25% void volume in the storage tank when the entire volume of liquid NaK is being stored. The storage tank has the required liquid NaK inlet and outlet lines, valves, trace heating, temperature instrumentation, liquid NaK level indications, and auxiliaries. In addition, the inlet and outlet inert cover gas supply lines are equipped with the pressure regulating valves required to maintain a constant internal storage tank pressure.

The 304 stainless steel storage tank has a design temperature of 900 F and pressure of 300 psig; it has a 90-inch inside diameter and a 168-inch-long cylindrical section. The tank will be equipped with the required mounting and supporting structures to ensure proper vertical and lateral restraint of the tank under all conditions.

Storage tank design data are summarized as follows:

Volume, ft ³	619
Inside diameter, in.	90
Height of inside heads, in.	168
Design temperature, F	900
Design pressure, psig	300
Material of construction	304 SS
Operating temperature, F	120
Operating pressure, psig	60
Operating liquid level, in.	12.5
Safety valves setting, psig	75

NaK Purification Unit

The NaK is purified by the NaK cold trap located in a bypass circuit in the cold leg of each loop. This cold trap will be designed for an NaK flow rate of 17,500 lb/hr to maintain the NaK purity requirements defined in Appendix 8-B. The cold trap will have shutoff valves for isolation during maintenance or when the system impurity level is so low that the trap is not required.

8.2.3. Principles of Operation

The decay heat removal system is started differently for each heat removal function; these are normal, emergency, and post-DBA heat removal. Figure 8-A-2 shows that the afterglow heat generation rate equals the thermal duty of the decay heat removal system at approximately 1000 sec after reactor shutdown. Afterglow heating in excess of 2% of the core thermal output is absorbed by the thermal inertia of the large sodium inventory. Normal on-line reactor operation places only one requirement on the system: it must remain filled with the liquid NaK heat transport medium. This requirement is necessary to ensure prompt, reliable startup and operation of the decay heat removal system under all operating conditions.

8.2.3.1. Normal Decay Heat Removal Startup

Following periods of intentionally initiated reactor shutdown, the decay heat removal system is started manually. After the manual startup is completed, the operating functions of the system are controlled automatically to maintain the primary sodium temperature at 800 F.

8.2.3.2. Emergency Decay Heat Removal Startup

Emergency startup of the decay heat removal system is required during periods of unanticipated reactor shutdown or when the cold leg of the reactor coolant system reaches 850 F.

After the automatic startup of the system is completed, the operating functions of the system are automatically controlled by the decay heat removal system's temperature control instrumentation, which maintains the system's operating conditions in the manner discussed under normal decay heat removal.

8.2.3.3. Post-DBA Decay Heat Removal Startup

Startup of the decay heat removal system during post-DBA decay heat removal periods is performed manually and requires information from the protection system's instrumentation and process instrumentation before startup. The plant operator has approximately 2.5 hours to actuate the system under DBA conditions. This information is evaluated to determine these items:

1. The operating capability of both the Na-NaK heat exchangers and the NaK piping inside the reactor building.
2. The functional ability of these components to operate under the DBA conditions.
3. That the radiation monitoring instrumentation or the liquid NaK level indicators in the surge tanks have automatically closed the Na-NaK heat exchangers inlet and outlet piping isolation valves.
4. The presence or absence of sodium in the argon gas annulus between the reactor vessel and the reactor pit liner.

In the event that the reactor vessel does not fail and there is no sodium in the annulus, the annulus is flooded with intermediate sodium from the intermediate system, and the inlet valve for the circumferential biological shield cooling coils is then opened.

8.2.3.4. Standby Operation

When the reactor is on line, this system is in a standby condition. Its primary function is to remove the decay heat produced by the core after shutdown under any condition.

In the standby condition, all the fans and pumps are shut down. The reactor building's isolation valves are open, and the biological shield cooling coils are filled with NaK but are effectively isolated from the system. The NaK-to-air heat exchangers are isolated on the air side, and the surge tank is partially filled. Level, temperature, and radiation monitoring instrumentation are operating and recording the condition of the system. During a startup, which can be either manual or automatic, the instrumentation will sequentially check the two fluid circuits to determine whether a particular circuit is ready to operate, that is, determine whether there is an adequate level in the surge tank, determine the position of the reactor building isolation valves, and the like. After this check, the electromagnetic pumps will be started sequentially until two of three pumps are operating. Temperature sensing instrumentation in the reactor vessel will adjust the motor-actuated air inlet vanes and will start the fan motors to maintain the reactor coolant sodium temperature at 800 F. In addition, the temperature instrumentation will be capable of signaling the heat exchanger bypass valve to open if the situation demands it. Should any of the components fail to react as programmed (e.g., two out of three electromagnetic pumps), then the system will automatically select the second loop for operation.

When heat addition to the reactor coolant system is required, a propane heater firing directly on the finned tubes in the NaK-to-air heat exchanger will be actuated.

Simultaneous operations of the loops will be accomplished manually under conditions of adequate power supply and within the established limits of heating and cooling rates. Simultaneous operation of the loops is not a safety requirement.

8.3. Auxiliary Cooling System

8.3.1. Introduction

This system consists of five subsystems: the reactor building, the control room, the office air cooling subsystems, the closed loop argon subsystem and the plant refrigeration subsystem. Figure 8-1 is a schematic drawing of the system.

8.3.1.1. System Function

The reactor building, the control room, and the office air cooling subsystem function to remove waste heat and maintain prescribed temperatures for personnel comfort and equipment protection.

The closed loop argon cooling subsystem removes heat from the annulus between the reactor vessel and the biological shield liner to maintain the temperature of the concrete below 200 F.

The plant refrigeration subsystem provides the cooling medium for the rejection of heat from the other four subsystems.

8.3.1.2. Summary Design Description

The reactor building, the control room, and the plant offices are cooled by individual freon coolers connected to a common heat sink (see Figure 8-1). Makeup air to the reactor building is filtered, dehumidified, and cooled to the proper temperature, if required, before entering the reactor building. Section 2 of this report describes the feed and bleed system, which is responsible for the preparation of air for the reactor building; however, the air cooling coils of this system are interconnected with the refrigeration subsystem discussed in this section.

The closed loop argon subsystem circulates argon around the uninsulated reactor vessel and then passes the gas through the freon coolers before returning it to the annulus. The gas flow removes enough heat from the vessel to maintain the concrete at temperatures below 200 F.

The plant refrigeration subsystem consists of five one-third-capacity centrifugal chillers and the necessary auxiliary equipment and piping for the rejection of waste heat to the service water system.

8.3.1.3. System Design Requirements

Performance

The closed loop argon subsystem is capable of limiting the temperature rise of the concrete biological shield surrounding

the reactor vessel to 200 F to prevent deterioration of the structural strength.

Using reactor building air as an interim heat sink, the reactor building cooling subsystem, which has a thermal capacity of 5×10^6 Btu/hr, can remove all of the heat dumped into the reactor building by the reactor coolant pump motors, cold traps, reactor vessel, intermediate coolant pipework, outside environment, and other minor contributors. The air temperature must not exceed 100 F. The minimum air temperature for the reactor building is 60 F. The maximum moisture content in the reactor building air will be equivalent to that existing at 80 dry bulb and 56.5 wet bulb. Temperature stratification in the reactor building will be minimized by appropriate duct work and forced circulation.

The control room cooling subsystem will maintain the control room temperature at 72 F by removing up to 1.1×10^6 Btu/hr.

The heat rejection plant refrigeration subsystem will be capable of removing approximately 8.7×10^6 Btu/hr of heat from the coolant and conveying it to the service water system.

Environment

Under normal conditions no cooler will be exposed to temperatures in excess of 200 F. The area in which the evaporator units, fans, ducting, and pipework for the closed loop argon system are located will have the same degree of integrity as the gas annulus.

Reliability/Availability

After the reactor vessel is filled with sodium, the heat removal capacity in the closed loop argon subsystem must be 100% at all times. When the feed and bleed ventilation system is not operating, an additional heat removal capacity of 15% must be available from the reactor building cooling subsystem. All subsystems will be coupled to the temporarily interruptible power supply.

Maintenance

All coolers will be located in areas where manned access is permitted. All coolers will have isolation valves.

Operation

Multiple heat rejection machinery will be connected in parallel so that any combination of units can be used.

Safety

Argon will be the cooling fluid circulated in the space between the reactor vessel and the biological shield liner.

Design Standards

The refrigeration system design will conform to all applicable codes enumerated in section 1 of this report. In case of a conflict, the most stringent code will apply.

Interface

Reactor building penetrations will be kept to a minimum. Refrigeration system components will be located in areas served by overhead cranes. Adequate service water must be available to act as a heat sink.

8.3.2. Detailed System Description

8.3.2.1. Closed Loop Argon Cooling Subsystem

Drawing A31078E* is a cross section of the reference design LMFBFR showing the relationship of the reactor vessel to the adjacent biological shielding. A 1-inch-thick carbon steel liner and a 12-inch-layer of argon gas separate the concrete shielding from the hot vessel which normally operates at 800 F. Common structural-grade concrete must not be exposed for prolonged periods of time to temperatures above about 200 F if strength and radiation attenuation ability are to be preserved; therefore, a means of producing a 600 F temperature difference across the gap between the biological shield liner and the reactor vessel must be provided.

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* B&W drawing A31078E is placed in Volume 1 of this report.

A closed-loop, forced circulation of argon through coolers just below the operating floor level was selected as the means to produce the temperature difference. No insulation was applied to the reactor vessel, even though it would have reduced the thermal duty, since it would not permit vessel inspection or emergency core cooling through the vessel wall in the event that the normal heat removal coolers failed.

It is recognized that certain problem areas still remain to be investigated before this concept can be fully accepted. These areas include, among others, a means for distributing the argon flow around the vessel, the effect of temperature gradients on the vessel, and definition of system requirements with respect to the DBA.

Equipment functional specifications were developed from preliminary heat transfer analysis in an effort to establish the approximate sizing and costs as related to the technical and economic feasibility.

Engineering data for this subsystem are summarized as follows:

Number of units	3
Capacity per unit, % of total required capacity	50
Thermal duty per unit, Btu/hr	0.9×10^6
Number of 100% fans per unit	2
Argon flow rate per fan, cfm	8150
Inlet argon temperature, F	190
Discharge argon temperature, F	50
Refrigerant flow rate per unit, gpm	282

8.3.2.2. Reactor Building Cooling Subsystem

The reference design reactor building is 100 feet in diameter and rises to an elevation of 183 feet. The space above the 57-foot elevation is empty except for air of normal composition at a slightly negative pressure. The total enclosed volume of this space

is approximately 1 million ft.³ The reactor coolant pump motors, the refueling machine, and some minor components utilize the reactor building air as a heat sink to reduce the number of building penetrations. The cooling of machinery, coupled with the solar heat load, the heat losses from the intermediate loop pipework, and the heat transmitted through the reactor vessel cover structure, will require a maximum of about 5.0×10^6 Btu/hr.

To remove this heat, a cooling package consisting of two identical 50%-capacity air-to-R-11 containment coolers are located just below the operating floor level. Each cooler is rated at 212 tons and contains eight rows of cooling coils with a face area of 74 ft². Mounted with these coils are two full-capacity fans rated at 59,000 cfm and 62 bhp each. One fan is kept in continuous operation and the other is on standby. Appropriate duct work for air distribution suppresses the tendency for thermal stratification to occur in the building. Additional reactor building cooling is accomplished by the continual feed and bleed of air for personnel comfort.

Data on the cooling coils used in the preparation of purge air are included in the following summary:

Number of cooling units	2
Capacity per unit, % of total required	50
Thermal duty, Btu/hr	5×10^6
Number of 100% fans per unit	2
Air flow rate per fan, cfm	59,000
Inlet air temperature, F	100
Discharge air temperature	60
Refrigerant flow rate per unit, gpm	798
Number of ventilation cooling units	1
Capacity of ventilation cooling units, % of total required	100
Refrigerant flow rate, gpm	45

8.3.2.3. Control Room Cooling Subsystem

The control room cooler is a single 90-ton-capacity air-conditioning unit with two full-capacity fans rated at 30,000 cfm and with a power requirement of 25 bhp each. One fan operates continuously and one is on standby.

Engineering data for this subsystem are summarized as follows:

Number of units	1
Capacity per unit, % of total required	100
Number of full-capacity fans per unit	2
Air flow rate per fan, cfm	30,000
Refrigerant flow, gpm	338

8.3.2.4. Plant Office Cooling Subsystem

The office building cooler is a 57-ton-capacity air-conditioning unit with one full-capacity fan rated at 20,000 cfm and with a power requirement of 15 bhp. The required refrigerant flow to this unit is 217 gallons.

8.3.2.5. Plant Refrigeration Subsystem

Five one-third-capacity centrifugal chillers are the main components in the heat rejection subsystem. Each unit is rated at 250 tons and requires an input of 300 bhp at this capacity.

The working fluid in the chiller is vaporized in an evaporator as it absorbs heat from the R-11 coolant circulated to and from the reactor building and other areas. This vapor is compressed in a centrifugal compressor before entering a condenser. Service water flowing on the tube side of the condenser extracts sufficient heat to cause the working fluid to condense. The working fluid in liquid form is then returned to the evaporator through a throttling valve, thus completing the cycle.

In addition to the five centrifugal chillers, there are five one-third-capacity pumps for circulating the R-11 coolant.

Each pump is rated at 958 gpm and absorbs 35 bhp at this condition. A piping network ties all of the plant coolers into a common system so that any three chillers can carry the load.

Important engineering data on this subsystem are as follows:

Number of centrifugal chillers	5
Capacity per chiller, % of total required	33-1/3
Service water flow rate, gpm	2300
Refrigerant flow rate, gpm	2760
Refrigerant inlet temperature, F	60
Refrigerant discharge temperature, F	40

8.3.3. Principles of Operation

8.3.3.1. Closed Loop Argon Cooling Subsystem

The biological shield is cooled by circulating argon through the space between the reactor vessel and the shield liner. Argon leaves the coolers at 50 F and is ducted to the bottom center of the reactor vessel. On the upward pass through the pot annulus, heat received from the reactor vessel raises the temperature of the argon to 190 F. At the top of the space, motor-driven circulators force the hot argon through finned coils cooled by R-11 refrigerant. The cooled argon is then ready for another cycle through the closed loop system.

8.3.3.2. Reactor Building, Control Room, and Office Cooling Subsystem

The reactor building cooling subsystem, control room cooling subsystem, and plant office cooling subsystem are in the category of comfort cooling, since the temperature limits of the air involved are dictated by human requirements although most of the heat load comes from equipment. In all three areas, hot air not exceeding 100 F is circulated over finned coils carrying R-11 refrigerant and is cooled to 60 F before being dumped back into the respective area from which it was withdrawn.

8.3.3.3. Plant Refrigeration Subsystem

Service water which passes through the heat exchangers of the centrifugal chillers at a maximum rate of 2300 gpm is the ultimate heat sink for all of the waste heat removed from the reactor complex by the refrigeration system. Figure 8-1 is a schematic drawing of the entire system showing the rated capacity of each cooler and the required flow. The centrifugal chillers extract heat from the R-11 coolant and reject it to the service water. The centrifugal pumps in the R-11 circuit maintain the flow in the closed loops.

8.3.4. Safety Precautions

If a malfunction occurs in the fans of the coolers, standby fans are available and would automatically assume the fan load except for the office unit. Since the biological shield cooling system is required to be the most reliable system under normal plant conditions, an additional 50%-capacity unit is available. If the control room cooler is non-operational over 8 hours, the office cooler will assume 66-2/3% of the control room requirement. The centrifugal chillers have two units as backup to ensure continuity of operation and subsystem reliability.

When the reactor building cooling subsystem is shut down, the reactor may operate at full power for 30 minutes, allowing personnel access to the operating floor, and for 90 minutes before the design temperature of the containment is reached. The reactor may operate at zero power for 7 hours, allowing personnel access, and for 20 hours before the containment temperature reaches the design value. With the containment ventilation section working and the reactor building cooler shut down, the allowable full-power operation time is increased by 15%. When the reactor building coolers are shut down, zero-power operation of the reactor is not limited if the ventilation section is operating.

8.3.5. Maintenance

Since no unusual items of equipment are included in the refrigeration system, maintenance will follow generally accepted industrial practice.

Figure 8-1. 1000-MWe LMFBR -
Refrigerant System
Diagram

The diagram illustrates the Argon Gas System, showing the flow of gas from the blanket gas subsystem through various processing and storage stages. The system is divided into two main sections: the upper section (containing the blanket gas subsystem) and the lower section (containing the waste gas handling and sludge processing).

Upper Section (Blanket Gas Subsystem):

- FROM BLANKET GAS SUBSYSTEM (SEE REACTOR COOLANT SYSTEM):** This line feeds into the **SODIUM VAPOR TRAPS**.
- SODIUM VAPOR TRAPS:** These traps lead to the **GAS COMPRESSORS**.
- GAS COMPRESSORS:** The compressed gas then passes through an **AFTER COOLER**.
- AFTER COOLER:** The cooled gas is then directed to the **ARGON DECAY TANKS**.
- ARGON DECAY TANKS:** There are three **ARGON DECAY TANKS**, each with a capacity of **3000 FT³**.
- RADIATION MONITOR:** A radiation monitor is positioned to monitor the gas flow from the decay tanks.

Lower Section (Waste Gas Handling and Sludge Processing):

- WASTE GAS SURGE TANK 3000 FT³:** This tank receives gas from the blanket gas subsystem and the **ARGON DECAY TANKS**.
- RADIATION MONITOR:** A radiation monitor is located on the line between the surge tank and the **50 SCFM PUMP VACUUM PUMP**.
- 50 SCFM PUMP VACUUM PUMP:** This pump draws gas from the surge tank and feeds it into the **ABSOLUTE FILTER**.
- ABSOLUTE FILTER:** The gas passes through an **ABSOLUTE FILTER** and then through **ACTIVATED CHARCOAL FILTERS**.
- ACTIVATED CHARCOAL FILTERS:** These filters lead to the **STACK**.
- STACK:** The gas is exhausted through the **STACK**.
- STACK FAN (STEAM GENERATOR BUILDING EXHAUST) 50,000 CFM:** This fan draws gas from the stack and discharges it to the atmosphere.
- WASTE GAS FAN VARIABLE DISCHARGE TO 300 CFM MAX.:** This fan is also connected to the stack.
- TO HIGH & INTERMEDIATE-LEVEL STORAGE TANK INLET HEADER:** Gas from the surge tank is also directed to this header.
- TO CIRCULATING WATER DISCHARGE FLUME:** Gas from the surge tank is also directed to this flume.
- 25,000 GAL. HIGH-LEVEL HOLDUP & STORAGE TANK:** This tank receives gas from the surge tank.
- 25,000 GAL. INTERMEDIATE-LEVEL HOLDUP & STORAGE TANK:** This tank receives gas from the surge tank.
- 25,000 GAL. STANDBY HOLDUP & STORAGE TANK:** This tank receives gas from the surge tank.
- RADIATION MONITOR:** A radiation monitor is located on the line between the surge tank and the high-level holdup tank.
- FROM FILTERS:** Gas from the filters is directed to the **FILTER SLUDGE STORAGE TANK**.
- FILTER SLUDGE STORAGE TANK:** This tank stores sludge from the filters.
- RADIATION MONITOR:** A radiation monitor is located on the line between the filter sludge storage tank and the **50 GPM PUMP**.
- 50 GPM PUMP:** This pump draws sludge from the filter sludge storage tank and feeds it into the **CENTRIFUGE**.
- CENTRIFUGE:** The centrifuge separates the sludge from the liquid.
- 50 GPM PUMP:** This pump draws liquid from the centrifuge and feeds it into the **EVAPORATOR 100 GPH**.
- EVAPORATOR 100 GPH:** The evaporator processes the liquid from the centrifuge.
- 50 GPM PUMP:** This pump draws liquid from the evaporator and feeds it into the **DRUM FILLING STATION**.
- DRUM FILLING STATION:** This station fills drums with the processed liquid.
- DRUM STORAGE AREA:** The filled drums are stored in this area.
- HYDRAULIC BALER:** This baler is used to handle the sludge from the centrifuge.
- MISC. SOLID WASTE:** This waste is directed to the **DRUM STORAGE AREA**.

Legend:

- INSTRUMENT ELECTRIC CONTROL LINE

8.4. Radioactive Waste Disposal System

8.4.1. Introduction

8.4.1.1. System Function

The radioactive waste disposal system processes and subsequently eliminates from the plant all gaseous, liquid, and solid waste matter generated during plant operation.

Liquid Wastes

The radioactive liquid waste disposal subsystem collects, monitors, segregates, stores, and exports all active and inactive liquid wastes. Liquid waste may be discharged to the river only if the activity level is below the limits established by state and federal regulations.

Gaseous Wastes

The gaseous radioactive waste disposal subsystem collects, monitors, segregates, stores for decay, filters, dilutes, and dispenses all waste gases in accordance with state and federal regulations.

Solid Wastes

The solid radioactive waste disposal subsystem compacts, stores, and packages solid wastes before they are shipped off-site for disposal.

8.4.1.2. Summary Design Description

The radioactive waste disposal system comprises three subsystems and incorporates components necessary to process all disposable liquid, gaseous, and solid waste matter in conformance with applicable state and federal regulations. The components required are storage tanks, filters, liquid-solid separators, such as evaporators, activity monitors, pumps, piping, valves, and means for disposal. Figure 8-2 is a schematic diagram of the system.

Liquid Wastes

It is estimated that 200,000 gallons of aqueous and nonaqueous waste that could conceivably contain radioactive materials

will be produced annually. The sources of such wastes include equipment decontamination, showers and wash basins, laundry, and chemical and health physics laboratories.

From these sources the waste matter is essentially in a liquid state, and the densities depend on the solids content. Approximately 75% of the liquid waste will be diluted and disposed of following an appropriate holdup to allow reduction in the activity levels. The remaining 25% will be concentrated by evaporation and processed for off-site disposal.

The liquid waste disposal system includes the following components: three 25,000-gallon stainless steel holdup tanks for high and intermediate level waste storage, one 30,000-gallon carbon steel storage tank for low-level waste storage, one 100-gph package evaporator, four 100-gpm transfer pumps, seven 50-gpm transfer pumps, and associated piping, valves, instrumentation, and radiation monitors.

Gaseous Wastes

The radioactive gaseous waste disposal subsystem processes radioactive gases originating in both the reactor blanket gas subsystem and the liquid waste subsystems. Gaseous wastes from the liquid waste subsystem are allowed to decay in a surge tank and are subsequently filtered, diluted, and released to the atmosphere in compliance with state and federal regulations. Radioactive reactor blanket gas is held for decay in three decay tanks in the reactor building before being released to the environment. Blanket gas activity is discussed in the Vented Versus Non-Vented Trade-Off Study.

The gaseous waste disposal subsystem consists of the following components: two sodium vapor traps, three 150-scfm compressors, one after cooler, three 3000-ft³ stainless steel argon decay tanks, one 3000-ft³ stainless steel waste gas surge tank, one 50-scfm (1 hp) vacuum pump, two absolute filters of 1000 scfm capacity for removal of particulates, two charcoal filters for removal of ¹³¹I, one 30-to-300 cfm waste gas fan, one 50,000-cfm stack fan, and associated piping, valves, radiation monitors, gas analyzers, and instrumentation.

An alternate scheme for the processing of radioactive waste gas is presented in Appendix 8-C.

Solid Wastes

High-level solid wastes, such as those that might originate from evaporator residues or filter sludges, are compacted by centrifugation and then mixed with cement in 55-gallon drums which are stored before off-site shipment. Low-level solid wastes, such as laboratory breakage, discarded clothing, etc., are compacted in 55-gallon drums by a hydraulic baler. After a volume reduction factor of approximately 10 is achieved, the low-level waste-containing drums are stored and subsequently shipped off-site.

The solid waste disposal system consists of the following components: one 50-gpm-capacity centrifuge with associated hopper, piping, and valves, one 2-hp hydraulic baler, remote-controlled conveyor-driven drum filling machine, and miscellaneous radiation monitors, instrumentation, and 55-gallon drums.

8.4.1.3. System Design Requirements

Performance

All waste transfer, metering, and discharge pumps are designed for a liquid with a specific gravity of one. Design discharge heads for these pumps will be 120 feet.

Fluid velocities in pipework transporting liquid waste will not exceed 20 fps. The liquid waste disposal portion of the radioactive waste disposal system will be capable of handling a throughput of approximately 200,000 gallons of aqueous and nonaqueous fluids per year.

The gaseous waste disposal portion of the radioactive waste disposal system will be capable of handling a throughput as set by the maximum site release rate of ^{85}K . The allowable limit on this release is 5000 $\mu\text{C}/\text{sec}$. ^{85}Kr produced in excess of the maximum release rate will be either retained in the fuel or stored in the reactor blanket gas space.

The solid waste disposal portion of the overall system will be capable of handling solids centrifuged from liquid waste or contaminated clothing, and any other solid or semisolid waste generated in the plant.

Environment

Most of the radioactive waste handling equipment will be located in a typical industrial atmosphere. Average temperatures will probably not exceed 100 F.

Reliability/Availability

Multiple pumps, filters, and other critical components of the radioactive waste disposal system will be provided to ensure uninterrupted service.

Maintenance

All parts of the system will be accessible for maintenance and will be fitted with sufficient valving to by pass or shut off gaseous or liquid flow.

Operation

The system will be capable of uninterrupted operation while maintenance, repair, component replacement, or waste sampling are being performed.

Safety

All components of the radioactive waste disposal system will be located within limited access, shielded compartments with adequate ventilation to minimize the spread of contaminated material in the event of leaks. All areas will have monitors.

Drain tank inlets are designed to prevent reverse flow of liquids and radioactive vapors. River and atmospheric discharging of aqueous and gaseous wastes will be continuously monitored to ensure that activity levels in the effluents are below the limits specified by state and federal regulations. These monitors operate on a two out of three principle for closing two valves in the discharge path.

Facilities to flush leakages into radioactive waste disposal system drains will be provided in all compartments or rooms housing radioactive waste handling equipment. Care will be taken to limit the volume of the flushing fluid.

Design Standards

The radioactive waste disposal system is designed in accordance with generally recognized codes, standards, and regulations that are applicable to the individual components. Equipment exposed to corrosive fluids will be constructed of stainless steel or other suitable corrosion-resistant materials.

Interface

Sizing of the radioactive waste disposal facilities will reflect the disposal needs of all areas or functions producing waste. Among these sources are:

1. The reactor blanket gas subsystem.
2. The laundry.
3. Equipment decontamination facilities.
4. Chemical laboratory.
5. Building drains.

8.4.2. Detailed System Description

8.4.2.1. Liquid Wastes

Chemical Laboratory Neutralizer Drain Tank

The liquid wastes received by this tank originate within the chemical and health physics laboratories of the plant. Therefore, it is probable that wastes from these areas will contain high concentrations of radioactive materials, as well as highly corrosive chemicals. Because of the corrosion problem, 304 stainless steel has been selected as the tank material. This 5000-gallon tank is 7.25 feet in diameter and 17 feet long. A tank wall thickness of 0.4 inch was established to prevent collapse by inadvertent evacuation.

To neutralize the contents of the tank, chemicals are dispensed into the tank through a remotely controlled hopper. A motor-driven mixing device with dual propellers is installed to aid in the distribution of the neutralizer chemicals. The centrifugal motion imparted to the liquid by the mixer is controlled by four equally spaced vertical baffles welded to its inner circumference. The tank baffles and the portions of the mixer in contact with the corrosive materials are made of 304 stainless steel.

Additional equipment includes a 304 stainless steel vent pipe equipped with a mechanism allowing inward containment air flow to equalize the tank's internal pressure by counteracting the low pressure created by the pump; this arrangement prevents any out-flow of contaminated vapors. A manhole is provided for maintenance accessibility. Drain backflow is prevented by check valves.

Fluid in excess of a prescribed level, as indicated by a level sensing device, is withdrawn from the tank by one of the 100-gpm liquid waste transfer pumps and routed through a radiation counter. Disposition of the radioactive fluid to the 25,000-gallon holdup tanks will depend on the level of activity in the fluid.

Tank design data are summarized as follows:

Capacity, gal	5000
Operating temperature, F	150
Operating pressure, psia	14.7
Material	304 SS, ASTM A-240
Type of mixing device	Mechanical
Location	Grade floor

Laundry Drain Tank

This tank receives liquid wastes of relatively low activity from the laundry, personnel decontamination, and shower drains. The carbon steel tank has a capacity of 1500 gallons, an outside diameter of 5 feet, a height of 11 feet, and a wall thickness of 0.1875 inch. The tank is contained within a grade-level compartment which requires no further shielding. The tank operates with a slightly negative pressure relative to atmospheric pressure. It has a carbon steel vent to permit inward air flow and prevent the release of radioactive vapors. A manhole is provided for internal maintenance accessibility.

When the maximum capacity of the tank is reached, the contents are drained and routed to the low level storage tank by a 100 gpm liquid waste transfer pump.

Pertinent tank design data are summarized as follows:

Capacity, gal	1500
Operating temperature, F	150
Operating pressure, psia	14.7
Material	Carbon steel, ASTM A-283, Gr C
Location	Grade floor

Intermediate- and High-Level Storage Tanks

Three 25,000 gallon tanks fabricated of 304 stainless steel are designated as the intermediate- and high-level storage tanks. These storage units receive liquid wastes from the equipment decontamination tank, the chemical laboratory neutralizer drain tanks, the emergency river bypass, and the centrifuge. Storage functions are as follows:

1. One tank stores high-level wastes until the activity decays to a point where treatment and release is possible.
2. One tank stores intermediate-level wastes until the activity decays to a point where treatment and release is possible. Intermediate-level activity is slightly in excess of the permissible release value and too hot to store in the low-level waste tank.
3. The third tank is installed as a backup for either of the first two tanks.

Each tank has a carbon steel vent pipe for venting released gases to the waste gas surge tank, a chemical feed funnel with strainer cover, a manhole for maintenance accessibility, and check valves that prevent back flow. Each tank has an outside diameter of 16 feet, a height of 22 feet, and a wall thickness of 9/16 inch.

Tank design data are summarized as follows:

No. of high-level storage tanks	3
Capacity per tank, gal	25,000
Operating temperature, F	150
Operating pressure, psia	14.7
Material	304 SS, ASTM A-240
Location	Grade level

Low-Level Storage Tank

The low-level storage tank located on the grade floor collects low-level liquid wastes from the turbine building and laundry drain tanks. This tank is designed for a maximum capacity of 30,000 gallons, which permits storage for 3 months before the wastes are processed and disposed of in the river. A carbon steel vent pipe is provided for pressure equalization. The vent mechanism is designed to allow containment air to enter the tank but prevent contaminated vapors from escaping. Internal accessibility for maintenance is by a manhole. Additional equipment consists of a chemical feed funnel with a strainer cover and a check valve to prevent back flow. An externally mounted sump pump transfers the contents of the tank to the river sampling tank.

Tank design data are summarized as follows:

Capacity, gal	30,000
Operating temperature, F	150
Operating pressure, psia	14.7
Material	Carbon steel, ASTM A-283, Gr C
Location	Grade floor

Equipment Decontamination Drain Tank

Liquid wastes from equipment decontamination are collected in this stainless steel tank before being transferred to either the high- or intermediate-level waste storage tank. The maximum capacity of the tank is 20,000 gallons. The tank is equipped with a chemical feed system similar to that of the chemical neutralizer drain tank, a manhole providing access for internal maintenance, a stainless steel vent pipe to allow containment air to enter but prevent contaminated vapors from leaking, and a check valve to prevent a flow reversal. The equipment decontamination drain tank is located within a shielded grade-floor compartment next to the neutralizer drain tank.

Tank design data are summarized as follows:

Capacity, gal	20,000
Operating temperature, F	150
Operating pressure, psia	14.7
Material	304 SS, ASTM A-240
Location	Grade floor

River Sampling Tank

This 3000 gallon carbon steel tank receives processed low-level liquid wastes before their release to the river. It is equipped with a chemical feed funnel, manhole, check valve, and a specimen drain valve.

Design data for the tank are as follows:

Capacity, gal	3000
Operating temperature, F	150
Operating pressure, psia	14.7
Material	Carbon steel, ASTM A-283, Gr C
Location	Grade floor

Evaporator

The evaporator concentrates dissolved solids in the liquid waste before they are separated by centrifugation. This unit, the AMF Ray-Di-Pak evaporator package, has distinct advantages over the conventional steam-heated evaporator: since no steam is utilized in the system, there is no possibility of airborne contamination due to a leak. Operating temperatures are below the atmospheric boiling point of water since the unit operates under a vacuum. Any leakage in the concentrator shell passes into the evaporator section and thereby precludes the spread of radioactive material.

The operation of this skid-mounted package system may be divided into two cycles:

1. Waste Cycle — Liquid waste from the high- and intermediate-level storage tanks is pumped into the hot waste feed tank. This waste is then continuously circulated through the evaporator

section of the concentrator by the feed tank pump, in which it is boiled at low temperature and high vacuum until the desired solids concentration is reached. The rate of evaporation and distillate production is controlled by throttling the heating and cooling water valves. When the desired waste solids concentration has been reached, the concentrate is pumped to the centrifuge for further separation. Solids are then mixed with cement and placed in 55-gallon drums before shipment for off-site disposal.

2. Distillate Cycle — Vapors produced in the evaporator section of the concentrator pass through separators to remove droplets, thereby preventing the carryover of contaminated particulates. The vapor is subsequently condensed on the condenser tube bundle and withdrawn by an eductor in the distillate circulating line. When the distillate is of sufficient purity, as determined by a conductivity cell, it is circulated through a mixed-bed ion exchange column and then pumped to the river.

Engineering data on the evaporator unit are summarized as follows:

Capacity, gph	100
Type	Low pressure, water heated
Operating temperature, F	110
Operating pressure, in. Hg	27
Service requirements	
Water, gpm	75 at 85 F, 80 at 185 F
Air	1 cf/hr
Power	440 volt, 3 phase, 60 cycle
Location	Mezzanine floor

Transfer Pumps

The liquid waste disposal system has 11 waste transfer pumps which route waste products through the system. Four 100-gpm pumps transfer liquid waste from the drain tanks to the storage tanks. Two of these pumps transfer the liquid waste from the laundry drain tank to the low-level liquid waste storage tank. The other two

pumps transfer the liquid wastes from the chemical laboratory neutralizer drain tank and the equipment decontamination drain tank to the high- and intermediate-level waste storage tanks. Each pump is rated at full capacity, and one serves as a standby unit.

The remainder of the pumps (7) are rated at 50 gpm and function as follows:

Two transfer liquids from the high- and intermediate-level storage tanks to the evaporator concentrator system.

Two transfer wastes from the evaporator to the centrifuge. (These units, along with the two listed above, are rated at full capacity, and one is on standby.)

One transfers the residue from the filter sludge storage tank to the centrifuge.

One transfers liquid effluents from the centrifuge back to the high- and intermediate-level waste storage tanks.

One, an externally mounted sump pump on the low-level waste storage tank, transfers low-level wastes to the river sampling tank.

Design data for the transfer pumps are summarized as follows:

Liquid Waste Transfer Pumps (Type 1)

No. of pumps	4
Capacity, gpm	100
Power required, bhp	3
Pump duty, % of system requirements	100
Location	Two adjacent to equipment decontamination drain tank and chemical neutralizer drain tank; two adjacent to turbine building floor drain tank and the laundry drain tank.

Liquid Waste Transfer Pumps (Type 2)

No. of pumps	7
Capacity, gpm	50
Power required, bhp	1.5
Pump duty, % of system requirements	100
Location	One (sump) inside 30,000-gallon low-level liquid waste storage tank; two adjacent to 25,000-gallon high-level holdup tanks; two adjacent to the evaporator; two adjacent to the centrifuge.

Waste Discharge Pump

One 100-gpm waste discharge pump located near the river sampling tank transfers waste originating in the low-level storage tank to the river. This pump is manually operated to prevent erroneous discharge to the circulating water flume when the plant is not operating. A 3-hp motor drives this pump.

8.4.2.2. Gaseous Wastes*

Waste Gas Surge Tank

A 3000-ft³ stainless steel tank in a shielded compartment on the mezzanine floor is used to store and hold up gaseous effluents originating in the high- and intermediate-level liquid waste storage tanks. Charcoal and particulate filters are also located in this compartment. Any condensation occurring in the tank is gravity drained through a 2-inch stainless steel conduit back to the high-level storage tank on the grade level. A check valve in the drain permits only the downflowing liquid to pass. A remotely operated, solenoid shutoff valve serves as a maintenance aid for both the surge and the liquid waste high-level holdup tank.

Following holdup within the surge tank, the waste gas is filtered for particulates and iodine-131, diluted, monitored, and released to the atmosphere through the exhaust stack.

Design data for the waste gas surge tank are as follows:

Capacity, ft ³	3000
Design pressure, psia	50
Material	304 SS, ASTM A-240
Location	Mezzanine floor

Reactor Blanket Gas Decay Tanks

Three 3000-ft³ stainless steel decay tanks, 8 feet in diameter and 15 feet high, are located within the reactor building

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* See Appendix 8-C for a discussion of an alternate radioactive waste gas system.

for storing the radioactive argon and gaseous fission products from the reactor blanket gas subsystem. A design pressure of 75 psia gives each tank a storage capacity of 15,000 SCF. A maximum permissible activity release rate of 5000 $\mu\text{c/sec}$ of ^{85}Kr was established as a design limitation on the waste gas disposal system. It was also concluded that, with a 0.1-CFM flow rate from the equilibrium gas mixture in the reactor vessel, a period of 246 days is required to decay all the isotopes except ^{85}Kr to insignificant contributors. Since the half-life of ^{85}Kr is so long (10.5 years), it is impractical to retain the blanket gas any longer than the stated period.

The three tanks are arranged in series with internal baffling to provide a restricted and directional flow path for the gas. A flow rate of 0.1 SCFM for a period of 246 days results in a total volume of 35,424 SCF; the combined storage volume is 45,000 SCF. Therefore, the total residence time is 312 days. This should be ample storage, considering a certain inefficiency (flow bypass) in the baffles. The radioactive gases are monitored, filtered, and released to the atmosphere through the exhaust stack.

Each decay tank is individually housed within a limited-access, concrete-shielded compartment with 48-inch-thick walls. This arrangement reduces the radiation level near each tank and provides the isolation required during maintenance.

Pertinent design data for the decay tanks are summarized as follows:

No. of decay tanks	3
Capacity per tank, ft^3	3000
Operating pressure, psia	75
Material	304 SS, ASTM A-240
Location	Containment building

Waste Gas Fan

A variable discharge waste gas fan of 300 cfm maximum capacity transports filtered waste gases to the stack before dilution, final monitoring, and release. This fan is located on the mezzanine floor and requires a 5-hp motor to attain rated capacity.

Stack Fan

A 50,000-cfm stack fan furnishes air from the steam generator building ventilation system for dilution of the waste

gases before their release to the environment. Fan bearings are air cooled. A single-flanged 44-inch-diameter inlet and a 42-inch-diameter duct outlet characterize the construction of this fan, which is located on the mezzanine floor.

Waste Gas Vent Stack

A 42-inch ID, 1/4-inch-thick carbon steel stack receives the exhaust gas from the stack dilution fan and discharges it to the atmosphere at an elevation of 165 feet. Activity levels are limited in compliance with state and federal regulations. Any liquid condensation will drain to one of the intermediate-level liquid waste storage tanks through a 2-inch stainless steel conduit. The drain has a check valve which allows the condensed fluid to flow to the liquid waste storage tank, but prevents any contaminated vapors from flowing up the vent from the tank to the stack.

Vacuum Pump

A 50-scfm vacuum pump transports decayed waste gases from the waste gas surge tank to the particulate filters before the gases are diluted and released. This pump is located on the mezzanine floor and requires a 1-hp-drive motor.

Absolute Filters

Two high-efficiency particulate filters in parallel remove radioactive dust particles. When a differential pressure transducer indicates a differential pressure in excess of 1.5 inches of water, the flow is switched to a standby filter. The filters are fire-proof and resistant to moisture damage at 100% relative humidity. Both are contained in a zinc-plated sheet-steel housing with provisions for simple removal and replacement of filter elements.

Engineering data for these filters are as follows:

No. of filters	2
Capacity per filter, scfm	1000
Type	99.97% efficient for +0.3 micron
Filter medium	Glass
Static pressure drop, in. of H ₂ O at full capacity	1
Location	Mezzanine floor

Charcoal Filters

Two high-efficiency activated charcoal filters in parallel (for reliability through redundancy) remove iodine vapors. A differential pressure transducer similar to those on the particulate filters indicates the condition of the filter. To eliminate a fire hazard, thermocouples will be installed to ensure that the filter is not operated near the ignition point of the activated charcoal.

Design data for these filters are summarized as follows:

No. of filters	2
Capacity per filter, scfm	1000
Type	99.9% efficient*
Filter medium	Activated coconut shell
Static pressure drop, in. of H ₂ O at full capacity	1.5
Location	Mezzanine floor

Sodium Vapor Traps

Two air-cooled, NaK-filled sodium vapor traps will remove any sodium vapor from the waste gas before it reaches the decay tanks. The traps are located in the reactor building within isolated shielded compartments.

Gas Compressors

Three 15-SCFM gas compressors are installed in parallel to force the blanket gas withdrawn from the reactor into the high-pressure storage tanks. Each compressor is capable of producing a discharge head of 75 psia.

After Cooler

The after cooler removes the heat of compression from the gas. The compressed gas flows through cooling coils which are cooled by the forced circulation of reactor building air. A fan generates reactor building air motion.

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* True efficiencies may be as high as 99.9% ; however, regulatory people acknowledge credit for only 90% .

8.4.2.3. Solid Wastes

Hydraulic Baler

A hydraulic baler compacts low-level solid wastes, such as laboratory breakage, discarded clothing, etc., in 55-gallon drums to a volume reduction factor of approximately 10. The baler is manually operated from a local station near the shipping and receiving overhead door on the grade floor.

A standard 55-gallon drum is used if no shielding is required; if shielding is required, a 15- or 30-gallon drum is placed inside an empty 55-gallon drum, and the annulus is filled with concrete or other suitable shielding material.

Pertinent data on the baler are summarized as follows:

Dimensions, ft	
OD	4
Height	8.5
Power required, bhp	2
Piston compacting force, lb (8.3 psi)	2300
Location	Grade floor

Filter Sludge Storage Tank

Spent filter sludge originating in the liquid waste disposal system is stored in a 1000-gallon carbon steel tank. The sludge is pumped to the centrifuge for separation. Solids are mixed with cement in 55-gallon drums before shipment for off-site disposal. Tank data are summarized as follows:

Capacity, gal	1000
Material	304 SS, ASTM A-240
Operating temperature, F	150
Operating pressure, psia	14.7
Location	Mezzanine floor

Centrifuge and Drum-Filling Machine

The centrifuge separates radioactive sludges from the filter sludge storage tank and the evaporator into a semidry radioactive material. A 50-gpm transfer pump forces the liquids through

a radiation monitor and recirculates them to the high- and intermediate-level storage tanks for further holdup or to the river sampling tank depending on the activity level. The concentrated solid sludge drops from the centrifuge via gravity through a vertical chute and into a vibrating solids collection hopper. The hopper has a 50-ft³ volume and is equipped with water spray nozzles to wash out any material adhering to the inside of the hopper after it has been emptied. Hopper discharge is through a 12-inch-diameter, 3/16-inch-thick drum feed pipe. A standard 55-gallon open-top drum is positioned under this pipe for filling operations. All materials are 304 stainless steel.

The remotely operated centrifuge is located on the mezzanine floor and is capable of unattended operation; i. e., it will not be accessible during operation. The removal efficiency of solids will be at least 98% of the solids in the feed, and the resultant sludge will contain from 40 to 50% water by weight.

Waste Drum Conveying and Storage Assembly

This system transports a 55-gallon drum from the empty drum storage area, through the filling, mixing and capping stations, and on to the storage area before shipment by truck or rail for off-site disposal. The system comprises three subsystems: the empty drum conveyor, the transfer car, and the storage conveyor.

After an empty drum is manually loaded onto the empty conveyor, it is transported to the transfer car. The transfer car moves the drum to the feed pipe where it is partially filled with radioactive sludge from the vibrating solids hopper. To account for impact loads when wastes are dumped into the drums, the wall mounted transfer car and its support is designed for a 4500 pound static load. The drum is then moved to the mixing station where cement is added and the contents of the drum are mixed. Following the mixing operation, the transfer car moves to the capping station where the drum is capped by a remotely operated capping machine. The capped drum is then transferred to the storage conveyor which moves it to a shielded storage area.

Engineering data for the drum handling system are summarized as follows:

Transfer Car

Capacity (loaded drum), lb	750
Type	Wall mounted
Elevation	Top of drum, 5 ft above grade floor
Drum dimensions, in.	
Lid OD	24
Base OD	22-7/8
Height	34-3/4
Drum weight, lb	52
Speed, ft/min	≈20
Positioning	Automatic at any stop-position with maximum tolerance of ±3/8 in. either along or across car run

Empty Drum Conveyor

Capacity, lb	300
Type	Powered rollers, remotely controlled
Elevation	Matches transfer car
Speed, fpm	≈20

Storage Conveyor

Capacity, lb	750
Type	Power rollers, remotely controlled
Elevation	Matches transfer car
Speed, fpm	≈20

8.4.2.4. Instruments, Controls, Alarms, and Protective Devices

Liquid Waste Disposal System

The liquid radwaste control panel and motor control center are located in the northwest corner of the steam generator building on the mezzanine floor. Each of the system's tanks is equipped with a locally mounted level transmitter having an associated level indicator mounted on the control panel. The tank's also have level-sensing devices which actuate level alarms on annunciators mounted on the control panel and in the main control room. The alarm is an audible signal which is terminated by a button when the alarm signal is acknowledged. Also mounted on the control panel is a blinking trouble light which indicates a tank with a high level. Locally mounted pressure gauges are provided downstream of all liquid waste transfer pumps. A radiation detector mounted in an in-line instrument well monitors the

effluent from the low-level liquid storage tank before it is released to the river. The detector output is amplified by a preamplifier and transmitted to the main control room where it is indicated on a meter and recorded by a multipoint recorder.

Gaseous Waste Disposal System

The gaseous radwaste control panel and motor control center are located adjacent to the liquid radwaste control panel. The activity level of gases about to be released to the environment is monitored by control panel and in the main control room. The protection system actuates a diversion valve when particulate and/or halogen activity exceeds $7 \times 10^{-6} \mu\text{C/cc}^*$ or when noble gas activity exceeds $0.5 \mu\text{C/cc}^{**}$. Audible and visual alarms are activated on the gaseous radwaste control panel and in the control room when the aforementioned concentrations are approached. Since these trip settings are based on the maximum rated flow capacity of the waste gas fan (300 cfm), a flow recorder is provided in the fan discharge line to trigger an alarm on the gaseous radwaste control panel and in the control room when the fan discharge flow exceeds 330 cfm. Pressure gauges monitor the pressure in the argon decay tanks and the waste gas surge tank.

Solid Waste Disposal System

The control panel governing the operations of the empty drum conveyor, transfer car, and storage conveyor is located on the north shield wall of the drum loading aisle. Indicator lights on the control function as follows:

1. Indicate when a drum is in the process of passing through each escapement unit.
2. Indicate when each storage lane is full.
3. Indicate drum position at filling, mixing, and capping stations.

A digital counter with manual reset is located at each storage lane and counts drums in and subtracts drums out. An interlock prevents an empty drum from entering the transfer car until

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*Based on "milk cycle" limit for ^{131}I .

**Based on most restrictive MPC for noble gas.

the preceding drum has been discharged from the car and the car has returned to its initial position to accept the next drum.

The hydraulic baler is manually operated and requires no particular control or instrumentation device. It has a fan and filter system which is connected to the steam generator building exhaust ventilation system by an air duct in order to remove any air-borne particulates released during the compaction process.

8.4.3. Principles of Operation

8.4.3.1. Normal Operation

Liquid Waste Disposal System

Liquid wastes collected in the laundry drain tank are pumped by one of two 100-gpm liquid waste transfer pumps to the 30,000-gallon low-level liquid storage tanks. The estimated activity concentration of the laundry drain tank contents varies from a normal operation level of 10^{-5} $\mu\text{C/cc}$ to a maximum operating level of 10^{-3} $\mu\text{C/cc}$. If the activity of the liquid in the low-level storage tank is less than 10^{-5} $\mu\text{C/cc}$ as determined by liquid sampling probes, then an externally mounted sump pump pumps the liquid through a particulate filter and in-line liquid radiation monitor and on to the river sampling tank. Liquids with activities below 10^{-5} $\mu\text{C/cc}$ will be below the maximum permissible concentrations* when diluted by the discharge circulating water; therefore, such liquids will be pumped into the circulating water discharge flume and subsequently discharged to the river.

Liquid wastes collected in the chemistry laboratory's neutralizer drain tank and the equipment decontamination drain tank are pumped through a radiation monitor which operates a diverter valve. The diverter valve permits separation of the liquid wastes based on the indicated activity. High-level waste is routed to the 25,000-gallon high-level holdup tank, and intermediate-level waste is forced into the intermediate level holdup tank. The estimated activity concentration of the contents of the chemistry laboratory's neutralizer drain tank varies

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* 1×10^{-7} for unidentified mixtures in unrestricted areas, since it is known that ^{129}I , ^{226}Ra , and ^{228}Ra are not present.

from a normal operating level of 10^{-5} $\mu\text{C/cc}$ to a maximum operating level of 10^{-3} $\mu\text{C/cc}$; the estimated activity concentration of the contents of the equipment decontamination drain tank varies from a normal operating level of 10^{-3} to 10^{-4} $\mu\text{C/cc}$ to a maximum operating level of 10^{-1} $\mu\text{C/cc}$. After being held up for decay in the intermediate- and high-level storage tanks, the waste is pumped to the evaporator by one of two liquid waste transfer pumps. Purified distillate from the evaporator is pumped to the river discharge line, while concentrated sludge is pumped to the centrifuge. Centrifuge effluents are either pumped to the river discharge line, recirculated to the high- and intermediate-level storage tanks, or recirculated to the centrifuge.

Gaseous Waste Disposal System

Radioactive reactor blanket gas from the reactor blanket gas subsystem is passed through electrically heated vent lines to the NaK-filled sodium vapor traps. Following the sodium vapor removal process, the blanket gas is compressed by one of the three 150-scfm compressors and cooled by an after cooler to remove the heat of compression. Following holdup, the gas passes through a radiation monitor after which it may be recirculated to the decay tanks (via the compressors) or bled to the waste gas discharge line. Gases originating in the high- and intermediate-level storage tanks are bled to the waste gas surge tank where they are held up for decay. Following holdup, the gases are pumped through a high-efficiency particulate filter, a charcoal filter, a radiation monitor, and discharged through a 165-foot-high stack. Another radiation sensing element located in the stack at elevation 65 feet closes a flow control valve in the event of an increase in the gamma radiation level.

Solid Waste Disposal System

Solid-laden sludges from the evaporator and the filter sludge storage tank are pumped to the centrifuge. After centrifugation, the waste is stored in a vibrating hopper and gravity fed into 55-gallon drums. After the drums are partially filled with waste, cement is added, the contents are mixed, and the drum is capped and conveyed to a shielded storage area to await shipment for off-site disposal.

8.4.3.2. Emergency Operation

Liquid Waste Disposal System

An in-line radiation monitor between the particulate filters and the river sampling tank activates a valve which diverts the river discharge flow to the inlet header of the high- and intermediate-level liquid storage tanks in the event that an activity exceeding 10^{-5} $\mu\text{C/cc}$ is detected.

A radiation monitor downstream of the centrifuge activates a four-way valve in accordance with the radiation level of the liquid passing through it. For a radiation level of 10^{-3} to 10^{-5} $\mu\text{C/cc}$, the flow is recirculated to the inlet of the centrifuge. For radiation levels in excess of 10^{-3} $\mu\text{C/cc}$, the flow is pumped back to the inlet header of the high- and intermediate-level storage tanks.

Gaseous Waste Disposal System

A radiation monitor located downstream of the waste gas surge tanks diverts the gaseous discharge back to the tank upon sensing an abnormally high radiation level.

A radiation monitor located between the waste gas fan and the stack and one in the stack itself monitor the activity of the gas before and after its dilution. After an abnormally high radiation level is detected, the vacuum pump and the waste gas fan are shut off, thus isolating the source of the high activity.

A radiation monitor downstream of the blanket gas decay tanks controls the operation of another valve. A signal caused by an abnormally high radiation level activates this valve to divert the flow back to the gas compressor inlet.

Solid Waste Disposal System

An emergency situation, such as the tipping over of a solids-containing drum in the loading and transfer area, would be remotely handled; a series of water nozzles located throughout the drum-filling and storage area would flush the radioactive solids into drains. The resulting radioactive sludge would be pumped to the equipment decontamination drain tank.

8.4.4. Safety Precautions

Hazards associated with the operation of the radioactive waste disposal system are due to spills in the drum loading area and leakage from piping and tanks. Spillage in the drum loading area resulting from the tipping over of a solids-containing drum is remotely handled; a series of water nozzles located throughout the drum-filling and storage area flush the radioactive solids into drains. The flushed solids entering the drains are pumped to the equipment decontamination drain tank.

Any radioactive liquid released from pipes or tanks is first detected by area monitors. Steps are then taken to isolate the contaminated area, and operators close the valves necessary to prevent further contamination. If the released radiation level permits personnel access, fire hoses can be used to flush the spill into suitably located floor drains. If personnel access is impossible, the fire protection sprinkler system can be used to partially flush the waste down the floor drains and thus permit personnel access for more thorough decontamination. Wastes entering the floor drains are pumped to the equipment decontamination drain tank.

A radioactive gaseous release is handled similarly; area monitors alert operators so that the area can be sealed off and steps can be taken to halt the release through the operation of certain valves. The released wastes are then processed by the building ventilation exhaust system or by other means, depending on the amount of radioactive gases released. The means for disposing of large quantities of radioactive gases released accidentally or intentionally, have not been devised. A candidate method is discussed in Appendix 8-C.

Permanent area radiation monitors are provided to warn personnel of above-normal radiation levels in areas most subject to irradiation from equipment, tanks, or pipelines. Areas that are subject to gross contamination and consequently are shielded, do not contain permanent area monitors. When entrance into such areas is required, health physics procedures subject to administrative control are applied. The following facilities are provided in the control room for each detector:

1. Adjustable trips for dose rate alarm and loss of current.
2. Audible and visual alarms, both locally and in the control room.

3. Pushbutton to actuate detector check source.
4. Dose rate indicating meters and recorder.

8.4.5. Maintenance

8.4.5.1. Liquid Waste Disposal System Components

Maintenance on tanks is expected to be infrequent. In the event of leaks or defects, appropriate valving is provided to evacuate the tank if it requires internal maintenance. Each tank has a leakproof manhole to allow personnel to enter if it is mandatory. The liquid waste system is designed to allow drainage of a tank. The effluent is monitored and returned to another storage and holdup tank if the activity level exceeds the allowable disposable limit.

Maintenance of the pumps should be infrequent. In the event of a malfunction, the standby pump, which is in parallel with the defective pump, is started. Flow to the defective pump is valved off, allowing the pump to be removed or repaired.

Multiple filters provide for uninterruptible service in the event that component repair or replacement is necessary. Each filter has valves to allow isolation and removal of the components from the system.

In the event of an evaporator malfunction, the incoming liquid wastes from the holdup and storage tanks are by-passed, or the 50-gpm waste transfer pumps and valves are turned off to halt drainage of the storage tanks.

8.4.5.2. Solid Wastes

The components in this system require very little maintenance. However, the drum-filling area conveyors and remotely operated drum capping machine are annually inspected, cleaned, and repaired or replaced if necessary.

8.4.5.3. Gaseous Wastes

The design of multiple vapor traps, compressors, decay tanks, and absolute and charcoal filters provides for uninterruptible service in the event that component repair or replacement is necessary. Other components are maintained by curtailing discharges from the surge tank or the blanket gas decay tank.

8.4.5.4. Instruments, Controls, Alarms, and Protective Devices

The reliability of these components must be extremely high. They must be periodically tested, calibrated, and replaced according to the manufacturer's specifications.

8.5. Preheat System

8.5.1. Introduction

All of the systems handling or receiving liquid sodium must be preheated and maintained at a temperature in excess of 208 F to prevent solidification of sodium. Preheating needs can be divided into two areas: system or loop restart, and plant commissioning. The restart preheating equipment is "permanent" and will be available during plant commissioning (for example, the intermediate loop pipework heaters are in this category). Such prestart preheating may take the form of raising the temperature of a system being held on standby in a subcooled condition. During plant commissioning the capability of the permanently installed preheating equipment must be augmented because both the reactor coolant system and the intermediate coolant system are "dry" and normal internal heat generation via the pumps or the core cannot be utilized. Only selected portions of the preheat system are described in this section since some of the equipment can be treated more logically as a part of the major system to which it is attached.

8.5.1.1. System Function

The preheat system raises the temperature of LMFBR systems handling or receiving molten sodium to a point that will preclude solidification, thermal shock, and flow blockages in critical areas by impurity precipitation.

8.5.1.2. Summary Design Description

The preheating system utilizes permanently installed systems, such as the decay heat removal system, and especially designed supplementary equipment to complete the capability of raising the sodium heat transport loops to the desired operating temperature from a dry condition. The argon circulators for the reactor vessel are the only major components that will not have a continuing function after plant commissioning. Components or systems used for

preheating, in addition to those already mentioned, include the intermediate loop pipework heaters, the package boiler, and the main cool- and pumps (optional).

8.5.1.3. System Design Requirements

Performance

The reactor preheating system will be capable of raising the temperature of the reactor vessel and reactor internals to 400 F.

Inert gas circulation will be sufficient to limit temperature variation within the reactor vessel to a maximum of ± 20 F.

The preheating capacity of the reactor system will be based on raising the reactor system to 800 F with a full inventory of sodium.

Environment

Inert gas circulation blowers will be exposed to an argon atmosphere bearing sodium vapor at 400 F.

Reliability/Availability

Once the first batch of sodium is injected into the reactor vessel, the reactor preheat system must stay on line until enough sodium is loaded into the vessel to submerge the reactor coolant pumps.

Maintenance

All induction heating coils and resistance heaters will be accessible for replacement.

Operations

Provisions will be made to operate the reactor coolant and intermediate pumps as a heat source during system preheating.

Safety

When the plant is commissioned, there will be no sodium radioactivity; however, typical safety precautions will be in effect for the handling of the chemically reactive sodium.

Design Standards

All components in the preheat system will conform to the applicable design codes enumerated in this report.

Interface

Argon circulators will not require special openings in the reactor vessel cover structure.

8.5.2. Detailed System Description

LMFBR reference design systems or components having permanently installed electrical heating units are as follows:

1. Intermediate loop pipework and valves.
2. Intermediate loop blanket gas vent, makeup and equalizing lines.
3. Intermediate coolant transfer pipework.
4. Nonradioactive sodium storage subsystem.
5. Sodium unloading subsystem.
6. Nonradioactive sodium purification subsystem.
7. Intermediate sodium purification subsystem.
8. Steam generator and reheater rupture disks.
9. Radioactive sodium storage subsystem.
10. Reactor blanket gas subsystem.

After the reactor coolant system has been filled to a predetermined level above the primary pump inlets, heat can be added at a maximum rate of 16.5×10^6 Btu/hr by the pumps operating at full speed against the design system head. The design head, however, must be simulated artificially to get the full benefit of this method of heating, since the core will not be loaded at this stage of commissioning. The intermediate pumps can be used in a similar manner after the system has been partially filled with sodium. The maximum rate of heat input to an intermediate loop (9.7×10^6 Btu/hr) can be achieved without special flow restriction devices. Since the pumps cannot be operated until sodium is admitted to the systems, and the permanently installed electrical heating units are incapable of bringing the coolant loops up to temperature, special provisions must be made for preheating.

8.5.2.1. Reactor Vessel Preheat Subsystem

The decay heat removal system described in section 8.2 of this report will be used to preheat the reactor vessel, reactor internals, and cover structure to a uniform temperature of 400 F by circulating hot NaK through the two shell-and-tube heat exchangers in the reactor vessel. Propane heaters outside of the reactor building will be used to adjust the temperature of the flowing NaK up to the design point of the decay heat removal system (900 F) although such high temperatures will probably not be needed.

The maximum NaK inlet temperature will depend upon three factors: the heat loss from the reactor system, the desired rate of heat up, and the effectiveness of the tube bundle heat transfer surface. Temporary blowers will be installed on each heat exchanger bundle to obtain the required heat transfer coefficient, since the bundle surface area is fixed at 5930 ft² as a result of its primary function. If required, additional blowers and heaters will be positioned in the vessel to ensure uniform thermal soaking of the entire system before sodium is admitted.

8.5.2.2. Steam Generator and Reheater Preheat Subsystem

The steam-raising equipment will be preheated to 400 F by superheated steam from the plant package boilers before sodium is admitted from the nonradioactive sodium storage vessel. The valves, pipework, and other components that are required to execute this operation will be a permanent part of the plant installation because these units will have to be preheated after any maintenance activity involving loop drainage.

The combined external surface area of a reheater and a steam generator is 4382 ft². The estimated heat loss from this surface is 132,000 Btu/hr with 4 inches of typical medium-temperature block insulation covering a metal skin temperature of 400 F. A steam flow rate of 1200 lb/hr from the package boiler (rated at 60,000 lb/hr, 190 psia, and 600 F) would be sufficient to make up these thermal losses. Since more auxiliary steam is available than needed, the heatup rate can be adjusted to ensure compatibility with interfacing systems.

Argon gas at 10 psig in the steam generator and reheater will convey heat by natural circulation from the tube bundles to the shells and internal structural members.

8.5.3. Principles of Operation

8.5.3.1. Reactor Preheating

In explaining the preheating phase of reactor startup, it is assumed that plant commissioning has advanced as follows:

1. All of the reactor internals except the core have been installed.
2. The closed loop argon coolers are operational for the protection of the biological shielding.
3. The temporary sodium transfer line has been installed between the reactor vessel and the nonradioactive sodium storage subsystem.
4. 100,000 gallons of pure sodium at 400 F is standing by in the nonradioactive sodium storage subsystem.
5. The reactor internal surfaces have been cleaned using the best available techniques.
6. The reactor vessel has been purged and filled with argon to a pressure of 5 inches of water.
7. All dry operational tests, leak tests, and inspections have been made on the reactor and interfacing systems.
8. Temporary argon circulating blowers have been installed in the reactor vessel.

The reactor system will initially be brought up to the normal operating temperature of 800 F as follows:

1. Start the reactor vessel argon circulating blowers.
2. Gradually raise the temperature of the NaK circulating in the decay heat system by increasing the heat input from the propane heaters.
3. When the reactor vessel skin reaches 200 F, place the closed loop argon cooling system in operation to protect the biological shielding.
4. Continuously add heat to the reactor to obtain complete thermal soaking of the system at a temperature of 400 F.
5. At 400 F, purge the reactor vessel with fresh argon until the water vapor and gaseous impurities that were baked off the internal surfaces reach an acceptable concentration.
6. Pressurize the nonradioactive sodium storage vessel and transfer 100,000 gallons of sodium into the reactor vessel via the temporarily installed pipeline between the two systems. As the sodium enters the reactor vessel, the displaced argon is vented to the atmosphere to keep the vessel pressure below 5 inches of water.
7. Increase the heat input to the reactor coolant via the decay cooling system to bring the system up to the normal operating temperature of 800 F, so that the cold traps can complete the purification process.
8. Once the vessel has been filled, remove the argon blowers and temporary sodium transfer pipework. The temperature can then be manipulated by the pumps and the submerged decay heat removal system.

8.5.3.2. Intermediate Heat Transport Loop Preheating

There will be very few differences between the initial intermediate loop startup procedure and any subsequent startups after the plant is placed in service. For any startup, it is assumed that:

1. Enough pure sodium at 400 F is standing by in the nonradioactive sodium storage vessel to completely fill one intermediate loop.

2. The steam raising equipment and intermediate loop pipework have been thoroughly cleaned, tested, and inspected.

3. The entire loop has been purged of air and filled with argon at a pressure of 10 psig.

A loop will be brought up to temperature and charged with sodium as follows:

1. Energize the pipework electrical heaters.

2. Admit a controlled quantity of steam from the package boiler to the steam generator and reheater. Carefully survey the combined effects of the two preheating methods to preclude excessive thermal stresses.

3. When the entire system reaches 400 F, pressurize the nonradioactive sodium storage vessel and open the loop drain valve. Sodium fills the loop from the bottom, permitting atmospheric vent of the displaced argon.

4. When the impellor of the intermediate pump is submerged, begin low-speed operation to assist in temperature stabilization.

5. Begin cold-trapping of the impurities leached from the loop internal surfaces as soon as practicable.

6. When the system is completely charged, use a combination of intermediate pumps and steam from the package boiler to manipulate the temperature up to at least 600 F.

8.5.4. Safety Precautions

Care must be exercised in preheating operations to prevent excessive thermal stresses due to uneven heating. Strategically located temperature instrumentation will provide sufficient data to preclude adverse conditions.

8.5.5. Maintenance

Maintenance aspects for the equipment used in system preheating are discussed in the section describing the prime function of the particular item. The only exception to this is the argon circulators, which are not expected to give trouble during their limited period of use.

APPENDIX 8-A
Sizing of the Decay Heat Removal System

The decay heat removal system prevents excessive reactor temperature buildup and core damage by heat additions from the decay of radioactive isotopes produced by fission and transmutation. In B&W's LMFBR reference design, fission products are present in the core and in the spent fuel assemblies stored in the reactor vessel. Figure 8-A-2 is a plot of reactor thermal power generation as a function of time following the initiation of a scram from full power.* The effects of delayed neutrons and spent fuel in storage are included in this curve. Heat release by spent fuel has been conservatively factored into the curve by assuming an infinite core operation before shutdown.

Because of the large inventory of sodium in the primary system of the reference design and the high core neutron flux in a fast breeder reactor, the decay of sodium isotopes produced by transmutation contributes a portion of the total afterglow heat that must be removed from the system to maintain prescribed temperatures. Figure 8-A-1 gives the heat generation rate as a function of elapsed time after shutdown for the ^{24}Na isotope under the assumption that all decay energy is deposited in the reactor coolant. The absolute magnitude of this curve is a function of the core neutron flux and the sodium inventory, and was plotted for the reference core flux of $5 \times 10^{15} \text{ n/cm}^2/\text{sec}$ assuming that sufficient time had elapsed to produce an equilibrium isotopic concentration. Heat generated by other sources activated by the core neutron flux has not been computed because it will be negligible. To obtain the maximum total heat generation rate in the reactor vessel, the ordinates of the curves in Figures 8-A-1 and 8-A-2 can be added for each point in time following reactor shutdown. A plot of the combined curves is not included, since heating by sodium decay amounts to only a small portion of the total decay heat and would not appreciably alter the shape of the curve in Figure 8-A-2.

An examination of Figure 8-A-2 reveals that thermal power generation in the reactor system drops very rapidly, but not rapidly enough to eliminate the need for core flow inertia. Examination also brings up

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* Gravity insertion only—for safety reasons, the control rods are followed mechanically into the core to prevent incomplete insertion.

the question of the sizing of the decay heat removal system. Since B&W's reference design has a large primary sodium inventory and, hence, considerable thermal inertia, the size of the decay cooling system can be reduced. There is sufficient thermal inertia in the 2.5×10^6 lb of primary sodium to absorb all heat during the shutdown transient in excess of the 2% rated capacity (171×10^6 Btu/hr) of the decay heat removal system without exceeding reactor design temperatures. An added advantage of having a great deal of thermal inertia is the elimination of the need to get the decay heat removal system on line quickly. There is sufficient time to actuate the system manually from local control stations if a malfunction of the automatic control system occurs.

Sodium flow through the reactor core must be sufficient to limit the maximum fuel pin temperature to 1250 F. Minimum flow requirements would follow the decay heat generation curve shown in Figure 8-A-2. During a scram caused by a loss of pump power, flow decay would begin at time zero and not at the beginning of control rod insertion (see dashed curve in Figure 8-A-2). Calculations show that a rise of less than 1 F in the core bulk outlet temperature occurs under these conditions because the pump inertia is great enough to sustain a substantial portion of the design flow through this critical period. At 250 seconds, pony motors must be brought on line to drive the pumps, so that a minimum flow of 11,100 gpm can be sustained through the core. Maintenance of this minimum flow throughout normal decay heat removal operation is considered mandatory because no credit has been taken for flow caused by natural circulation.

Under DBA conditions, decay heat will be removed by coils embedded in the biological shielding surrounding the reactor vessel if failures occur in the normal decay heat removal paths. To obtain the required thermal communication, a large quantity of intermediate system sodium will be dumped into the 12-inch annulus surrounding the reactor vessel. The additional sodium, coupled with a higher permissible reactor system temperature rise (to 1300 F), prolongs the period during which no heat removal is required to approximately 2.5 hours. Because of this increased time, the required decay heat removal capacity is reduced to 85.5×10^6 Btu/hr.

Figure 8-A-1. Ratio of ^{24}Na Decay Heat Generation Rate to Reactor Thermal Power

Figure 8-A-2. Ratio of Decay Heat Generation to Reactor Thermal Power* and Ratio of Coastdown Flow to Design Reactor Flow

*The decay heat generation curve is plotted for the reference design core at the end of its lifetime, so that heat generated in the fuel storage racks can be accounted for in a simple manner. The actual heat generation rate during a core cycle will depend somewhat on the spent fuel handling scheme. The decay heat generation rate shown above will be conservative.

APPENDIX 8-B
Purity Requirements for the Liquid
NaK Heat Transport Medium

The main concern for the liquid NaK (56% Na, 44% K) heat transport medium is not for corrosion, since adequate allowances can be provided in the system without significantly degrading performance or increasing costs, but for the plugging effects caused by high impurity concentrations. Although the system will operate at temperatures in excess of 600 F, it must be designed for a plugging temperature of no more than 250 F, which is considered mandatory for trouble-free system operation under all expected operating conditions. The tentative operating specifications established for the liquid NaK heat transport medium are as follows:

Normal Chemistry Control

<u>Element</u>	<u>ppm</u>
B	10
C (total)	40
Cyanide	--
Co	10
Cr	25
Cu	5
Fe	40
H	5
N	20
O	10

Prohibited Additions Except as Residuals

Ag	Mg
Al	Pb
Au	S
Ba	Sb
Ca	Si*
Cd	Sn
Halogens	Zn
Hg	

* Allowable to 0.75% in metals and alloys.

APPENDIX 8-C
Alternate Radioactive Gas Handling Scheme

The production of radioactive noble gases is a direct function of the fission rate and hence the thermal power output of a reactor core. As the installed capacity for a single site increases, the problems of safely disposing of the gaseous radioactive wastes are increased. Presently, there is a practical activity release limit of approximately 5000 $\mu\text{C}/\text{sec}$ for ^{85}Kr in a single area. Exact limits depend on meteorological conditions and the location of the reactor. Pollution control trends point toward even more strict limitations than are now imposed. By the time that a 1000-MWe LMFBR goes on line the release of radioactivity may be even more restrictive.

Sodium-cooled breeder reactors tend to compound the problems of waste disposal, especially gaseous waste, as follows:

1. Require large-capacity plants to be economically competitive with other power producing systems.
2. Have a very sensitive direct relationship of fuel cost to clad thickness and thus favor a vented fuel design.
3. Have numerous unsolved seal problems in the reactor vessel cover structure and other critical areas.
4. Require large volumes of inert gas to blanket the highly reactive sodium. Candidate blanket gases are expensive, cause metallurgical problems, or become activated.
5. Have activated sodium vapor in the blanket gas atmosphere.

The alternatives are quite clear for B&W's reference design. If adequate seals (dynamic and static) can be developed to limit the reactor blanket gas purge rate to less than 0.1 scfm and the leakage of the radioactive blanket gas out of the vessel into the reactor buildings to 4.1 cc/day, then a holdup for decay scheme will work satisfactorily (assuming a release rate of 5000 $\mu\text{C}/\text{sec}$ for ^{85}Kr).* Otherwise, a continuous purge of the blanket gas followed by noble gas separation will be required. An alternate scheme to the holdup and decay system has been proposed by Oak Ridge National Laboratory.

The proposed scheme involves the separation of the gaseous effluents in a cascade operation using permselective membranes of dimethylsilicone rubber. Figure 8-C-1 is a diagram of this operation. These membranes are 1-mil-thick sheets suitably protected on each side by porous Dacron mats. A number of such sheets are assembled

*See "Vented Vs Non Vented Fuel Pin Trade-Off Study, Volume 4, Section 4, this report.

in a compact package to constitute one stage, and this in turn is housed in a steel enclosure for containment. The membranes are able to dissolve gases and allow diffusion of these gases through the membrane under the influence of a difference in partial pressure. Each gas has its characteristic permeation rate, and if these rates are different for gases in a mixture, then the faster permeating gases will concentrate on the downstream side of the membrane. In this instance, xenon has the highest permeation rate, krypton the next highest, and argon the third highest.

Sizing of the separation system and other calculations were made by ORNL for the B&W reference design data shown in Table 8-C-1; the calculations are summarized in Table 8-C-2. A parametric study of the product gas rate for a cover gas purge rate of 10 scfm was included in the calculations. Product rates chosen were 1, 0.1, and 0.02 scfm. Low values of the product gas rate are desirable because this gas, which contains the bulk of the krypton and xenon activity, must be stored for long periods of time for the decay of ^{85}Kr . The governing criterion for each calculation was that the recycle (or discharged) argon contains no more than 5000 $\mu\text{C/sec}$ of Kr + Xe activity. This value was considered to be a permissible rate for continuous discharge. Radioactive decay of nuclides while in transit through the cascade was not taken into account.

The data in Table 8-C-2 represent the results of calculations with concentration factors (product/feed) of 10, 100, and 500 for Kr + Xe activity. Estimated plant costs are \$289,000, \$342,000, and \$366,000, respectively. These capital costs are installed costs including membrane, compressor, cooler, piping, instrumentation, electrical, engineering, inspection, and contingency costs. Not included is the cost of building space to house the plant. The volume occupied by the plant would be primarily that required for compressors and coolers, one of each for each stage. The cascade itself occupies little space. Component costs are based on reliable data for current costs of equipment and labor with the possible exception of the membrane cost. Membranes are not a current commercial item, and a projected mass production cost of \$10/yd² was used. This figure was estimated by the General

Electric Company, manufacturers of the membrane. On this basis, the membrane cost is about 15% of the total cost.

To obtain an essentially minimum-cost cascade, the design criterion adopted was that the ratio of permeated flow in a stage to its inlet flow be identical for all stages in a cascade. A multicomponent, cascade-productivity computer code was used to establish the minimum number of enricher and stripper stages conforming to this criterion. The complete number of stages is the number required to yield xenon and krypton concentrations in the recycle stream, so that the specified maximum activity in it will not be exceeded.

There are some approximations and uncertainties inherent in the estimates presented. First, the size of a membrane cascade and therefore its cost are strongly dependent upon the relative and absolute permeability of the membrane to each of the gases present, and the design calculations are based on preliminary permeability measurements which are the only ones available. Second, the stage model assumes a cross-flow path for the permeated stream, which may not be representative of an actual stage design. Third, pressure drop and mixing inefficiency in the high-pressure steam have been neglected.

The results furnished here required the efforts of a number of people. In particular, Babcock & Wilcox wishes to acknowledge the work of Mr. Samuel Blumkin of ORGDP, Mr. R. H. Rainey of ORNL, and Mr. W. L. Carter of Chemical Technology Division of ORNL. Mr. Blumkin prepared the computer program and made most of the calculations. Mr. Rainey's laboratory research furnished the basic permeability data, and Mr. W. L. Carter incorporated the information from many sources into a comprehensive and practical system design.

Figure 8-C-1. Removal of Kr and Xe From Reactor Cover Gas Using Permeable Membranes
(ORNL Dwg 68-3577-R2)

Table 8-C-1. Description of Argon Cover Gas System for a 1000-MWe Sodium-Cooled Fast-Breeder Reactor

Physical Data:

Volume of argon plus rare gases, ft ³	25,000
Gas temperature, F	800
Gas pressure, atm	1
Volume of krypton, ft ^{3(a)}	1
Volume of xenon, ft ^{3(a)}	34
Gas purge rate, scfm	10

<u>Nuclide</u>	<u>Half-life</u>	<u>Activity, curies</u>
⁸³ Kr ^m	1.86 hr	9,880
⁸⁵ Kr ^m	4.4 hr	39,800
⁸⁵ Kr	10.76 yr	43
⁸⁷ Kr	76 min	70,000
⁸⁸ Kr	2.8 hr	108,000
⁸⁹ Kr	3.2 min	9,660
¹³¹ Xe ^m	12 d	16,300
¹³³ Xe ^m	2.3 d	89,400
¹³³ Xe	5.27 d	4,470,000
¹³⁵ Xe ^m	16 min	66,200
¹³⁵ Xe	9.2 hr	356,000
¹³⁷ Xe	4.2 min	33,900
¹³⁸ Xe	17 min	258,000
⁴¹ Ar	1.83 hr	2,200

Total: 5,530,000

(a) Total volume of radioactive plus stable isotopes.

Table 8-C-2. Performance and Cost of Several Cascades of Permselective Membranes for Separating Krypton and Xenon From Argon

(Gas Feed Rate to Cascade = 10 scfm)

	Product gas rate		
	1 scfm	0.1 scfm	0.02 scfm
Cascade			
Number of enriching stages	9	15	20
Number of stripping stages	19	23	24
Membrane area, yd ²	4,630	5,020	5,140
Cascade volume, ft ³	~9	~10	~11
Power requirement, kW	126	136	140
Largest compressor, hp	9.2	7.9	7.7
High pressure side of membrane, psig	150	150	150
Low pressure side of membrane, psig	0	0	0
Feed Gas			
Kr concentration, at. %	0.004	0.004	0.004
Xe concentration, at. %	0.136	0.136	0.136
Kr activity, curie/ft ³	24.4 (a)	24.4	24.4
Xe activity, curie/ft ³	542.6(a)	542.6	542.6
Recycle (or vented) Gas			
Kr concentration, at. %	0.47×10^{-5}	0.49×10^{-5}	0.46×10^{-5}
Xe concentration, at. %	0.53×10^{-9}	0.51×10^{-10}	0.25×10^{-10}
Kr + Xe activity, curie/ft ³	0.029	0.030	0.028
Product Gas (to be stored)			
Kr concentration, at. %	0.04	0.4	2
Xe concentration, at. %	1.4	14	68
Ar concentration, at. %	98.6	85.0	30
Kr activity, curie/ft ³	244	2,440	12,200
Xe activity, curie/ft ³	5,426	54,260	271,300
Concentration factor, product/feed	10	100	500
Number of storage cylinders per week	50	5	1
Capital Cost			
Membrane	46,000	50,000	51,000
Compressors	70,000	84,000	90,000
Coolers	7,000	8,000	8,000
Installation	62,000	68,000	71,000
Instrumentation	42,000	57,000	66,000
Electrical	3,000	3,000	3,000
Engineering and inspection	23,000	27,000	29,000
Contingency (15%)	39,000	45,000	48,000
Total	\$289,000	\$342,000	\$366,000

(a) Based on 32F and 1 atm., i.e., standard conditions.

B&W Drawing A31117E —
 1000-MWe LMFBR - Reference
 Concept Decay Heat Removal
 System, Schematic Diagram

LEGEND

- ⋈ VALVE NORMALLY OPEN
- ⋈ VALVE NORMALLY CLOSED
- ⋈ VALVE NORMALLY THROTTLED
- ⋈ STOP-CHECK VALVE
- ⋈ ELECTRICALLY OPERATED VALVE
- ⋈ RELIEF VALVE
- ⋈ ELECTROMAGNETIC PUMP
- ⋈ FAN, ELECTRIC MOTOR DRIVEN

NOTES

1. ALL PIPING INSIDE REACTOR BUILDING EXCEPT CBSCC WILL BE DOUBLY CONTAINED
2. ALL PIPING OUTSIDE REACTOR BUILDING TO HAVE 4" THK. INSULATION AND ELECTRICAL TRACE HEATING.
3. LOOP No. 2 IS IDENTICAL TO LOOP No. 1

Section 9

INTERMEDIATE COOLANT SYSTEM

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The following B&W drawings are placed at the
end of this section:

A31113E	1000-MWe LMFBR - Reference Concept Intermediate Coolant Pump
A31114E	1000-MWe LMFBR - Reference Concept Intermediate Coolant System Piping, Isometric View
A40507D	1000-MWe LMFBR - Reference Concept Intermediate Cold Trap

9. INTERMEDIATE COOLANT SYSTEM

9.1. Introduction

9.1.1. System Function

The intermediate coolant system transports heat received from the reactor coolant¹ in the intermediate heat exchangers to the steam-raising equipment² for all normal modes of plant operation. The intermediate heat transport system separates the reactor coolant sodium from the feedwater, thereby (1) restricting the highly radioactive sodium to the reactor building, (2) eliminating the need to route feedwater to the reactor building, and (3) eliminating the possibility of the products of a sodium-water reaction from degrading fuel performance.

9.1.2. System Summary Description

The intermediate coolant system physically connects the reactor coolant system with the steam generating system. The system consists of three independent loops, each servicing two intermediate heat exchangers, one steam generator, and one reheater (see Figure 9-1). The sodium is circulated in each loop by one vertically mounted, free-surface, single-suction centrifugal pump located in the cold leg. A constant-speed induction motor is connected to the pump through an eddy current coupling which is used for varying the pump speed under partial load operation of the plant.

The piping in each loop is arranged so that the intermediate coolant pump discharges into a single pipe which then splits before entering the reactor building and connecting to the IHXs. Return piping from the IHXs join outside the reactor building before continuing to the steam generator and the reheater. Each loop has two sodium inlets for both the steam generator and the reheater, and a throttle valve between the units is used to proportion the sodium flow. A sodium bypass line is provided around the steam-raising components for control of partial load.

Figure 9-1. Intermediate Coolant System Flow Diagram

Isolating valves are located outside the reactor building on the suction and discharge lines to and from the IHX. Discharge isolation valves are also provided for the steam generators and reheaters.

Each loop contains its own gas blanket system which provides for makeup, venting, and relief of hydrogen resulting from a sodium-water reaction. Sodium purification is provided in each intermediate loop by an air-cooled cold trap which is located in a bypass loop containing an electromagnetic pump.

The intermediate coolant system also includes the incoming plant sodium unloading facilities and the nonradioactive sodium storage tanks. The unloading facilities provide the hot oil used to liquify tank car sodium and the transfer lines and pumps used to deliver the sodium to the nonradioactive storage tanks. The storage facilities comprise two 50,000-gallon carbon steel tanks, a cover gas system, and a circulating cleanup system. The tanks can accept or discharge sodium to or from the intermediate loops and interfacing components.

The subsystems required to place the intermediate system in service, to control the purity of the coolant, or to support the maintenance activities are as follows:

1. Intermediate blanket gas.
2. Intermediate sodium purification.
3. Sodium unloading.
4. Nonradioactive sodium storage.
5. Nonradioactive sodium purification.
6. Intermediate pipework heating.

These subsystems are considered as part of the intermediate cooling system and are discussed in this section.

The intermediate blanket gas subsystem can be divided into three identical portions corresponding to the three intermediate coolant loops. Each loop has vent lines between the gas spaces in the intermediate pump, the steam generator, and the reheater. Pressure-regulating valves continuously sense the blanket gas pressure and admit fresh argon from the plant manifold when the pressure drops below 10 psig, primarily because of the dynamic seals on the intermediate pump shaft. Excess argon pressure during loop purging will be relieved by a pressure-regulating valve exhausting to the atmosphere through check valves. Also, the intermediate blanket gas subsystem maintains an

inert atmosphere in the safety stacks which exhaust the hydrogen generated in a sodium-water reaction.

The intermediate sodium purification subsystem contains an air-cooled cold trap for each independent loop. These traps are positioned near the top of the steam generators for safety and ease of maintenance (see drawing A31114E^{*}). Oxide-laden sodium is withdrawn from the discharge line of the steam generator, and forced by an EM pump back to the top of the unit where it enters the cold trap. Purified sodium is pumped back into the hot leg of the intermediate loop.

The sodium unloading subsystem comprises a propane-fired heater with integral controls, which heats the oil circulated through the tank car and thus melts any solidified sodium, a heated carbon-steel pipe which transfers sodium from the tank car to the nonradioactive sodium storage vessel, a controlled argon supply which effects sodium transfer, and a sodium sampling system.

Before the integrity of an intermediate loop is violated, the contents of the loop are transferred to the nonradioactive sodium storage subsystem. Two 50,000-gallon, electrically heated carbon-steel tanks are the main components in this subsystem. A network of heated Croloy pipes with strategically positioned blocking valves connects the three intermediate loops to the common storage facility. Vent lines connecting the blanket gas spaces in the steam-raising equipment with the blanket gas space in the storage vessels permit gravity drainage of a loop when desired. A secondary purpose of the nonradioactive storage system is to provide an interim sump for newly purchased sodium while it is being purified before injection into the heat transport systems.

The nonradioactive sodium purification system includes a filtration module and an air-cooled cold trap. A 100-gpm electromagnetic pump withdraws sodium from the nonradioactive storage vessel and forces it through the purification system. An axial-flow fan powered by a constant speed, 20-hp, a-c motor passes a stream of cool air over the finned cold trap. The temperature of the trap is controlled

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^{*}B&W drawing A31114E is placed at the end of this section.

by a damper below the fan motor. The sodium oxide capacity for the cold trap is approximately 250 pounds.

Pertinent data for the reference design LMFBR power plant are summarized in Table 9-1.

Table 9-1. Intermediate Coolant System Design Data

Intermediate Pumps

Type	Free surface, single suction, centrifugal
Number of units	3
Capacity per pump, gpm/lb per hr	71,000/30.70 $\times 10^6$
Developed head, ft at 700 F	205
Design temperature, F	900
Brake horsepower	3780
Operating temperature, F	700
Rated speed, rpm	825
Normal speed range, rpm	231 to 825
Impellor diameter, in.	41.0
Diameter suction/discharge, in.	36/36

Pump Drivers

Type of motor	Vertical induction
Motor speed, rpm	900
Motor horsepower, bhp	3800
Type of motor cooling	Air
Pony motor horsepower, bhp	5

Pump Speed Control

Type	Vertical eddy current coupling
Design eddy current coupling slip, %	3
Type of eddy current coupling controller	Solid state
Torque at rated output, lb-ft	~24,000

Table 9-1. (Cont'd)

Type of field coil cooling	Circulating oil
Cooling oil flow rate, gpm	180
Max. duty for oil-water HX per coupling, Btu/hr	2.3×10^6
Approximate dimensions of pump, motor and coupling envelope, ft	
Diameter	12
Height	19

Intermediate Pipework & Valves

General

Number of independent intermediate coolant loops	3
Number of intermediate heat exchangers	6
Number of steam generators	3
Number of reheaters	3
Sodium flow rate/IHX, lb/hr	15.35×10^6
Sodium flow rate/SG, lb/hr	22.20×10^6
Sodium flow rate/RH, lb/hr	8.50×10^6
Hot leg sodium temp, F	1000
SG discharge sodium temp, F	852
RH discharge sodium temp, F	825

Sodium Piping

Material	Croloy 2-1/2
Maximum sodium velocity, ft/sec	25
Pipe wall thickness, in.	3/8
SG inlet pipes	
Number	2
Dia, in.	28

Table 9-1. (Cont'd)

Reheater inlet pipes	
Number	2
Dia, in.	18
Reactor building penetrations	
Number	6, 6
Dia, in.	22, 26
Average thrust/moments, lb/lb-in.	1200/52,000
Insulation	Medium-temperature block
Insulation thickness, in.	4
Insulation surface temp, F	≈140
Assumed environmental temp, F	80
Estimated specific pipe- work loss, Btu/hr-ft ²	150
Estimated total inter- mediate coolant system surface area, ft ²	34,000
Estimated total inter- mediate coolant system heat loss, Btu/hr	5.1×10^6

Valves

Location of intermediate coolant system isolation valves	Outside reactor building
Reactor building isolation valves	
Number	6, 6
Size, in.	22, 26
Reheater throttling valves	
Number	3
Size, in.	22
Bypass throttling valves	
Number	3
Size, in.	14

Table 9-1. (Cont'd)

Intermediate Coolant System Auxiliaries

Intermediate Blanket Gas Subsystem

Intermediate loop blanket gas pressure, psig	10
---	----

Intermediate Sodium Purification Subsystem

Number of intermediate system cold traps	3
Cold trap circulating pump	EM, linear induction
Cold trap flow rate, gpm	100
Intermediate cold trap coolant	Air
Intermediate cold trap blower power, hp	20
Cold trap blower motor	Constant speed, induction
Cold trap temperature control	Inlet air dampers

Sodium Unloading Subsystem

Sodium tank car circulating oil supply capacity, gal.	500
Oil heater	Propane fired
Minimum thermal duty of thawing system, Btu/hr	1.2×10^6
Tank car blanket gas pres- sure, psig	15

Nonradioactive Sodium Storage Subsystem

Number of nonradioactive sodium storage tanks	2
Total nonradioactive sodium storage capacity, gal.	100,000
Orientation of nonradioactive sodium storage tanks	Horizontal
Nonradioactive sodium stor- age vessel design pressure, psig	50

Table 9-1. (Cont'd)

Nonradioactive sodium storage vessel design temperature, F	1000
Nonradioactive sodium storage vessel insulation thickness, in.	4

9.1.3. System Design Requirements

9.1.3.1. Performance

1. The intermediate coolant system shall respond to programmed temperature and flow changes throughout the plant's lifetime without inducing excessive thermal stresses or thermal fatigue.
2. The capacity of the intermediate pumps must be variable from 20 to 100% of the rated plant conditions.³
3. The head capacity characteristics of all intermediate pumps must be matched within $\pm 5\%$ of the design rating.
4. The average sodium velocity in the intermediate loops shall not exceed 25 ft/sec.
5. The intermediate sodium purification subsystem shall be designed to hold the oxygen concentration in the coolant to a maximum of 10 ppm.
6. The sodium unloading subsystem shall be capable of raising the temperature of a carload (40 tons) of sodium to 240 F in 16 hours.⁴
7. The nonradioactive sodium storage subsystem shall have sufficient capacity to raise the temperature of the fully loaded tanks evenly at a rate of 10 F per hour up to a maximum temperature of 400 F, and exist in the deenergized state at temperatures up to 1000 F.

8. The nonradioactive sodium purification subsystem shall be capable of filtering and purifying a full inventory of sodium in the nonradioactive sodium storage tanks to an oxygen concentration of 10 ppm in 100 hours and of maintaining this impurity level until loop charging is completed.

9. Gravity drainage of an intermediate loop shall be possible in less than 1 hour.

10. The intermediate pipework heating subsystem shall be capable of raising the temperature of all conduits filled with sodium to their prescribed limits in not more than 100 hours, starting from a frozen state at 60 F.

9.1.3.2. Environment

The intermediate coolant system shall be enclosed in a typical industrial building, and temperatures shall be controlled by a ventilation system.

9.1.3.3. Reliability/Availability

1. Routine maintenance on loop components should be done during refueling shutdowns so that plant availability is not influenced.

2. Clearance between rotating and stationary parts must be adequate to prevent seizure when operating up to 900 F.

3. At least one independent intermediate sodium purification unit shall be provided for each heat transport loop.

4. The oxide holdup capability of a cold trap shall be designed for a minimum of one year. Plant availability shall not be compromised as a result of replacement of the purification components.

5. All sodium transfer pipework and argon vents must be equipped with some means of heating to prevent flow blockage of freezing.

6. Redundant pipework heating elements shall be provided in inaccessible areas.

7. The intermediate sodium pipework shall be capable of safe operation under the schedule of temperature transients expected in the 30-year lifetime of the plant.

9.1.3.4. Maintenance

1. Seals at the interface of the intermediate coolant pump driver shall be replaceable without removal of the driver.

2. Pump drivers shall be easily disconnected from the pumps for maintenance or replacement.

3. All components and pipework in the intermediate coolant system shall have provisions for gravity drainage.

9.1.3.5. Operations

1. At least three samples shall be taken from each load of sodium purchased in order to provide initial data for a continuing analysis of the interplay of the sodium purification system.

2. All on-line loops shall be operated within $\pm 1\%$ power output.

3. A sodium bypass line around the steam-raising components should be provided for plant control at partial loads in conjunction with the variable speed pumps.

9.1.3.6. Safety

1. The elevation of the free sodium surface in the intermediate coolant pumps, the steam generators, and the reheaters should be approximately the same.

2. The sodium blanket gas pressure in the pump shall equal the blanket gas pressure in the steam-raising units. To guarantee pressure equality, heated vent lines connecting the steam generator, the reheater, and the pump shall be installed in each loop.

3. The intermediate sodium purification systems shall be near the top of the steam generators as a safety precaution against the possibility of losing the intermediate coolant during replacement of the cold trap.

4. The electromagnetic pump supplying the cold trap shall be placed low in the cold leg of the intermediate loop to ensure priming and prevent burnout.

5. The nonradioactive sodium storage subsystem shall be below the level of all other intermediate system components to permit a gravity drain of the loop contents in an emergency.

6. Driver inertia is guaranteed to supplement the pump inertia to ensure a gradual flow coastdown during power failures.

7. The intermediate loop isolation valves shall be designed to perform satisfactorily in the event of a pipe rupture directly adjacent to the steam generator side of the valves.

8. Normally, the isolation valves will be opened and closed by motor-driven actuators powered from the (diesel backed) interruptible bus. Manual operation shall be included for extremely adverse conditions.

9. Isolation valves shall not be used as throttling valves.

10. The isolation valves shall form a part of the reactor building and be positioned so that radiation hazards to personnel outside the containment building would be negligible after an accident in which radioactive sodium were able to diffuse into the intermediate system and reach the valve.

11. The intermediate coolant pipework within the reactor building shall be enclosed by a secondary barrier as a positive means of reducing the possibility of a sodium fire in the area.

12. Inert gas shall be supplied to the sodium delivery vehicle at a pressure not to exceed the tank's rating.

13. Flexible hoses shall not be used in transferring sodium from the delivery vehicle.

14. The integrity of the intermediate system must not be compromised by the use of mechanical pumps requiring dynamic seals, except in the case of the main coolant pumps.

9.1.3.7. Design Standards

1. Pump parts in contact with sodium shall be constructed of Croloy 2-1/4, with the exception of bearings and wear ring materials.

2. All intermediate loop sodium conduits shall be made of standard-weight Croloy pipe and be fusion welded in accordance with ASTM Code A155-65.

3. All pipework shall be of single-wall construction except in areas in which there is a mixing of flow between two streams of sodium at different temperatures.

4. All system pipework shall be designed for an internal pressure of 175 psia. Hot leg pipework shall be designed to operate at a maximum temperature of 1100 F. Cold leg pipework shall be designed to operate at a maximum temperature of 900 F.

5. Piping stress will be taken by the piping configuration rather than packed slip joints, bellows, or other devices which may be less reliable.

6. The intermediate blanket gas subsystem shall receive gaseous argon from a liquified argon source at a pressure not to exceed 150 psig. Purity requirements as analyzed for commercial grade argon are satisfactory:⁵

<u>Impurity</u>	<u>Maximum conc, ppm</u>	<u>Dew point, F</u>
O ₂	5	
N ₂	20	-76
Other	5	

7. Sodium shall be the lubricant for the lower bearing in the intermediate coolant pump.

8. The storage vessel for nonradioactive sodium shall be designed to withstand the maximum pressure that can be imposed on the intermediate system.

9. Temperature gradients in the intermediate coolant system pipework shall not be severe enough to cause a thermal stress failure.

10. Commercial grade sodium with a reduced calcium content shall be sufficiently pure for bulk delivery to the plant site.

11. An operating cold trap shall be provided in each loop of the intermediate system for operation with a plugging temperature of 300 F or less. The tentative operating specifications established for the intermediate sodium are as follows:⁶

Normal Chemistry Control

<u>Element</u>	<u>ppm</u>
Boron	10
Carbon (total)	40
Cyanide	--
Cobalt	10
Chromium	25
Copper	5
Iron	40
Hydrogen	5
Nitrogen	20
Oxygen	10

Prohibited Additions (Except as Residuals)

Aluminum	Mercury
Antimony	Potassium
Barium	Silicon (allowable to 0.75%
Cadmium	in metals and alloys)
Calcium	Silver
Gold	Sulfur
Halogens	Tin
Lead	Zinc
Magnesium	

12. To ensure quality control and comply with license stipulations, all components of the intermediate coolant system

shall be designed in accordance with the applicable codes given in Volume 1, section 2 of this report.

9.1.3.8. Interface

1. The intermediate pump motors shall be air cooled.
2. Heat released in the intermediate coolant pump's eddy current couplings shall be transferred outside the steam generator building by the cooling oil to an oil-to-water heat exchanger. See Table 9-1, Pump Speed Control.
3. Fresh argon shall be supplied to the intermediate blanket gas subsystem from the plant manifold.
4. Vital components, such as the pumps, in the intermediate loops shall be protected from damage by freezing sodium with current supplied from the guaranteed interruptible (diesel backed) power supply.
5. Pump supports shall transmit pipework thrusts and moments to the steam generator building.
6. The intermediate coolant system shall be compatible with the reactor coolant system and the steam generation system.

9.2. Detailed System Description

9.2.1. Components

9.2.1.1. Intermediate Pumps

The pumping duty of the intermediate coolant system is divided evenly among three free-surface, vertical, single-suction centrifugal machines. Each pump turning at 825 rpm will develop a 206 foot head of 700 F sodium while delivering a rated capacity of 71,000 gpm. A sketch of the pump and driver assembly is shown in drawing A31113E.*

Sodium enters the pump via a 36-inch-OD, 3/8-inch-wall Croloy pipe rising vertically from the lower levels of the building housing the steam-raising equipment. The discharge flow is handled by a similar pipe located in a horizontal plane.

The basic load (pipework reaction and physical weight) carrying members of the pump are the outer barrel and the diffuser casing. This assembly is bolted to a suspension structure which is grounded to the building floor at the 71-foot elevation.

To facilitate maintenance, pump internals are mounted on an internal barrel which can be removed. The lower pump shaft is aligned in the radial direction by a hydrostatic sodium bearing which is near the 41-inch-diameter impellor. Sodium for this bearing is obtained from the high pressure side of the impellor. The upper seals are oil-lubricated and can be replaced without removing the pump driver.

The intermediate pump is manifolded to the blanket gas on the sodium side of the steam generator and reheater (see section 9.2.2.1). An argon injection port and an isolation valve to remove the pump from the common manifold are provided to permit purging of the pump.

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*B&W drawing A31113E is placed at the end of this section.

9.2.1.2. Intermediate Pump Drives

The intermediate pumps are driven by 3-phase, 60-cycle, 900-rpm, vertical induction motors rated at 3800 brake horsepower. The speed of the pumps is controlled by variations in the field excitation of an eddy current coupling⁷ between the output shaft of the constant-speed motor and the pump shaft.

Physically, an intermediate pump motor and coupling can be enclosed in an envelope 12 feet in diameter and 19 feet high. The motors are cooled by circulating building air through the windings. Bearing lubricants are cooled by a forced-convection, oil-to-air heat exchanger which is mounted as an appendage.

The Kingsbury-type thrust bearings used in the motor require an oil film between the rubbing surfaces during startup and shutdown to reduce friction. The oil film is provided by a mechanism known as the lift system, which is also an appendage of the motor. The major components in the lift system are two servomotor-driven pumps, one ac and one dc; each unit is capable of slightly more than full pumping power, and each is equipped with pressure switches, a pressure gage, a relief valve, and an oil-metering manifold assembly. A zero-leakage check valve is positioned at each thrust bearing shoe to prevent reverse flow and loss of the hydrodynamic oil film during normal bearing operation when the lift system is not in operation.

Mounted on the top of the main pump drives are 5-hp pony motors which are energized to maintain some loop circulation if a power failure occurs. Each pony motor is connected to the main motor shaft through a set of speed-reducing gears and overriding ball clutch. These motors are connected to the guaranteed (diesel backed) interruptible power.

The eddy current coupling is connected to the pump motor through an electrically insulated rigid coupling and is separated from the pump to provide access to the argon atmosphere seals of the pump without removing the driver assembly. The 900-rpm couplings can transmit 24,000 pounds-feet of torque at 3% slip. The normal operating speed ranges from about 231 rpm up to a maximum of 873 rpm available at 3% slip, but the maximum is not needed at the rated pump speed.

The coupling has three major components: the rotor, the rotor drum, and the housing. The rotor drum is mounted on the input shaft (motor side), and the rotor operating inside the rotor drum is connected to the output (pump side) shaft. The field coils and encircling magnets are part of the coupling's housing. The excitation coils are stationary surrounding the rotor-drum and therefore are not subjected to the centrifugal stresses of rotation. There is no need for brushes, slip rings, or commutators. The input shaft supporting the rotor drum is free to rotate concentrically within the field magnets and is separated from them by the parasitic air gap.

During operation the input assembly turns at the speed of the prime mover, although the output assembly will not turn until current flows in the field coils. Magnetic lines of force then flow through the field body, rotor, and drum and are interrupted by the relative differential in speed between the drum and the rotor. As a result, eddy currents are developed in the inner surface of the drum opposite the rotor poles. A second magnetic field is developed by the flow of eddy currents in the drum. The resultant interaction between the two magnetic fields provides torque-transmitting ability between the drum and the rotor. Varying the excitation current in the field coil produces a corresponding change in torque or speed at the load.

The field coils are cooled by the circulation of a high-grade, low-viscosity turbine oil to the heat exchangers which are supplied with water from the service system. Bearing lubrication is supplied through separate pumps, coolers, and strainers from a reservoir common to both the cooling and the lubrication systems. Both the cooling and the lubricating oil supply systems have two motor-driven pumps in parallel circuits: an a-c pump for normal operation and a d-c pump for emergency operation.

A torsional shear section in the input shaft is designed to twist off at 250% of rated torque. A solid-state controller unit supplies d-c excitation current to the field of the eddy current coupling.

9.2.1.3. Intermediate Pipework and Valves

The number of intermediate loops and their configuration were selected as a result of trade-off studies on the primary

and intermediate heat transport loops (see Volume 4, section 3). These results indicated a preference for a three-loop intermediate system having three steam generators, three reheaters, three intermediate pumps, and six intermediate heat exchangers. An isometric drawing of the east loop of the pipework system designed by Sargent & Lundy Engineers is shown in drawing A31114E. Since the west loop is a mirror image of the east loop and the center loop is virtually the same, they are omitted from the drawing.

Twenty-six-inch (OD) lines from each of the six intermediate heat exchangers transport the hot sodium (1000 F) through the reactor building's wall to the steam generator bay. Isolation valves are placed on the outside of the building's penetrations. Two of the 26-inch-OD pipes are combined into one 36-inch line which feeds the two 28-inch leads on the steam generator and the two 18-inch leads on the reheaters. The cold sodium leaving the steam generator and the reheater are combined in a special mixing section before entering the 36-inch-OD suction line of the intermediate pump. The intermediate pump discharges the sodium to a 36-inch-OD header which splits into two 22-inch-OD lines before entering the reactor building. Each 22-inch line then feeds one intermediate heat exchanger.

The intermediate piping inside the reactor building is encased in a mild steel pipe to prevent the possibility of fire from small leaks. The annulus between the piping is filled with argon and contains leak-detecting instrumentation.

Data used in determining the piping sizes are as follows:

Hot sodium temperature, F	1000
Steam generator discharge temperature, F	652
Reheater discharge temperature, F	825
Intermediate pump discharge temperature, F	700
Steam generator flow, lb/hr	22.20×10^6
Reheater flow, lb/hr	8.50×10^6
Intermediate pump flow, lb/hr	30.70×10^6
Intermediate heat exchanger flow, lb/hr	15.35×10^6
Maximum sodium velocity, ft/sec	25

The sodium piping is routed in the steam generator bay with ample long-radius elbows to adjust for the thermal

expansion due to the temperatures involved. Calculations have been made, and it has been determined that the thrusts and moments of the equipment are about 12,000 lb and 52,000 lb-in., respectively. These values are compatible with the design philosophy. However, additional investigation is needed to consider the design problems associated with the mixing of the steam generator and the reheater discharges.

All intermediate pipework and valves are covered with 4 inches of commercially available, medium-temperature block insulation. Heat losses from the piping are estimated to be 150 Btu/hr-ft² with an outside insulation temperature of 140 F and an ambient temperature of 80 F. The order of magnitude of the heat losses for the entire intermediate coolant system is shown in Table 9-2.

Table 9-2. Intermediate Coolant Piping Data

Pipe ID, in.	14	18	20	22	26	28	32	36
Insulation OD, in.	22	26	28	30	34	36	40	44
Surface area per ft, ft ² /ft (accounting for the 4 inches of insulation)	5.76	6.80	7.33	7.85	8.90	9.42	9.95	11.52
Pipe length, hot leg, ft	30	108	75		340	180		480
Pipe length, cold leg, ft	45			480	275		125	540
Total pipe length, ft	75	108	75	480	615	108	125	1020
Pipe surface area, including insulation, ft ²	432	735	550	3768	5474	1017	1244	11750
Number of 90-deg long-radius bends	6	12	3	9	9	12	6	60
Surface area per 90-deg long-radius bend, including insulation, ft ²	15.8	24	28.8	33.9	45.4	51.8	58.5	81.4
90-deg long-radius bend surface area, ft ²	142	288	86	305	409	622	351	4884

(Total surface area in straight pipe and long-radius bends = 31,034 ft².)

The surface area in the straight piping and the long-radius bends totals 31,034 ft². After adding a conservative estimate for the surface area in the valves, tees, reducers, and other components

in the system, the total surface area approaches 34,000 ft². Under normal plant operating conditions, approximately 5.1×10^6 Btu/hr would be dumped into the pipework gallery from this source.

A combination of induction and resistance heating will be used to preheat the intermediate system before the coolant is charged and to hold the system at a prescribed temperature during loop inactivity.

9.2.2. Intermediate Coolant System Auxiliaries

9.2.2.1. Intermediate Blanket Gas Subsystem

The intermediate blanket subsystem has four basic components: the vent network, the argon makeup unit, the argon exhaust port, and the safety stack. The vent network is a system of heated pipes connecting the gas voids in the steam generator, reheater, intermediate pump, and nonradioactive sodium storage vessel. Motor-operated blocking valves between each space permit equipment isolation for maintenance or other reasons. Vent pipework will be arranged so that any condensation of sodium vapor will drain from the lines.

The argon makeup unit is near the top of the steam generator and is plumbed into the main plant's argon supply. A pressure-regulating valve automatically admits fresh argon into the common blanket gas system via the steam generator space when the pressure drops below 10 psig. A bypass valve is included for purging and higher blanket gas pressures when desired.

The argon exhaust port consists of a blocking valve upstream of an adjustable pressure-regulating valve in a tube penetrating the steam generator head.

Both the steam generator and the reheater are equipped with rupture disks and safety stacks so that gases liberated in a large sodium-water accident can be safely discharged to the atmosphere. A flapper valve must be installed at the stack outlet so that the stack can be filled with inert gas. This arrangement will prevent an explosive reaction between the liberated hydrogen and the atmospheric oxygen within the steam generator building. A safety stack inerting arrangement consisting of a pressure-regulating valve and a blocking valve is installed on each stack and draws argon directly from the main plant manifold.

An evacuation nipple and a bypass line around the pressure-regulating valve are also provided in each stack for purging and leak testing.

9.2.2.2. Intermediate Sodium Purification Subsystem

The intermediate-loop sodium will be purified by an air-cooled cold trap similar to the Hallam design⁸ (see drawing A40507D*). Oxide-laden sodium at 1000 F is extracted from the hot leg of the intermediate system and is circulated by an electromagnetic pump to a cold trap at the top of the steam generator via a 2-inch Croloy pipe. The piping is arranged to allow inlet sodium to the cold trap to originate in the cold leg when the intermediate coolant loop is shut down. The cooling process begins as the impure sodium flows through the regenerative heat exchanger section before being pumped back into the hot leg of the intermediate system. On leaving the economizer, the oxide-laden sodium flows toward the bottom collection chamber through an annular passageway containing stainless steel packing. Heat from the sodium is extracted by an external air stream passing over axially oriented fins. When the sodium reaches the lower collection chamber the temperature is approaching 300 F and has lost most of the precipitable impurities in the stainless steel packing. Clean sodium then flows upward in an annular passageway toward the economizer coil. Heat received from the impure sodium in the inner annular passageway and the economizer coils raises the temperature of the clean sodium to within approximately 50 F of its initial temperature.

An axial-flow fan powered by a constant speed, 20 hp, a-c motor forces a stream of cool air over the finned cold trap. The temperature of the trap is controlled by a damper positioned below the fan's motor. Temperature signals from several thermocouples imbedded in the trap are used to automatically adjust the cooling capacity to maintain maximum trap performance.

Most of the hardware for the cold trap is permanently installed. When a trap becomes ineffective, the inlet and exit sodium lines are cut, and the trap cannister is removed by the overhead crane and transported to the maintenance area.

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* B&W drawing A40507D is placed at the end of this section.

9.2.2.3. Sodium Unloading Subsystem

The sodium unloading subsystem is on the west side of the plant complex near the nonradioactive sodium storage vessel and the railroad siding. New supplies of sodium will be delivered in special railroad tank cars equipped with an oil circulation system for thawing frozen sodium.

A propane-fired oil heater, a 500-gallon oil storage tank, an oil circulating pump, a 4-inch electrically heated transfer pipe, and an argon gas supply form the major components of the sodium unloading subsystem. The oil heater will be chosen from commercially available units and will have a minimum thermal duty of 1.2×10^6 Btu/hr. Flexible hoses will supply oil to the tank car. Gaseous argon will be supplied from the main plant manifold at a regulated pressure of 15 psig. Gas supply lines near the tank car will have heaters and vapor traps to prevent line blockage by freezing sodium. The end of the sodium transfer pipe that bolts to the tank car will be cantilevered for a distance great enough to permit vertical and horizontal motion, so that adjustment for minor deviations in the position of the tank car's hatch can be made without bellows or other flexible conduits. The transfer pipe discharges directly into the nonradioactive sodium storage vessel. Blocking valves are placed at each end of the pipe so that the contents of the pipe can be purged.

9.2.2.4. Nonradioactive Sodium Storage Subsystems

Two 50,000-gallon carbon steel tanks will be installed on the west side of the power plant complex to receive and store nonradioactive sodium from the intermediate heat transport loops or from the sodium unloading subsystem. Both tanks will be installed in the horizontal position and at a grade level that permits gravity drainage of any one of the intermediate sodium loops. The tanks will be electrically heated and designed for continuous duty at 600 F with intermittent periods of operation up to 1000 F. The design pressure at 1000 F will be 50 psig. Approximately 4 inches of removable insulation will be applied to the outside of the tank to reduce heat losses. Tank penetrations will be limited to a 4-inch sodium transfer pipe, an argon pressurization

and/or vent line, and an inlet and exit line for the nonradioactive sodium purification subsystem.

9.2.2.5. Nonradioactive Sodium Purification Subsystem

This subsystem contains an air-cooled cold trap identical in design to those in each of the intermediate sodium loops (see section 9.2.2.2) and a filtration unit to remove foreign matter transferred from the rail cars into the nonradioactive sodium storage vessel. The filtration unit is placed just upstream of the cold trap to prevent premature flow blockage of the trap by particulates.

9.3. Principles of Operation

The intermediate coolant system provides the thermal communication required between the reactor coolant system and the steam generation system. It transmits the energy demands by the steam generation system and therefore tends to decrease in temperature with an increase in plant load and to increase in temperature with a decrease in load. Its ability to maintain the programmed power output of the plant is controlled by variable-speed intermediate pumps and sodium bypass lines around the steam generator and the reheater.

During the startup phase of the plant operation, the intermediate system will be brought to temperature at a rate compatible with the systems that bound it. The intermediate pumps, together with electric heaters and steam from the package boilers, will raise the temperature of the system to a level equal to that of the reactor coolant system and the steam generation system. The reactor will then be started to provide the heat (if required) to raise the entire plant, except for the steam-raising equipment, to an isothermal condition of 800 F.

As the steam-raising equipment is placed on line, the cold leg temperature will drop and the hot leg temperature will rise simultaneously with an increase in the intermediate pump speed and a decrease in the steam generator bypass flow. The process will continue until the plant reaches 100% power, at which time the intermediate pumps will be operating at 100% of rated speed, the hot leg temperature will rise to 1000 F, and the mixed mean temperature of the cold leg will have dropped to 700 F.

During normal operation the intermediate system will share the entire plant load equally among the loops that are in operation. See section 8 for the control scheme that will integrate the intermediate system with the entire plant.

Of primary concern in the operation of the intermediate system is the effect of thermal transients not only on the intermediate system, but on the reactor coolant and steam generation system. Further investigation will be required to determine the effects of transients initiated by the steam generation system on the intermediate piping and the intermediate heat exchangers during a turbine trip not initiated by the reactor system and a reactor scram not initiated by the turbine plant.

9.3.1. Components

9.3.1.1. Intermediate Pumps

The three sodium pumps in the intermediate system are independent of each other and have no common flow paths. At full speed they operate under a constant available NPSH of approximately 70 feet. The level in the pump is maintained by an overflow line connected to the pump suction. At low loads, such as when the pony motor is in operation, the level is established by the loop inventory and is at the same elevation as the level in the steam generator and the reheater.

The hydrostatic sodium bearing is supplied with sodium from the high pressure side of the impellor. At low rpm the discharge pressure of the sodium to the bearing will have decreased, but will be sufficient because the radial load also will have decreased. Normal operation of the pump requires that the sodium be relatively free of sodium oxide and other solid impurities to prevent blockage of the flow channels leading to the bearing.

9.3.1.2. Intermediate Pump Drives

Since the basic principles of operation for constant-speed induction motors are widely understood, they are not discussed here. Eddy-current couplings, however, are not common in large machines and are treated in section 9.2.1.2 of the intermediate system design description.

A control analysis indicates that the intermediate coolant flow rate will be 28% of the maximum flow rate when the plant output drops to 20% of the rated capacity. The pump's speed for this mode of operation will be approximately 231 rpm while the hydraulic horsepower drops from 3180 to 70. The thrust bearing lift system will not be needed in the usual operating range of the power plant (20 to 100%), but it will be activated whenever the motor is below about 200 rpm.

An advantage of a coupling over other speed-control schemes is that the motor load can be reduced to zero for starting, thereby greatly reducing peak currents. With an eddy-current coupling, motor load can be applied at any desired rate throughout the designed speed range.

9.3.2. Intermediate Coolant System Auxiliaries

9.3.2.1. Intermediate Blanket Gas Subsystem

The intermediate blanket gas subsystem must function under four conditions: coolant charging, normal plant operation, coolant discharging, and the sodium-water accident. Figure 9-2, a schematic drawing of an intermediate loop, shows how the inert blanket gas supply is factored into the system.

Coolant Charging — Assuming that 100,000 gallons of purified sodium is available in the nonradioactive sodium storage vessel, an intermediate loop will be charged as follows:

1. The entire loop and both vent stacks are evacuated.
2. Fresh argon from the plant manifold is admitted to the loop and stacks.
3. The entire intermediate loop is preheated to a temperature of 400 F.
4. Pressurized nonradioactive sodium flows into the loop from the storage vessel. Argon displaced by the incoming sodium is vented to the atmosphere.
5. Once the prescribed sodium level is reached in the steam generator and the reheater, the pressure of the nonradioactive sodium storage vessel is reduced to 10 psig by venting the excess argon to the atmosphere.
6. Under normal operation conditions there is no need to vent blanket gas. This fact will be shown subsequently.

Figure 9-2. Intermediate Blanket Gas Subsystem

Normal Plant Operation — The normal operating range for the 1000-MWe LMFBR Reference Plant is between 20 and 100% of the rated capacity. At the maximum plant output the inlet sodium temperature to the steam generator is 1000 F. When the plant output drops to 20% the inlet sodium temperature falls to 960 F. Since the blanket gas pressure is so low (25 psia), the change in pressure due to the 40 F temperature drop amounts to only 0.75 psi. It may therefore be concluded that very few difficulties will be encountered in maintaining a stable blanket gas pressure. The system would be self regulating if zero leakage could be guaranteed for the intermediate pump shaft dynamic seals. Any leakage of argon through these seals would be automatically made up from the plant manifold, which will be set to hold 10 psig in the gas space. Equalization of the blanket gas pressures in the steam generator, reheater, and intermediate pump by a vent network eliminates the need for separate supply systems.

Coolant Discharging — During the lifetime of the plant, occasions will arise that require the complete drainage of an intermediate loop. For safety, the storage vessel has been placed below the level of any component in the intermediate sodium system so that sodium can be transferred by gravity alone. Since this feature is already built into the system, it will be the usual means of sodium transfer. As sodium flows into the storage vessel, displaced argon gas flows into the intermediate loop through the vent network.

Sodium-Water Reaction Accident — Sodium and sodium-water reaction products will be drained from the intermediate system by the same procedures used for a routine coolant discharge. However, during the accident, large volumes of hydrogen gas will be generated and must be safely vented to the atmosphere. If the shell side pressure in the steam generator or the reheater exceeds 200 psig, then the rupture disk in the top of the affected piece of equipment will fail and admit hydrogen to the safety stack, which is filled with argon at 10 psig. As the pressure rises, a flapper valve in the stack exit opens and discharges the hydrogen-argon mixture to the atmosphere. When the stack pressure drops below a predetermined limit, remotely operated argon inlet valves will be opened to purge the remaining hydrogen gas from the

stack. Subsequently, the pressure regulating valves will hold a blanket gas pressure of 10 psig until the loop can be drained.

9.3.2.2. Intermediate Sodium Purification Subsystem

The intermediate purification cold trap will be operated during plant operation at all loads as required. In-line instrumentation and sample analysis will determine the operating conditions as well as the period of operation for the trap.

9.3.2.3. Sodium Unloading Subsystem

Railroad tank cars of new sodium are pushed into the unloading bay on the west side of the power plant complex. The sodium will be transferred from the tank cars as follows:

1. The cantilevered transfer pipe is bolted to the discharge hatch of the tank car.
2. The transfer pipe is purged to remove all gases that are reactive with sodium and any water vapor that might have been admitted during the coupling operation.
3. Oil heater and pressurizing gas connections are made with the tank car.
4. Thawing of the sodium is begun.
5. Before the actual transfer of the tank's contents, the transfer pipe is brought up to approximately the same temperature as the sodium being handled, and the non-radioactive sodium storage vessels are evacuated to 28 inches of mercury.
6. The blocking valve in the transfer pipe near the nonradioactive storage vessel is opened.
7. Argon gas pressure (15 psia) is applied to the tank car to effect the sodium transfer.

After the entire contents of the tank car have been emptied, the empty car is disconnected as follows:

1. The blocking valve in the transfer pipe near the tank car is closed.
2. An argon purge is used to force the residual quantities of sodium into the tank car or into the nonradioactive storage tank.
3. The blocking valve near the nonradioactive storage tank and the rail car valve is closed.

4. Heater power is removed from the transfer pipe, the tank car is pressurized to 5 psig, and the propane furnace is shut down.
5. After cooling, all connections are removed and a new rail car is brought up for unloading.

9.3.2.4. Nonradioactive Sodium Storage Subsystem

Sodium can be added or removed from the non-radioactive sodium storage subsystem in two ways: by differential cover gas pressure, or by gravity-induced differential pressure, depending upon the particular operational function. New sodium arriving by railroad tank car will be transferred to the nonradioactive storage tanks by pressurizing the molten sodium in the delivery vehicle and evacuating the inert gas in the storage tanks to obtain the necessary driving head. Sodium from the nonradioactive sodium storage subsystem will be transferred to the reactor vessel via the temporarily installed transfer pipework by pressurizing the storage vessel and venting the gas from the reactor vessel.

Sodium can be transferred from the intermediate heat transport loops to the storage vessel by either of the methods mentioned above. During a loss of electric power, a loop can be drained to the nonradioactive storage subsystem by gravity. Blanket gas spaces in both the storage tanks and the intermediate coolant loop being emptied must be kept at an equal pressure. To do this, a vent line common to all spacers is installed in the system.

Sodium can be forced from the storage tanks back into the heat transport loops only by differential cover gas pressure, since the tanks are below the level of the intermediate sodium loops. The inert gas in the heat transport loop is vented to the atmosphere as it is displaced by the entering sodium.

9.3.2.5. Nonradioactive Sodium Purification Subsystem

Because the nonradioactive sodium purification subsystem is almost identical to the intermediate sodium purification subsystem, the principles of operation are essentially the same and are described in section 9.3.2.2 of this report. The only operational

difference lies in the fact that a filtration unit has been added to the bypass flow circuit. This unit mechanically removes particulates by directing the flow of sodium through layers of stainless steel screen whose mesh gets progressively smaller as the fluid approaches the exit.

9.4. Safety Precautions

9.4.1. Components

9.4.1.1. Intermediate Pumps

The location of the sodium pump in the 1000 MWe LMFBR reference plant absolutely prevents a gravity-induced loss of sodium through the gas seals. A vent line connecting the gas voids in the steam generator, reheater, and pump in common relieves the possibility of leakage due to pressure differentials.

9.4.1.2. Intermediate Pump Drives

The pump's speed must be accurately controlled during startup, shutdown, and power maneuvers to prevent radical thermal transients from causing critical thermal stresses in certain heat transport components. A computerized automatic control system will direct normal operations so that the likelihood of extreme conditions is remote. However, during a loss of power, motor inertia will play a significant role in causing a gradual letdown of intermediate coolant flow. The combined motor, pump, and coupling inertia will adequately prolong the coastdown period following a power failure. To preclude the loss of the motor component of the total inertia, the coupling excitation current will be supplied from the guaranteed noninterruptible bus. Pony motors connected to the guaranteed interruptible bus will take over the pump driving duties as required.

9.4.1.3. Intermediate Pipework and Valves

Sodium fires of any origin in the reactor building could have serious consequences. Therefore, all intermediate pipework carrying sodium into or out of the reactor building is enclosed in a secondary steel jacket.⁹ These protective jackets are welded into the steel reactor building liner and into the steel cap covering the intermediate heat exchangers.

Under accident conditions, it is theoretically possible for the intermediate loop's flow conduits to become escape avenues for dangerous quantities of radioactive materials. Isolation valves on each intermediate loop pipe penetrating the reactor building

give plant personnel the option of positively controlling this factor affecting safety. Furthermore, manual valve operation is provided as a backup to the remotely controlled motor-driven operation.

9.4.2. Intermediate Coolant System Auxiliaries

9.4.2.1. Intermediate Blanket Gas Subsystem

Failures in the intermediate blanket gas subsystem fall into three categories: loss of argon pressure, argon overpressure, and the presence of a hazardous safety stack atmosphere.

Normally, a pressure regulating valve maintains a 10 psig argon blanket gas pressure over the free surfaces in an intermediate loop. Failure of this valve to make up leakage from the system would result in a gradual drop to atmospheric pressure. In the absence of a pressure differential across the pump's dynamic seals, any contamination of the blanket gas would be limited to that caused by diffusion. Pressure transducers in the blanket space would indicate the malfunction in the control room, so that timely corrective action could be taken. A malfunction of this nature does not constitute a potential safety hazard from a sodium-air reaction.

If a failure in the argon makeup controller caused over-pressurization of the blanket gas space, it would be relieved by a safety valve. Pressure transducers in the blanket gas space would indicate the malfunction in the control room so that corrective action could be taken.

The presence of air in the safety stack would definitely pose a threat to plant safety. At ambient temperature the concentration of air in an air-argon-hydrogen mixture must be kept below 20% to preclude the possibility of an explosion.¹⁰ This limit is considerably reduced as the temperature of the mixture rises. To guarantee the safe expulsion of the hydrogen produced in a sodium-water reaction, the air concentration will be kept below 5%. Although the operating procedure calls for evacuation of the safety stack followed by argon pressurization to 10 psig, samples will be withdrawn and analyzed periodically as an additional safety precaution.

9.4.2.2. Intermediate Sodium Purification Subsystem

The design of the intermediate cold trap was heavily influenced by considerations of the safety in the handling of highly reactive sodium metal. Specific features contributing to plant safety are as follows:

1. The secondary cold trap is positioned near the top of the steam generator to eliminate the possibility of losing loop sodium in the event that a cold trap tank or a valve fails.

2. The electromagnetic circulation pump is placed low in the intermediate system in order to boost the pressure of the secondary cold trap's sodium supply, to ensure pump priming at all liquid levels in the steam generator, and to preclude pipe failures from overheating.

9.4.2.3. Sodium Unloading Subsystem

Water vapor-sodium reactions and sodium fires will be prevented by maintaining a dry argon blanket over all free sodium surfaces. Generous purging of the transfer pipework will remove any residual sodium before a depleted tank car is disconnected. The use of a cantilevered transfer pipe eliminates the need for failure-prone bellows or other types of flexible conduits.

9.4.2.4. Nonradioactive Sodium Storage Subsystem

Safety measures for the nonradioactive sodium storage subsystem will follow the guidelines that are generally accepted for handling liquid sodium. Vessels and pipework will be built to high standards to preclude the exposure of sodium to water, water vapor, or atmospheric oxygen. Certain passive measures, however, will be taken to limit the consequences of an accident involving the rupture of any part of this subsystem. Some of these passive measures are as follows:

1. There is no entrance way or penetration into the storage vault below the liquid level corresponding to a complete dump of the entire intermediate sodium inventory. This design will confine any fires to the vault area.

2. All storage tank penetrations are in the top of the vessel to eliminate the possibility of drainage due to pipework failures or procedural errors in repair or maintenance activities.

3. Although the storage vault housing the nonradioactive storage tanks will not have an inert atmosphere, the consequences of a sodium fire will be limited by minimizing the atmospheric oxygen available for a reaction with sodium. To do this, provisions will be made to isolate the storage vault from the plant air-conditioning system under accident conditions. Also, the vault's volume will be the minimum allowed for access by maintenance and inspection personnel.

9.4.2.5. Nonradioactive Sodium Purification Subsystem

The intermediate sodium purification units are placed at the top of the steam generator to eliminate the possibility of draining the loop's contents inadvertently while replacing the cold trap cartridge. The same general philosophy is applied to the nonradioactive sodium purification subsystem to prevent the loss of sodium from the nonradioactive sodium storage vessel in the event that a failure occurs anywhere in the filter or the cold trap device. Both the filter and the cold trap will be mounted above the maximum level of sodium expected in the tank.

9.5. Maintenance

9.5.1. Components

9.5.1.1. Intermediate Pumps

Maintenance of the intermediate sodium pumps will not be especially difficult because of the following design features:

1. A manway permits access to the upper bearings and seals of the pump shaft without removing the drive assembly.
2. Pump internals can be removed without severing the inlet and the discharge pipework.
3. At the option of the purchasing utility, a special canister can be built to permit withdrawal of the pump internals without draining the intermediate sodium.

9.5.1.2. Intermediate Pump Drives

Large motors have reached a high degree of development in that maintenance is low and reliability is high. This is

especially true of brush-free motors. The prescribed maintenance for a typical large induction motor manufactured by a reputable firm involves the inspection of one pad in each thrust bearing and an oil change every 2 years, and the replacement of all seals in the lift system every 10 years.

Since no large, vertical eddy-current couplings have been built to date, representative maintenance data are not available. However, since this type of coupling is a relatively simple device whose problems are similar to those of large motors, it could probably be maintained as easily as a large motor could.

9.5.1.3. Intermediate Pipework and Valves

Pipework heaters and large sodium valves will probably account for most of the flow-conduit maintenance. Conservative design, coupled with features for easy repair, will be used to minimize problems in this area.

9.5.2. Intermediate Coolant System Auxiliaries

9.5.2.1. Intermediate Blanket Gas Subsystems

Although very little maintenance is anticipated for this subsystem, periodic inspections and performance tests will probably be specified in the plant operational procedure. Appropriate features permitting these activities will be incorporated into the design.

9.5.2.2. Intermediate Sodium Purification Subsystem

Maintenance is considerably influenced by the design of the intermediate sodium purification subsystem. Specific features that alleviate some of the handling problems are as follows:

1. Spent cold traps will be removed by cutting the inlet and exit sodium lines and lifting the cold trap out of the sheet metal shroud with the overhead crane installed to handle the steam generator and re-heater bundles.
2. The cold trap cartridge may be scrapped or cleaned, depending on which is the more economical approach. The cartridge internals are designed to be withdrawn from the outer shell when the top bulk-head is separated from the shell.
3. All electrical components are permanently installed.

9.5.2.3. Sodium Unloading Subsystem

The as-purchased sodium is expected to have a relatively high concentration of plug-forming impurities. If a filtration system were installed in the transfer pipe of the sodium unloading subsystem, then frequent plugging would probably occur; this would result in unacceptable time delays for filter maintenance and would increase the hazards. The two storage vessels will hold approximately one intermediate loop volume. However, during commissioning only one tank (50,000 gal.) will be filled and then processed to the other tank before being introduced into either the reactor coolant or the intermediate systems. Plans are to unload enough sodium at one time to fill the storage vessels and then batch-filter and purify it with the nonradioactive sodium purification subsystem before injection into the heat transport loops.

9.5.2.4. Nonradioactive Sodium Storage Subsystem

Very little maintenance is anticipated for this subsystem, but periodic inspections will probably be called for in the plant operational procedure. Appropriate features, such as easy removal of insulation, will be incorporated into the design to expedite these activities.

Since the nonradioactive sodium storage subsystem will be kept at a temperature high enough to permit emergency dumping of an intermediate heat transport loop, heating elements are likely to fail occasionally. Provisions will be made for easy replacement of these devices.

9.5.2.5. Nonradioactive Sodium Purification Subsystem

Since this subsystem is almost identical to the intermediate sodium purification subsystem, reference should be made to section 9.5.2.2 of this report. The only difference in the two subsystems is the addition of a filtration unit to prevent premature blockage of the cold trap flow by particulates received in the as-purchased sodium. During plant commissioning, this subsystem will purify all of the sodium that will be used in all systems of the plant. Therefore, a tabulation of the entire plant inventory is given:

Reactor coolant system, lb	2.85×10^6
Intermediate system, lb	0.68×10^6
Steam generation system, lb	<u>1.64</u>
Total Na inventory of entire plant	5.17×10^6

Sodium will be introduced into the reactor coolant system through temporary piping. Approximately 60 railroad tank car loads of new sodium will be received, so that frequent replacement of the filter material will be necessary. Easy regeneration of these filters will be essential to the expeditious preparation of as-purchased sodium before its transfer to the heat transport loops; therefore, regeneration will be given special attention in the design.

REFERENCES

Section 9

- ¹ 1000-MWe LMFBR Follow-On Study, Task II and III Final Report, Babcock & Wilcox, BAW-1328, Volume 2, Section 6, Reactor Coolant System, Lynchburg, Virginia, December 1968.
- ² BAW-1328, Volume 3, Section 10, Steam Generating System.
- ³ Burris, J. R. and Trost, R. J., 1000-MWe LMFBR Follow-On Study, Control Study, Babcock & Wilcox, BAW-1330, Lynchburg, Virginia, November 1968.
- ⁴ Handling Metallic Sodium on a Plant Scale, U.S. Industrial Chemicals Co., New York, New York, 3rd Ed. (1959).
- ⁵ Sales Catalogue Sheet No. 325, p. 1, Air Reduction Co., August 1963.
- ⁶ Materials Standards for Fast Reactors, Babcock & Wilcox, Std. No. AS4, Lynchburg, Virginia, April 15, 1968 (Company Confidential).
- ⁷ Hanna, J. H. and Bilisoly, C. S., "Operating Experience With Eddy-Current Couplings for Boiler Feed-Pump Drive, Proceedings of the American Power Conference, Vol XXVII, pp 477-483, Illinois Inst of Technology, Chicago, Illinois (1965).
- ⁸ Liquid Metals Handbook, Section 6, LMEC, June 30, 1967.
- ⁹ BAW-1328, Volume 3, Section 14, Safety System (Appendix 14-A).
- ¹⁰ Yevick, J. G. and Amorosi, A., Fast Reactor Technology, p 143, Massachusetts Inst. of Technology, Cambridge, Massachusetts (1966).

B&W Drawing A31113E —
 1000-MWe LMFBR - Reference
 Concept Intermediate Coolant
 Pump

NOTE:

ONE LOOP OF 3

HOT LINE

COLD LINE

ALL PIPE DIMENSIONS ARE O.D.

B&W Drawing A31114E —

1000-MWe LMFBR - Reference
Concept Intermediate Coolant
System Piping, Isometric View

Section 10

STEAM GENERATING SYSTEM

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The following B&W drawings are placed at the
end of this section:

A25586F 1000-MWe LMFBR Steam Generator, General Arrangement
A25587F 1000-MWe LMFBR Sodium-Heated Reheater, General
Arrangement

10. STEAM GENERATING SYSTEM

10.1. Introduction

10.1.1. System Function

The main function of the steam generating system is to produce superheated steam at constant temperature and pressure over the operating power range of the plant. The steam that drives the turbine in the energy conversion system is produced from heat generated in the reactor core and transferred by the intermediate coolant system to the steam generators. To fulfill this function, the system must operate safely and reliably and must produce steam at the lowest total power cost within the restraints imposed by technical limits and overall plant control requirements. The system will generate steam at a predetermined rate in response to load demands while on line over the service life of the plant.

10.1.2. Summary Description of System

The steam generating system consists of three steam generators, three sodium reheaters, and the associated auxiliary systems and components (see Figure 10-1). It includes the feedwater and steam piping in the steam generator building and the steam piping as well as the associated valves, up to and including the turbine stop valve. The relationships between the steam generating system components, the heat source, the heat transport loops, and the energy conversion system of the plant are shown in the overall schematic diagram, A20510F.^{1-4*} The physical layout of the steam generator building is shown in the overall plant system description in Volume 1. The steam generator and

* B&W drawing A20510F is placed at the back of Volume 1.

reheater, which are the principal components of the system, are shown in drawings A25586F* and A25587F.*

One steam generator and one reheater are located in each of the three intermediate sodium loops. Each loop circulates 30.70×10^6 lb/hr of sodium at 100% load, with 22.20×10^6 lb/hr going to the steam generator and 8.50×10^6 lb/hr going to the reheater. The sodium enters the steam generator and the reheater at 1000 F and leaves the units at 652 and 825 F, respectively. The exit streams are mixed and enter the IHX at 700 F. Table 10-1 summarizes pertinent design and performance data for the reactor coolant, intermediate coolant, steam generating, and energy conversion systems.

Three pipes carrying feedwater at 484 F enter the steam generator building from the turbine building. Each line is connected to a single steam generator only. In the steam generator the feedwater passes through the boiling and superheating sections in a single pass at a rate of 2.45×10^6 lb/hr. The steam lines leaving the steam generator are connected individually to the turbine stop valves of the high-pressure turbine. Steam is delivered at 2400 psig and 950 F.

Steam exhausted from the last stage of the high-pressure turbine at 585 psig and 610 F is recirculated by individual steam lines back into the steam generator building and is passed through the three sodium reheaters. After the steam is reheated to 950 F it is piped back, at 535 psig, into the turbine house at the rate of 2.28×10^6 lb/hr at 100% load. Here connections are made to the turbine stop valves of the intermediate-pressure turbine.

The steam generator is a once-through design on both sodium and water sides and features single-tube-wall construction. It is vertically mounted and bottom-supported. The steam generator has hemispherical heads and an internal shell liner. The heat transfer surface is a helical-coil, single-bundle design with counterflow of water and sodium. The upper head is flanged to permit the removal of internal parts for maintenance and repair.

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*B&W drawings A25586F and A25587F are placed at the back of Appendix 10-A.

Figure 10-1. Steam Generating System

Intermediate sodium enters the steam generator through two sodium inlet nozzles in the upper head, flows through a distributor system, and is discharged evenly over the upper row of tubes of the superheater and boiler tubes. The sodium leaves through the sodium outlet nozzle in the lower head.

Feedwater enters through the feedwater inlet tubesheet at the top and flows down past the outside of the boiler coils in annular rows of downcomers; then it turns and flows upward through the tubes of the boiler and superheater. Superheated steam passes upward through the superheater riser tubes and leaves through the superheater outlet tubesheet at the top of the steam generator.

The sodium reheater is similar in design to the steam generator. The primary differences between the units are their dimensions.

Instrumentation, controls, and relief and auxiliary systems are provided for safe, reliable operation of the system.

Table 10-1. Summary of Data for the Overall Plant and for the Steam Generating System at Full-Load Conditions

Overall Plant

Net generator output, MWe	1008
Turbine—tandem-compound, 6-flow output, MWe	1051
Steam cycle, psig/F/F	2400/950/950
Core power, MWt	2448
Total core flow, lb/hr $\times 10^6$	92.4
Core inlet temp, F	800
Core temp rise, F	300
No. of intermediate loops	3
Total intermediate coolant flow, lb/hr $\times 10^6$	92.4
No. of IHXs	6
Intermediate coolant temp to IHX, F	700
Intermediate coolant-IHX temp rise, F	300
Power transferred across each IHX, MWt	409
No. of intermediate pumps	3
No. of SG/loop	1
No. of RH/loop	1

Table 10-1. (Cont'd)

Steam Generating System

Steam generator

No. of units	3
Heat load, MWt/unit	688
Design life, yr	30

Sodium side operating conditions

Sodium flow per unit, lb/hr $\times 10^6$	22.20
Sodium inlet temp, F	1000
Sodium outlet temp, F	652
Avg sodium pressure, psig	40

Shell-side operating conditions

Design temp, F	1100
Design pressure, psig	200

Water-steam-side operating conditions

Steam flow per unit, lb/hr $\times 10^6$	2.45
Steam outlet temp, F	958
Feedwater inlet temp, F	484
Outlet steam pressure, psig	2544
Pressure drop, psi	500

Reheaters

No. of units	3
Heat load, MWe/unit	130
Design life, yr	30

Sodium-side operating conditions

Sodium flow per unit, lb/hr $\times 10^6$	8.50
Sodium inlet temp, F	1000
Sodium outlet temp, F	825
Avg sodium pressure, psig	40

Reheater shell design conditions

Temp, F	1100
Pressure, psig	200

Steam-side operating conditions

Steam flow/unit, lb/hr $\times 10^6$	2.28
Steam outlet temp, F	950
Steam inlet temp, F	610
Steam outlet pressure/pressure drop, psig/psi	535/40

Table 10-2. Instrumentation Data

Variable	Fluid	Function	Location
Flow	Feedwater	Control	SG inlet
	Sodium	Performance	SG inlet
	Sodium	Performance	SG outlet
	Sodium	Performance	IHX inlet
Temperature	Sodium	Control	SG inlet
	Sodium	Control	SG outlet
	Sodium	Control	RH inlet
	Sodium	Control	RH outlet
	Sodium	Performance and protection	IHX inlet
	Sodium	Performance and protection	IHX outlet
	Feedwater	Performance and protection	SG inlet
	Steam	Performance and protection	SG outlet
	Steam	Performance and protection	RH inlet
	Steam	Performance and protection	RH outlet
Differential pressure	Feedwater	Control	Across feedwater control valve
Pressure	Feedwater	Control	SG inlet
	Steam	Control	SG outlet
	Argon	Protection	In SG
	Argon	Protection	In RH
Level	Sodium	Performance and protection	In SG
	Sodium	Performance and protection	In RH
Hydrogen concentration	Argon + H ₂	Protection	SG & RH
Relief valve position	Argon, Na vapor, H ₂	Protection	SG & RH
Rupture disc integrity	Argon, Na vapor, H ₂	Protection	SG & RH

10.1.3. System Design Requirements

10.1.3.1. Performance

The steam generating system will be designed to deliver 7.35×10^6 lb/hr of steam to the high-pressure turbine throttle at 2400 psig and 950 F, in accordance with the steam cycle selected in the trade-off study.⁵ Reheat steam will be delivered to the intermediate-pressure turbine at 535 psig and 950 F. The system will be capable of operating down to 20% load and be capable of power maneuvers up to 2% per minute. Step changes in power demand of 5% per minute will also be accommodated. The system will also be required to operate from a cold startup to a cold shutdown and be capable of a hot restart after a turbine trip. Table 10-2 lists the instrumentation provided to monitor the protection, control, and performance functions of the steam generating system and its components. An excess heat transfer area of approximately 15% will be provided to allow for tube plugging and fouling.

10.1.3.2. Environment

The steam generators and reheaters will be located in individual areas separated by fire-resistant walls and above steel drip pans. All free sodium surfaces will be exposed to an atmosphere of commercial-grade argon.

10.1.3.3. Lifetime and Availability

The allowable membrane stresses for material operating in the creep range will reflect the operating environment and a 30-year lifetime.

10.1.3.4. Maintenance

Headers and tubesheets will be designed to provide access to all tube ends for leak detection and tube plugging without entry to the sodium side. The system will drain, by gravity, all sodium (as required) for active or preventive maintenance. Provisions will be made to purge all water from the tube side. A nitrogen pressurizing system will be provided to prevent leakage of sodium into a steam generator or reheater tube after a failure and an internal pressure

Table 10-3. Safety Relief Valves

<u>Component</u>	<u>Fluid</u>	<u>Type</u>	<u>Number - size, in.</u>	<u>Setpoint, psig</u>	<u>Total relieving capacity, lb/hr</u>
Steam generator	Steam	Spring-loaded	2 - 2.5	2790	252,000
Steam generator	Steam	Electrically operated	2 - 2.5	2700	568,300
Steam generator	Argon, H ₂ Na vapor	Rupture diaphragm	1 - 30	125	
Steam generator	Argon, H ₂ Na vapor	Spring-loaded	1 - 4	75	
Reheater	Steam	Spring-loaded	3 - 3	--	
Reheater	Argon, H ₂ Na vapor	Rupture diaphragm	1 - 20	120	
Reheater	Argon, H ₂ Na vapor	Spring-loaded	1 - 4	75	

reduction. The steam generators and reheaters will be designed so that the entire tube bundles can be removed for maintenance or replacement.

The design of the support structures will consider access for insulation and instrument maintenance, as well as shell inspection. The design will include isolation valves to adequately isolate the steam generators and reheaters for maintenance.

10.1.3.5. Operation

The system will be designed to operate on one, two, or three intermediate loops. The system will be capable of placing an intermediate loop into operation or taking a loop off line without shutting down the plant. In any case, all steam generators and all reheaters that are in service are to be operated at the same power level. The design will preclude the occurrence of deleterious instabilities during normal, startup, or shutdown operations.

10.1.3.6. Safety

Because of the nature of the sodium heat transfer fluid and its inherent reactivity with water, the sodium-water boundary is considered a critical area that requires special attention. To this end, the components of the system that contain sodium will have special instrumentation and relief devices to detect, relieve, and isolate any sodium-water reaction. The back flow of air into the component during its relieving function will be prevented. A rupture disc will eliminate the hydrogen pressure buildup caused by a large sodium-water reaction, and a smaller relief valve will be provided for smaller leaks. The system will incorporate a means to isolate and dump the steam side of both the steam generators and reheaters to the waste blowdown tanks in order to stop a detected steam water reaction.

Spring-loaded and electrically operated relief valves on the steam side of the steam generators will prevent overpressurization of the components and piping (see Table 10-3).

10.1.3.7. Design

The shell and steam sides of both the steam generator and the reheater will be designed in accordance with the latest

Table 10-4. Design and Operating Temperatures for Croloy 2-1/4 Components

Section	Operating pressure, psig	Design pressure, psig	Operating temp, F	Design temp, F	Allowable stress, psi ^(a)
<u>Steam Generator</u>					
Shell	50	200	1000	1100	3,000
Inlet tubesheet	2950	3250	484	550	15,000
Downcomer	2950	3250	484	550	15,000
Economizer and boiler	2925	3225	875	950	7,600
Primary superheat	2900	3200	925	1050	4,000
Secondary superheat	2900	3150	960	1100	3,000
Risers	2450	2750	958	1100	3,000
Outlet tubesheet	2450	2700	958	1000	5,500
Piping to HP turbine	2450	2700	950	1000	5,500
<u>Reheater</u>					
Shell	50	200	1000	1100	3,000
Inlet tubesheet	575	650	610	675	15,000
Downcomer	535	610	610	675	15,000
Reheater	535	610	900	1000	5,500
Outlet leg	535	610	950	1050	4,000
Outlet tubesheet	535	610	950	1050	4,000
Piping to turbine	535	610	950	1050	4,000

(a) Approximately 70% of 1962 ASME Boiler and Pressure Vessel Code, Section VIII, Stresses.

Table 10-5. Moments and Thrusts for Sodium Piping

	Moments, lb-in.	Thrust, lb
To steam generator	52,000	12,100
From steam generator	27,000	2,600
To reheater	26,900	5,500
From reheater	30,500	2,300

(a) These results are based on a sodium velocity of 20 fps and a hot leg operating temperature of 1025 F. Reducing the temperature to 1000 F and increasing the velocity to 25 fps will further reduce the values shown above.

edition of the ASME Pressure Vessel Code, Section III for Class C vessels (see Table 10-4 and Figure 10-2). External steam piping is designed in accordance with the ASA B 31.1 Pressure Piping Code, 1955 edition, with latest addenda and USAS B 31.1.0 Power Piping Code, 1967. Pressure relief valves on the steam-raising components will be designed according to Section I and the latest addenda of the ASME Code. The sodium nozzles on the steam generator and reheater will be designed to withstand the maximum piping thrusts and moments generated at the nozzle collection manifolds (Table 10-5). All nozzles will have prepared weld ends of the same material as the connecting pipe in order to avoid any field welds of dissimilar metals.

The basic design guide for seismic analysis is the AEC publication TID-7024, "Nuclear Reactors and Earthquakes." The design earthquake conditions are those defined in Volume I, section 1. In addition to normal loads, the tube supports, tube bundle supports, and the like will be capable of sustained loads caused by handling and shipping.

The feedwater and steam headers of both the steam generator and the reheater will be designed to ensure satisfactory flow distribution to each tube. The tube lengths for all parallel paths will be such that the pressure drop is approximately equal in each path.

The support structure will be capable of supporting dead-weight loads of 1,320,000 pounds for the steam generator and 594,000 pounds for the reheater. Lateral restraints will be designed to resist the seismic loadings mentioned above and any net piping reactions.

10.1.3.8. Interfaces

The steam generating system must be compatible with the intermediate cooling system and the energy conversion and feedwater systems.

The system control requirements will be cognizant of the characteristics of the feedwater regulation valves, the feedwater temperature limitations, the intermediate sodium temperature, and the steam temperature and pressure requirements of the turbines. The system will accept sodium from the storage facility using

Figure 10-2. Steam Generator Design and Operating Pressures

differential cover gas pressure and will be capable of gravity drainage to the sodium storage tanks.

10.2. Detailed Description of the System

10.2.1. General

The steam generating system comprises three steam generators, three sodium reheaters, and their auxiliary systems and components. It includes the feedwater and steam piping in the steam generator building and the associated valving, including the turbine stop valves. The system is primarily located in the steam generator building; the equipment arrangement is shown in Figures 4, 5, 7, and 9 in Volume 1 of this report. Some facets of the steam, turbine generator, and feedwater systems are described in the description of the energy conversion system. Section 12, Instrumentation and Control System, includes a detailed description of the overall plant instrumentation and control.

10.2.2. Steam Generator

One of the three identical steam generators is located in each intermediate loop (see A25586F). Each steam generator is of the once-through design on both the sodium and water sides, with single-tube-wall construction. The steam generators have been sized to meet the functional requirements of the heat generating and energy conversion systems using cost trends developed under a separate contract.^{6,7}

The steam generator consists of two sections—the lower shell and liner, and the upper head and tube bundle. Each section will be fabricated separately and assembled before shipment. A flanged connection joins the two sections so that the tube bundle can be removed for repair or replacement. A welded seal membrane provides a leak-proof seal at the flanges.

The lower section consists of the shell flange, straight shell section, lower hemispherical head, sodium outlet nozzle, support skirt, and shell liner. This section is bolted to foundation steel. It is made of Croloy 2-1/4, which was selected to be compatible with the design pressure and temperature in the sodium environment.

Figure 10-3. Steam Generator Temperature Enthalpy Diagram

The upper section consists of the head flange, upper hemispherical head, sodium inlet nozzles, feedwater inlet nozzles, superheater outlet nozzles; sodium distributor system, tubesheets, steam generator tube bundle (with its risers and downcomers), and nozzles for the sodium relief devices. The tube bundle and shrouds are supported from the upper head. The upper section is also of Croloy 2-1/4.

10.2.2.1. Functional Description (Figure 10-3)

Sodium Flow Path (100% Load) — Incoming sodium (at 1000 F) enters through two sodium inlet nozzles near the top of the vessel, flows downward through the sodium distributor system and is uniformly distributed over the top of the tube bundle with no bypassing. (The sodium distributors provide uniform distribution with as little turbulence and gas entrainment as possible.) The sodium then flows downward in cross-flow over the helical-coil tube bundle and leaves the steam generator through a single outlet nozzle at 652 F.

Steam and Water Flow Path (100% Load) — Feedwater enters the steam generator at 484 F through four inlet nozzles in the top head and flows downward in downcomer tubes arranged in the annular space between the outer shroud and the shell liner. The feedwater flows to the bottom of the tube bundle in these tubes and, after turning, flows upward in the helical-coil tube bundle, where it is changed to steam and superheated to 958 F. The steam leaves the tube bundle via the riser tubes to the superheater outlet headers. Each tube is a continuous flow path from the inlet header to the outlet header.

The water/steam side is essentially crevice-free. The sodium side has crevices in areas where no welding can be performed, e.g., between tube-to-tube supports, and where parts are mechanically assembled.

Argon Gas Blanket — The downcomer tubes are enclosed to maintain a virtually stagnant pool of sodium bounded on the outside by a liner and on the inside by a shroud. The liner is separated from the shell by an annular space filled with argon. The gas volume in the annulus and above the sodium level operates at a pressure of about 10 psig. The argon gas provides an inert atmosphere in contact with the sodium and serves to dampen pressure excursions that may

result from a sodium-water reaction. Since the tubesheets, vessel flanges, and support skirt-to-shell attachment are sensitive to rapid temperature changes, these parts are arranged so that they contact argon gas rather than sodium; thus they are protected from the rapid temperature changes that are typical of a sodium-heated steam generator.

10.2.2.2. Design Description

Outer Pressure Shell — The pressure shell is approximately 156 inches in ID and 70 feet long, including the upper and lower hemispherical heads and the support skirt. Two 28-inch-diameter sodium inlet nozzles are located in the upper head, and one 32-inch-diameter sodium outlet nozzle is located in the lower head. Four 6.5-inch-diameter feedwater inlet nozzles and four 6.5-inch-diameter superheater outlet nozzles are arranged alternately in the top head.

The shell and the lower hemispherical head are 1.5 inch thick; a 2.75-inch annular argon space separates the two. Horizontal support plates position the liner, which protects the pressure shell from rapid thermal transients. The liner also acts as a barrier to protect the shell from the corrosion that results from a sodium/water reaction. The upper hemispherical head is approximately 3.75 inches thick. The shell has a seismic brace in the same horizontal plane as the vessel center of gravity; it is attached to a fixed reference point.

Tube Bundle — The coiled-tube arrangement has a very compact heat transfer surface, and the steam generator is as compact as possible within the limits of other design criteria. The material for all tubes is Croloy 2-1/4. The downcomer tubes have an 0.88-inch OD and a wall thickness of at least 0.097 inch; each feedwater tubesheet contains 85 tubes. The tube bundle is made up of 340 helical-coil tubes, each 1.5 inch in OD, which form a circular bundle around a 28.25-inch-diameter center shroud. The tubes are stepped in three segments with minimum wall thicknesses of 0.170, 0.248, and 0.335 inch for the economizer and boiler, primary superheat, and secondary superheat sections, respectively. Thicknesses are determined by the pressures and temperatures in each section. The developed length of each tube in the bundle is approximately 410 feet. Riser tubes having a 0.302-inch minimum wall thickness connect the bundle to the four outlet tubesheets.

In the helical-coil design the risers and downcomers have enough flexibility to take care of movement between the coiled bundle and the shell. Each coil of tubes is pinned at the top and free to expand at the bottom. Except at the tube supports, the tubes are unrestrained, but each tube is clamped securely in its tube support to prevent vibration. The tubesheets are not immersed in sodium and see only water/steam transients; no rapid sodium transients are imposed on these parts. The vessel head and shell in the vicinity of the inlet and outlet sodium nozzles are protected by thermal sleeves. Table 10-6 and drawing A25586F provide additional data and heat transfer results.

10.2.2.3. Relief Systems

Sodium Side — The steam generator is designed to withstand the initial effects of a large sodium-water reaction (which includes a credible number of tubes failing simultaneously) without becoming a hazard to nearby personnel and equipment. Although the unit described here has not been quantitatively analyzed, the results of previous analyses on a similar sodium-heated steam generator indicated that 25 tubes could fail simultaneously without endangering plant safety.⁸ After the initial excursion, the products of the large sodium-water reaction must be relieved, since a feasible design to contain the reaction would be unattainable.

A 30-inch rupture diaphragm is installed to relieve large reactions safely; it is set for 125 psi, which is 75 psi below the vessel design pressure of 200 psi. In addition, a 4-inch safety valve (set at 75 psi) is installed in the upper head. The safety valve, set at a lower pressure than the large rupture discs, will relieve minor pressure surges due to smaller sodium-water reactions or to malfunction of the cover gas pressure control system. After it has relieved the pressure, the safety valve will reseal, whereas the rupture disc must be replaced.

Table 10-6. 1000-MWe LMFBR Steam Generator Heat Transfer and Pressure Drop Results at Full-Load Conditions

Operating Conditions

Heat load per SG, Btu/hr $\times 10^8$	23.48
Sodium flow per SG, lb/hr $\times 10^6$	22.20
Sodium inlet temp, F	1000
Sodium outlet temp, F	652
Steam flow per SG, lb/hr $\times 10^6$	2.45
Feedwater temp, F	484
Outlet steam temp, F	958
Outlet steam pressure, psig	2544

Materials

Tubes	Croloy 2-1/4
Shells, head	Croloy 2-1/4
Tubesheets, nozzles	Croloy 2-1/4

Heat Transfer Results

Tube section	Number	OD, in.	Minimum wall, in.	Effective length, ft
Economizer	340	1.5	0.170	107
Nucleate & film boiling	340	1.5	0.170	71
Primary superheat	340	1.5	0.248	81
Secondary superheat	340	1.5	0.335	150
Boiler inlet tubes				
Size—OD/min wall, in.				0.875/0.097
Length, ft				55
Boiler outlet legs				
Size—OD/min wall, in.				1.5/0.302
Length, ft				13
Fouling factor, Btu/hr-ft ² -°F				10,000
Corrosion allowance, in.				0.010
Minimum wall tolerance, %				10
Transverse tube spacing, in.				2
Parallel tube spacing, in.				2
Overall heat transfer coefficient, Btu/hr-ft ² -°F				
Economizer section				536
Nucleate boiling section				634
Film boiling section				495
Primary superheat section				385
Secondary superheat section				266

Table 10-6. (Cont'd)

Heat transfer surface, ft ²	
Economizer section	14,100
Nucleate boiling section	5,810
Film boiling section	3,510
Primary superheat section	10,700
Secondary superheat section	<u>19,700</u>
Total heat transfer surface	53,820
 <u>Pressure Drop</u>	
Steam side, psi	
Downcomer tubes	10.0
Economizer section	4.8
Nucleate boiling section	5.4
Film boiling section	6.8
Primary superheat section	39.4
Secondary superheat section	397.0
Riser tubes	<u>36.6</u>
Total	500.0
Sodium side, psi	17.7
 <u>Nozzles</u>	
Number and diam per SG, No. /in.	
Sodium inlet	2/28.0
Sodium outlet	1/32.0
Rupture disc	1/30.0
Feedwater inlet	4/6.5
Steam outlet	4/6.25

Steam Side — Relief devices for the steam side of the steam generator are sized according to requirements of Section 1 and the latest addenda of the ASME Code. The spring-loaded safety valves are Maxiflow 171OD or the equivalent. The two 2.5-inch valves are to be set at 2790 psig with an orifice size of 0.994 inch. The total capacity is 252,000 lb/hr at design temperature and pressure conditions.

To afford greater reliability to the spring-loaded safety valves, a power relief valve system is also installed. Two 2.5-inch, electrically operated relief valves, having a total capacity of

Figure 10-4. Reheater Temperature Enthalpy Diagram

Capacity, Btu/hr $\times 10^8$	4.45
Steam Flow, lb/hr $\times 10^6$	2.28
Sodium Flow, lb/hr $\times 10^6$	8.50

1000 F
 T = 950 F
 P = 535 psig
 h = 1492 Btu/lb

568,300 lb/hr at design conditions, are used. The valves are set at 2700 psig. The additional valves prevent the spring-loaded safety valves from lifting under normal conditions and thus substantially reduce safety valve maintenance. The valves required are Consolidated Type 1538-VX or the equivalent.

10.2.3. Reheater

Three identical reheaters are used in this system; each is of single-tube-wall construction (see drawing A25587F). The reheater has been designed to be compatible with and to meet the functional requirements of the intermediate-pressure turbine and the intermediate coolant system. It is similar in design to the steam generator, except that it does not incorporate the stepped tube bundle used in the steam generator design. Like the steam generator, the reheater consists of two sections: the lower shell and liner, and the upper head and tube bundle. A full flanged connection joins the two sections so that the tube bundle can be removed for repair or replacement. The two sections consist of the same material and parts as those described for the steam generator in section 10.2.2.

10.2.3.1. Functional Description (Figure 10-4)

Sodium Flow Path — Sodium from the IHX (flowing in parallel with the steam generator) enters at 1000 F through two sodium inlet nozzles near the top of the reheater. Inside the reheater the sodium flows downward to the distribution system and over the tube bundle. The distribution system provides uniformity of flow with minimal turbulence and gas entrainment. Sodium flows downward over the helical-coil tube bundle in cross-flow and leaves the reheater at 825 F.

Steam Flow Path — Steam enters the reheater at 610 F through four inlet nozzles in the top head and flows downward in downcomer tubes between the outer shroud and the shell liner. It flows to the bottom of the tube bundle in these downcomer tubes, then turns and flows upward in the helical-coil tube bundle, where it gains enthalpy and is superheated to 950 F. The steam leaves the tube bundle via the riser tubes to the four steam outlet nozzles.

Table 10-7. 1000-MWe LMFBR Reheater Heat Transfer and Pressure Drop Results at Full-Load Conditions

Operating Conditions

Heat load per RH, Btu/hr $\times 10^8$	4.45
Sodium flow per RH, lb/hr $\times 10^6$	8.50
Sodium inlet temp, F	1000
Sodium outlet temp, F	825
Steam flow per RH, lb/hr $\times 10^6$	2.28
Inlet steam temp, F	610
Outlet steam temp, F	950
Outlet steam pressure, psig	535
Inlet steam pressure, psig	575

Material

Tubes	Croloy 2-1/4
Shells, heads	Croloy 2-1/4
Tubesheets, nozzles	Croloy 2-1/4

Heat Transfer Results

Number of tubes	292
Tube size - OD/min wall, in.	2.0/0.111
Tube effective length, ft	114
Transverse tube spacing, in.	2.5
Parallel tube spacing, in.	2.5
Overall heat transfer coefficient, Btu/hr-ft ² -°F	249
Fouling factor, Btu/hr-ft ² -°F	10,000
Corrosion allowance, in.	0.010
Min wall tolerance, %	10
Total heat transfer surface, ft ²	17,400
Inlet tubes	

Size - OD/min wall, in.	2.0/0.111
Length, ft	22

Outlet tubes

Size - OD/min wall, in.	2.0/0.111
Length, ft	12

Pressure Drop

Steam side, psi

Inlet leg	5.6
Reheater	28.0
Outlet leg	6.4

Total	40.0
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Sodium side, psi	9.2
------------------	-----

Nozzles

Number/diam per RH, No./in.

Sodium inlet	2/18.0
Sodium outlet	1/26.0
Rupture disc	1/20.0
Steam inlet	4/14.0
Steam outlet	4/16.0

Argon Gas Blanket — The argon gas blanket serves the same purpose for the reheaters as it does for the steam generators described previously.

10.2.3.2. Design Description

Outer Pressure Shell — The pressure shell is approximately 146 inches in ID and 53 feet long, including the upper and lower hemispherical heads and the support skirt. Two 18-inch-diameter sodium inlet nozzles are placed in the upper head, and the lower head has one 26-inch-diameter sodium outlet nozzle. Four 14-inch-diameter steam inlet nozzles and four 16-inch-diameter steam outlet nozzles are arranged alternately in the top head.

The shell and the lower hemispherical head are 1.5 inches thick; a 2.75-inch annular argon space separates the two. Horizontal support plates are provided to position the liner, which protects the pressure shell from rapid thermal transients. The upper hemispherical head is about 4.25 inches thick. The shell has a seismic brace located in the same horizontal plane as the vessel's center of gravity; it is attached to a fixed reference point.

Tube Bundle — The coiled-tube arrangement is one of the most compact heat transfer surface arrangements possible. The reheater is made as compact as possible within the limits of other design criteria. The material for all tubes is Croloy 2-1/4. The down-comer tubes have a 2-inch OD and an 0.111-inch minimum wall thickness. Each steam inlet tubesheet contains 73 tubes. The tube bundle is made up of 292 helical-coil tubes, which form a circular bundle around a 31.25-inch-diameter center shroud.

The tubes have a constant minimum wall thickness which is determined by the pressures and temperatures in the bundle. The developed length of each tube in the bundle is approximately 114 feet. Riser tubes having a minimum wall thickness of 0.111 inch connect the bundle to the four outlet tubesheets.

The problem of thermal expansion in the tubes and shell, support of the tubes, and thermal transients are similar to those of the steam generator and are handled as described for that component. Table 10-7 and drawing A25587F provide additional data and heat transfer results.

10.2.3.3. Relief Systems

Sodium Side — The reheater is designed to withstand the initial effects of a large sodium-water reaction (which includes a credible number of tubes failing simultaneously) without becoming a hazard to nearby personnel and equipment. The relief system for the reheater is common with the steam generator, since the probability of having a sodium-water reaction in both units at the same time is small.

A 20-inch rupture diaphragm is installed to relieve large reactions safely; it is set for 125 psi, which is 75 psi below the vessel design pressure of 200 psi. In addition, a 4-inch safety valve (set at a lower pressure than the large rupture disc) will relieve minor pressure surges due to smaller sodium/steam reactions or to malfunction of the cover gas pressure control system. After it has relieved the pressure, the safety valve will reseal. Section 10.2.2 includes a description of the vent system.

Steam Side — Relief devices for the steam side of the reheater are sized according to the requirements of Section 1 and the latest addenda of the ASME Code. The spring-loaded safety valves are Maxiflow 1755D or the equivalent. The three 3-inch valves are to be set at 580 psig with an orifice size of 3.341 inches. The total capacity is 249,270 lb/hr at design temperature and pressure conditions.

10.2.4. Piping

10.2.4.1. Main Feed and High Pressure Steam

The feedwater piping to each steam generator consists of a 14-inch-ID $\times$ 2.10-inch-thick carbon steel pipe which branches into two 9-inch-ID $\times$ 1.35-inch-thick pipes. These are attached to a feedwater inlet ring which encircles the steam generator. Attachments for the four 6.5-inch feedwater nozzles are connected to the ring.

The steam outlet piping has a similar layout. Four 6.5-inch outlet nozzles are attached to an outlet header which encircles the steam generator. From this ring, two 8.75-inch-ID $\times$ 1.5-inch-thick pipes lead to a single 12.25-inch $\times$ 2.0-inch-thick pipe.

10.2.4.2. Reheat Piping

Piping in the immediate vicinity of the reheater includes one 24-inch schedule 40 pipe from the high-pressure turbine which branches into two 18-inch schedule 40 pipes terminating in a ring encircling the reheater. From the ring, four pipes exit, connecting to the four 14-inch steam inlet nozzles. The four 16-inch steam outlet nozzles also connect to a ring encircling the reheater. Two 22-inch schedule 60 pipes exit from the ring and continue to the intermediate-pressure turbine.

10.2.5. Instrumentation

10.2.5.1. Sodium Level Indication

The sodium level indicator will be a double-coil, eddy-current instrument that depends on liquid sodium as a coupling medium. The coil is completely enclosed in a stainless-steel tube. As sodium surrounds the tube, there is a coupling between the two coils through the sodium; this coupling is read out on instruments that indicate sodium level. Two level indicators are used in each steam generator and each reheater.

As a backup for the primary level indicators, there will be an externally mounted, float-type level indicator of stainless-steel construction. Connection to the gas space will be through the upper hemispherical head. The sodium connection to the instrument will be in the sodium pipe leaving the bottom of the steam generator.

10.2.5.2. Sodium-Water Reaction Detection

Two types of sodium-water reaction are of concern in a steam generator or reheater. One is a large reaction resulting from the failure of the maximum credible number of tubes. The other type of reaction that may be encountered is a small reaction resulting from a pin-hole tube leak. The instrumentation needed to detect a large reaction is available and is listed here for the proposed steam generator and reheater:

1. Gas chromatograph for detecting hydrogen in the cover gas.

2. Pressure sensors for detecting rapid increases in cover gas pressure.
3. Devices to detect rupture disc failure or relief valve lifting.

The products of a small reaction are highly corrosive and must be detected rapidly before the leak increases significantly. There is also some question as to whether the products of the reaction will remain in the sodium or be forced into the water/steam side by the pressures of local reactions. Additional work is required to determine the distribution of the products of a small reaction.

Instrumentation to detect the presence of sodium in steam rapidly is available, but present methods of detecting reaction products on the sodium side are inadequate for small leaks. These methods include those installed for detecting large reactions plus constant sampling of the sodium for its sodium hydroxide content.

In addition to the development of the instruments required for small leak detection, more effort must be expended in determining where such instruments should be placed within the unit for maximum effectiveness.

10.3. Principles of Operation

10.3.1. Cold Startup

This kind of startup assumes the following:

1. The reactor coolant system is filled and is circulating at approximately 800 F.
2. Decay heat is being removed by the decay heat removal system as required.
3. The intermediate system is clean, dry, and inerted.
4. Feedwater at 300 F is circulating via the startup pump through one steam generator, by-passing the turbines and the main condensers, and through the six feedwater heaters.

Sodium will be added to the intermediate system at approximately 300 F until the entire system is filled. Since the shell side of the IHX is at 800 F, and the tube-side sodium was introduced at 300 F, heat will be transferred to the intermediate system. Operation of

the intermediate pumps at approximately 2% of full rpm will provide adequate distribution of temperatures during heatup.

To prevent heat from being transferred from the steam generator to the circulating boiler feedwater during heatup of the intermediate system, the steam generator will be isolated from the intermediate system. Proper feed and bleed of hot sodium to the steam generator will maintain its sodium temperature between 350 and 400 F.

Meanwhile, the sodium in the intermediate piping and reheater of the intermediate loop will be brought up to the temperature of the circulating reactor coolant, i. e., about 800 F.

Before it produces steam, the feedwater system will be realigned to include the condenser hot wells and condensate pumps.

The actual production of steam will require the addition of enough heat to increase the temperature level of the sodium in the steam generator as well as supplying the heat that is being extracted from the steam generator by the feedwater. This will be accomplished by throttling the sodium flow to the reheater and simultaneously feeding and bleeding hot sodium to and from the steam generator. Steam produced by the steam generator will by-pass around the turbines until the proper temperature and pressure are achieved.

10.3.1.1. Summary of Cold Startup Requirements

1. Twenty percent flow is required to prevent steam flow reversals (instability) in the steam generators.
2. Power levels greater than 2% of the full thermal power output of the core at less than operating temperatures are undesirable.
3. The increase or decrease of average system temperatures shall not exceed 100 °F per hour.
4. The IHXs, reheaters, and steam generators are not to be slugged with sodium—this would cause stressing due to large temperature differences.

10.3.1.2. Approach

1. Circulation of feedwater at 20% satisfies item 1 above.

2. Isolation of the steam generator prevents the transfer of large blocks of power (item 2 above) while the intermediate system is coming up to temperature in accordance with item 3.

3. Introducing the sodium slowly at 300 F will prevent slugging the IHXs (as in item 4). The steam generator and reheaters will not be affected by the low-temperature sodium.

4. Operation of the intermediate pumps at a very low circulation rate should not be harmful to the IHXs.

5. Feeding high-temperature sodium to the steam generator to maintain or raise its temperature is acceptable, since incoming sodium first strikes a pool of sodium before impinging on the tubes.

After the first steam generator is producing steam for the high-pressure turbine, exhaust from its stage will be introduced to the reheater. Sodium flow will be adjusted to yield the proper steam reheat temperatures. The remaining steam generators will be brought on line, one at a time, following these same general procedures.

10.3.2. Hot Restart After a Scram

Here our concern is to place the plant in a standby condition pending the decision to restart. The standby condition will be independent of the number of loops that were in operation at the time of shutdown. It will be assumed that all three loops had been operating at full power before the reactor scram.

Reviewing the plant conditions immediately after a scram, we should find the following to be true:

1. Approximately 20% of the entire reactor coolant system inventory (5×10^5 lb) will be at 1100 F, making the average temperature 860 F.

2. Assuming that the turbine trips simultaneously with rod insertion, the intermediate system would be thermally isolated at an average temperature of 845 F.

3. All feed pumps will be secured to prevent (1) dragging the steam generator and reheater sodium temperatures down at an unpermissible rate, and (2) the possibility of liquid water hitting the superheater steam outlet tubesheets.

4. The emergency decay heat cooling system would come on line to remove the decay heat and the sensible heat in the reactor coolant and intermediate coolant system.

5. The reactor coolant and intermediate coolant system flow will be reduced to produce an isothermal condition.

6. Power for all plant operations, except the emergency power supply systems, will be supplied from the utility grid network.

Once it has been decided that the plant can come back on line, the following general procedure will be initiated.

1. The steam generator sodium side will be isolated.
2. The intermediate system circulation will be maintained.
3. All three steam generators will be gravity-drained.
4. The steam generators will then be cooled down at approximately 100 °F per hour until they reach 300 F.
5. The sodium in the storage tanks will be cooled to 300 F.
6. At this point, feedwater can be introduced into the steam generators with the startup pump. The feedwater will then be circulated by passing the condensers.
7. The slow addition of hot sodium will heat the steam generator; the heat for this purpose will come from the reactor core decay heat.
8. When the temperature of the water leaving the steam generator is high enough to start steaming, the condensers will be cut into the feedwater circuit.

10.3.2.1. Summary of Hot Startup Requirements

In addition to the requirements stated in the cold startup summary, these factors are required:

1. Once the feedwater flow has been shut off from the steam generators, flow cannot be re-established if a difference of more than 100 F exists between the inlet tubesheet and the feedwater temperatures.
2. Both the reactor coolant and intermediate system should be operated as close to 800 F as possible in order to prevent deleterious temperature differences.

10.3.2.2. Approach

1. The startup pump, which can supply 20% flow, prevents boiler instabilities.
2. Securing the feedwater pumps prevents both the reactor coolant system and the intermediate system from dropping in temperature at a prohibitive rate.
3. Isolation sodium drainage and cooldown of the steam generators is necessary to minimize the temperature difference between the inlet tubesheet and the feedwater when feedwater flow is re-established.

10.3.3. Normal Operation

10.3.3.1. Feedwater Chemistry

The recommended limits on feedwater chemistry requirements are listed in Appendix 10-A. These limits are established by the requirements of all of the equipment in the steam cycle, including the turbine, rather than by the steam generators alone. The limits stated are common in modern power plant practice.

10.3.3.2. Level Variation

The normal sodium level in the steam generator and the reheater is 4 feet 9 inches above the tube bundles at full load. In both components the level may increase by about 12 inches or decrease by about 30 inches without impairing the performance of the units.

The argon cover gas pressure will change slightly with sodium level variation. Although detailed analyses have not been made, it appears from previous experience that sodium levels and gas pressure can be established at 20% load and will change only slightly at greater power levels. This will minimize the need for feeding and bleeding the cover gas to maintain proper levels.

10.3.3.3. Load Control

The output of the steam generating system is controlled by the master control system described in section 8. This system will determine steam demand and will provide output for the control of feedwater and sodium to the steam generating system.

10.3.4. Emergency Operation

In any sodium-heated steam-raising equipment it is possible that a leak, with a resulting sodium-water reaction, will occur. To bring the reaction under control, either the sodium or the water must be dumped from the steam generator; faster results can be obtained by dumping the water. In addition, the dumping of sodium presents the possibility of carrying the reaction into the sodium dump tank.

To speed the dumping of the water, hydraulically operated, quick-acting isolation valves are used in the steam line and feedwater line to each steam generator. Positioned between the feed isolation valve and the steam generator is a dump line with a hydraulically operated, quick-acting valve. During normal operation the two isolation valves are open and the dump valve is closed. To prevent initiation of the dump system during normal operation, both isolation valves and the dump valve fail in the "as-is" position. The dump system can be initiated manually if the automatic controls should fail.

When a sodium-water reaction is detected by the monitoring instruments described in section 11.2.5.2, the steam generator is automatically isolated and the water side is dumped. The isolation valves on the steam outlet and feedwater inlet close to isolate the leaking steam generator from external sources of pressure, and the dump valve opens to provide a flow path for the water, which is forced out by steam pressure. These three valves are interlocked to ensure that the dump line flow path is available before the steam generator is isolated.

Figure 10-5. Dump System for One Steam Generator System, Schematic Diagram

When the water is gone, the steam pressure will decrease rapidly, and the sodium-water reaction will cease. To impede the flow of sodium into the water side, a nitrogen purge system will automatically come into service when the steam pressure is reduced to about 200 psi. The dump line valve will close shortly after the purge is started to keep the water side at a higher pressure than the sodium side.

An isolation and dump system is also provided for each reheater. The configuration and principles of operation of the system are similar to those of the steam generator system. A schematic diagram of the dump system for one steam generator or reheater is shown in Figure 10-5.

10.4. Safety Precautions

10.4.1. Hazards

It is considered credible that at some time during the 30-year life of the steam generator, some type of leak will develop in which water will react with sodium. This condition has been considered in designing the sodium-heated steam generator. The design of the steam generator is such that the shell is protected by a liner separated from the shell by an inert gas annulus. A gas cushion is provided over the sodium in each steam generator, and relief valves and rupture discs are provided for the safe relief of the products of a sodium-water reaction. The two areas of greatest concern in providing for the safe handling of a sodium-water reaction in a steam generator are

1. Small leaks — It is important to detect small leaks as early as possible and to take corrective action before corrosion (resulting from the sodium-water reaction) has damaged the steam generator and before the small leak has had a chance to become a large leak. Rapid and reliable detection techniques are extremely important. This is an area where additional investigation is required.

2. Large leaks — In the case of large leaks, it is important to isolate the steam generator from the feedwater and the steam headers and to vent the water to the dump system. The more water that is removed through an isolation and dump system, the less water will be available to react with the sodium. Relief devices must be provided to

vent the products of the sodium-water reaction safely without allowing the pressure in the steam generator to build up higher than it can safely withstand. The steam generator shell must be capable of withstanding the maximum pressure peaks that may occur at any time during the reaction. This is imperative for the safety of personnel and equipment and is a prime design objective.

B&W has analyzed the strength of steam generators to determine the extent of damage due to the pressures and temperatures predicted in a cooperative effort by APDA and B&W. The analysis for a similar full-sized steam generator is described in the preliminary design report Full-Size Steam Generator Sodium-Water Reaction (BW 67-2). As a result of this study, it was concluded that the subject steam generator will withstand the simultaneous failure of a credible number of tubes without becoming a hazard to nearby personnel and equipment. Very conservative assumptions were used throughout, from the prediction of pressures and temperatures by APDA to the dynamic analyses of the steam generator shell and internal parts.

The basic design features of the steam generator presented in this report (section 10.2.2) are similar to those of the unit analyzed for sodium-water reaction. However, both the shell and tube designs have been simplified; this should improve the performance of the unit under conditions of a sodium-water reaction accident.

10.5. Maintenance

10.5.1. Steam Generator and Reheater

The sodium sides of both the steam generator and reheater can be completely drained. This permits bundle removal from the shell for inspection or repair by opening the flanged head. The necessary handling equipment and maintenance facilities, described in section 7, are provided.

The water/steam side of the units is not drainable, but most of the water can be removed as discussed in section 10.3.4. The desirability of a drainable water/steam side was traded off in this concept against the advantages of having the tubesheets out of sodium for protection against thermal transients, and having the tube bundle removable from the shell for major inspection, maintenance and repair.

Normal preventive or (minor) corrective maintenance can be accomplished without removing the tube bundle from the shell. Solids are cleaned from the water/steam side of the tubes in the once-through steam generator without removing the tube bundle from the shell. Acid cleaning and subsequent flushing with demineralized water cleans the solids from the water/steam side of the tubes in the once-through steam generator. Inspection and maintenance of the tubesheet areas, including plugging of individual tubes, are performed using handholes in each inlet and outlet header. If additional access is required, the piping can be cut.

10.5.2. Support Systems

The remaining components and piping in the steam generating system do not require unusual maintenance. Preventive maintenance for these systems involves normal steam plant items, such as safety valve setpoint checks and valve actuating mechanism checks. No special provisions are required for corrective maintenance.

REFERENCES

Section 10

- ¹ 1000-MWe Follow-On-Study, Task II and III Final Report, Babcock & Wilcox, BAW-1328, Volume 2, Section 4, Reactor Core System, Lynchburg, Virginia, December 1968.
- ² BAW-1328, Volume 2, Section 6, Reactor Coolant System.
- ³ BAW-1328, Volume 3, Section 9, Intermediate Coolant System.
- ⁴ BAW-1328, Volume 3, Section 11, Energy Conversion System.
- ⁵ BAW-1328, Volume 4, Section 5, Steam Cycle Selection Trade-Off Study.
- ⁶ Full Size Steam Generator Sodium-Water Reaction, Preliminary Design Report, Babcock & Wilcox, BW-67-2, AEC Contract No. AT (11-1)-1280.
- ⁷ Sodium to Steam Heat Exchanger Elements and Research, Babcock & Wilcox, BAW-1280-46, AEC Contract No. AT(11-1)-1280.
- ⁸ Preliminary Design Report — Prototype Steam Generator, Babcock & Wilcox, BAW-1280-28, Barberton, Ohio, November 1965.

APPENDIX 10-A
Recommended Feedwater Chemistry Requirements

The pH value should be adjusted to obtain a maximum of 0.01 ppm iron, which will normally require a pH between 8.5 and 9.2 at 77 F.

Item	Limit, ppm
Maximum total dissolved solids	0.05
Suspended solids	Minimum - preferably zero
Hardness	0.0
Organic	0.0
Free caustic	0.0
Maximum dissolved oxygen	0.007 - preferably zero
Carbon dioxide	Minimum - preferably zero
Maximum total silica (as SiO ₂)	0.02
Maximum total iron (as Fe)	0.01
Maximum total copper (as Cu)	0.002

In modern central stations this feedwater purity is maintained by eliminating condenser leakage, demineralizing makeup water, careful control of the condensate system chemistry, and full-flow condensate polishing demineralizers.

B&W Drawing A25586F —
 1000-MWe LMFBR Steam Generator,
 General Arrangement

5

Section 11

ENERGY CONVERSION SYSTEM

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11. ENERGY CONVERSION SYSTEM

11.1. Introduction

11.1.1. System Function

The primary function of the energy conversion system is to convert thermal energy that is supplied as steam into electrical energy economically and efficiently. To do this, heat generated in the primary system is transferred via the primary and intermediate coolant system to the steam generating system to produce the steam that drives the turbine-generator unit. To fulfill the prime function, the system must generate electricity to meet the demand of a utility company's load-following system.

The closed condensate and feedwater subsystem provides the means for condensing, transporting, deaerating, heating, polishing, and pressurizing the condensate water for admission into the steam generator.

11.1.2. System Summary Description

The energy conversion system comprises the turbine generator unit, condensate and feedwater equipment, and the associated auxiliary systems. The gross electrical output is 1033.9 MWe; based on the estimated auxiliary power requirements of 25 MWe, the net output is 1008.9 MWe and the overall net plant efficiency is 41.2%. The heat balance diagram is shown on Figure 11-1, the steam condensate and feedwater piping diagram is shown on Figure 11-2, and the principle system parameters are summarized in Table 11-1. The turbine plant arrangement is shown in Figures 5 through 11, Volume 1, section 2.

The turbine unit is a single-shaft, multi-cylinder, tandem-compound, six-flow reheat machine whose last stage is a 30-inch-diameter bucket arrangement. The unit consists of a single-flow,

high pressure turbine operating from 2400 psig to 600 psia, a double-flow intermediate reheat turbine operating from 540 to 149 psia, and three identical double-flow low-pressure turbines operating from 142 psia to 1.5 inch Hg absolute.

The generator unit is a 3600 rpm machine, and the rotating rectifier exiter is a totally enclosed self-ventilated assembly located on the generator shaft. Power is generated at approximately 25,000 volts, 3 phase, 60 cycles.

The turbine generator set is arranged longitudinally in the turbine hall and is carried on a separate reinforced concrete foundation block. The condenser is mounted longitudinally under the low-pressure turbine, and the feed pump, heaters and chemical plants are located at various levels below the turbine.

Steam at 2400 psig and 950 F flows from the three steam generators and through four control valves into the single-flow high-pressure turbine at the rate of 7.35×10^6 lb/hr. Extraction steam is taken from the high-pressure turbine at the rate of 0.46×10^6 lb/hr for operation of the final stage of the high-pressure feedwater heaters. The remaining steam, including losses, is taken at the rate of 6.84×10^6 lb/hr from the last stage of the high-pressure turbine at 600 psia and 610 F and is directed to the three sodium reheaters where it is reheated to 950 F. The steam is then directed back to the double-flow intermediate pressure turbine (IPT) which operates from 540 to 149 psia. Extraction steam is taken from the IPT at a rate of 0.28×10^6 lb/hr for operation of the boiler feed pump turbine drive, and 0.36×10^6 lb/hr for operation of the deaerating heater. In addition, 0.48×10^6 lb/hr of steam at a pressure of 379 psia is bled from the IPT for operation of the initial stage of the high-pressure feedwater heaters. The remainder of the steam from the IPT, including losses, is taken at the rate of 5.71×10^6 lb/hr from the last stage of the IPT and is directed to the three identical double-flow low-pressure turbines which operate at 142 psia to 1.5 inch Hg absolute. Extract steam from the LPT passes into the three identical two-pass condensers which operate in parallel with a total heat duty of 4.79×10^9 Btu/hr.

GENERATOR GROSS OUTPUT =
1,033,926 Kwe
AUX. POWER = 25,000 Kwe
NET POWER OUTPUT = 1,008,926 Kwe

$$* \text{ GROSS HEAT RATE} = \frac{7,350,000(1427.8 - 469.3) + 6,837,000(1492.08 - 1298.1)}{1,033,926 + 25,201} = 7903.9 \text{ BTU/KW-HR}$$

Table 11-1. Principal Parameters of the Energy Conversion System

<u>Steam Cycle</u>	Rankine, reheat, six-stage regenerative
<u>Power</u>	
Reactor plant output, MWt	2450
Gross electric power generation, MWe	1034
<u>Turbine-Generator</u>	
Designation	TC6F-30 reheat
Number of casings	5
Speed, rpm	3600
Turbine rating, Hg absolute	1.5
Turbine makeup, %	0
Number of heater stages	6 extraction
Steam conditions at turbine throttle, psig/F	2400/950
Steam flow at turbine throttle, lb/hr	7,350,000
Steam conditions at reheat turbine, psia/F	540/950
Steam flow at reheat turbine, lb/hr	6,837,000
<u>Condensers</u>	
Number of units	3
Total heat rejected, Btu/hr	4.790×10^9
Surface area (each), ft ²	172,788
Number of tubes (each)	16,500
Cooling water supplied (each), gpm	120,000
Tube dimensions	
OD, in.	1
Thickness, BWG	18
Length, ft	40
Cooling water temperature (avg min/avg max), F	40-75
<u>Condensate Pumps</u>	
Number of units	3 (1-standby)
Flow (each), gpm	5000
Inlet head, ft of H ₂ O	25
Discharge head, ft of H ₂ O	560
Driver, hp	(Electric motor)
Low-pressure drain pumps	3

Table 11-1. (Cont'd)

Low-Pressure Heaters

Total number	7
Number of stages	3
Number per stage	3 stage A 2 stages B & C
Type	Closed shell and tube, horizontal, stage A vertical, stages B & C
Terminal temperature, F	289

Deaerating Heater

Total number	1
Type	Direct contact, horizontal
Storage tank capacity, gal	75,000
Elevation head, ft above BFP suction	155
Terminal temperature, F	354

Boiler Feed Pumps

Number of units	2
Flow (each), gpm	8300
Inlet head, ft of H ₂ O	150
Discharge head, ft of H ₂ O	8150
Driver type	Condensing, 17,100 hp, extraction, steam-turbine drive
Emergency pump (20% rating - electric drive)	1

High-Pressure Heaters

Total number	4
Number of stages	2
Number per stage	2
Type	Closed shell and tube, vertical
Terminal temperature, F	484

The condensate is transferred at 92 F by three half-size condensate pumps and is directed through the steam jet air ejector and steam packing exhauster to the full-flow demineralizer system. The

condensate is heated in three stages of feedwater heating before entering the direct-contact deaerating heater. The turbine-driven boiler feed pumps take suction from the deaerating heater storage tank and pressurize the water to a head of 7450 feet. The feedwater is heated to its final temperature of 484 F by passing through two stages of high-pressure heaters. High-pressure piping transports the feedwater to the three steam generators.

The main condensers are cooled by river water entering at 57 F and flowing at a rate of 120,000 gal/min to each condenser. Before reaching the condenser, the river water is subjected to coarse and fine screening, chlorination, and ice melting if required.

11.1.3. System Design Requirement

11.1.3.1. General

The design requirements are identical to those stated in Volume 1, section 2.1.4, with the following additions.

11.1.3.2. Performance Requirements

The following requirements established during the steam cycle selection trade-off study (Vol 4) will hold for this system:

Main steam pressure at turbine throttle, psig	2400
Main steam temperature, F	950
Main steam flow at 100% load, lb/hr	7,350,000
Reheat turbine, reheat temperature, F	950
Final feedwater temperature, F	484

11.1.3.3. Environment Requirements

All of the components will be designed for the seismological conditions as stated in Volume 1, section 2.1.4.

The feedwater, steam and condensate system will operate in an environment established by the feedwater chemistry requirements as stated in Volume 3, section 10, Appendix A. Under normal operating conditions, there will be no radioactive contaminants in the steam and power conversion system. This system could become contaminated only through a combined IHX and steam generator tube leak

in conjunction with a low-pressure feedwater system. In the event of IHX leakage, monitoring of the steam generator's shell side sample points will detect any contamination. Therefore, no radiation shielding will be required for the components of this system.

11.1.3.4. Reliability/Availability Requirements

All of the components will be designed for the 30-year lifetime of the plant. The reliability and availability of the system will be high, equaling that established for similar conventional plants.

11.1.3.5. Maintenance Requirements

Standard preventive maintenance will be scheduled for all items consistent with that normally scheduled for such plants. Frequent functional operational checks will be made on vital valves, control systems, and protective equipment since such checks are essential to the successful operation of modern power stations.

11.1.3.6. Operating Requirements

The turbine will be safe for continuous operation at 105% of rated pressure with the control valves in the wide-open position and all extractions in full operation. The generator will be capable of withstanding, without damage, a 10-second, 3-phase short circuit at the terminals when operating at the rated kVA and rated power factor, and at a 5% overvoltage with fixed excitation. It will also be capable of withstanding, without damage, any other short circuit at its terminals lasting for 10 seconds or less, provided that the machine phase currents under fault conditions are such that the negative phase sequence current (I_2), expressed in terms of the unit stator current at rated kVA and the duration of the fault in seconds (T), is limited to values giving an integrated product ($I_2^2 T$) equal to, or less than, ten.

The maximum phase current will be limited by suitable reactance or resistance to a value that does not exceed the maximum phase current obtained from the 3-phase fault. The generator will operate successfully at rated kVA, frequency, power factor, and gas pressure at any voltage not more than 5% above or below the rated

voltage, but not necessarily in accordance with the standards of performance established for operation at rated voltage.

For a loss of full load, all of the energy existent or produced in the reactor coolant system will be dissipated through steam relief valves to the condenser and the atmosphere. The unit will be designed to maintain the station auxiliary load without a reactor trip when a loss of full load occurs. The steam bypass to the condenser and the atmospheric relief valves will be used as necessary to achieve such a load reduction.

Following a turbine trip, the protection system will rapidly reduce the reactor power output to 20% load. The safety valves will relieve excess steam until the output is low enough for the steam bypass to the condenser to handle all of the steam generated.

If a single feedwater pump fails, there will be an automatic setback of the power demand. The one feedwater pump remaining in service will carry approximately 60% of the full-load feedwater flow. If both feedwater pumps fail, the turbine will be tripped, and the emergency feedwater pump will be started. If the conditions of the reactor coolant system reach trip limits, the reactor will trip.

If a condensate pump fails, the spare condensate pump will be automatically started.

Hydrazine will be added to the feedwater for oxygen control, and ammonia will be used to maintain the pH at the optimum value for the system's materials of construction.

11.1.3.7. Safety Requirements

If there is a sudden loss of load and a-c auxiliary power, the necessary sections of the system will remain in operation to enable the safe rundown of the turbine.

Trips, automatic control actions, and alarms will be initiated by deviations of system variables within the steam and power conversion system. When automatic corrective action in the steam and power conversion system occurs, appropriate corrective action will be taken to protect the reactor coolant system.

11.1.3.8. Design Requirements

The following parameters will be used in the design of this system:

Average river water temperature, F	57
Turbine exhaust pressure, in. Hg abs	1.5
Condenser tube velocity, fps	7.0
Condensate hot well temperature, F	91.7
Low-pressure heater tube velocity, fps	5.5
High-pressure heater tube velocity, fps	8.0
Heater tube design pressure, % above pump shutoff head	5
Heater shell design pressure, % above normal operating pressure	20
Deaerator storage tank	
Capacity, gal	75,000
Emergency supply, minutes	4.5 at 100% flow
Condensate piping velocity, fps	10
Feedwater piping velocity, fps	20

The turbine-generator equipment will conform to applicable USAS, ASME, and IEEE standards. The design, materials, and details of construction of the feedwater heaters will be in accordance with the ASME Code, Section VIII, Unfired Pressure Vessels, and the Standards of Feedwater Heater Manufacturers Association, Inc.

The system will be capable of accepting a 5% step change and a 2% per minute ramp/change in power over the operating range of 20 to 100% of rated power.

Impurities in the steam and power conversion system will be controlled to maintain the specified water purity for the steam generator. The condensate will be treated by a separate full-flow demineralizer.

11.1.3.9. Interface Requirements

This system is in direct contact with the steam generating system, the land, structures, and conventional systems and with the electrical grid of the utility; it is also indirectly coupled to the reactor system.

The system must operate economically and must meet the requirements of the utility grid within the generating capability of the reactor primary and intermediate coolant system, ensuring satisfactory, safe operation of the plant.

Pressure relief is required at the system design pressure, and the first safety valve bank in the steam generating system will be set to relieve at this pressure. Additional safety valve banks in the steam generating system will be set at pressures as allowed by the ASME Code. The pressure relief capacity will be set so that the energy generated at the reactor high-power level trip setting can be dissipated through this system.

11.2. Detailed System Description

11.2.1. General

This system description is subdivided into three parts: the turbine-generator unit, condensate and feedwater, and heat rejection system. However, because of the conventional nature of the energy conversion system and the reduced emphasis on this system in B&W's contractual work scope, major attention was not devoted to this part of the plant.

All of the components in the energy conversion system are present state-of-the-art equipment, and their performance is based on manufacturer's data and operating experience at existing steam generation facilities. The components in the system are summarized in Table 11-1.

Although the need for chlorinating the circulating water to maintain cleanliness of the condenser is subject to a specific analysis of the site conditions, liquid-type chlorinators for intermittent chlorine feed have been installed in the crib house.

11.2.2. Turbine Generator Unit

The turbine is a multicylinder, tandem, compound six-flow machine employing reheat (for selection evaluation, see Appendix 11-A). It consists of a single-flow, high-pressure turbine operating at pressures from 2400 psig to 600 psia, double-flow intermediate reheat turbine operating at pressures from 540 to 149 psia, and three identical double-flow, low-pressure turbines operating at pressures from 142 psia to 1.5 in. Hg absolute.

The five turbines, the generator, and the exciter operate on a single shaft at a speed of 3600 rpm. Automatic extraction steam is provided at pressures of 600.1, 379.4, 149.3, 64.4, 21.9, and 5.37 psia to heat the feedwater and operate the boiler feed pump turbine. The turbine has an electrohydraulic control system, complete oil pumping system, a gland-sealing system, a motor-operated turning gear, and various protective devices, gauges, and instruments.

The turbine is provided with a 25% flow steam bypass system from the turbine inlet valves to the main condensers. This bypass contains pressure-reducing valves to reduce the steam pressure entering the condenser to 250 psia and desuperheaters to reduce the temperature of the steam to 400 F. The discharge of the boiler feed pumps provides cooling water for the desuperheaters. The bypass valves are set to bypass steam at pressures above 105% of full power flow. These set points are 2520 psig for the main steam and 565 psig for the hot re-heat steam.

The turbine is started, loaded, and operated from the central control room; records are made on the data logging equipment in the control room, and alarms warn of abnormal conditions. In addition, indicators are mounted on instrument panels near the machine.

Significant malfunctions or faults causing trips, automatic actions, or alarms in the steam and power conversion system are as follows:

Turbine Trips

1. Generator/electrical faults.
2. Loss of condenser vacuum.
3. Thrust bearing wear.
4. Loss of generator coolant capability.
5. Loss of both feedwater pumps.
6. Turbine overspeed.
7. Reactor trip.
8. Loss of lubrication capacity.

Principal Alarms

1. Low pressure at feedwater pump suction.
2. Low vacuum in condenser.

3. Low water level in condenser hotwell.
4. High water level in condenser hotwell.
5. High pressure to turbine throttle valves.
6. Low pressure to turbine throttle valves.
7. Low feedwater temperature.
8. Feedwater conductivity (or resistivity).
9. Feedwater O₂ content.
10. Low level in deaerator tank (diminished safety margin for cooling).

The generator is a totally enclosed, self-ventilated, non-salient pole-type, alternating current unit with a conventional shaft-driven, direct-current exciter. It is a 3600 rpm machine rated at 1,150,000 kVA at a 0.9 power factor. Power is generated at approximately 25,000 volts, 3 phase, 60 cycles, and the short circuit ratio will be not less than 0.58 at the rated kVA. The rotor winding is cooled by hydrogen, and the stator winding is cooled by water. The generator losses are 16,864 kW.

The rotating rectifier exciter, which is totally enclosed and self-ventilated, is located on the generator shaft. This exciter eliminates the generator field breaker, exciter field rheostat, and main lead bus wiring between the exciter and the a-c generator. No instruments are required for indicating or recording the field temperature, voltage, and current of the a-c generator.

The rotating rectifier exciter includes the following shaft mounted components:

1. Permanent-magnet pilot exciter — This high-frequency permanent-magnet generator provides power to the voltage regulator which controls the stationary field of the a-c exciter.

2. a-c exciter — The output from the rotating armature of the a-c exciter is fed along the shaft to silicon diodes mounted in a wheel on the shaft. The exciter's enclosure is pressurized to prevent oil vapor leakage from the bearings.

3. Diode wheel — The output of the a-c exciter is rectified by silicon diodes mounted on the shaft, and the resultant direct current is carried along the shaft to the rotating field of the a-c generator.

There are no commutators, brushes, or collector rings. Continuous operation of this system is possible with 20% of the diodes per phase out of service.

4. Fuses — The fuses that protect the system against diode failure are located on the fuse wheel.

11.2.3. Condensate and Feedwater System

11.2.3.1. General

The condensate and feedwater system provides the means for condensing, transporting, deaerating, heating, polishing, and pressurizing the condensate water for admission into the steam generator system. The flow diagram showing the interconnecting pipe sizes and schedule for plant operation between 20-100% is shown on Figure 11-2.

The condenser hot wells discharge 92 F condensate to three half-size condensate pumps rated at 5000 gpm each, with a discharge head of 560 feet. Each condensate pump is driven by an 800 hp electric motor drive. Condensate flows through the steam jet air ejectors and the gland steam condenser to the full-flow demineralizer system.

The condensate polishing demineralizer system comprises six 2000 gpm regenerative demineralizers. Additional capacity is provided to ensure continuous operation while any one demineralizer vessel or demineralizer auxiliary is being serviced. The resins from each condensate demineralizer are automatically transferred to the resin regeneration tanks when required.

After passing through the demineralizer system, the condensate flows to the initial condensate heater in the neck of each condenser. This stage of condensate heating consists of three one-third capacity, horizontal, extraction steam-condensate heat exchangers operating at a pressure of 5.1 psia. The terminal temperature of this heating stage is 158 F. The remaining two stages of condensate heating each consist of two 50% capacity, vertical, extraction steam-condensate heat exchangers. They are operated at 20.8 and 61.2 psia, respectively. The condensate temperature after the third stage of condensate heating is 289 F. The heater drains from each heater are cascaded to the first heating stage and combined with the condensate flow.

Figure 11-2. 1000-MWe LMFBR -
Condensate and Feedwater
Piping System, Flow
Diagram

Note: The steam generators and reheaters are shown for clarity but are not part of this system.

Following the third stage of condensate heating, the condensate is routed to a single full-capacity, horizontal, direct contact, deaerating heat exchanger with a 75,000 gallon storage tank. This deaerating heater is operated at a pressure of 141.8 psia and a final temperature of 354 F.

The deaerating heater storage tank provides the net suction head required by the two 50% capacity boiler feed pumps. These pumps are rated at 8300 gpm with a discharge head of 7450 feet. They are driven by 17,100 hp turbine drives using 141.8 psia extraction steam which is condensed at 2.0 inches of Hg in auxiliary condensers. The high-pressure feedwater is discharged through two stages of feedwater heating, each consisting of two 50% capacity, vertical, extraction steam-feedwater heat exchangers. These high-pressure feedwater heaters operate at 360.4 and 571.5 psia, respectively. The final feedwater, at a temperature of 484 F, is routed to the three steam generators to complete the cycle. The following sections give additional details on the individual components in the system.

11.2.3.2. Condensers

There are three identical condensers, one for each low-pressure turbine. Each condenser is a double-pass, split water box with 16,500 Admiralty metal tubes having a surface area of 172,788 ft²; these 18-BWG tubes are 40 feet long and 1 inch OD. A 76-inch pipe supplies 120,000 gal/min of circulating water to each unit. The condensers have expansion joints and are capable of reversing the direction of the circulating water flow to maintain the cleanliness of the tubes. The condensers are designed to receive 25% of rated flow bypass damp steam from the main steam generators for a limited duration.

11.2.3.3. Condensate Pumps

Each of the three identical condensate pumps can deliver one half of the condensate flow. Each pump is a vertical "canned" pit unit with a rate of 5000 gal/min and a discharge head of 560 feet. These pumps are driven by vertical, 800-hp electric motor drives.

11.2.3.4. Air Ejectors

Each of the three identical steam-jet air ejectors is capable of removing the noncondensable gases from one condenser. With the cooling water at 57 F and the air ejectors in operation, a pressure of 1.5% Hg absolute is maintained in the condensers. Each air ejector is a two-stage steam-jet-air unit with a combined inter and aftercooler of the surface type. A total of 330 lb/hr of main steam is condensed by 3350 gal/min of condensate in each air ejector after cooling of the condensate.

11.2.3.5. Gland Steam Condenser

A single surface-type gland steam condenser condenses the turbine gland seal steam. This condenser is designed for a pressure of 400 psig and is cooled by the full flow of condensate water. The tubes in this condenser are 90-10 copper-nickel.

11.2.3.6. Water Treatment

The water-treatment equipment includes chemical additive storage and mixing tanks, chemical injection pumps, makeup water supply storage, makeup water deionization equipment, and condensate polishing demineralizers. The condensate polishing demineralizer system contains six 2000 gal/min regenerative demineralizers. Five of these demineralizers polish 100% flow of condensate. The sixth unit ensures continuous operation while any one demineralizer vessel or demineralizer auxiliary is being serviced. The resins from each condensate demineralizer are automatically transferred to the resin regenerant tank when required.

11.2.3.7. Low-Pressure Heaters

Seven low-pressure heaters are arranged in three stages: stage A (three heaters), stage B (two heaters), and stage C (two heaters). Heater stage A consists of three horizontal shell-tube heat exchangers located in the condenser neck. These identical units operate in parallel, each heating 3500 gal/min of condensate. They operate at 5.10 psia with 5.37-psia saturated extraction steam having an average enthalpy of 1090 Btu/lb. The shell's design pressure is 50 psia,

and the tube's design pressure is 400 psia. The heater's terminal water temperature is 158.1 F.

These heaters collect the cascading heater drainage from heater stages B and C and discharge their contents to the suction of the heater drain pumps, which inject the heater drains forward into the condensate flow. Each of these two-pass heaters contains 560 U-tubes (18 BWG) with an effective straight length of 86 feet and an OD of 3/4 inch. The surface area of each heater is 9400 ft².

Heater stage B comprises two vertical shell-tube heat exchangers operating in parallel; each identical unit heats 6000 gal/min of condensate. They operate at 20.78 psia with extraction steam of 21.90 psia and 275 F, having an average enthalpy of 1177 Btu/lb. The shell's design pressure is 50 psia, and the tube's design pressure is 400 psia. The heater's terminal water temperature is 224 F. These heaters collect drainage from heater stage C and discharge the contents to heater stage A. Each of these two-pass heaters contains 1050 U-tubes (18 BWG) with an effective straight length of 84 feet and an OD of 3/4 inch. The surface area of each heater is 17,200 ft².

Heater stage C consists of two vertical shell-tube heat exchangers operating in parallel; each identical unit heats 6000 gal/min of condensate. They operate at 61.2 psia with extraction steam at 64.4 psia and 465 F, having an average enthalpy of 1265.5 Btu/lb. The shell's design pressure is 100 psia, and the tube's design pressure is 400 psia. The heater's terminal water temperature is 289 F. These heaters discharge their drainage to heater stage B through an integral drain cooler. Each of these two-pass heaters contains 1050 U-tubes (18 BWG) with an effective straight length of 78 feet and an OD of 3/4 inch. The surface area of each heater is 16,100 ft².

11.2.3.8. Deaerating Heater

The full-capacity, horizontal, direct-contact deaerating heater and storage tank collects the condensate, the high-pressure heater drains, and extraction steam. It operates at 141.8 psia with 149.3 psia and 645 F extraction steam, having an average enthalpy of 1345 Btu/lb. The heater's design pressure is 200 psia, and the discharge temperature is 345 F. The horizontal storage tank has a 75,000 gallon capacity.

11.2.3.9. Boiler Feed Pump

Each of the two identical boiler feed pumps can deliver one-half of the feedwater flow. Each pump is a horizontal multistage, turbine-driven unit rated at 8300 gal/min; each has a net developed head of 8000 feet and a discharge head of 8150 feet. They are driven by direct-coupled turbine drives developing 17,100 hp and are operated on extraction steam at 141.8 psia and 645 F. Each turbine-drive exhaust is condensed in its own auxiliary condenser, which rejects 134,500,000 Btu/hr. These single-pass condensers operate at 2.0 inches of Hg absolute and contain 2220 straight tubes (18 BWG) which are 3/4 inch OD and 15 feet long. The surface area of each condenser is 7000 ft², and 16,250 gal/min of circulating water is supplied to each condenser from the circulating water system.

One electric motor-driven boiler feed pump is provided for shutdown, low load, and startup operations. This pump is sized for 20% feedwater flow and is rated at 3300 gal/min at a discharge head of 7000 feet.

11.2.3.10. High-Pressure Heater

The four high-pressure heaters are arranged in two stages: stage E (two heaters), and stage F (two heaters). Heat stage E consists of two vertical shell-tube heat exchangers operating in parallel, each heating 8300 gal/min of feedwater. These identical units operate at 360.4 psia with extraction steam at 379.4 psia and 865 F having an average enthalpy of 1448 Btu/lb. The design pressure of the shell is 475 psia, and the design pressure of the tube is 4000 psia. The heaters have a desuperheating section, a condensing section, and an integral subcooling section. The heaters' terminal water temperature is 434.4 F, and their drainage is discharged to the deaerator at a subcooled temperature of 371 F. Each of these two-pass heaters has 1350 U-tubes (10 BWG) with an effective straight length of 86 feet and an OD of 3/4 inch. The surface area of each heater is 22,700 ft².

Heater stage F comprises two vertical shell-tube heat exchangers operating in parallel; each identical unit heats 8300 gal/min of feedwater. They operate at 571.5 psia with extraction steam at 600.1 psia and 610 F, having an average enthalpy of 1298.1 Btu/lb.

The shell's design pressure is 725 psia, and the tube's design pressure is 4000 psia. The heaters have a desuperheating section, a condensing section, and an integral subcooling section. The feedwater is heated to its final temperature of 484 F, and the drainage is discharged to heater stage E at a subcooled temperature of 444.4 F. Each of these two-pass heaters has 1350 U-tubes (10 BWG) with an effective straight length of 61 feet and an OD of 3/4 inch. The surface area of each heater is 16,000 ft².

11.2.4. Heat Rejection

The heat rejection system contains pumps, piping, valves, and the like which supply the cooling circulating water to the main turbine condensers to remove the heat of condensation of the power cycle.

TID-7025, AEC Guide to Nuclear Power Plant Evaluation, Section 110, specifies that the river water must vary from an average maximum temperature of 75 F to an average minimum of 40 F, and that the condenser must be designed for a circulating water temperature of 57 F. The resultant arithmetic average is undoubtedly not the mean temperature; thus the temperature of 57 F and the backpressure of 1-1/2 inches of Hg absolute may not be optimum values. However, since the information available for the site is limited and a detailed optimization of the condenser is not within the scope of this report, the design temperature specified by TID-7025 was used in designing the condenser and the circulating water system.

Three one-third-size circulating water pumps, five traveling screens with basket widths of 10 feet each, and equipment for trash removal, chlorination and ice melting are contained in the crib house. Each of the three one-third-size vertical multistage circulating water pumps delivers 120,000 gpm at a 50 foot discharge head. These pumps are driven by 1750 hp induction motor drives. They discharge into a 10-foot-diameter underground circulating water pipe. The discharge of the circulating water from the condenser is collected in a 10-foot-diameter discharge pipe and returned to the river in an open flume.

11.3. Principles of Operation

11.3.1. Normal Operation

The turbine-generator normally operates to produce 1,033,926 kWe at 100% throttle flow. Four control valves admit 7,350,000 lb/hr of 2400 psig-950 F superheated steam to the single-flow high-pressure turbine. After being reheated from 610 to 950 F, 6,837,000 lb/hr of the high-pressure turbine's exhaust is admitted to the two-flow intermediate turbine. Also, 5,717,041 lb/hr of the intermediate turbine's exhaust is admitted to three identical two-flow low-pressure turbines. Each of the six last stages of the low-pressure turbines exhaust 780,944 lb/hr of steam at 7-1/2% moisture and 1.5 inches of Hg absolute. The overall net plant efficiency is 41.2% with a heat rate of 8096.6 Btu/kWh based on the generator's gross output. Extraction steam for operating the feedwater heaters and the boiler feed pump's turbine drive is extracted from the high-pressure turbine's exhaust, the intermediate-pressure turbine's 379.4-psia stage and exhaust; and the low-pressure turbine's 64.4-psia, 21.9-psia, and 5.37-psia stages.

Steam from the final stage of the low-pressure turbine is discharged to the three-shell, two-pass condenser at 1.5 inches of Hg. This condenser has three separate shells with a total heat duty of 4,790,000,000 Btu/hr. A total of 120,000 gal/min of circulating water at 57 F is supplied to each section. The condenser hot wells discharge 92 F condensate to the condensate pumps. The condensate is used to cool the steam-jet-air ejectors and the gland steam condenser. It is demineralized in the water treating system and heated to a final feedwater temperature of 484 F in the six heating stages.

When steam is available to the turbine, the turbine start-up section of the startup program is initiated. This program is very similar to sequence monitoring, but many of the functions are done automatically. The program checks to determine whether the various drain valves are open and will automatically open them if they are closed. The program also checks the status of the turbine interlocks and, if they are incorrectly set, will either reset them to the correct startup position or inform the operator to do so. The computer will then operate in

conjunction with the electrohydraulic control system to start the turbine.

The computer will calculate the acceleration rate and heat soak periods based on the steam temperature available to the turbine and the metal temperature of the turbines. The computer will then initiate outputs to the electrohydraulic control to automatically accelerate the turbine to 3600 rpm. The computer will instruct the operator to synchronize the unit and afterward will perform the initial loading, hold the turbine load until the initial load heat soak is completed, and then load the turbine.

11.3.2. Infrequent Operation

The turbine output is based on a 100% flow of 7,350,000 lb/hr. However, to provide for manufacturing tolerances and variation in flow coefficients, the turbine will be designed with a margin 5% above the flow required to meet the guaranteed output. The turbine will be safe for continuing operations at 105% of rated pressure with the control valves in the wide-open position and all extraction in full operation. The turbine may operate at a flow less than the rated flow. However, the manufacturer does not recommend operation at a flow less than that required to produce the plant auxiliary power of 25,000 kWe.

The pressure at the turbine main steam valves will be controlled so that it does not exceed 105% of rated pressure. During abnormal conditions, the pressure may briefly exceed rated pressure by as much as 30%, but the aggregate duration of such brief swings beyond 105% of rated pressure should not exceed 12 hours per 12-month operating period. The pressure at the high-pressure turbine exhaust will not be greater than 25% above the exhaust pressure that exists when the high-pressure turbine is passing design flow with rated steam and operating conditions. Relief valves will be provided to meet these criteria. The turbine is automatically tripped as the superheated steam pressure drops to a value corresponding to the low initial pressure setting of the turbine, which is approximately 2000 psia.

The steam temperature at each steam lead to the turbine's main steam valves should not average more than rated temperature over any 12 month operating period. If this 12-month average is

maintained, then the temperature should not exceed rated temperature by more than 15 F except during abnormal conditions resulting in temperatures that are not in excess of rated temperature by more than 25 F for operating periods of not more than 400 hours per year, nor in excess of 50 F above rated temperatures for swings of 15-minute duration, or less, aggregating not more than 80 hours per year. The variation of temperature between the three steam leads should not exceed 20 F. It is acceptable to operate with temperature differences not in excess of 75 F for periods aggregating not more than 400 hours per year. The average main steam temperature should not be less than 50 F of the average reheat steam temperature, nor 75 F greater than the average reheat temperature at loads of 50% or higher. The maximum difference between the main steam temperature and the reheat temperature should not exceed 300 F at zero load. The turbine has a low main steam temperature trip, which will function when the main steam temperature drops to 800 F.

11.3.3. Low Load Operation

One electric motor-driven boiler feed pump capable of supplying 20% of the feedwater flow is supplied for operation at low load. A drain pump for heater E is also supplied to boost the high-pressure drains to the deaerator when the operating pressure of heater E is insufficient to overcome the static head. This pump is rated at 450 gal/min with a discharge head of 275 feet and operates at loads of less than 60% of rated output.

11.4. Safety Precautions

No special safety precautions are envisioned for this system; normal precautions for conventional electric generating utilities will be observed.

11.5. Maintenance

Major equipment is served by the turbine room crane, which is described in Volume 2, section 1.2.4. This crane has a main hook capacity of 200 tons and an auxiliary capacity of 25 tons.

The manufacturer suggests that a warranty inspection of the entire turbine-generator be made after 1 year of operation. It is estimated

that such an inspection would require eight men and a total of 1300 man hours. At 4-year intervals thereafter the turbine-generator should be given a 100% inspection, which would require approximately eight men and 2000 man hours. However, 100% inspection could be performed on a yearly basis by inspecting one-fourth of the turbine-generator components per year. It is estimated that this would require eight men and approximately 900 man hours each year. The anticipated forced outage of this unit is 1% of the operating time per year.

APPENDIX 11-A
Turbine-Generator Selection

Four turbine-generator units were evaluated to compare cost factors:

1. A tandem-compound six-flow machine with 30-inch last-stage buckets using a six-heater feedwater cycle.
2. A tandem-compound six-flow machine with 30-inch last-stage buckets using a seven-heater feedwater cycle.
3. A tandem-compound six-flow machine with 33-1/2-inch last-stage buckets using a seven-heater feedwater cycle.
4. A cross-compound four-flow machine with 43-inch last-stage buckets using a six-heater feedwater cycle.

Gross outputs were established for these four units as follows:

<u>Unit</u>	<u>Output, kWe</u>
1	1,033,926
2	1,037,186
3	1,045,883
4	1,054,002

These units were analyzed to compare the value of the additional installed capacity, the fuel savings for a lower heat rate, and the difference in the installed capital cost for the various units. In this evaluation, the ground rules were as follows:

1. Fixed charge rate, %	12
2. Fuel cost, mils/kWh	0.9
3. Plant operating factor, %	80
4. Value of installed capacity over 1,000,000 kW, \$/kW	80
5. Lifetime of plant, yr	30

A breakdown of the economic analysis is tabulated on the following page. The tabulation clearly indicates that a tandem-compound six-flow machine with 30-inch last-stage buckets utilizing a six-heater feedwater cycle is more economical than the other units evaluated.

<u>Turbine unit</u>	<u>TC6F-30 in. 6-htr</u>	<u>TC6F-30 in. 7-htr</u>
Gross output, kWe	1,033,926	1,037,186
Gain in output, kWe	Base	3,260
Value of addition, \$	--	261,000
Net heat rate, Btu/kWh	8096	8071
Gain in heat rate, Btu/kWh	Base	25
Fuel savings, \$	--	182,000
Total cost savings, \$	--	443,000
Capital cost difference, \$	Base	500,000
Net loss, \$	--	57,000

<u>Turbine unit</u>	<u>TC6F-33.5 in. 7-htr</u>	<u>CC4F-43 in. 6 htr</u>
Gross output, kWe	1,045,883	1,054,002
Gain in output, kWe	11,957	20,076
Value of addition, \$	956,000	1,601,000
Net heat rate, Btu/kWh	8020	7954
Gain in heat rate, Btu/kWh	76	142
Fuel savings, \$	555,000	1,040,000
Total cost savings, \$	1,511,000	2,641,000
Capital cost difference, \$	1,812,000	4,587,000
Net loss, \$	301,000	1,946,000

Section 12

INSTRUMENTATION AND
CONTROL SYSTEM

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12. INSTRUMENTATION AND CONTROL SYSTEM

12.1. Introduction

12.1.1. System Functions

The following three primary functions of the instrumentation and control system are performed continuously under normal and abnormal operating conditions:

1. The system determines the plant condition by monitoring selected variables. The core, heat removal systems, steam-electric plant, engineered safety features, buildings, discharge paths, and environs are monitored so that the operation and the safety of the plant are determined at all times.

2. The system provides for maintaining selected variables within prescribed limits in two ways: First, current values of significant parameters are displayed to the operator for his guidance; second, an automatic control system normally maintains the plant in the desired operating state above 20% power.

3. The system ensures automatic prevention or suppression of conditions that could result in exceeding acceptable fuel damage limits or in a public safety incident. If the values of certain designated variables exceed the safe limits, control rods are inserted and engineered safety features are actuated automatically as necessary.

12.1.2. Summary Description

The instrumentation and control system mainly comprises two broad logical systems: the control system (described in section 12.2) and the protection system (described in section 12.3). These systems are served by a central digital computer system (section 12.4) and by many specialized instruments, (section 12.5). The

Figure 12-1. Instrumentation and Control System Organization

computer and some of the instruments also perform routine plant monitoring and data logging beyond the protection and control needs.

Figure 12-1 shows the organization of the instrumentation and control system. Instruments measure all the plant parameters needed for control, protection, and data logging, and transmit electrical outputs to the computer system. The computer performs computations, automatically actuates control and protection devices (such as control rods and valves), and displays and records data. The operator(s) can supplement automatic control or protection and can override automatic control (but not protection). The control instruments and the computer also display to the operator(s) the data necessary for manual operations, such as plant startup, since automatic control is limited to normal maneuvering above 20% power.

Insofar as the computer and instruments serve the control and protection systems, they become parts of those systems. For example, the protection functions in the computer and all instruments that perform measurements for the protection system are subject to the rigid design requirements of the protection system specified in section 12.3.1.3. For the sake of clarity and consistency, however, the computer and the instruments are covered in separate system descriptions.

12.1.2.1. Control System

The plant control system automatically causes the reactor plant to follow the turbine generator, which responds to the megawatt demand of the utility grid system. The core power and coolant flow rates are controlled so that the heat transferred to the steam plant from the intermediate loops matches the steam demand of the turbine generator, and important variables are held at selected values as a function of plant load. The basic control method is known as the "plant following the turbine mode" and is typical of modern fossil-fired and nuclear central-station plants.

Core power and flows are varied to maintain constant steam pressure and temperature to the high-pressure turbine, constant steam temperature to the intermediate-pressure turbine, constant reactor inlet temperature, and a variable reactor outlet temperature. Temperatures and temperature differentials are minimized

in the remainder of the plant, including the core. Control is effected by coordinated feedforward and feedback loops which receive information from several instruments and which adjust the following: turbine throttle valves, reheater-steam temperature control valves, feedwater control valves, feedwater pump turbine throttle valves, feedwater pumps, control rods, reactor coolant pumps, intermediate coolant pumps, and intermediate bypass control valves.

12. 1. 2. 2. Protection System

The protection system automatically acts to maintain core integrity and to preserve public safety whenever necessary. Plant parameters indicative of a direct or indirect threat of fuel damage or of a nuclear incident are continuously monitored and evaluated. Instruments and actuation commands are separate from those used for control. Although protection and control share the same computer system, separate input/output interfaces and a rigid internal priority structure preserve the required separation of these functions. Coincidence and redundancy techniques minimize the possibility of failing to act when necessary, while reducing the possibility of spurious protective action. Although protection system components receive power from non-interruptible vital buses, loss of power in general will result in protective action.

Protective actions consist of core power reduction by control rod insertion and/or the actuation of engineered safety features, as required. Three types of power reduction are provided: scrams (maximum-capability insertion of all control rods), fast setbacks (rapid power reduction by controlled insertion of 18 shim/regulating rods), and rundowns (2%/minute power reduction by controlled insertion of three regulating rods).

12. 1. 2. 3. Computer System

Although predominantly analog equipment would be satisfactory, it is expected that plant control and protection will be accomplished by a digital computer system supplemented by the minimum amount of analog circuits necessary to condition detector outputs for acceptance by the computer. This choice is made primarily

because digital equipment can handle the large quantity of data (especially from the fuel assembly outlet thermocouples) with greater efficiency and reliability, and because it is believed that by 1980, digital control and possibly digital protection will generally predominate in central-station plants.

The digital computer system has two main frames which share control and routine data logging functions and which are operated in parallel to provide the redundancy necessary for protection. Each main frame has input and output devices, memory, and a central processor. The basic operating mode is that of multi-programming, which permits the high-speed capabilities of the digital computer to be used with the relatively simple program structure necessary for on-line use. Any operation in either main frame will be interrupted upon demand for the performance of a higher-priority function and will later be resumed at the point of interruption.

The computer system also tests itself and the other protection and control system components automatically. The capability of frequent automatic testing adds significantly to reliability.

12.1.2.4. Instruments

Although there are approximately 100 different kinds of instruments in the LMFBR plant, Figure 12-1 shows that they all share the common function of measuring the value of a particular plant parameter. It is convenient to describe the instruments in terms of measurement types, rather than in the control-monitoring-protection categories shown in the figure. Accordingly, the instrument system descriptions are divided into three sections: nuclear, core, and process and radiation-monitoring instrumentation. Each of these three sections may include control, monitoring, and/or protection instruments.

The nuclear instruments measure core leakage neutron flux from natural source level to 150% power; they are periodically calibrated against heat balance measurements to serve as a measure of instantaneous core power level for both control and protection. The high power scram from the nuclear instruments is the only protective function that does not pass through the digital computer.

The core instrumentation includes those instruments whose detectors are located in or across the reactor core. Neutron flux, temperature, and flow distributions are monitored for protection and performance analysis.

The process and radiation monitoring instrumentation includes all the other instruments in the plant. Included are the traditional categories of primary plant, intermediate plant, steam plant, failed fuel, and radiation monitoring instruments. Outputs are used for control, monitoring, and protection.

12.1.2.5. Operating Control Stations

Control stations, actuators, and indicators necessary to start up, operate, and shut down the plant are located in the control room. Control functions necessary to maintain safe conditions after an accident are initiated from the control room. Controls for some auxiliary functions may be located at remote stations, which are placed under the direction of the chief operator by the intercommunications system.

The computer system and instrument cabinets are located in the computer room directly beneath the control room. However, the operator input/output interfaces are in the control room. In addition to the information normally displayed in the control room, these interfaces permit the operator to request the display of any other variable in the computer. Nearly all equipment maintenance is performed in the computer room.

The control and computer rooms are shielded against the design basis accident (DBA) and other conditions (such as a sodium-water reaction in a steam generator, or adverse weather) to preserve operator safety and equipment operability. The use of flammable material will be restricted to charts, records, manuals, and some of the electronics, so that a fire resulting in the loss of redundant protection channels is not considered possible.

12.1.3. Design Requirements

12.1.3.1. General Requirements

The design will be in accordance with the standards of the National Electrical Manufacturers Association, the

National Fire Protection Association publication, the Instrument Society of America, the International Electrotechnical Commission, the United States of America Standards Institute, the Institute of Electrical and Electronic Engineers, the American Society for Testing Materials, the U.S. Atomic Energy Commission's Code of Federal Regulations - Title 10, and the "General Design Criteria, Liquid Metal Cooled Fast Breeder Reactor using 1980 Technology" (see Appendix 14-A with section 14, Safety System Description), all of the issues in effect on the date of invitation for bids. In addition, specifications developed under the LMFBR Program Plan will be utilized as much as is practical.

12.1.3.2. Interface Requirements

In general, the instrumentation and control system interface requirements consist of detector access to sensed variables and control and protection system access to actuating devices. These requirements are obvious from inspection of the inputs and outputs in the individual system descriptions to follow, and they involve virtually every system in the plant. It is necessary that the land, structures, and conventional systems must provide housing as needed to satisfy the general design criteria. Intimately close interfaces exist with the reactivity control and safety systems. The following interface requirements are less obvious:

1. The installation of the nuclear instrument system detectors requires interfaces with the reactor vessel, reactor shielding, and possibly the reactor support systems.
2. The installation of the in-core neutron detectors requires interfaces with the design of the fuel assembly in the core system and with the reactor vessel, reactor auxiliaries, and possibly the reactor support systems.
3. The installation of several primary sodium detectors employing remote transducers involves non-obvious interfaces with the reactor vessel, reactor auxiliaries, and possibly the reactor support systems.

12. 1. 3. 3. Detailed Requirements

Detailed requirements for the instrumentation and control system are covered in the introductory section of each individual system description to follow.

12. 2. Plant Control System

12. 2. 1. Introduction

12. 2. 1. 1. System Function

The plant control system provides the necessary instrumentation and control equipment for the safe and reliable operation (both manual and automatic) of the nuclear steam supply system or plant. Numerous plant variables are monitored for operational status information and control purposes.

The controlled variables are maintained constant or on variable programmed functions, depending on the particular parameter. These control actions are performed during certain phases of plant operation from startup to shutdown. The variable monitors are in operation during all plant operations.

12. 2. 1. 2. Summary Description

The nuclear power generation plant is designed to be incorporated into an electrical utility's grid system. The plant must be able to follow the megawatt demand of the electrical grid system. The nuclear steam plant consists of three basic systems in the heat transport train from the nuclear core to the turbine generator. These are the reactor coolant system, intermediate coolant system, and the feedwater/steam systems; they are shown schematically on Figure 12-2a. The turbine generator megawatt demand power level must be balanced through this entire thermal train in each of these systems; it must be transported to the turbine generator, through the steam generators-reheaters, through the IHXs, and removed from the nuclear core. At steady-state power levels, these heat rates are equivalent except for the net heat losses.

The mode of control is the plant (reactor) following the turbine generator megawatt demand. This control concept is based on the modern fossil-fired and nuclear utility plants. The plant

systems and components follow the demands of the turbine generator. The turbine generator responds to the demands of the utility grid, operating in the turbine-following mode of control. This arrangement combines features of both the "plant following the turbine generator" mode and the "turbine generator following the utility grid" mode. The system is an integrated control arrangement.

Basically, these control systems must maintain constant steam pressure and temperature to the high-pressure turbine and constant reheat steam temperature to the intermediate-pressure turbine. The integrated control system provides the coordinated feedforward and feedback control loops to enable the plant to follow the turbine generator megawatt demands. The overall control schematic is shown in Figure 12-2b. This integrated system controls the following major dependent variables: reactor power level, turbine throttle pressure, steam generator superheater outlet steam temperature, reheater steam temperature, feedwater flow, steam generator intermediate outlet temperature, reactor coolant inlet temperature, and reactor outlet temperature. These variables are regulated by adjusting the turbine throttle valve positions, reheater steam temperature control valve positions, feedwater control valve positions, feedwater pump speeds, control rod positions, reactor coolant pump speeds, intermediate coolant pump speeds, and the intermediate bypass control valve positions.

The normal power maneuvering range for the plant is between 20 and 100% of the turbine generator load. In this power range the plant is capable of both manual and automatic control and responds to the megawatt demands of an automatic dispatch system. The plant is limited to 2%/minute load change and 5% step changes between 20 and 100% load.

The main control room(s) is the center of plant operations. The normal mode of operation is automatic control, and both manual and automatic operations may be directed from the main control center. The main control console and auxiliary panels contain indicators, recorders, annunciators, a data logging printer, and selector stations. A digital computer is included for data processing, data display, plant control, and plant protection.

Figure 12-2a. LMFBR Schematic Flow Diagram

Figure 12-2b. Overall Plant Control Schematic Diagram (A31110E)

12.2.1.3. System Design Requirements

The plant control system shall be designed in accordance with the general design requirements stated in 12.1.3. Specific requirements for performance, operation, safety, and interfaces are listed below:

1. Performance Requirements — The nuclear steam system must respond to the operational requirements of the electrical grid system.

2. Operation Requirements — The LMFBR plant must be capable of operating in the plant (reactor) following the turbine generator mode of control. This overall plant control system integrates the various control system loops in the plant so that the total plant can be automatically controlled as a unit from one main control center. The plant operation must be consistent with utility requirements and standard utility practices.

The control system is designed for plant startup, normal steady-state plant maneuvering (transient), and shutdown operations. In addition, these systems are arranged to maintain acceptable operating conditions for special, infrequent, and emergency operations, including (but not limited to) heatup, cooldown, loss of system heat removal capability, and turbine generator trip.

The heat load during partial- and full-load operations must be balanced between the parallel components and loops in the major heat transport systems. In addition, the steam temperatures from the three parallel steam generators and parallel reheaters must be maintained relatively constant.

During and after load transients, certain system variables are regulated to protect the reactor core and the other plant components. Basically, the system must maintain constant steam pressure and temperature to the high-pressure turbine and constant steam reheat temperature to the intermediate-pressure turbine. These steam conditions are 2400 psig/950 F/950 F. The following list includes the basic control requirements:

- a. Balance reactor power level with the turbine generator megawatt demand.
- b. Maintain constant turbine throttle steam pressure and temperature to the high-pressure turbine at 2400 psig and 950 F.
- c. Maintain constant reheat steam temperature to the intermediate-pressure turbine at 950 F.
- d. Maintain reactor outlet temperature on the program (function of plant load).
- e. Maintain constant reactor inlet temperature at 800 F.
- f. Maintain the IHX intermediate outlet temperature on the program (function of plant load).
- g. Maintain a constant difference between the steam generator intermediate sodium outlet temperature and the feedwater temperature.

These variables and other resulting system temperatures and flows are given in section 12.2.3. Basically, these operating conditions (programs) enable the nuclear plant to regulate the proper power level and thermal head for maintaining the proper steam conditions to the turbines in the normal power maneuvering range—20 to 100% of full load. In addition, the variable flows and temperatures in the reactor and intermediate coolant systems minimize temperatures and temperature differentials in the system components and the core.

The plant is limited to load changes of 2%/minute and to step changes of 5% between 20 and 100% of plant load.

3. Safety and Interface Requirements —

By controlling plant variables on designated programs and setpoints, the control system tends to avoid situations that require protective action. The control system operation is compatible with the protection system.

12.2.2. Detailed Description of the System

12.2.2.1. Integrated Control System

The integrated control system consists of numerous control loops which enable the plant to respond to the demands of an electrical utility's grid system. The control system provides for the safe and reliable operation of the plant in all operating conditions. These operating conditions include startup, normal steady-state power,

normal transient, shutdown, abnormal, special or infrequent, and emergency plant operations.

The nuclear steam supply system consists of the reactor and the three major heat transport systems (see Figure 12-2a). These components and systems make up the heat transport train from the reactor to the turbine generator; they must respond to the demands of the turbine generator. As the generator moves from one power level to another, these plant systems follow and seek another steady-state operating condition at the termination of the load transient. The systems contain parallel components and loops, and the overall controls provide for balancing the heat load in each of the operating loops during normal operations (see Figure 12-2b).

The feedwater-steam systems will operate in a conventional arrangement. The controls provide constant steam pressure to the high-pressure turbine and constant steam temperatures to the high- and intermediate-pressure turbines. The steam conditions are 2400 psig/950 F/950 F.

The intermediate coolant system operates with some variable and some constant temperatures and variable flows at the various plant loads. The controls regulate the IHX outlet hot leg temperature and the steam generator sodium outlet temperature on a program (function of plant load) in the normal power range. The mixed IHX inlet cold leg temperature is maintained constant as a result of the controls mentioned above. These control loops produce a variable intermediate total coolant flow as a function of plant load (see Figure 12-12).

The reactor control and the reactor inlet temperature control regulate the reactor power level, maintain the reactor outlet temperature on a program (function of plant load), and maintain a constant reactor inlet temperature in the normal power range. The temperature program maintains the proper thermal head to maintain constant steam temperatures to the high- and intermediate-pressure turbines. These controls result in a variable non-linear reactor coolant flow as a function of plant load (see Figure 12-12).

The system temperatures and flow rates for partial- and full-load, steady-state operation are regulated. The

control system loops work in parallel with combinations of feedforward and feedback control loops to balance the thermal power of the reactor core and the turbine generator steam demands. The system temperatures and flow rates (functions of plant load) are shown on Figures 12-11, 12-12, and 12-13 for normal full-power and partial-load steady-state operations. Criteria for full- and partial-load operation are given in section 12.2.3.

The load delivered to the turbine generator is approximately equal to that transported through the systems and that generated in the reactor core to maintain thermal equilibrium in the plant. During load transients, each of these systems and the coupling heat exchangers move toward the new power level. The integrated control system provides the means for accomplishing this and for maintaining the balanced plant conditions mentioned above.

The control concept is the "plant (reactor) following the turbine generator megawatt demand" mode. The nuclear steam supply components and systems and their auxiliary supporting systems must be able to follow the demands of the turbine generator. The turbine generator must respond to the demands of the utility grid; the turbine operates in the turbine-following mode of control. This control concept is based on modern fossil-fired utility plants and commercial pressurized-water reactor (PWR) plants with once-through steam generators. It combines the features of the turbine-following mode and the old boiler-following mode of control. Turbine following provides more stable pressure control, whereas the boiler-following mode has the faster load-changing characteristic. The pressure error is supplemented by the turbine governor speed control. The turbine is operated in the turbine-following mode with the pressure error signal and megawatt demand signal fed to the turbine throttle valve controls. A frequency signal supplements this demand signal to correct for the grid system frequency variations. The control arrangement has features of both the plant following the turbine generator and the turbine generator following the electrical grid. (The latter is an inherent requirement of any electrical power generating station.) Thus the system is an integrated control arrangement.

Figure 12-3. Reactor Control, Flow Diagram

The various plant control loops are described in more detail in the following subsections. The instrument measurements that feed into the control loop are described in sections 12.5.1 and 12.5.3.

12.2.2.2. Control Loops

1. Reactor Control — The reactor control system loop balances the reactor power level with the megawatt demand and maintains the reactor outlet temperature on the program. The heat rate generated in the nuclear core is transferred in the heat transport train from the core through the IHXs, through the steam generators and reheaters, and on to the turbine generator, where the electric power is generated. The heat load will be shared equally in each of the respective parallel components and loops in the reactor coolant, intermediate coolant, and the feedwater/steam systems. In addition to the necessity for balancing the nuclear core and the turbine generator heat loads, the reactor outlet temperature must be maintained at an absolute value high enough to provide the proper thermal driving head to maintain the proper steam temperatures to the high- and intermediate-pressure turbines. These steam temperatures are maintained constant over the normal power maneuvering range of 20 to 100% of plant load.

The three selected regulating control rod positions are varied to provide the reactivity adjustment to control the reactor. The reactivity is adjusted to balance the difference between the actual neutron flux level and the reactor demand in accordance with the deadband (see Figure 12-3). The reactor demand signal consists of the combination of the megawatt demand and the two-mode controller action (proportional plus reset actions) on the reactor outlet temperature error. The reactor demand is described from the relations below. The temperature error is

$$\delta T_1 = T_1 - T_{1p} \quad (12-1)$$

Figure 12-4. Reactor Inlet Temperature Control,
Flow Diagram

The combined reactor demand signal is

$$N_d = K_1 MW_d - K_2 \delta T_1 - \frac{K_2 R_2}{60} \int \delta T_1 dt \quad (12-2)$$

The symbols in the equations are defined in section 12.2.4 (see Figure 12-2a). The regulating control rods move into the core, decreasing reactivity, when the temperature increases (when T_1 is above the programmed value). Conversely, the rods move out of the core when the megawatt demand increases. The reactor demand signal is compared with the nuclear instrument flux measurement (calibrated with heat balance data) and the resulting difference is the error which acts through the reactivity control system. The error is given by

$$E_{cr} = N_i - N_d \quad (12-3)$$

This error will operate through a deadband of $\pm 1\%$ power and actuate the reactivity control system to position the regulating rods. The regulating rods move further into the core, decreasing reactivity, when the control rod drive error increases (when N_i is greater than N_d). Conversely, the regulating rods move further out of the core on a decreasing error. The regulating control rods are inserted or withdrawn from their position in the core at a constant speed of 15 inches per minute.

2. Reactor Inlet Temperature Control —

The reactor coolant flow rate is regulated via pump speed regulation to control the reactor inlet temperature at a constant value. Since the flow is related directly with the speed of a centrifugal pump, the flow through the pump is regulated by its speed. The reactor outlet temperature is held on a programmed function, and the reactor power level is maintained by the reactor control loop. Consequently, the speeds of the reactor coolant pumps are varied to maintain a constant reactor inlet temperature (see Figure 12-4). The error is the difference between the measured reactor inlet temperature and a fixed setpoint.

$$E_{p\omega} = T_2^* - T_{2sp} \quad (12-4)$$

Figure 12-5. IHX Intermediate Outlet Temperature Control,

The reactor coolant pump speeds are arranged to decrease when the error signal increases (reactor inlet temperature is above setpoint). The demanded change in pump speed signal is fed to each of the six reactor coolant pump eddy-current couplings; all six pumps change speed together. The speed of each pump should be maintained relatively equal to that of the others. The demanded change in the pump speeds is given by the two-mode controller action on the error.

$$\Delta\omega_{pd} = -K_3 E_{p\omega} - \frac{K_3 R_3}{60} \int E_{p\omega} dt \quad (12-5)$$

The pump speeds change until the error signal is corrected.

3. IHX Intermediate Outlet Temperature

Control — The speed of the intermediate coolant pump in each of the three intermediate coolant loops is regulated to maintain the IHX intermediate outlet temperature in the respective loop on the program. The IHX intermediate outlet temperature is programmed similarly and functions with the reactor outlet temperature to maintain the proper absolute temperature for providing constant steam temperatures to the turbines. The intermediate hot leg temperature for each intermediate loop is controlled by varying the total loop coolant flow by regulating the speed of the loop's coolant pump (see Figure 12-5). The error applied to a typical intermediate coolant pump is the difference between the measured hot leg temperature and the temperature program. It is given by

$$E_{i\omega} = T_3 - T_{3p} \quad (12-6)$$

The intermediate coolant pump speed is arranged to increase when the error signal increases (hot leg temperature is above the program). The heat transport loops in the intermediate coolant system are physically independent from heat source to heat sink; the control loops also operate independently. The demanded change in a typical intermediate coolant pump speed is given by the two-mode controller action applied to the error above (limiting action will prevent excessively fast or slow pump speeds).

Figure 12-6. Steam Generator Intermediate Outlet Temperature Control

$$\Delta\omega_{id} = K_4 E_{i\omega} + \frac{K_4 R_4}{60} \int E_{i\omega} dt \quad (12-7)$$

The pump speed changes until the error signal is corrected.

4. Steam Generator Intermediate Outlet Temperature Control — The intermediate bypass valve (v_1) for an intermediate loop is positioned to maintain a constant temperature difference between the related steam generator intermediate outlet and the feedwater inlet temperatures (see Figure 12-6). Both of these temperatures are measured and compared. In the normal maneuvering power range this temperature differential is conveniently maintained constant in each loop consistent with the other flow and temperature programs in the intermediate coolant system. The feedwater temperature in this operating range results from the conventional feedwater heating train and decreases as load decreases. The intermediate bypass control valve will be positioned to proportion the sodium coolant flows between the steam generator and the bypass line for the loop (see item 5 for reheater sodium flow control). The error applied to the bypass valve is the difference between the steam generator intermediate outlet temperature and the feedwater temperature and a fixed-difference setpoint given below:

$$E_{v1} = T_7 - T_9 - \delta T_7 - \delta_{sp} \quad (12-8)$$

The intermediate bypass valve position is arranged to open further when the error signal increases (T_7 is greater than the difference setpoint from T_9). The demanded change in the bypass valve stroke position results from the two-mode controller action supplied to the error above. The demanded change in valve position is

$$\Delta S_{v1d} = K_5 E_{v1} + \frac{K_5 R_5}{60} \int E_{v1} dt \quad (12-9)$$

The valve position in each loop continues to change until its error signal has been corrected.

Figure 12-7. Reheater Steam Temperature Control

5. Reheater Steam Outlet Temperature

Control — The reheater steam temperature control valve in each of the intermediate loops regulates the intermediate sodium flow through the loop reheater to maintain its reheat steam at a constant temperature. This control regulates the heat transferred for reheating the steam from the high-pressure turbine before it enters the intermediate-pressure turbine (see Figure 12-7). The error applied to the reheater steam temperature control valve (v_2) is the difference between the reheat steam temperature from the associated reheater and a fixed setpoint. The error is

$$E_{v_2} = T_{13} - T_{13sp} \quad (12-10)$$

Like other control loops in the intermediate coolant system, this control will work in each loop independently. The reheat steam temperature control valve is arranged to close more when the error increases (T_{13} is above its setpoint). The demanded change in valve stroke position is given by the two-mode controller action upon the error. The demanded change in the position is

$$\Delta S_{v_2d} = -K_6 E_{v_2} - \frac{K_6 R_6}{60} \int E_{v_2} dt \quad (12-11)$$

The valve continues to move until the error is corrected.

6. Feedwater Flow Control — The feedwater flow to the once-through steam generator is controlled in a similar manner to the feedwater flow control in modern fossil-fired utility boilers and the commercial PWR once-through steam generators—by varying the feedwater regulating control valve (v_3) position and regulating the speed of the feedwater pumps. These two controls are cascaded together to balance the feedwater flow and the steam flow and to regulate a constant steam pressure to the main steam throttle valves.

Each of the variable-speed feedwater pumps is driven by a steam turbine, and the speed of a pump is changed by varying the steam flow to its turbine driver. The driving steam flows

Figure 12-8. Feedwater Flow Control, Flow Diagram

Legend

- H/A Hand/Automatic Selector
- E/P Electric/Pneumatic Converter
- Σ Summing
- Δ Difference
- Valve Positioner Valve Positioner
- < Selecting Lower Signal (Auctioneer)
- $\% + \int dt$ Proportional + Reset Controller
- P Pressure
- F Flow
- ΔP Delta Pressure
- f_{gen} Generator Frequency
- Δ Difference

are taken from one of the intermediate-pressure turbine exhausts and passed to the feedwater pump turbines through the feedwater pump steam throttle valves (v5). Therefore, each pump speed is regulated by varying the position of its throttle valve.

The feedwater flow control is regulated by these two cascaded controls shown in Figure 12-8. The initiating signal for a feedwater flow change is applied directly to the feedwater regulating control valve. This signal (the feedwater flow demand) is compared with the actual feedwater flow. The feedwater flow demand consists of the megawatt demand (see equation 12-17) and the steam pressure error. The steam pressure is maintained constant at all loads in automatic control. The difference between the feedwater flow and the feedwater flow demand is the error applied to the feedwater regulating control valve. These signals and the error are given by

$$w_{9d} = MW_d - \Delta P_8 \quad (12-12)$$

$$E_{v3} = w_9 - w_{9d} \quad (12-13)$$

The pressure error is also applied to the main steam throttle valve controls. The control is arranged so that the feedwater valve position will close more when the error increases (feedwater flow is greater than feedwater flow demand). The demanded change in the feedwater valve position results from the two-mode controller action applied to the error given above. This demanded change in the valve stroke position is given by equation 12-14.

$$\Delta S_{v3d} = -K_7 E_{v3} - \frac{K_7 R_7}{60} \int E_{v3} dt \quad (12-14)$$

The feedwater pump speeds are varied to maintain a constant pressure drop across the feedwater control valves. The pump speeds are directly affected by the steam flow, which is regulated to the turbines by the feedwater pump throttle valves. The pressure drop across the feedwater control valves is measured and the lowest

of three (auctioneered) is compared with a setpoint. The error for each valve is

$$E_{v5} = \Delta P_{v3} - \delta P_{v3sp} \quad (12-15)$$

The control is arranged so that the throttle valve positions close more on an increasing error signal (pressure drop is greater than the setpoint). The demanded change in the steam throttle valve stroke position is given by the two-mode controller action applied to the error.

$$\Delta S_{v5} = -K_8 E_{v5} - \frac{K_8 R_8}{60} \int E_{v5} dt \quad (12-16)$$

The once-through steam generator has a load variation of the mass of water-steam in the secondary side. The mass of water-steam varies directly with the plant load and, subsequently, with the steam generator load. During a load transient, the controls must be able to change the water-steam mass in the steam generator and maintain a constant steam pressure to the throttle valve; these controls are arranged to accomplish this requirement. As the steam generator load decreases, the mass of water-steam mixture also decreases. The lower mass results in a more sensitive pressure characteristic for the steam generator. It has been shown that this sensitivity can be lessened, to a degree, by a variable gain on the pressure error signal. This gain, K_9 (see below), would be a direct function of the plant load or steam flow decreasing as the plant load is decreased.

Another arrangement for better pressure control is the use of derivative action for faster response. The action would act on the rate of change of the throttle valve position. This control feature gives an anticipatory action to the feedwater control.

These two alternate control features could both be incorporated into the feedwater demand. This alternate approach would alter the equation for the feedwater demand (12-12) as

$$w_{9d} = MW_d - K_9 \Delta P_8 + K_{10} \frac{d}{dt} S_{v4} \quad (12-12a)$$

This equation would simply replace equation 12-12 with all other controls unchanged.

7. Turbine Generator Megawatt Demand and Main Steam Throttle Valve Control — In the plant following the turbine generator mode of control, the load dispatch system or a manual demand initiates a turbine-generator load change through its control circuit to the main steam throttle valves. Generally, the demanded megawatt power level is input to this system by the load dispatch system, where this demand is compared with the actual megawatt generation rate and the difference is fed to the main steam throttle valve (v_4). See Figure 12-9.

The demand signal will normally come from the load dispatch system through a hand/automatic station demanding a change in the plant output. The turbine generator frequency error is summed with this demand, forming the megawatt demand.

$$MW_d = MW_{lds(m)} + K_{11}(f_g - f_{sp}) \quad (12-17)$$

The combined signal passes through limits of high plant load, low plant load, and high load rate of change and is compared with the actual megawatt level from the turbine generator. This error is given by

$$E'_{v4} = MW_{e_g} - MW_d \quad (12-18)$$

This difference is fed into the circuit combined with the steam pressure setpoint as a sliding steam pressure setpoint. This altered steam pressure setpoint is compared with the steam pressure. The error is given by

$$E_{v4} = E'_{v4} - K_{12} \Delta P_8 \quad (12-19)$$

Figure 12-9. Turbine Generator MW-Demand and Main Steam Throttle Valve Control

The error above is combined with the turbine generator governor speed control to position the main steam throttle valves via a hydraulic control system supplied with the turbine generator and main throttle valves. The demanded change in the main steam throttle valves is a function of the error E_{v_4} .

8. Steam Dump Valve Control (v_6) — The steam dump system bypasses steam around the turbines in the event of a sudden loss in plant load (see Figure 12-10). The dump valve will serve to relieve pressure from the steam generator in the event of a decrease in the plant load. The dump valves provide for the sudden loss in load of something in excess of the dump valve capacity without opening the main steam safety valves. The dump valve will open to full capacity when the steam generator steam pressure reaches a prescribed setpoint.

12.2.2.3. Main Control Center

Following proven central-station steam power plant philosophy, all central-station switches and indicators necessary for startup, operation, and shutdown of the plant are located in one main control center. The arrangement of the control room is based on the fact that certain systems normally require more attention from the operators than others; such controls are centrally located on an operator's console and are readily accessible to the operator. Automation is employed in the design to reduce to a minimum the number of operating personnel in the control room. The normal mode is the automatic control mode. Automatic and manual control will be directed from the main control center. Control selector stations generally will feature various combinations of "manual," "computer," and "automatic," as well as manual "start/stop" and "increase/decrease" pushbuttons. Color coordination and proper arrangement of selector stations, indicators, annunciators, lights, and pushbuttons help to avoid confusion during all plant operating conditions.

Data-logging typewriters are available to provide appropriate records of all pertinent data in compact, easy-to-read form. A demand data logger is available to print out data at the operator's request. This printer has several codes that enable the operator to examine trends in certain systems and loops without receiving

Figure 12-10. Steam Dump Valve Control,

a complete log of all variables. An annunciator-logging printer will print out annunciator actuations during plant operation. The operator may select important variables for visual examination on a limited number of cathode ray tube displays; all necessary indications are also provided by gages on the control panels. Strip chart recorders of pertinent variables are located in the main control room.

12.2.3. Principles of Operations

12.2.3.1. Plant Operations

1. Partial Load Criteria — In the normal maneuvering power range (20 to 100% of plant load) certain plant variables are regulated to ensure the proper operating conditions throughout the plant. The criteria for partial load operating conditions are given in reference 1. Basic to these operating conditions is the requirement to maintain constant steam pressure and temperature (2400 psig and 950 F) to the high-pressure turbine and constant reheat steam temperature (950 F) to the intermediate-pressure turbine in the normal power range. Throughout the heat transport train, the proper thermal head is maintained in the hot legs to hold the steam temperatures constant. The reactor inlet temperature is controlled to a constant value; likewise a resulting constant IHX intermediate mixed temperature is programmed. The latter two temperature restrictions ensure that the cold-end temperature differentials on the IHXs are maintained constant. This arrangement employs variable flow in both the reactor coolant and the intermediate coolant systems.

The integrated control system employed for the plant ensures that the various plant variables are maintained on their programmed values, as shown in Figures 12-11, 12-12, and 12-13. The sensed and regulated variables are (1) high-pressure turbine throttle pressure, (2) steam generator outlet steam temperature, (3) reheater outlet steam temperature, (4) feedwater flow, (5) neutron power level, (6) IHX intermediate outlet temperature, (7) reactor inlet temperature, and (8) reactor outlet temperature. With the variables controlled as listed above, and with balanced heat loads across the system, the remaining variables in the plant will operate as shown in Figures 12-11, 12-12, and 12-13. Some of these variables are directly controlled,

while others are the result of the controlled variables. In the figures the controlled variables are dashed lines, and the resulting operating conditions are solid curves.

Control is accomplished by adjusting the main steam throttle valve, reheater steam temperature control valves, feedwater control valves, feedwater pump speeds, control rods, primary sodium pump speeds, intermediate sodium pump speeds, and intermediate bypass control valves.

In the heat transport train, the heat rate generated in the reactor core must be balanced (less the net heat losses) to that required by the turbine generator. The heat transferred through the coupling heat exchangers must be equivalent; this heat rate varies directly with the product of the flow and temperature differentials on the hot and cold sides. The arrangement used is for both variable flows and variable temperature differentials.

2. Normal Power Maneuvering Operations — In the normal power-maneuvering range, the plant will normally operate in automatic control and will be capable of responding to the megawatt demands of the load dispatch system or to a manual signal. The plant is limited to 2%/minute load ramps and to 5% step changes in loads between 20 and 100% of plant load.

A central control console and auxiliary panel boards are located in a central control center, where both automatic and manual operations of the plant are directed. The normal plant operating mode is automatic control.

The overall plant has been simulated analytically as reported in reference 1. The control system described in this section has been simulated and connected to the plant model. These studies have been done in an open loop fashion. Typical plant response to the load changes of 2%/minute are illustrated in Figures 12-14 and 12-15.

Figure 12-11. 1000-MWe LMFBR Reference Design
Temperature Vs Plant Load

Figure 12-12. 1000-MWe LMFBR Reference Design Flows
Vs Plant Load (w_1 , w_4 , w_9 , w_{12})

Figure 12-13. 1000-MWe LMFBR Reference Design Flows
Vs Plant Load (w_4 , w_6 , w_{12})

Figure 12-14. Reactor-Primary System Response (85 to 25% Load Ramp — 2%/Minute)

Figure 12-15. Reactor-Primary System Response (25 to 85% Load Ramp — 2%/Minute)

12.2.4. Definition of Symbols

Symbol	Description
E	error Subscripts: cr - control rod drive, % i ω - intermediate pump speed, % p ω - primary pump speed, % v1 - intermediate bypass valve, F v2 - RH steam temperature control valve, F v3 - FW control valve, % v4 - main steam throttle valve, % v5 - FW pump steam throttle valve, psi
K	proportional gain Subscripts: 1 - gain on megawatt demand 2 - proportional gain on reactor outlet temperature error, %/°F 3 - proportional gain on reactor inlet temperature error, %/°F 4 - proportional gain on SG cold end delta temperature error, %/°F 5 - proportional gain on IHX intermediate outlet temperature error, %/°F 6 - proportional gain on RH steam temperature error, %/°F 7 - proportional gain on FW flow error 8 - proportional gain on feedpump steam valve delta pressure error, %/°F 9 - variable gain on steam pressure error, %/psi 10 - derivative gain on throttle valve position, sec 11 - scaling factor, MW/Hz 12 - scaling factor, MW/psi
MW _{lds(m)}	load dispatch system or manual demand, %
MW _d	megawatt demand, %
MWe _g	actual megawatt generation rate, %
n	neutron flux density, neut/cm ² -sec

Symbol	Description
n	neutron flux density, neut/cm ² -sec
N _d	reactor demand, %
N _i	neutron flux measurement, %
R	reset action
	Subscripts:
	2 - reset action on reactor outlet temperature error, repeats/min
	3 - reset action on reactor inlet temperature error, repeats/min
	4 - reset action on SG cold end delta temperature error, repeats/min
	5 - reset action on IHX intermediate outlet temperature error, repeats/min
	6 - reset action on RH steam temperature error, repeats/min
	7 - reset action on FW flow error, repeats/min
	8 - reset action on feedpump steam valve delta pressure error, repeats/min
S _{v4}	throttle valve stroke position, %
T	temperature, F
	Subscripts*:
	1 - bulk reactor outlet
	1p - bulk reactor outlet program
	2 - reactor inlet
	2sp - reactor inlet setpoint
	3 - intermediate IHX outlet
	3p - intermediate IHX outlet program
	7 - intermediate SG outlet
	9 - feedwater
	13 - RH steam outlet
	13sp - RH steam outlet setpoint
w	flow, %
	Subscripts*:
	9 - feedwater
	9d - FW demand

*See Figure 12-2a.

Symbol	Description
δT_1	difference between bulk reactor outlet temperature and its program, F^*
ΔP_{v3}	pressure drop across FW control valve, psi
δP_{v3sp}	pressure drop across FW control valve setpoint, psi
ΔP_8	steam pressure error (difference from setpoint), psi^*
ΔS_{v1d}	demanded change in intermediate bypass control valve stroke, %
ΔS_{v2d}	demanded change in RH steam temperature control valve stroke, %
ΔS_{v3d}	demanded change in FW flow control valve stroke, %
ΔS_{v5}	demanded change in FW pump speed control valve stroke, %
δT_{7-9sp}	difference setpoint between SG intermediate outlet and FW temperatures, F^*
$\Delta \omega_{id}$	demanded change in intermediate pump speed, %
$\Delta \omega_{pd}$	demanded change in primary pump speed, %

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 *See Figure 12-2a.

12.3. Protection System

12.3.1. Introduction

12.3.1.1. System Function

The protection system ensures automatic prevention or suppression of conditions that could result in exceeding the acceptable fuel damage limits or in a nuclear incident. This function is accomplished by monitoring selected variables, continuously comparing their values and certain of their relationships with acceptable limits, and automatically initiating corrective action (usually in the form of demanding control rod insertion) when these limits are reached. The protection system is separate and distinct from manual and automatic control actions and has priority over them at all times.

12.3.1.2. Summary Description

The protection system acts independently of the control system to perform the most important safety functions. The protection system extends from sensing instruments through final actuating devices, such as scram circuit breakers and motor contactors.

Various actions are initiated by the protection system. In the event that a condition is detected indicating a direct immediate threat of fuel damage or of a nuclear incident, the protection system will initiate a scram (maximum-capability insertion of all 25 control rods) or non-power-changing action, such as isolation of the containment atmosphere. Less-immediate or less-severe conditions will result in a fast setback (rapid reduction of power by controlled insertion of 18 shim/regulating rods) or a normal speed rundown (2%/minute power reduction by controlled insertion of three regulating rods).

The designs of the protection and reactivity control systems are closely coordinated to ensure that higher-level protective actions receive permanent priority over lower-level and non-protective actions. For example, the scram command interrupts all power to the control rod primary drive motors, thus overriding all other rod motion commands (which are accomplished by switching the primary motor power).

Figure 12-16. Typical Protection Function

Scrams are positive, non-interruptible, full shutdowns. In contrast, setbacks and rundowns usually reduce power to a level commensurate with the detected operating impairment. For example, if one intermediate pump fails, at least 50% of rated power can be removed through the two remaining intermediate loops, and a full shutdown is not required. However, it is necessary to reduce power rapidly to near 50% to avoid an eventual scram (such as high core outlet temperature). Accordingly, the loss of one intermediate pump will initiate a fast setback to 50% power. Setbacks and rundowns, like scrams, are positive and non-interruptible actions which go to completion, but in some cases completion will be a designated reduced power level instead of a full shutdown.

Figure 12-16 shows the redundancy employed in the protection system by illustrating a typical protection function (high core outlet temperature scram). The sensed parameter is monitored by three identical instruments (i.e., thermocouples). The output of each instrument is scanned by each main frame of the digital computer. Each main frame has stored in its memory pertinent data and programs necessary to determine whether the instrument readings require protective action (voltage-to-temperature calibration charts for each instrument, an 1130 F scram setpoint, and the coincidence requirement that the reactor must scram if the readings of two instruments exceed the setpoint). The main frames, through output buffers and bistables, control redundant relays which in turn control redundant final actuating devices (scram breakers interrupting power to the control rod primary drive motors). The redundant control relays are arranged in 2 of 2 coincidence; that is, both main frames must demand action before it will occur.

The indicated redundancy, together with the safe-failure requirements of section 12.3.1.3, assures that no single component failure will result in loss of protection or in spurious protective action. It further provides for at-power testing of any protection component or interconnection from the detector cable through and including the final actuating devices; only the detectors and their cables cannot be fully tested.

In addition to restricting core power level, the protection system also automatically initiates action of the engineered safety features associated with emergency core cooling and the release of radioactivity from the reactor plant.

The plant instrumentation also monitors many variables not associated with automatic protective actions. In many cases, these variables indicate undesirable and potentially unsafe conditions which can be corrected by operator action. In such cases, an alarm will be initiated automatically. These alarm functions are included in the protection system; however, neither the instrumentation nor the annunciation associated with the alarms is subject to the design requirements of section 12.3.1.3. In all cases the alarms are backed up by automatic protective actions that maintain plant safety.

12.3.1.3. Design Requirements

1. The protection system will meet the design requirements of the instrumentation and control system, section 12.1.3.

2. The protection system will meet the requirements of the "IEEE Criteria for Nuclear Power Plant Protection Systems," except that, where a conflict exists with one of the following specific design criteria, the specific criterion will govern.

3. The protection system will ensure automatic prevention or suppression of conditions that could result in exceeding acceptable fuel damage limits or in a nuclear incident. The specific conditions for which automatic action is required are listed in Table 12-1 (note also that the plant control system will usually act automatically to correct high core power and improper temperatures and flows, and that a number of interlocks are provided to mechanically prevent unsafe operations).

4. The protection system will monitor the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

5. The protection system will be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

6. Redundancy and independence designed into the protection system will be sufficient to assure that no single failure, or multiple failure resulting from a single event, or removal from service of any component or channel will result in the loss of a protection function. The redundancy provided will include, as a minimum, two channels of protection for each protection function. Different principles will be used where necessary to achieve true independence of redundant components.

7. The protection system will be separated from control instrumentation systems to the extent that the failure or removal from service of any control instrumentation component or channel, or of any single component or channel common to control and protection, leaves intact a system that satisfies all protection requirements.

8. The effects of adverse conditions to which redundant channels or components might be exposed in common, under either normal or accident conditions, will not result in loss of the protection function.

9. In the event that all normal power is lost, sufficient alternate power sources will be provided to permit the required functioning of the protection system.

10. Means will be provided for testing the protection system while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

11. The protection system and its individual channels will be designed to fail into a safe state or into a state established as tolerable on a defined basis if such conditions as disconnection, loss of energy, or adverse environments are experienced.

12. Subservient to the fail-safe requirement above, the protection system will employ such techniques as coincidence and redundancy to reasonably minimize the occurrence of spurious protective actions.

13. The protection system will monitor the performance of the reactor and the engineered safety features during and after accidents.

14. The protection system will provide for manual initiation, by the operator, of any type of action listed in Table 12-1, but only scrams will automatically go to completion after manual initiation.

Table 12-1. Protection System Functions

Scrams

1. High neutron power.
2. High rate of increase or decrease of neutron power (assembly voiding).
3. High neutron power/core flow ratio.
4. High bulk temperature, core outlet plenum.
5. High rate of temperature increase, two fuel assembly outlets (voiding).
6. Complete loss of intermediate flow.
7. Complete loss of feedwater flow or heat sink.
8. Sodium-water reaction.
9. Earthquake.
10. High source range count rate (for rod testing, shutdown).

Fast Setbacks

1. Loss of one primary pump (reduce power by one half).
2. Loss of one secondary pump (to 50% power, close IHX valve).
3. Loss of one feedwater pump (to 50% power).
4. Turbine generator trip/loss of electrical load (to 20% power, dump steam).
5. High startup rate in intermediate range (until cleared).

Normal Speed Rundowns

1. High core flux tilt (to 95% power).
2. High rate of temperature increase, one fuel assembly outlet with medium increase in one neighbor (full shutdown).
3. Sodium in interspace between reactor vessel and radial neutron shield (full shutdown).

Other Protective Actions

1. Loss of power to any primary pump (start pony motor).
2. High primary sodium temperature in pot (actuate emergency cooling system).
3. Low primary sodium temperature in pot (energize heater in NaK-air heat exchanger).
4. Very high cover gas pressure, implying DBA (isolate containment; do not close emergency cooling paths).
5. High stack activity (close stack discharge).
6. High radioactivity at containment atmosphere discharge (isolate containment; do not close emergency cooling paths).
7. High NaK activity (close Na/NaK heat exchanger isolation valves).

12.3.2. Detailed Description

The protection system must supplement the inherent safety features of the reactor plant to provide virtually absolute assurance that excessive fuel damage will not occur and that excessive radioactivity will not be released from the reactor site. The system is passive in the sense that it performs no actions under normal conditions other than the display of monitored variables. However, under emergency conditions, the protection system overrides all other means of plant control.

The design philosophy of the protection system is to monitor plant variables indicative of all credible accident conditions and to automatically correct any condition defined as unacceptable. The amount of corrective action taken is matched to the severity of the condition.

The protection system employs the computer system discussed in section 12.4 and a number of the instruments covered in section 12.5.

12.3.2.1. Protective Actions

Table 12-2 lists all protection actions that have been identified as necessary or desirable to satisfy the design criteria; a formal analysis was not performed to define these actions. Further safety and operation analyses and further design work on the protection system would result in changes to the list, but any such changes would probably be minor. It is to be noted that all fast setbacks are backed up by scrams; however, there is little penalty for including setbacks among protective actions. Also shown in the table are various characteristics of the protection actions, including coincidence logic and the total delay time from the instant that the sensed variable exceeds its setpoint until control rod motion begins.

Most of the actions in Table 12-2 are self-explanatory, and the following paragraphs will clarify those that are not.

The high neutron power scram is the only automatic protective action not handled through the digital computer; this is to avoid delaying the scram by a scanning interval at the computer input.

Table 12-2. Protection System Actions

Condition	Power reduction limit	Other action	Sensing instrument(s)	Coincidence	Normal limit	Setpoint	Time delay, sec
<u>Scrams</u>							
1. High N	Full shutdown	--	Power range	2/3	105%	107%	0.15
2. High or low $\Delta N/\Delta t$	Full shutdown	--	Power range	2/3 ratios	2%/min	10%/sec	0.25/0.35
3. High N/F	Full shutdown	--	Power range, core ΔP	2/3	--	Program	0.35/0.45
4. High T outlet	Full shutdown	--	Core outlet temp	2/3	1120 F	1130 F	5
5. High $+ \Delta T/\Delta t$, 2 assy	Full shutdown	--	Fuel assy outlet temp	2/288	1°F/min	50°F/sec	0.3/1.3
6. Complete loss of intermediate flow	Full shutdown	--	Intermediate pump power	3/3 loops	--	High or low pump power	0.25
7. Na-H ₂ O reaction	Full shutdown	--	SG and RH gas pressure	Any SG or RH	20 psia	50 psia	0.25
8. Complete loss of FW flow or heat sink	Full shutdown	--	FW pump tachometer, int. cold leg temp	3/3 (incl. aux. pump), any 1 loop	--, 700 F	--, 750 F	0.25, 5
9. Earthquake	Full shutdown	--	Seismograph	--	--	--	--
10. Manual scram	Full shutdown	--	Operator	--	--	--	--
11. High source range count rate (interlocked)	Full shutdown	--	Source range	1/2	~10 c/sec	10 ⁵ c/sec	~1
<u>Fast Setbacks</u>							
1. Loss of 1 primary pump	One-half power	--	Primary pump power	1/6 pumps	--	High or low pump power	0.25
2. Loss of 1 intermediate pump	50% power	Close IHX primary inlet shut-off rings	Int. pump power	1/3 pumps	--	High or low pump power	0.25
3. Loss of 1 FW pump	50% power	--	FW pump speed	1/2 pumps	--	High or low pump power	0.25
4. TG trip or loss of electrical load	20% power	Open steam dump valve	TG trip	1/1	--	--	0.25
5. High startup range, int. range (interlocked)	Until cleared	--	Int. range	1/2	~12 sec	10 sec	Varies
6. Manual setback	Until cleared	--	Operator	--	--	--	--

Table 12-2. (Cont'd)

Condition	Power reduction limit	Other action	Sensing instrument(s)	Coincidence	Normal limit	Setpoint	Time delay, sec
<u>Rundowns</u>							
1. High core flux tilt	95% power	--	Core neutron	2/3 any region	3%	4%	1.2/2.2
2. High $+\Delta T/\Delta t$, 1 Assy and med $+\Delta T/\Delta t$ 1 Assy	Full shutdown	--	Fuel Assy outlet temp	2/288	1 °F/min	50 + 30 °F/sec	0.3/1.3
3. Na in interspace	Full shutdown	--	Interspace Na	--	--	--	--
4. Manual rundown	Until cleared	--	Operator	--	--	--	--
<u>Other Protective Actions</u>							
1. Loss of power to any primary pump	No automatic power reduction	Start primary pony motor	Primary pump power	1/6 pumps	--	Low pump power only	--
2. High primary Na temp (pot)	No automatic power reduction	Actuate emerg. cooling system	Pot temp monitor	2/3	--	850 F	--
3. Low primary Na temp (pot)	No automatic power reduction	Energize heater in NaK-air HX	Pot temp monitor	2/3	--	300 F	--
4. Very high cover gas pressure (implies DBA)	No automatic power reduction	Isolate containment ^(a)	Cover gas monitor	2/3	--	~100 psia	--
5. High stack activity	No automatic power reduction	Close stack discharge	Stack activity monitor	2/3	--	--	--
6. High radioactivity at containment atmosphere discharge	No automatic power reduction	Isolate containment ^(a)	Containment discharge monitor	2/3	--	--	--
7. High NaK activity	No automatic power reduction	Close Na/NaK HX isolation valves	NaK activity monitor	2/3	--	--	--

^(a) But do not close emergency cooling paths.

The $\Delta N/\Delta t$ scrams limit the spread of existing fuel damage. Most fuel assemblies have a significant positive sodium void coefficient of reactivity that is sufficient to trip the $+\Delta N/\Delta t$ scram. In the event that a significant quantity of fuel is ejected from an assembly, the $-\Delta N/\Delta t$ scram will act. Because of the freedom from noise of the power range instruments and the digital scanner, the scram setpoints can be very tight— $\pm 2\%$ change in 0.2 sec.

The core outlet temperature measurement is obtained from 2 of 3 coincidence on the average of three sets of thermocouples distributed about the core outlet plenum.

The fuel assembly outlet $\Delta T/\Delta t$ actions are also associated with sodium voiding in individual assemblies. The outlet coolant from an assembly is expected to experience a large step temperature increase (approximately 500 F at full power) if the assembly voids. Protective actions are based on coincidence of thermocouples in separate fuel assemblies, so that redundant thermocouples are not necessary ($3 \times 288 = 864$, an excessive number); this approach is justified, since, if an assembly experiences a large temperature increase, then its neighbors will also heat up. The $\Delta T/\Delta t$ scram is, of course, redundant to the $+\Delta N/\Delta t$ scram for most fuel assemblies.

Sodium-water reactions are monitored by measuring the cover gas pressure in each steam generator; a high pressure indicates leakage and requires a scram. A more advanced technique for monitoring the sodium-water reaction will be employed if it becomes available.

Loss of heat sink is monitored by measuring the cold leg temperature of each intermediate loop. A high temperature indicates that sufficient heat was not removed from the associated steam generator. A scram is initiated, even though reduced power operation on the other two intermediate loops is possible, because the associated steam generator may be in danger of experiencing a sodium-water reaction.

The high source range count rate scram is manually activated during shutdown periods to prevent accidental rod withdrawal and to limit power during low-level testing; normally this scram is deactivated.

The fast setback on the loss of one primary pump is expected to act before, and to prevent, a power-to-flow scram. The scram margin and flow coastdown shape provide adequate time for fast power reduction (see the safety system description, section 14).

Turbine generator trip and loss of electrical load cause power reduction only to the steam-dumping capability of the plant, so that the complete plant is not shut down by brief interruptions. The operator then has the choice of maintaining that level or of further reducing power simply by overriding the automatic control or by adjusting the megawatt demand.

The intermediate-range, high startup rate fast setback is automatically disabled when the average power range signal reaches 10% power and is activated below that level.

The detection of sodium in the interspace (annulus) surrounding the pot is regarded as evidence of a sodium leak and requires a full shutdown.

Very high cover gas pressure implies the occurrence of a design basis accident, and the containment is isolated to prevent the spread of radioactivity.

12.3.2.2. Alarms

The protection system also initiates a number of alarms warning the operator of undesirable and/or potentially unsafe conditions monitored by many plant instruments. A partial listing of some of the more important alarms is given in Table 12-3. In general, neither the monitoring instruments nor the annunciation hardware associated with alarms is subject to the rigid protection system design criteria of section 12.3.1.3. However, in some cases, alarms are derived from protection instruments; since alarm functions are processed by the digital computer main frames, the integrity of the protection instruments is not compromised. Plant safety depends on neither the initiation of an alarm nor proper operator action following an alarm.

Table 12-3. Partial List of Alarms

1. Any scram, setback, rundown, or other protective action.
2. Hydrogen in sodium (pot or intermediate loop).
3. High temperature, neutron shield.
4. High pump temperature (primary or secondary).
5. Low temperature, cold trap.
6. Fission products in primary sodium.
7. High radioactivity level (many places: cover gas, discharges, Na, containment, etc.).
8. Core flux tilt.
9. High $\Delta T/\Delta t$, 1 fuel assembly outlet thermocouple.
10. High neutron count rate, fuel storage.
11. High or low cover gas pressure.
12. Smoke in containment.

12.3.2.3. Protection System Design

The protection system described herein depends heavily on the use of a digital computer. A major reason for this choice is the need for 288 fuel assembly outlet thermocouples for sodium voiding protection; this large number of inputs cannot be handled economically or efficiently with analog equipment under the coincidence and reliability requirements of the protection system. This factor, combined with the strong current trend of digital equipment taking over many of the more complex functions previously performed by analog equipment, and the resultant growth of the digital process control industry, indicates that digital core protection will be acceptable by 1980. It is expected that future protection systems will represent hybridizations of analog and digital equipment.

Clearly, an entirely-analog protection system controlling bistables and relays could be designed for this LMFBR application. The hybrid described herein is considered to be superior to all-analog in terms of reliability, testability, and cost.

Figure 12-17 is a block diagram of the protection system. As noted above, the high neutron power scram is not handled by the computer; hence, a scanning interval delay is not added to other time delays in this scram. However, the power range currents are fed into the computer for the following functions: (1) $\Delta N/\Delta t$, (2) power-to-flow, (3) intermediate range startup rate setback interlock, (4) termination of limited setbacks and rundowns, and (5) limitation of rundown power reduction rate.

All other protection instruments (see Table 12-2) feed directly into the computer. The amount of signal-conditioning circuitry interposed between the detectors and the computer varies among instruments; none is used with thermocouples, while a complete set of standard instrumentation is probably necessary for the nuclear instrument source range. In general, the amount of individual instrument circuitry provided is the minimum necessary to develop a signal that can be scanned by the computer.

Each protective function employs redundant channels throughout. Three instruments monitor each sensed variable except in the cases of the fuel assembly outlet temperature and core neutron instruments. Assuming the use of a digital computer, two main frames are employed and each instrument feeds both main frames redundantly. The appropriate calculations, setpoint comparisons, and coincidence logic (usually 2 of 3) for each instrument are redundantly and independently performed in both main frames. Each main frame controls actuating devices, such as scram breakers, which are also redundant.

The block diagram of Figure 12-17 suggests considerable interconnection of protection channels and, since the same computer may also control the reactor, possible interaction of protection and control. In fact, these various channels are rigidly separated in time. Any individual channel is connected to a computer main frame only when the input scanning switch is closed, and only one such switch is closed for that main frame at any given time. This separation is, of course, vital to the protection system, but the excellent capability of a digital computer for self-testing and fault diagnosis is unquestioned even in present-day machines.

Figure 12-17. Protection System, Block Diagram

Figure 12-18 shows the use of the protection relays of Figure 12-17 in controlling scram breakers and setback and rundown relays, which are described in the reactivity control system. Full redundancy is employed in that no failure of any single relay or breaker can prevent action or cause false action. Provision is also made for manual initiation of scram, fast setback, or rundown at the option of the operator.

The scram logic is shown in Figure 12-18a (see the reactivity control system description for a full description of the scram action). To scram, it is necessary to interrupt both power sources for the control rod primary drive motors. This is accomplished by opening one (or both) scram breaker(s) in each of two power lines; i. e., breaker 1 and/or 2 plus breaker 3 and/or 4. Each breaker receives control power from one of two non-interruptible vital buses through normally closed relay contacts and through the normally closed manual scram switch. The relay contacts are controlled by the digital computer in 2 of 2 coincidence (a and b, a' and b', etc.) and by the nuclear instrument power range channels A, B, and C in 2 of 3 coincidence (A, B, C; A', B', C', etc.). If either relay coincidence requirement (2 of 2 on a and b, 2 of 3 on A, B, and C) is satisfied by opening the appropriate relays, or if the manual scram switch contacts are opened, the affected scram breaker will lose control power and will open the motor power lead.

The fast setback logic is shown in Figure 12-18b (see the reactivity control system description for a full description of the fast setback action). It is similar to the scram logic with three exceptions: (1) setback action is accomplished by grounding control rod speed control contactors; (2) setback relays are employed instead of scram breakers; and (3) setback commands are obtained only from the computer or by manual initiation (2 of 2 coincidence). Multiple contacts are used for each of the 18 affected control rods.

The rundown logic is shown in Figure 12-18c (see the reactivity control system description for a full description of the rundown action). It would be identical to the fast setback logic, except that rundown action is applied to the regulating rod group selected for control action; otherwise, control action could offset the rundown by

Figure 12-18a. Scram Logic

Figure 12-18b. Fast Setback Logic (Typical of 18 Shim/Regulating Rods)

Figure 12-18c. Rundown Logic

raising the other regulating rod group. Rundown relays 1-and-2, and 3-and-4 have multiple contacts for each of the three regulating rods in each group.

Protection system outputs also include actuation of engineered safety features, as indicated in Table 12-2, and a full array of diagnostic alarm features. It is expected that the alarm will be communicated by a flashing light accompanied by an audible alarm and printout; operator acknowledgment would silence the audible alarm and change the light to steady-on until the condition is cleared. Provisions will be made for identifying the time sequence of up to perhaps six nearly simultaneous alarm conditions to aid in fault diagnosis.

12.3.3. Principles of Operation

The principles of operation of the protection system are discussed above and in the computer and individual instrument descriptions (sections 12.4 and 12.5).

12.3.4. Safety Precautions

The design requirement (12.3.1.3) and detailed description (12.3.2) sections above stress the redundancy and independence employed in the protection system design. If any single component fails, either it must fail in the direction of requiring safety action, or it must be automatically tested and its normal output automatically replaced by an action-requiring signal. The only exceptions to this requirement in the entire protection system are detectors and (in some cases) their lead wires. Indirect testing of the detectors and leads is permitted by their redundancy; this "testing" would be covered by procedure and performed manually.

Automatic testing of the protection system components is accomplished periodically by the digital computer main frames. Most vital is the testing of the main frames themselves, their input interfaces, and their output interfaces, relays, and actuating devices, since virtually all protection channels pass through them. Because 2 of 2 coincidence is applied to all of these components, each component may be fully tested, to the point of operating the actuating device, while the reactor is operating.

Bistables, relays, and breakers are all selected to provide the safety function when de-energized, so that loss of power does not reduce safety; however, non-interruptible vital bus power from three separate buses is employed to eliminate spurious actions caused by loss of power. (The equipment shown in Figures 12-17 and 12-18 needs only two vital buses; the third is employed so that triplicated instrument channels have separate power sources.)

The removal of equipment from service will initiate the normal loss-of-power function of that equipment, which will be backed up by interlocking instrument and computer modules. Physical isolation techniques and barriers are employed, as is typical of reactor protection systems.

Reliability is of the utmost importance in the protection system, and high-grade system design techniques and hardware are utilized. The use of the digital computer not only provides the extremely high reliability of digital equipment, but it also provides ready capability for testing the analog instrumentation as well. Although no detailed planning of automatic testing operations for this system was performed, it is immediately obvious that a significant improvement on current testing capabilities is readily achievable. Not only does the computer automatically take over the drudgery of routine testing, but it tests much more frequently than is possible or desirable manually. A significant improvement in reliability is expected.

12.3.5. Maintenance

Maintenance is keyed to rapid fault identification and location by automatic testing and rapid replacement of defective components or modules. The defective unit can then be repaired without interfering with reactor operation. This technique again increases reliability, since, in almost every instance of component failure, testing of components redundant to it must be stopped, on penalty of spurious protective action.

12.4. Computer System

12.4.1. Introduction

12.4.1.1. System Functions

The computer system accepts the outputs of all reactor and turbine plant monitoring instruments, conditions them as necessary, and provides the following outputs:

1. Operation of actuating devices for control of the plant.
2. Operation of actuating devices for protection of the plant.
3. Display of data necessary for manual control of the plant.
4. Recording of data necessary for logging plant operations.

The computer system also automatically tests itself, the instruments, and the actuating devices.

12.4.1.2. Summary Description

The computer system consists of two separate main frames, each including input and output devices, memory, and a central processing unit. The input devices monitor the outputs of all plant instruments, convert the data to usable binary form, and place them in memory. In general, analog circuits in the individual instruments are limited to the minimum amount necessary to develop a signal that can be handled by the computer. The memory contains instrument data, control and protection programs, calibrations, set points, and other information in indexed locations. The central processor is composed of two sections: an arithmetic-logical unit which performs the required calculations, and a control section which directs and coordinates the entire main frame. The output devices control actuators for plant control and protection and display and recording devices.

Each protection instrument channel output is redundantly scanned by both main frames for each protection channel. For example, each main frame compares each of the three core outlet temperature measurements independently with the scram setpoint and determines whether the 2 of 3 scram coincidence exists. Each main

frame operates its own protection actuation devices (such as scram breaker control relays), which are arranged in 2 of 2 coincidence, so that both main frames must demand action before it will occur.

Control instrument inputs are divided between the two main frames, as are the control calculation and actuating devices. (Further analysis may indicate advantages in grouping control functions in a single main frame, or even in making them redundant in both main frames). Routine data-logging functions are also divided between the main frames.

It is probable that the computer will also be used to perform on-site calculations, such as the rod motion required for power-flattening. No extensive effort has been made to identify such functions, but the list of possibilities (fuel management, reactivity calculations, rod height predictions, etc.) is lengthy. Detailed effort is required to optimize the value of performing such calculations on-site versus the cost of additional capacity. There is a significant trend, however, toward providing on-site simulators for these functions in large power reactors.

The computer system automatically tests itself, the instruments, and the actuating devices. Test signals are periodically generated and imposed on various points in the system; if the required response is not obtained, the fault will be localized and annunciated. Testing is performed from instrument inputs (detectors excluded) through the operation of protection actuating devices (the redundancy of the latter permits their testing without incurring shutdown).

The central processor of each main frame provides rigid priority and interrupt levels so that protection action is not delayed while the main frame is occupied with other functions. The basic operating mode is that of multi-programming, which permits the high-speed capabilities of the digital computer to be used with the relatively simple program structure and flexibility necessary for on-line protection and control.

The control and protection functions performed by the computer are described in the appropriate system descriptions, 12.2 and 12.3. Appendix 12-A of this system description briefly

comments on the application of a digital computer for reactor control and protection.

12.4.1.3. System Design Requirements

1. The computer system will meet the design requirements of the instrumentation and control system (section 12.1.3).
2. The system will meet the design requirements of the control system (section 12.2.1.3) and of the protection system (section 12.3.1.3) insofar as it performs functions of these systems.
3. A priority schedule will ensure that protection functions receive priority at all times and that control functions receive priority over routine data logging.
4. Safeguards will be established against performance impairments, such as destruction of programs.
5. Flexibility will be provided to permit software alterations or additions.
6. The computer system will automatically test itself, instruments, and output actuating, display, and recording devices and annunciate any identified failures. Where a failure affects a protection or control function, that function will be placed in the fail-safe mode.
7. Data not normally displayed under requirement 1 will be displayed on demand.

12.4.2. Detailed Description

12.4.2.1. General

The digital computer system can be divided into two separate main frames, each of which contains four types of functional units: input devices, output devices, storage, and central processing units.

Input devices read or sense recorded data in a form meaningful to the computer. Input data coming from analog-

type sensors are converted into digital form available to the computer. The digitalized input data are entered into the system memory. Output devices record or write information from the computer on cards, paper tape, magnetic tape, or as printed information on paper; or they provide direct control of actuating devices for valves, control rods, and other components. Eraseable output devices, such as cathode ray tubes, and a diagnostic alarm panel showing the existence, identity, and time sequence of alarms are also provided.

Storage is like an electronic filing cabinet, completely indexed and almost instantaneously accessible to the computer. All data must be placed in storage before they can be processed by the computer. Information is read into storage by an input unit and is available for internal processing. Each location, position, or section of storage is number-designated so that stored data can be readily located by the computer as needed.

The two central processing units are the controlling centers of the entire system. Each is made up of an arithmetic-logical unit and a control section. The arithmetic-logical unit performs such logical operations as addition, subtraction, multiplication, division, shifting, transferring, comparing, and storing. The control section directs and coordinates the entire main frame as a single multi-purpose machine. These functions involve controlling the input/output units and the arithmetic-logical operation of the central processing units and transferring data to and from storage.

Terminal and multiplexing units electrically connect the computer system to the plant. Signals from the plant are transmitted as electrical signals from instruments or transducers and are connected to pairs of terminals in these units. In addition to measured variables, the status (whether open or closed) of any contact or switch, or the presence or absence of a pulse connected to the terminal units can be sensed by the computer.

Analog-to-digital converters translate measurements into digital units for use by the computer. Conversely, computer setpoints or calculations are transmitted through digital-to-analog converters back to the actuating devices that control the plant.

The basic computer operations must be accessible (on a carefully controlled basis) for the complete and intended use of the system. An administrative control system will ensure that the necessary flexibility cannot be used to the detriment of plant safety. Automatic coding techniques are required and are supplied with the computer to aid the programmer in writing the various codes. There has been widespread adoption of compilers, such as Cobol and Fortran, to assist the programmer. These compilers relieve the programmer of using machine languages and take care of internal house-keeping. All of these coding techniques are generally referred to as "software." Built-in flexibility is required to make changes in the codes for the most efficient use of the computer system. A major effort is required in the area of software to realize an efficient and economical computer system in this application.

A clock is provided for each central processor to provide a calculation time base, to sequence computer programs, to record the time for periodic data logging, and to record the time of unusual or infrequent plant occurrences.

A more detailed description of the computer system is given in the following sections, supplemented by standard sources such as reference 2.

12.4.2.2. System Organization

The computer is organized as shown in Figure 12-19. It utilizes two central processors, which are multi-programmed to perform the specified input/output operations. The multi-program operations are divided into these five major functional operations:

1. Data acquisition and characterization.
2. Every scan cycle (automatic control, protection, and testing) calculations.
3. Demand calculations for performance monitoring.
4. Data alarm monitoring and reporting.
5. Numeric output display, printing, and trend recording.

An independent disc-core memory transfer channel, initiated by processor instruction, loads the requested block

Figure 12-19. Computer System Block Diagram

of words into core memory from the disc storage without utilizing central processor time (other than the extremely short period required to initialize the disc-to-core transfer control). Input data and recently computed information are stored in a reserved portion of the core memory. The central processor refers to the core-stored data as part of its execution routines to display data on the numerical indicator lights, print information on the logging typewriters, and provide output trend records. Each of the readout devices has its own independent output control hardware package (peripheral equipment buffer) which provides the buffering and timing required for information outputting. Each central processor needs only to set the data to be outputted into its outputting data register.

When the central processor completes the current program as determined by the time allocator, the memory transfer channel is again initiated. The current program is then replaced in the disc storage, and the next program is loaded into the core memory. The procedure continues until all programs have been worked on and the basic time cycle is complete, at which time the cycle begins again. It may be possible, instead, to copy programs from the disc storage, which would eliminate the need for replacing them after use by the central processor.

The application of multi-programming is designed to implement the speed capabilities of very fast central processors while retaining a relatively simple program structure utilizing functionally independent input and output programs.

12.4.2.3. Input Data Acquisition

Two methods are used to obtain inputs. The first method is the data acquisition function which scans all designated analog inputs and any digital inputs which are used as system data. The second method provides the capability for computer programs to read digital inputs directly without requiring previous memory storage, reading these data only when the inputs are required for immediate decisions and calculations or control programs.

The data acquisition function operates in parallel with, but independently of, the computing and output functions. The inputs, read under control of the data acquisition function, are

stored in the system data area of core memory. This data area is a "reference file" of the latest reading of each input variable. These values are available for reading by the central processor to perform the required computing and output functions independent of one another.

The scan sequence, which can be modified by an external condition (contact closures, system clock, interpretation of input data, etc.), is controlled by lists as described below. The program control of the data-acquisition function operates within the automatic multi-programming method. The computer system accepts inputs from all plant-monitoring instruments. In some cases, such as thermocouples, the transducer input is accepted directly.

Each main frame handles analog input data in the following manner:

1. System memory contains an individual control word (for each analog input) which specifies the required manipulation of the input variable. Individual preconditioning elements between the system terminals and multiplexers convert signal forms to the system standard (zero to perhaps 100 millivolts).
2. Selected analog signals are ranged or suppressed as specified by the control words.
3. The digital value of the measured variable from the analog to digital converter is conditioned (e.g., the square root extracted) by the input characterization routine as required to facilitate future operations. This routine for each data variable is contained in the individual control words. The input characterization number denotes special calibration for individual point numbers as required; this is in addition to a number of standard calibrations.

Digital pulse inputs are scanned and accumulated in the system data memory under control of the data acquisition program at a scan rate sufficient to accumulate each new pulse received. Computing programs can also read digital inputs directly. This feature is useful especially where on/off inputs are required for immediate control decisions. In addition, all digital inputs are buffered to prevent noise from entering the system through input signal lines.

To program the acquisition of data, the following information is specified for each point:

1. Point number - specifies where the datum is stored in the system data memory.
2. Input number - identifies the input.
3. Analog control number - specifies ranging and suppression (prior to digital conversion) to be applied to the input.
4. Digital characterization number - specifies characterization routine to condition the input (before storing in the system data memory).

The data acquisition scan sequence is specified by listing the point numbers in groups according to the interval at which each input is to be scanned. The programmer can select different scan intervals in groups. For example, one group may contain all the inputs to be scanned once per second, a second group contains all the points to be scanned at 10-second intervals, and so on. Other specified groups can be changed from one scan interval to another or scanned only when required, depending on signals which indicate a change in process status, operator information requirements, and the like. To remove any input from the input scanning cycle temporarily, a "flag" is added to the corresponding point number.

The programmer can change point numbers in the scan lists or the input control words while the system is on-line without disturbing operation. Major modifications to the data acquisition program will require temporary storage of the program operation. The user can place the input scan lists and the input control word for each input in the system memory. The real-time operating sequence, selection of input manipulation, etc., are performed by the system equipment in conjunction with a standard list processing routine which interprets the scan lists and control words.

The digital characterization of inputs is accomplished by a set of subroutines within the data acquisition program. Each subroutine characterizes a certain type of input. With appropriate tables of constants in each subroutine, the same subroutine can be used for standard or special calibrations. The input control word specifies which subroutine and set of constants are to be used.

Standard characterization subroutines are provided for common types of inputs. These are written so that they can be used in any application with a minimum of programming effort.

Constants for special calibrations can be readily modified without disturbing system operation. The input characterization function is open-ended in that non-standard subroutines for each application can be included among the standard subroutines. Such subroutines are limited in execution time by the data acquisition operation. Characterization subroutines which cannot be performed rapidly enough to meet the data requisition program time limits are programmed as part of the automatic calculations.

A partial list of typical standard input characterization subroutines is given below:

1. Thermocouples.
2. Resistance bridges.
3. Uncorrected flow from differential pressure.
4. Linear translation (constants per application).

12.4.2.4. Alarm Monitoring

As listed previously, one of the major functions executed by the central processor is alarm monitoring and reporting. The central processor executes an independent program which contains individually assigned high and low alarm limits for certain analog input data and computed results. Some of the alarm limits are modified under program control; that is, they are variable as a function of one of the system input parameters (for example, load). Input data and computed results are compared with the individually assigned high and/or low alarm limits and also with the previous alarm status of the particular point. When a variable changes its alarm condition—that is, it goes off normal or returns to normal—the computer initiates an alarm report. To prevent repeated alarm printouts of data at their assigned limit values, a predetermined deadband is provided. Alarm reporting, then, consists of logging the time of day, point identification, data variable value, and related information. In addition, audible and flashing light indications of alarms are initiated.

12.4.2.5. Computing

The computer system executes two separate, independent sets of programs which divide the computational functions into two categories:

1. Calculations initiated automatically on a timed or cyclic basis.
2. Calculations performed only on demand or under certain specific conditions, such as during turbine startup.

The programs for the computing functions are stored in separate functional areas and executed from core memory. It should be noted, however, that computations required for the digital linearization of input data would normally be contained within the data acquisition function. The resulting calculated value would then be stored in the system memory (e.g., an instantaneous measurement of flow rate). As part of the automatic calculation programs, this variable could be integrated over time intervals to produce the data necessary for logging historical information and/or computing plant performance.

12.4.2.6. Information Outputting

System data readout to the operator is accomplished using the following items:

1. Operator's console with numerical display lights, group alarms, trending CRTs, and meters.
2. Output typewriters.
 - a. Periodic log.
 - b. Alarm log.
 - c. Demand digital trend logs.

System data are converted to engineering units by the output characterization routine before alpha-numerical printout or display. Variables which are not called for by the central processor or the operator display or printout need not be converted to engineering units.

The computers enter data on the three typewriters under program control to generate the automatic (hourly) logs, alarm information, and operator-initiated trend logs of selected measurements and computations as required by the operator who selects the measurements.

12.4.2.7. Actuator Outputting

Digital-to-analog converters provide the means for transmitting the results of the computer operations back to the plant. Should the operator desire remote manual manipulation of a particular control device, he may direct the positioning by selecting it from the console. Local manual operation of final elements is possible through manual stations.

12.4.2.8. Programming

The computer system includes an independent input/output communication channel for the system bulk disc memories. Information entry into disc memory, and display or recording of information stored in disc memory, are made completely independent of the central processor, and therefore, they do not require any central processor time.

The ability to debug and modify system programs while a computer system is on line has proved to be a necessity in industrial computer systems. The computer system has several features which provide access to the system bulk disc memory without affecting operations; these are as follows:

1. The contents of any memory word can be read or printed without using central processor time.
2. The contents of any disc word can be changed (with a key lock) independent of the central processor.
3. Any functions in the automatic multi-programming cycle can be programmed and debugged independent of any other function, which greatly simplifies the use of off-line computations, including programming aids.

All system programs are stored in total in the bulk disc memory and are executed out of the core working memory by the central processor. Programs can be entered into the bulk disc memory using the following devices:

1. Input keyboard unit.
2. Input-type reader unit (e.g., paper, magnetic tapes).
3. Bit buttons on the programmer's panel mounted in the main cabinet assembly.

Administrative controls and in-built priority levels will prevent the use of these inputs in a manner that could interfere with plant safety.

12.5. Instruments

The preceding sections describe the computer system which processes plant performance measurements and the use of these measurements for reactor plant protection and control. Following is a description of the instruments that perform the measurements. The description is divided into these three sections: Section 12.5.1 covers the nuclear instruments, which monitor core leakage neutron flux from shutdown natural source level to 150% power. Section 12.5.2 covers the core instrumentation, which monitors neutron flux, temperature, and flow in and across the reactor core. Section 12.5.3 covers the process and radiation monitoring instrumentation, which performs all other measurements necessary for plant operation.

12.5.1. Nuclear Instrument System

12.5.1.1. System Function

The nuclear instrument system includes three ranges, which together monitor the core leakage neutron flux, and hence total core power level, from the shutdown natural source level to 150% power. The ranges are power, intermediate, and source.

Power Range — The power-range nuclear instrument continuously measures core leakage neutron flux on a linear scale from 0 to 150% power. This measurement is used for the following purposes:

1. Protection against unacceptably high total core power. This function also protects against unacceptable consequences of reactivity insertion transients (startup accident and fuel assembly voiding).
2. Protection against the spread of existing fuel damage by monitoring for sudden power increase or decrease resulting from a loss of coolant or fuel.
3. Protection against an unacceptably high ratio of core power to flow at any power level, in conjunction with the core differential pressure instrumentation.
4. Determination of core power level for use by the reactor plant automatic controller.

5. Determination of core power level for display to the operator.

Intermediate Range — The intermediate-range nuclear instruments continuously measure core leakage neutron flux on a logarithmic scale from 10^{-6} to 10% power and compute the reactor period (time for power level to increase by a factor of $e = 2.71828$) from this measurement. This measurement and computation are used for the following purposes:

1. Protection against an unacceptably high startup rate in the intermediate range (backed up by the power-range high power protection).
2. Providing the operator with relative core power level and startup rate information.

Source Range — The source-range nuclear instrument continuously measures core leakage neutron flux on a logarithmic scale from shutdown to approximately $10^{-5}\%$ power and computes the reactor period from this measurement. The relative core power level and startup rate are displayed to the operator. No source-range automatic protection is provided; backup protection is furnished by the intermediate- and power-range nuclear instruments.

12.5.1.2. Summary Description

Power Range — Six identical power-range neutron detectors are installed at a common location outside the reactor core. Three detectors are included in identical protection channels, one is included in an information and automatic control channel, and two serve as spares. The detectors are replaceable at shutdown; the installed spares ensure only brief channel outage in case of an at-power detector failure. The common location of the power-range detectors derives from separation of core power level and core power shape in the protection system concept.

The power-range signal is used as a fast-response, continuous linear measure of total core power level from 0 to 150% power. To ensure the validity of this use, the gain of the power range amplifiers will be periodically adjusted to correspond with a calorimetric heat balance continuously computed by the digital computer.

The three identical power-range protection channels are arranged in 2 of 3 coincidence logic, so that each channel can be automatically tested at frequent intervals during power-range operation, and so that failure (including detector failure) in a single channel will cause neither loss of protection nor false scram. The detectors (as well as the amplifiers) are tested by comparison with the intermediate range in the 1-to-10% power overlap range during reactor startup.

Three scram signals, each with 2 of 3 coincidence, are derived from the power-range instrument: (1) high core power (directly initiated), which also protects against the consequences of reactivity insertion accidents; (2) high rate of change of power, plus or minus (derived by the computer), which protects against the spread of fuel assembly voiding; and (3) high core power/flow ratio (derived by the computer in conjunction with the core differential pressure instrument). The three power-range protection channels are also used by the computer in terminating power-limited setbacks and rundowns and in adjusting the rundown rate to 2% power per minute.

The information and control channel, although identical to the protection channels, is separate and distinct to avoid compromising the reliability of the protection system. Calibration of this channel is the same as for the protection channels. The on-line reliability of the information and control channel can be improved, if desired, by providing for automatically switching to a spare detector in case of detector failure; or the spare detector can be connected into a complete second channel that can be auctioneered or otherwise made available. This technique will not be used in the protection channels, because the required additional switching would reduce the reliability of the protection system.

Intermediate Range — The intermediate-range detectors are installed with the power-range detectors at a common point outside the reactor core. Installed spare detectors are not planned, since the detectors are replaceable at shutdown and the intermediate range is not employed at power; however, if desired, a spare could be added to reduce startup delays caused by a failed detector.

Two identical intermediate-range channels logarithmically monitor core leakage neutron flux from approximately 10^{-6} to 10% reactor power. An overlap of approximately one decade of core power will exist with the source range at the lower end and the power range at the upper end of this span. The intermediate range channels will compute startup rate $\left(\frac{1}{n} \frac{dn}{dt}\right)$ over the seven-decade range.

The intermediate-range channels are tested and calibrated by automatic testing during shutdown combined with overlap information. Both core power (on a relative scale) and reactor startup rate are displayed for guiding the operator during reactor startup. In addition, either channel can initiate an alarm to request the operator to reduce the startup rate if it exceeds a designated rate or can initiate a controlled setback if it exceeds a designated (faster) rate.

Source Range — Two identical source-range detectors are installed with the power- and intermediate-range detectors at a common point outside the reactor core. As for the intermediate range, an installed spare detector is not planned, since the detectors are replaceable at shutdown, but one could be provided if desired. The detectors will be pulse-type fission counters and will be stationary (not retracted at power).

Two identical source-range channels monitor core leakage neutron flux from the natural source shutdown level to approximately $10^{-5}\%$ power; to ensure this capability, the monitored range will be from approximately 1 to 10^6 counts per second. The channels will integrate the detector pulses to form an analog neutron level signal, which then is differentiated as in the intermediate range to compute startup rate. The source range channels are tested and calibrated by on-scale response to core natural source, automatic testing, and overlap with the intermediate range.

Both core power (on a relative scale) and startup rate are displayed to guide the operator during startup and also to monitor the core during shutdown and fuel handling. Because of the tendency of pulse channels toward noise susceptibility, the reduced reliability of the complex circuitry, and the slow response required at low counting rates, automatic protective signals will not be derived.

from the source range. The intermediate- and power-range instruments will protect against the consequences of accidents initiated in the source range. An alarm for operator guidance will be initiated on high startup rate during reactor startup, and on high count rate during shutdown and fuel handling. Either channel can initiate either alarm. However, for conservatism, a high count rate source-range scram can be activated during shutdown periods. This scram can be initiated by either channel.

12.5.1.3. General Design Requirements

The general design requirements shall be as specified in 12.1.3 for the instrumentation and control system and in 12.3.1.3 for the protection system.

12.5.1.4. Specific Design Requirements

1. Operating range - neutron flux, nv, all energies:
 - a. Power range - 10^{10} to 10^{13} , amplifier gain adjustable to provide for linear 1 to 150% power span within this range.
 - b. Intermediate range - 10^5 to 10^{12} .
 - c. Source range - 1 to 10^6 .
2. Detector environment:
 - a. Temperature - 0 to 900 F.
 - b. Pressure - 10 to 100 psia.
 - c. Neutron flux - 0 to 10^{13} nv, all energies.
 - d. Gamma dose rate - 0 to 5×10^5 R/hr.
 - e. Environmental material - air or sodium.
 - f. Useful life - 5 years minimum (10 years desired).
 - g. Replacement - at shutdown.

12.5.1.5. Detailed Description

Power Range — The power-range instrument is similar to those used in conventional water reactor power

Figure 12-20. Nuclear Instrument Power Range,
Schematic Diagram

ranges; Figure 12-20 is a schematic diagram of the instrument. The detector is located outside the core for several reasons:

1. The increased detector reliability associated with lower temperatures and radiation levels.
2. The calibration stability resulting from negligible burn-out of detector neutron-sensitive material.
3. The relative freedom of fast reactors from the factors that produce significant power tilting and oscillation in thermal reactors.
4. The fact that the core can be refueled without disrupting the detectors or adding temporary detectors for monitoring during refueling.

The power-range detectors may be either conventional, boron-loaded ion chambers or uranium-loaded fission chambers; in either case the detector is a simple d-c chamber with a thermal neutron sensitivity of about 10^{-13} to 10^{-14} amperes/nv thermal. However, the detector must operate at approximately 800 F; this temperature is somewhat higher than the rating of detectors that are now commercially available, and a relatively low-risk development effort is required. Note that while the 800 F temperature would apply continuously in the alternate neutron shield location, it would apply to the preferred location in the concrete biological shield only during a design basis accident for a relatively short time.

The 800 F temperature also requires high-temperature electrical cable for connections to the detector. Since the d-c signal does not require cable with high-quality pulse transmission characteristics, currently available ceramic-insulated, metal-sheathed, coaxial cable would be suitable. The cable should be integral with the detector (rather than joined by a connector) for maximum simplicity and reliability. The integral cable can extend all the way through a suitable low-leakage penetration in the reactor containment, or it can be joined to another cable by a connector in the cover gas.

Core neutron flux monitoring is based on the premise that core power shape and total core power level can be controlled separately. Thus, the in-core neutron instrumentation limits power tilting to acceptable levels, and the nuclear instrumentation need only monitor total core power level. This separation permits placing all of the power range detectors at a common location, with the resulting simplicity of access and installation.

The nuclear instrument detectors require a full-power thermal neutron flux of approximately 10^{10} nv. Shielding calculations indicate that, with suitable local flux shaping, the desired flux can be provided in either of two locations: in the containment concrete underneath the core, or in the first row of the lower radial neutron shield alongside the core. Preliminary calculations indicate that the azimuthal flux shape variation and the sensitivity to control rod motion in the latter location are relatively small opposite any one of the six small core flats, as indicated in Figure 12-20. Both the under-core and beside-core locations are considered feasible, and a final decision between them would depend on detailed installation considerations.

A preliminary assessment of installation feasibility tends to favor the under-core detector location. The sodium containment is not penetrated and each detector can be installed in a dry horizontal pipe with a virtually straight run to an accessible loading area at the outside of the concrete. Generally vertical pipes could provide similar access to the neutron shield location, but a lateral offset is necessary so that the pipe entrances are in the stationary area of the operating floor (not in a rotating plug). This arrangement would partially block the refueling mechanism travel where the pipes extend above the top of the neutron shield but would not block its access to any fuel assembly; access to a portion of the upper radial neutron shield would also be blocked by the pipes as they curve through it. As an alternate means of access to the neutron shield location, the refueling mechanism or an offset handling mechanism could be used to insert the detectors in sodium-wetted holes; special tools would withdraw the free ends of the cables through ports in the stationary portion of the operating floor.

Six power-range detectors are employed: Three are included in 2 of 3 coincidence channels, and one is installed as a spare protection detector, so that if a protection detector fails, the resulting 1 of 2 scram coincidence will exist only until the installed spare can be connected to the third channel. This technique further increases the capability of the 2 of 3 coincidence for minimizing false scrams. The other two installed power-range detectors provide core power information to the reactor plant automatic controller; one detector can be employed strictly as an installed spare, or both can be

connected with the controller selecting one for input. All six detectors are replaceable at shutdown.

Power supplies and amplifiers for all power range channels are identical. The power supply is a conventional, moderately well regulated, solid-state unit rated at 2 mA at approximately 1000 Vdc (the voltage must be sufficient to keep the detector saturated at maximum design current). Since the detector output will be well over one microampere throughout the power range, the amplifier requires little gain, but it must allow for the following range of gain adjustment:

1. Error in predicting full-power flux at detectors.
2. Flux variation from one detector location to another.
3. Sensitivity variation among detectors.
4. Calibration changes (with time).

Although none of these variations will be large, there is little penalty in providing a wide range of adjustment.

The power-range amplifier gains will be periodically adjusted to correspond with the most accurate calorimetric heat balance information available (probably from the steam plant). It is convenient for the digital computer to perform the calculations and determine the proper gain settings with a frequency of once per hour or faster. A limited component development effort may be necessary for a reliable, long-lived (in terms of many adjustment changes) potentiometer for this purpose.

Each protection channel amplifier has two outputs—one connected to a high-power scram bistable and one to the digital computer for power-to-flow comparison, control of setbacks and rundowns, and determination of power rate-of-change—each of these functions has 2 of 3 coincidence. The scram delay time from the introduction of a high flux into the detector to the beginning of control rod scram motion will be 150 milliseconds or less. The control channel amplifier output is displayed on meters and connected to the reactor plant automatic controller.

The protection channels will be tested automatically by superimposing a test signal on the detector current input to the amplifier and noting trip action of the individual channel

scram bistable. Test duration will be much less than test frequency (say, one millisecond and 100 milliseconds) so that the testing will contribute negligibly to the possibility of false scrams. Automatic testing of the control channel will check only the amplifier, and the output will be disconnected from the controller and monitored by the tester during testing. Additional design effort would probably lead to improved testing methods (for example, pulsing the power supply output voltage). Since the power-range circuits are conventional, adaptation of testing methods developed over the next several years will be readily feasible.

Intermediate Range — The intermediate instrument consists of two identical channels. Each channel is of the mean-square-voltage (MSV) or "Campbell" type and logarithmically covers seven decades of reactor power from approximately 10% of power downward. The MSV method is chosen over the conventional logarithmic d-c intermediate-range technique because detector and cable leakage currents at 800 F can be expected to exceed considerably the 10^{-11} -ampere lower limit of the d-c system. Although they are relatively new, MSV circuits have been successfully tested,^{3,4} and the availability of control-grade MSV circuits by 1980 seems a certainty.

The detector in the MSV channel is an uncompensated ion chamber or fission chamber identical to the power-range detectors. The ceramic-insulated cable suitable for the power range, however, is probably not acceptable for the intermediate range because of poor a-c characteristics. Reference 3 reports development of a quartz and S-glass cable which is expected to be satisfactory if it is enclosed in a solid metal sheath. Reference 3 also contains a detailed description of an MSV channel, and reference 4 provides an excellent summary of the operating principles. Very briefly, the MSV circuits perform the following duties:

1. Block the d-c component of the detector output signal.
2. Pass the a-c component through a band-pass amplifier.
3. Square the amplified a-c component.
4. Average the squared signal.

Two important benefits are derived from these operations. First, blocking the d-c signal virtually eliminates leakage current as a noise source; almost the only limit on leakage is the ability of the detector power

supply to keep up with it. This is significant, because at the 800 F temperature most cables exhibit leakage currents well above the lower limit of a d-c intermediate range. Second, the squaring operation increases the neutron-to-gamma signal-to-noise ratio by a factor of 1000,³ which is better gamma rejection than can be obtained in service from a gamma-compensated ion chamber. This is significant because detector response to sodium-24 activity (which is not proportional to present power level) could be well above the bottom of a d-c intermediate range in an LMFBR.

Figure 12-21 is a simple schematic diagram of the intermediate-range instrument; both channels are identical. The circuits, other than the MSV, are again conventional, solid-state circuits. Level is displayed for operator information only. Period is displayed, alarmed at about 10 to 15 seconds, and initiates a controlled rod insertion (setback) at 5 to 10 seconds; the setback will clear when this period increases to above 10 to 15 seconds. If desired, the alarm can be accompanied, or buffered from the setback, by a trip which prevents rod motion; however, this complication is not considered necessary. Either intermediate-range channel can initiate these functions.

The principal reasons that scrams are not initiated from the intermediate range are as follows:

1. The intermediate-range circuits are slow.
2. The intermediate-range circuits are complex and cannot reasonably be made as reliable as power-range circuits (because of wide range and small low-end signals).
3. Adequate protection against accidents beginning below the power range is obtained from the high power-to-flow and overpower scrams.

Automatic testing of the intermediate-range channels is performed only during reactor shutdown and power-range operation.

Source Range — The source-range instrument consists of two identical channels, as shown in Figure 12-22. The channels are of the type conventionally used in power reactors, except that the detectors are fission counters rather than the more common boron-trifluoride (BF_3) proportional counters. Fission counters were selected for the following major reasons:

1. The 800 F temperature is probably too high for more sensitive proportional counters, but improvement of the temperature rating of existing fission counters is considered straightforward and feasible.

2. The shutdown gamma level at the detectors may be several hundred R/hr, which is probably too high for proportional counters (even boron-lined); even if fission counters cannot tolerate the shutdown gamma level, a minimum of gamma shielding is required if they are used.

3. The high natural source level of the fast breeder core and the low neutron attenuation in sodium are expected to permit on-scale monitoring of the shut down core, even with the lower sensitivity of fission counters.

The source-range detectors will be located with the intermediate- and power-range detectors. The detectors will be stationary (except for the capability of replacing them with the reactor shut down); motion will not be required during source-range operation, and retraction will not be required at higher power levels. No spare source- or intermediate-range detectors are installed, on the premise that the detectors are replaceable at shutdown, and these ranges are only used at shutdown or startup.

The source-range detectors are identical to the fission chambers that might be used in the power and/or intermediate ranges. The source-range detector is a pulse instrument only because the input circuit time constant is so short, and the range of monitored neutron flux is so low, that individual neutron events in the detector are resolved. Thus it is feasible for the source and intermediate ranges to share two detectors, each feeding one channel of each range. It is also feasible for these same two detectors to serve the power-range information and control channel. Reference 4 reports successful operation of a single detector with pulse, MSV, and d-c circuits to monitor the complete range from reactor shutdown to full power. Further analysis is required to determine whether this approach is preferred. Note that the power range safety channel detectors would remain separate and distinct in any event, to preserve the simplicity and reliability of these channels.

Figure 12-21. Nuclear Instrument Intermediate-Range Channel, Schematic Diagram

```

graph LR
    Detector["Detector  
(In Radial Neutron Shield)"] --> Preamplifier["Pulse Preamplifier  
(In Cover Gas)"]
    PowerSupply["Power Supply"] --> Preamplifier
    Preamplifier --> Amplifier["Pulse Amplifier"]
    Amplifier --> Discriminator["Pulse Height Discriminator"]
    Discriminator --> Integrator["Pulse Integrator"]
    Discriminator --> OM1["To Operator Display Meters"]
    Integrator --> LogCircuit["Log Circuit"]
    LogCircuit --> Junction1(( ))
    Junction1 --> PeriodCircuit["Period Circuit"]
    PeriodCircuit --> OM2["To Operator Display Meters"]
    PeriodCircuit --> Bistable1["Bistable"]
    Bistable1 --> Alarm["Alarm"]
    Bistable1 --> Bistable2["Bistable"]
    Bistable2 --> Scram["Scram"]
    Bistable2 --> Shutdown["Close Only During Shutdown"]
  
```

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Because of high temperature, conventional electrical cable is, again, not usable in the source range. (The detectors and cables could, of course, be force-cooled with gas or liquid, but this approach is so undesirable that it would be considered only if 800 F detectors and cables cannot be developed, and that possibility appears remote.) The quartz-insulated coaxial cable developed for in-core counters as reported in reference 3 appears to be satisfactory for the source range, which has critical pulse-transmission requirements. If common detectors are used in different ranges, this type of cable should be suitable for all ranges. Here again, cables integral with the detectors are desirable but not absolutely necessary.

As indicated in Figure 12-22, the source-range circuits are completely conventional transistorized equipment, although extension of the upper count rate limit to 10^6 (random) counts per second is a relatively recent development.³ The use of a fission counter as the detector requires only a few hundred volts from the power supply, rather than the 2000+ volts required for a BF_3 counter. The pre-amplifier may have to be located in the cover gas above the pot sodium to minimize cable attenuation of unamplified detector pulses; for this reason, it should be a high-reliability unit—the type developed by ORNL, for example⁵—since access for maintenance will not be very convenient.

The source range circuits compute both count rate and period. Count rate is logarithmically displayed for operator information only; period is displayed and also alarmed at about 10 to 15 seconds. Automatic protection is not initiated by the source range for the same reasons that the intermediate range does not: it is slow at low levels, complex, and unnecessary. Source-range channels are also traditionally noisy instruments. An exception is the provision for a high count rate scram (at perhaps 10^5 counts per second, a relatively noiseless region) for use during shutdown periods.

The basic check for operability of the source-range channels is on-scale response to the shut-down core. Automatic testing of most of the circuits is also provided.

12.5.1.6. Principles of Operation

The power-range detector is a conventional, gas-filled ionization chamber, rendered neutron-sensitive by including either enriched boron-10 or enriched uranium-235. Except for temperature rating and the desirability of using integral cables, the detector is identical to the ion chambers or fission chambers traditionally used in power-range channels.

As discussed above, the circuits used in the power range are also conventional. However, the best available automatic testing techniques should be used to minimize the possibility of false scrams and undetected component failures.

Except for the MSV circuits described above, the intermediate-range channels are conventional. Detector operation is the same as in the power range. The log-taking circuits will be transistorized.

The operating principles of the source range are conventional and straightforward. Pulses from the detector (microvolts amplitude) are amplified by the preamplifier to override attenuation in the long cable run to the amplifier, where the pulses are amplified to a usable level. The discriminator is, in fact, a biased uni-vibrator which is triggered by neutron pulses but not by smaller gamma pulses. The discriminator output is a series of identical pulses bearing a one-to-one time correspondence to the detector neutron pulses (but free of the detector pulse amplitude and shape variations). These pulses are integrated to form a direct current proportional to neutron count rate, and the logarithm of this current is formed by a transistor network because of the wide range of count rates measured by the source range. The log count rate is displayed to the operator and is also differentiated for period display.

12.5.1.7. Safety Precautions

Automatic testing is depended on to avoid possibly hazardous conditions resulting from power-range malfunction. Although the detectors are not fully tested for power range operation, additional confidence in them can be gained by comparing amplifier gain

settings, comparing currents in the redundant channels, and observing their overlap with the intermediate range during startup.

Automatic testing and overlap with the source range are the basic preventatives against putting a defective intermediate-range channel into service. Setback coincidence of 1 of 2 (in effect), plus operator observation of level and period meters protect against the loss of a channel during startup. Ultimate protection, however, is obtained from the high-level scrams.

The source-range safety precautions are similar to those of the intermediate range, except for the existence of direct on-scale response to the shutdown core. It is noted that the detector high-voltage power supply need not be de-energized at high power levels to prevent detector damage, as is necessary with BF_3 counters.

12.5.1.8. Maintenance

Maintenance of the nuclear instrumentation other than detectors will be straightforward and conventional. None of the circuits are more complicated than present-day source-range channels, and the application of advanced automatic testing techniques and modular construction will greatly facilitate alignment and repair operations.

Detector replacement will be generally comparable to present-day power reactor practice in complexity and radiation hazard, except that the alternate installation method employing the refueling machine may be more complex.

12.5.2. Core Instrumentation System

12.5.2.1. System Function

The core instrumentation system includes four subsystems which monitor core performance. The subsystems are core neutron, fuel assembly outlet temperature, fuel assembly flow, and core differential pressure instrumentation.

Core Neutron Instrumentation — The core neutron instrumentation continuously measures neutron flux at 22 points within the reactor core and periodically measures axial neutron flux profiles through these points. These measurements are used for the following purposes:

1. Maintaining flat power distribution in the as-built core.
2. Confirming performance characteristics of the as-built core.
3. Providing information for optimum fuel management.

Fuel Assembly Outlet Temperature Instrumentation — The fuel assembly outlet temperature instrumentation continuously measures the coolant outlet temperature for each of the 288 fuel assemblies. These measurements are used for the following purposes:

1. Detection of progressive flow blockage in any fuel assembly.
2. Detection of coolant voiding in any fuel assembly.
3. Determination of power generation in selected fuel assemblies containing flow meters (in conjunction with the flow meter and core inlet temperature measurement) for fuel management purposes.

Fuel Assembly Flow Instrumentation — The fuel assembly flow instrumentation continuously measures the coolant flow through eight selected fuel assemblies. This measurement is combined with assembly differential temperature measurements obtained outside of this subsystem to determine assembly power. The assembly power measurements will be used for fuel management purposes.

Core Differential Pressure Instrumentation —

The core differential pressure instrumentation continuously measures the static differential pressure between the core inlet and outlet plenums, and hence, the total coolant flow through the core. This measurement is used for the following purposes:

1. Protection against unacceptable low flow through the core, independent of power level.
2. Protection against an unacceptably high ratio of power to flow at any power level, in conjunction with the power-range nuclear instrumentation.

12.5.2.2. Summary Description

Core Neutron Instrumentation — Eighteen min-

iature neutron detectors are symmetrically distributed throughout the core, and four more are asymmetrically added in a single 1/6 section. The sensitive volume of each detector is located at the fuel mid-height. The detectors are located in permanently installed dry thimbles and are replaceable at power. Prompt detector response permits continuous determination of core power density distribution.

A single additional miniature neutron detector can be inserted in normally-empty, dry thimbles adjacent to the 22 stationary detectors. The movable detector is used for periodic measurement of axial neutron flux profiles in the core and for periodic accurate intercalibration of the stationary detectors. These normally-empty thimbles will be available for inserting special materials (such as resonance energy flux wires), and the complete drive system will be available for special detectors (such as a gamma ion chamber).

Power distribution information from the stationary detectors will be digitally processed and indicated to the operator. If any two azimuthal 1/6 sections of the core differ by more than a designated amount, an alarm will request the operator to reshape the power distribution by moving individual control rods; a greater difference will initiate a power rundown to 95% power. The core neutron instrumentation initiates no scram action and no automatic rod withdrawals.

Data from the core neutron instrumentation will also be digitally tape recorded for use in fuel management planning and as-built core performance analysis.

Fuel Assembly Outlet Temperature Instrumentation — One thermocouple is placed in the coolant outlet flow path of each of the 288 fuel assemblies. Each thermocouple is installed in a dry guide tube that terminates just above each fuel assembly. Thus, each thermocouple continuously monitors the outlet coolant temperature of a single fuel assembly. The thermocouples are replaceable at shutdown.

Each outlet thermocouple will be digitally scanned once per second. If any thermocouple output shows a significant increase, that thermocouple and its six adjacent neighbors are switched to fast scan at 0.2-second intervals. If both the first thermocouple and any one neighbor then show significant increases, a power reduction or scram is initiated. Operation with a failed thermocouple is permitted by fast-scanning its six neighbors and acting if any two show increases. The philosophy behind both of these operations is that if an assembly voids, its neighbors will also experience temperature increases due to heat conducted from the voided assembly.

Additional protection against the consequences of assembly voiding is provided by the high power and rate scrams of the power range nuclear instrument, which responds rapidly to the reactivity increase associated with the voiding of most assemblies.

For the eight fuel assemblies that contain flow meters, the outlet temperature measurement is digitally combined with core inlet plenum temperature measurement and flow measurement to continuously determine the volume-integrated power generation in each assembly, which is used in fuel management.

Fuel Assembly Flow Instrumentation — Six individual flow meters are located at the inlets to fuel assemblies symmetrically arranged in the core, and two more are asymmetrically added in a single 1/6 section. The inlet location was chosen so that the flow meters are exposed only to the inlet sodium temperature, and so that the leads may be kept clear of fuel handling operations. Although the flow meters are essentially non-replaceable, they could be replaced at shutdown with extreme difficulty.

As noted above, the assembly flow meter measurements are combined with temperature measurements to calculate

power for each monitored assembly for fuel management purposes. Comparison of this coolant enthalpy-derived local power with local power derived from the core neutron instrumentation is facilitated by performing both types of measurements in three fuel assemblies.

Core Differential Pressure Instrumentation —

Three identical core differential pressure detectors are provided to permit 2 of 3 coincidence logic. Each detector comprises an out-of-pot electromechanical differential pressure transducer connected by metal tubing to the core inlet and outlet plenums; thus the in-pot components are restricted to mechanical hardware, which will require only minimal maintenance.

The differential pressure measurement is calibrated for flow determination by comparison with the primary pump instrumentation, both initially and periodically thereafter. This procedure assures accurate flow measurement, even if the differential pressure/flow relation changes because of crud buildup or other aging effects. The individual channels will be automatically tested at frequent intervals.

The digital computer compares the core flow and power range nuclear instrument signals with a power-to-flow program and will initiate high power-to-flow scram.

12.5.2.3. General Design Requirements

The general design requirements will be as specified in 12.1.3 for the instrumentation and control system and in 12.3.1.3 for the protection system, except that the latter requirement will not apply to the fuel assembly flow instrumentation.

12.5.2.4. Detailed Design Requirements

Table 12-4 specifies the design requirements of the in-core instrumentation subsystems. Each subsystem will monitor its particular sensed variable over the indicated range, and it must perform this function with its detector in the environment formed by any combination of the other variables.

Table 12-4. Detailed Design Requirements for the
In-Core Instrumentation Subsystems

Variable/subsystem	Core neutron	Fuel assembly outlet temperature	Fuel assembly flow	Core differential pressure (transducer)
Temp, operate, F	800 - 1400	800 - 1600	700 - 900 (bottom) 700 - 1200 (top)	1000
Temp, withstand, F	0 - 1400	0 - 1600	0 - 1000 (bottom) 0 - 1600 (top)	0 - 1000
Pressure, psia	10 - 100	10 - 100	10 - 100	10 - 100
Differential pressure, psid	--	--	--	0 - 40
Max neutron flux, nv (all energy)	10^{16} $10^{14} - 10^{16}$ operate	10^{16}	10^{16}	10^8
Gamma dose rate, R/hr	5×10^8 (75% proportional to power-oper)	10^8	10^8	10^5
Environment	Inert gas	Inert gas	Na	Na or Ar
Useful life, yr	5 ^(a) 10 desired	5 10 desired	30	5
Calibration provisions	yes (relative)	Not in oper range	No	Yes
Replaceable	Any time	Shutdown	No	Shutdown

(a) Movable detector has 0.01 service factor (in core only 1% of time).

12.5.2.5. Detailed Description

Core Neutron Instrumentation — Stationary in-

core neutron detectors are located at 22 positions in the core, as shown in Figure 12-23. Eighteen of the positions are symmetrically dispersed among the control rods in an inner ring of 6 and an outer ring of 12.

These 18 locations follow complete six-fold symmetry and no more than two (in most cases, one) fuel assemblies lie between an instrumented assembly and the most distant assembly. The other four positions are located in the same $1/6$ section of the core, as follows: one in the fuel assembly adjacent to the central control rod, one in the assembly provided with a flow meter, and two in assemblies which are in fact three assemblies removed from the symmetrically instrumented assemblies. Since the latter two assemblies are repeated in twelve-fold symmetry, in effect, no assembly in this heavily instrumented $1/6$ section is more than two assemblies removed from an in-core neutron detector (that is, there is at most only one fuel assembly between any assembly and the closest in-core neutron detector). Further study might lead to a different arrangement having perhaps more or fewer locations, but it probably would not differ substantially from the present arrangement.

A vacant dry thimble is located immediately adjacent to each of the 22 dry thimbles that contain stationary detectors. The vacant thimbles provide for periodic introduction of a movable neutron detector which very accurately (approximately 1%) intercalibrates the stationary detectors, and which at the same time obtains axial flux maps at the 22 locations. The vacant thimbles also permit the introduction into the core of other detectors and materials such as proton recoil counters (hopefully) and threshold foils or wires for measuring more detailed core performance.

Each thimble is installed in a wet guide tube which replaces one fuel pin and is an integral part of the fuel assembly. A mating guide tube extension passes from the lower core support structure (to which it is welded) through the core vessel wall and up to the stationary portion of the operating floor, where it is seal-welded to a box recessed into the floor. The thimble is jacked into the guide tube

Figure 12-23. In-Core Instrument Arrangement

for operation and is withdrawn into the guide tube extension (approximately a 6-foot stroke) if that fuel assembly is to be replaced. The thimbles extend continuously to a small drive room located above or beside the reactor vessel, where drive, thimble indexing, and readout equipment is located. Backup protection against thimble failure, and disposition of the small quantity of radioactive inert gas generated inside the thimbles, is achieved by pressurization and valving in the drive room. Sliding seals control leakage around the thimbles inside the box recessed in the operating floor, and a positive seal is maintained at the top of this box. Thimbles are jacked using a portable short-stroke jack at the top of the box.

An in-vessel minimum bend radius of approximately 8 feet will be maintained to facilitate thimble jacking, and thimble replacement can be accomplished through the access box when necessary. (Replacement would be occasioned by thimble leakage and possibly on a periodic basis, depending on the ability of the thimble material to resist radiation embrittlement.) Since each fuel assembly is handled only once in three years, and thimbles are disturbed only in assemblies being replaced, thimble jacking is infrequent.

Thimbles for the stationary detectors can be handled in the same manner as is used with the movable detectors. Alternately, these thimbles can be terminated in hermetically sealed electrical connectors at the stationary portion of the operating floor, with the signal transmitted to the readout equipment in the drive room by conventional coaxial cable. As noted in section 12.5.2.8 (Maintenance), the longer thimble path facilitates the replacement of a radioactive failed detector.

Drive equipment for the movable detector can be of the type recently supplied by General Electric Company for the Pathfinder and KBR reactors, for example. A synchronous electric motor of approximately 0.25 horsepower acts on a small wire spiral-wrapped on the detector cable through a rack-and-pinion drivebox. Drive and indexing equipment is expected to be thoroughly proved by 1980 because of its widespread application in the large thermal reactors now being designed.

Figure 12-24. In-Core Instrumentation, Schematic Diagram

Detectors may be either of two types: miniature ion chambers or self-powered detectors. The first choice between these two would be a prompt-response, self-powered, fast neutron detector with a high neutron-to-gamma signal-to-noise ratio, if such a detector can be developed. There is no such detector now, but the possibility of its successful development is encouraging. Self-powered in-core detectors offer the following advantages for this application, compared with ion chambers:

1. Reliability should be greater because of simpler construction and the absence of hermetic seals (except for sheath-closure welds).
2. The cost should be less.
3. Detector diameter is less (about 0.040 to 0.060 inch versus about 0.188 inch for a high quality ion chamber), which permits multiple-detector installation in place of a single fuel pin.
4. The absence of the rigid cylinder formed by an ion chamber allows the self-powered detector to negotiate tighter bends in the thimble path, which simplifies installation.
5. The absence of a power supply and the associated 100 to 200 volts in the system simplify circuit requirements and facilitate data processing.

These advantages indicate the desirability of developing a prompt-response, self-powered fast neutron detector.

If an in-core neutron ion chamber were used, it would probably employ MSV (Campbell) circuitry to eliminate the effects of high-temperature leakage current and increase the signal-to-noise ratio, as discussed in the nuclear instrument system intermediate-range detailed description. The required ion chamber development effort would include selecting an optimum neutron-sensitive material, upgrading the temperature rating from approximately 800 to 1400 F (allowing for radiation heating in the dry thimble), and proving by test the radiation resistance of the required low-capacitance cable.

The stationary and movable in-core neutron detector channels are shown schematically in Figure 12-24 (upper half). Data processing will be performed by the digital computer. In addition

to digital data handling, the axial flux profiles generated by the movable detector can be directly plotted (versus core height) by the X-Y plotter. Variable gain adjustments for the stationary detectors are varied to provide intercalibration based on their sequential comparison with the movable detector at periodic intervals of perhaps one week to one month.

Sensitivity reduction of the stationary detectors caused by the burnup of their neutron-sensitive loading will not be as great as might be expected. A fission chamber, for example, would undergo burnup comparable to the fuel—perhaps 10% in one core lifetime. Detectors using lower-cross-section materials (as would be the case with self-powered detectors) would experience even less sensitivity change because of depletion. The damages inflicted on the material by large, integrated fast neutron doses, however, are unknown at this time. Since the movable detector is only in the core a small part of the time, burnout and aging effects in it will take place very slowly.

It is noted that the inclusion of 22 normally-vacant thimbles in the core, and a drive system capable of reaching any of these thimbles, provides ready capability for performing additional measurements in the core. For example, if a suitably small proton recoil counter is developed, it can be used for occasional neutron energy spectrum measurements in the three-dimensional matrix provided by these thimbles; such measurements are useful in determining as-built core performance and in fuel management planning. As another example, gamma detectors can be used to determine at-power gamma distribution (and if necessary to correct for gamma background signal in the neutron detectors), and to determine the shutdown fission product activity distribution as a means of inferring fuel depletion distribution. Further, a high-sensitivity neutron counter or chamber could be used to obtain sub-power-range flux distributions if such detectors are available for the application. Also, foils or flux wires can be conveniently inserted in the core for measuring resonance and threshold fluxes, which give additional detailed assessment of nuclear performance.

The stationary neutron detectors are used to maintain a flat core power distribution on a six-fold symmetry basis. Since each 1/6 section contains three symmetrically repeated detectors, a pseudo-coincidence is obtained in the sense that if one detector fails,

the remaining two detectors in that section can still be compared with symmetrically located detectors in the other five sections. Flux measurements by the stationary detectors are also used for confirmation of as-built nuclear performance and for fuel management planning. Finally, it is expected that the detector response will change if a nearby fuel assembly voids (because of the change in local neutron energy spectrum); this would aid in diagnosing this accident condition.

Automatic testing comparable to that used for the nuclear instrument power range will be employed for the in-core nuclear instruments. Circuit functions can be readily incorporated into the digital computer (except high-voltage power supplies for ion chambers, if they are used).

Fuel Assembly Outlet Temperature Instrumentation — One thermocouple is located in the coolant outlet flow path above each of the 288 fuel assemblies. For each fuel assembly a thermocouple guide tube extends from the top of the inner rotating refueling plug, down the side of one of the 25 control rod guide tubes, across the top of the core holddown assembly, and downward through flow holes in the hold-down plates, terminating just above the appropriate fuel assembly. A free span between the refueling plug and control rod guide tube takes up the 12-inch stroke of the holddown assembly. The thermocouple tip bottoms in the guide tube, just above the sloping coolant discharge path from the fuel assembly upper end fitting. This position "sees" about one-third of the coolant from the assembly. This fraction is sufficient for the basic void-monitoring function of the thermocouples, but calculations are needed to relate thermocouple reading to true mean outlet temperature for assembly enthalpy measurement.

Dry-interior guide tubes are used to facilitate thermocouple replacement and to improve their reliability, although these are achieved at the expense of increased response time. Careful detail design must ensure good repeatable contact between the thermocouple tip and the guide tube. The thermocouples have enclosed ungrounded junctions, again to increase reliability at the cost of poorer time response. The combination of this junction with dry guide tubes in a relatively straight installation path is expected to maximize reliability

and minimize failure rate in terms of both drift and gross failure. (The effect of the increased response time is discussed below.)

Chromel-alumel thermocouples will be used, unless by 1980 another type is shown to be superior. Although the long term performance of chromel-alumel in combined high radiation (particularly fast neutron) and temperature has not been demonstrated, neither has it been demonstrated for other types. Chromel-alumel shows the best balance of performance in the following areas: high output, ready availability in satisfactory size and sheath material, proven out-of-core capability at the required temperatures, proven in-core capability in water power reactors, low theoretical tendency toward calibration changes caused by transmutation,⁶ high sensitivity, and a well-behaved emf versus temperature response curve shape.

The dry guide tubes and sheathed thermocouples terminate in sealed boxes on the inner rotating refueling plug. Electrical leads penetrate the boxes through hermetically sealed electrical feed-through connectors, which are disconnected before refueling. The extension leads extend to a reference junction box and then to the scanner input to the digital computer. The computer scans the thermocouples in time sequence and performs the voltage-to-temperature conversion.

The principal uses of the fuel assembly outlet temperature instrumentation are the detection and location of temperature increases associated with fuel assembly flow blockage or voiding (the secondary function of assisting in fuel assembly power measurement is discussed on page 12-108). Since the accuracy and response time requirements are different for progressive flow blockage and for sudden blockage or voiding, the two cases are considered separately.

Progressive flow blockage occurs gradually, over a duration of days or longer; crud buildup is a major probable cause. In this application the thermocouple time response is an insignificant factor, but accuracy with respect to long-term drift is important. As previously mentioned, the thermocouple type and installation design are chosen to minimize drift. However, in the absence of data to the contrary, it is estimated that long-term drifts of 20 to 50 F may occur. With this

possibility, long-term accuracy depends on the ability to calibrate the thermocouples during their service life.

Out-of-core calibration of the thermocouples during life is undesirable, because (1) it would not necessarily reflect in-place performance; (2) the required handling would undoubtedly reduce reliability; and (3) it would require handling radioactive thermocouples. Core isothermal conditions can be approximated after shutdown with the primary pumps operating at full speed; then the calibration error is determined by the amount of decay heat remaining in the fuel. For a shutdown time of approximately one day, the decay heat error would be approximately 3 F for most thermocouples, and somewhat larger for assemblies having some flow blockage. This calibration would be at approximately 800 F. For longer shutdown periods, decay heat would cause negligible error, but the pot temperature would be only 300 to 400 F.

The calibration methods cited above are the most accurate obtainable, but the temperatures fall far short of the 1100-1600 F range in which information is required from the thermocouples. Thus it is desirable to compare the distribution of full-power thermocouple readings (near 1100 F) with power distributions obtained by calculation and confirmed by the in-core neutron detectors. Significant discrepancies can then be resolved by replacing the questionable thermocouples, since deviations could be caused by local flow or power variations as well as by thermocouple error. The following tabulation summarizes the estimated capability for in-place calibration of the thermocouples, including allowance for gamma heating errors:

<u>Temperature, F</u>	<u>Calibration error, $\pm^\circ\text{F}$</u>
300 to 400	3 to 5
400 to 800	5 to 10
800 to 1100	10 to 20
1100 to 1600	20 to 40 (extrapolated)

Since progressive flow blockages will generally require temperature measurements in the range from 1100 to 1150 or 1200 F, an accuracy of perhaps ± 25 F is expected. Of course, this estimate must be tempered by experimental drift data as they become available.

Sudden flow blockage or assembly voiding is a rapid effect, in contrast to the progressive voiding situation. The resulting temperature increase is so large and so fast that, with respect to accuracy, the thermocouple need do little more than produce an increased output as the temperature rises. Since the command signal from the thermocouples consists of temperature increase between successive scans, and since the temperature increase is faster than the guide-tube-and-thermocouple response time, the command signal magnitude is dominated by response time rather than accuracy. Furthermore, long-term drift effects become negligible, since only very-short-term changes are significant.

To a first approximation, thermocouple detection of assembly voiding consists of an exponential response to a step input, where the step magnitude is 500 F and the exponential is defined by the time constant of the guide tube and thermocouple. In this case, the rate at which the indicated temperature rises is initially very fast and decreases with time. Assuming a 1-second scan interval, the increase in indicated temperature during the first second is shown below for some representative time constants (where time constant is defined as the time to reach 63% of the step amplitude, or 315 F):

<u>Time constant, sec</u>	<u>ΔT after 1 sec, F</u>
1	315
3	140
5	90
10	50
15	30

Similarly, for the 0.2-sec fast scan interval, the temperature increase after 0.2 sec, as a function of the instrument time constant, varies as follows:

<u>Time constant, sec</u>	<u>ΔT after 0.2 sec, F</u>
1	90
3	30
5	20
10	10
15	6

It is reasonable to establish setpoints in the vicinity of 30 to 50 °F/sec, or 6 to 10 °F/0.2 sec. These values lie well above normal operation values and provide reasonable tolerance for noise. Accordingly, with a liberal allowance for thermocouple errors, time constants in the neighborhood of 5 to 10 seconds are satisfactory. It is also noted that a 10-second time constant corresponds to a 10-to-90% response time of approximately 20 seconds, which is practically achievable with 1/16-inch thermocouples in dry wells.

The preceding discussion assumes that when a fuel assembly voids, the thermocouple is exposed to very hot sodium for perhaps a tenth of a second. This would not be the case if an assembly inlet were suddenly, completely, and permanently blocked. In this event, "percolation" of sodium from the core outlet plenum might not cause sufficient thermocouple indication change for some time. However, sudden, complete, and continued blockage of flow to any assembly is extremely unlikely because of the inlet geometry.

The inability to anticipate fuel assembly voiding or to effectively act to prevent it is conceded (except in the special case of long-term progressive blockage). However, continued power operation with molten fuel is considered undesirable. As discussed below, the thermocouples will initiate a power setback or scram if any fuel assembly in the core voids. In most cases this action merely serves as backup for the nuclear instrument system power-range rate scram, which is expected to act if any fuel assembly voids other than those in the outermost ring (voiding an assembly in the outermost ring would produce negligible reactivity effects). The thermocouples, of course, would identify the voided assembly, which the power range cannot do.

Protective action based on the thermocouples employs coincidence techniques, two scan rates, and two distinct setpoints. A high setpoint, at approximately 50 °F/sec in slow scan and 10 °F/0.2 sec in fast scan, corresponds to voiding a fuel assembly. A low setpoint, at approximately 20 °F/sec and 4 °F/0.2 sec, corresponds to an assembly temperature increase as a result of voiding an adjacent assembly. The uses of these setpoints are as follows:

All thermocouples are scanned once per second routinely. If any reading exceeds the low setpoint, an alarm is initiated, and that thermocouple and its six immediate neighbors are transferred to fast scan for perhaps 1 minute. A normal speed rundown terminating in a full shutdown is initiated upon one thermocouple reaching the high setpoint and one reaching the low setpoint during fast scan; a scram is initiated if two reach the high setpoint.

In the case of a failed thermocouple, its six neighbors are routinely placed on fast scan. An alarm will be initiated if any of the six reaches the low setpoint, a normal speed rundown if any two reach the low setpoint, and a scram if one reaches the high setpoint and one reaches the low setpoint. In this manner, provision is made for operation with failed thermocouples. Assuming that a 100% increase in thermocouple scanning capability is accepted as a penalty, operation with as many as eight failed thermocouples is permitted. A further prudent operating restriction is that no two adjacent thermocouples may be failed. It is expected that these restrictions will result in an extremely low probability of plant outage because of failed thermocouples, and that replacements can be performed during scheduled routine shutdowns.

Thermocouples were selected to monitor assembly voiding, whereas flow meters would appear to be a more logical choice. Thermocouples were chosen because of their expected higher reliability, easier installation and replaceability, simpler readout, and capability for pseudo-redundancy through coincidence; it is further expected that their cost will be less than that of flow meters.

Fuel Assembly Flow Instrumentation — Flow meters are located at eight fuel assemblies, as shown in Figure 12-23. Six of the meters are symmetrically arranged in a ring approximately

mid-way between the two rings of in-core neutron detectors. The other two are located at assemblies that contain neutron detectors—one in each ring; these two locations are in the same 1/6 core section most heavily instrumented with neutron detectors.

Each flow meter measures the coolant flow through one fuel assembly. Flow meter location at the assembly inlet beneath the core offers two important advantages: the electrical lead paths can be kept completely free from interference with fuel handling operations; and the sodium temperature is only 800 F (compared with the full-power outlet temperature of 1100 F). The flow meters are welded to the lower support grid structure.*

Individual fuel assembly flow measurements will be combined with enthalpy rise measurements obtained from common inlet plenum temperature measurement and individual outlet

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* Two other flow meter assembly locations were considered and discarded:

1. The meters could be attached to the top grid structure, with the advantages of somewhat easier access and the assurance that loose parts would not be swept into the core (although they could fall in with no flow). These advantages are considered overridden by the higher temperature and lead routing complications.

2. The meters could be attached to the fuel assemblies. Although this approach would greatly facilitate flow meter maintenance and replacement, it requires either an under-sodium electrical disconnect or manipulation of the electrical leads extending from the fuel assembly during refueling (preliminary assessment of the handling problems associated with the longer fuel assembly length indicates that little complication may be involved). Since the flow meters are non-vital, neither of these complications appears necessary. However, it is noted that development of a reliable under-sodium electrical disconnect method could justify reassessment of the flow meter mounting.

The selection of the type of flow meter to be used, on the basis of demonstrated performance, must be somewhat arbitrary. As discussed in reference 7, permanent magnet flow meters appear most desirable at the present time. Orifice or flow nozzle meters may also be desirable, since the electromechanical portion can be remote from the core and hence more accessible for maintenance. Future development efforts and experience with the FFTF and demonstrator plants will dictate the type of in-core flow meters to be procured in 1980. The circuitry to be used is, of course, determined by the type of flow meter.

temperature measurements (which are otherwise available) to determine the power generation in each of the monitored assemblies.

Three fuel assemblies have both neutron detectors and flow meters (one in each of three rings as shown in Figure 12-23); all three are in the same 1/6 section of the core, and each is in one of the three core-encircling rings of neutron detectors or flow meters. The three "common" assemblies provide for direct comparison of the pointwise neutron flux measurements with individual assembly enthalpy rise measurements; thus, in effect, they provide a measured calibration of the neutron detectors in terms of local power (axially integrated). The ring of flow meters further provides measurement of axially integrated local power around the core.

Neutron detector "power calibration" by means of individual assembly enthalpy rise measurements is less accurate than calibration by physics calculations. Assuming a flow meter error of $\pm 5\%$ and a temperature measurement error of ± 25 °F in 300 °F, the root-mean-square error in assembly enthalpy rise is nearly $\pm 10\%$. This calibration is nonetheless considered useful for two reasons: (1) it is based on measurements and thus provides confirmation of calculation (assuming agreements), and (2) it is expected that changes in local flux-to-power ratio can be determined within only a few percent error.

No provision is planned for calibrating the flow meters after they are installed. Accuracy depends on pre-installation calibration and on the inherent ability of the flow meter to retain this calibration or at least to drift in a predictable manner. Calibration can also be inferred with changes in the fuel assemblies above the flow meters, particularly if the assembly flow characteristics are measured before and after service.

Core Differential Pressure Instrumentation —

Three identical core differential pressure detectors are provided for protection. Each detector consists of a differential pressure transducer connected by metal pipe (approximately 1 inch inside diameter) to the core inlet and outlet plenums, and located in an accessible region where the lengths of the connecting pipes are minimized. The transducer contains a diaphragm or bellows to convert differential pressure to a motion which is electrically sensed by such techniques as a linear

differential transformer or a strain gage. Plenum pressures are transmitted to opposite sides of the diaphragm or bellows by essentially stagnant primary coolant in the connecting pipes.

The remote transducer location is selected so that it is accessible for zero adjust, bleeding, flushing, and replacement as necessary. However, since the detector and the connecting pipes are filled with primary sodium, the location should be close to the pot, and provision must be made for bleeding, flushing, and spilling waste discharge. Proximity to the hot sodium in the pot is also desirable to facilitate maintaining the connecting pipes at core plenum temperatures (to prevent cold-trapping effects in the pipes).

The core differential pressure instrumentation will be calibrated initially by running the primary pumps at various speeds and comparing measured pressure differentials with total flows determined from pump head measurements and pre-installation head-versus-flow calibrations of the pumps. At the same time, differential pressure measurements across elbows in the pipes between the pumps and the core inlet plenum will be calibrated in terms of flow. These measurements then become the long-term standard for periodic calibration of the core differential pressure instrumentation. (Neither core ΔP nor pump ΔP is expected to remain an accurate measure of flow permanently, because of such effects as corrosion and crud buildup.) It is noted that complete, sudden flow blockage in a primary pipe or in the core is not credible; thus, the core differential pressure measurement is satisfactory for power-to-flow scram purposes, provided that it is calibrated at intervals of perhaps one week to one month. A calibration accuracy of about ± 3 to 4% is expected. Calibration results and a power-to-flow scram setpoint schedule are stored in the digital computer.

The computer compares core power, as measured by the power-range nuclear instrument, with core flow; if their relation lies above the scram setpoint schedule stored in the computer, a high power-to-flow scram is initiated with 2 of 3 coincidence.

A response time of less than one second is desired. This response is available with the selected detector but probably not with volumetric (capillary) detectors, which would otherwise be satisfactory.

12.5.2.6. Principles of Operation

Core Neutron Instrumentation — If in-core ion chambers are used, they will be similar to the nuclear instrument system power-range detectors, except for their size, which will be approximately 0.188 inch diameter by 2.5 inches long. A polarizing electrical field obtained from an external power supply collects ions that result from the neutron-induced fission of fissile material in the detector. Self-powered detectors operate differently. In devices available now, the signal is generated directly by beta decay of a material rendered radioactive by neutron capture; time response is controlled by the half-life of the radioactivant, which presently exceeds 30 seconds. In the desired developmental detector, prompt response would be obtained perhaps from an (n, α) or (n, p) reaction. In self-powered detectors, the ionizing particle travels through solid ceramic insulation from the emitting electrode to the collecting electrode by virtue of its own kinetic energy; the gas-filled ionization volume and polarizing electrical field of the ion chamber are not needed.

As noted in the previous section, the ion chambers require d-c power supplies; the large d-c leakage of the 1400 F detector and cable probably require MSV circuitry, which is described under the nuclear instrument system intermediate range. The MSV outputs will be scanned, digitalized, and processed by the digital computer. Further, in the case of the movable detector, the synchronous drive motor permits direct immediate printout of axial flux profiles on an X-Y recorder. The virtues of this feature include the ability to immediately repeat any questionable or particularly interesting data by manually overriding the semi-automatic drive system.

It is expected that the self-powered detector outputs could be directly connected to the digital computer's scanner without intervening circuitry. In this case, the computer would generate the prompt axial flux profiles.

Fuel Assembly Outlet Temperature Instrumentation — A thermocouple essentially consists of a closed electrical circuit formed by wires of two different metals—in this case chromel and alumel. If one junction (the "cold" or reference junction) of the dissimilar

metals is held at constant temperature, then the voltage in the circuit is determined uniquely by the temperature of the other junction (the "hot" or sensing junction). Standard reference calibrations for the thermocouple materials permit direct conversion of measured voltage to reference junction temperature. With digital scanning, the voltage-to-temperature conversion table is stored in the computer memory; each measured voltage is compared with the stored calibration to determine the corresponding temperature.

In 1980, it is presumed that scanning will be accomplished by solid-state amplifiers and multiplexers which will be developed and proved in the intervening years. We expect to obtain very high reliability and high scanning speeds using these techniques.

Fuel Assembly Flow Instrumentation — In a magnetic flow meter, sodium flowing between the two pole faces of a magnet generates an electrical field proportional to the velocity of flow. The electrical field is detected by electrodes mounted perpendicular to the magnet faces. There are no moving parts.

The core lower grid plate contains orifices that cause pressure drops to control flow distribution. In seven of the eight flow meter locations, the desired loss is about 5 psi, and in the eighth location it is about half of that. Since the flow meters constrict the inlet flow channel, it appears that the installation of flow meters that provide the desired pressure loss would be convenient in order to increase sodium velocity through the meter and to provide space for the magnet and housing. Since most of the pressure drop in the meters is recoverable, the working pressure drop will be much greater than 5 psi.

Core Differential Pressure Instrumentation — Core differential pressure is converted to an electrical signal as described above, amplified, and calibrated against the flow measured by primary loop instrumentation. The amplified signal is scanned by the digital computer, converted to a digital signal, and conditioned by the computer to provide the necessary protective functions.

12.5.2.7. Safety Precautions

Core Neutron Instrumentation — Automatic testing and intercalibration with the movable detector are the primary assurances of satisfactory operation. Coarse checks on detector operability can also be obtained at any time during power operation by comparing the detector readings with each other and with the out-of-core nuclear instrument power range. Operating procedures will prohibit operation with no less than two operating channels in each 1/6 core section to prevent loss of power distribution monitoring. The ability to calibrate detectors (including fresh replacements) at power with the movable detector permits accurate checking of the complete system whenever desired.

Fuel Assembly Outlet Temperature Instrumentation — Automatic testing and periodic thermocouple calibration are depended on to assure satisfactory operation. Procedures will prohibit operation with more than two adjacent or approximately eight total thermocouples failed. Nevertheless, it is probable that some difficulty will be encountered in determining long-term progressive flow blockage because the capability for calibration is quite poor.

Fuel Assembly Flow Instrumentation — Since the fuel assembly flow instrumentation provides neither protection nor operational control signals, its safe operation is ensured primarily by a detector which cannot block flow in case of failure. A solidly encased magnetic flow meter, welded in place, meets this criterion. The use of data from the flow meters will be well tempered by comparison with calculations and with neutron flux measurements.

Core Differential Pressure Instrumentation — Automatic circuit testing, direct calibration of the entire instrument against an independent flow measurement, and 2 of 3 coincidence are depended upon to ensure operation.

12.5.2.8. Maintenance

Core Neutron Instrumentation — The most significant maintenance item is replacing failed detectors. The fissile content of an ion chamber and its in-core location may lead to radiation

levels of thousands of R/hr after long service. The removal and disposition of a failed detector is most safely and conveniently accomplished if the in-core thimbles extend continuously to the movable detector drive room (as mentioned in the detailed description above). A rubber-wheeled friction drive box that can be used for any thimble, and a two-position switch in each thimble path leading to a common discharge thimble, permit remote removal of a failed detector from the core and its discharge to a suitable location, such as the equipment storage pit. The discharge thimble can terminate at a shear which cuts the radioactive detector and cable into convenient lengths and drops them into a disposal container. The friction drive system can also be used for detector installation. Note that friction drive is not used for the movable detector, because frequent use of the latter would require a more trouble-free drive system at somewhat greater cost.

With the detector replacement system just described, a failed detector can be removed, a replacement installed, and the replacement calibrated, all with the plant at power and within less than one hour. Further, these operations are accomplished remotely, without personnel radiation exposure, although it may be necessary to limit access to regions near the path of the failed detector during its discharge.

Alternatively, if thimbles for the stationary detectors terminate at the operating floor, detectors must be replaced with the plant shut down, and special procedures will be required to minimize personnel radiation exposure. The replacement detector, then, could not be calibrated until the plant returned to power.

No special maintenance provisions are necessary for the circuitry. Periodic preventative maintenance is required for indexing switches, valves, and similar components; this can be performed during scheduled plant shutdowns.

Fuel Assembly Outlet Temperature Instrumentation — The most significant maintenance item is the replacement of failed thermocouples, which requires penetrating the cover gas containment boundary and handling the radioactive thermocouple. Suitable handling procedures are expected to permit smooth thermocouple replacement during scheduled shutdowns.

Fuel Assembly Flow Instrumentation — Assuming the use of welded-in-place magnetic flow meters, maintenance is limited to routine circuit alignment, calibration, and overhaul. Detector malfunction or failure results in a loss of information. Conceivably, detectors could be replaced with the plant shut down, but this would be a hazardous and difficult operation. It is noted that if an overriding need for assembly flow meters should arise during plant life, they could be installed in the fuel assemblies.

Core Differential Pressure Instrumentation — Transducers require periodic maintenance, as discussed in the detailed description. The circuitry is conventional and presents no special maintenance problems.

12.5.3. Process and Radiation Monitoring Instrumentation

12.5.3.1. System Function

Process Instrumentation — The process instrumentation monitors variables throughout the reactor plant and buildings, except for measurements in the reactor core or measurements of radiation levels, as required to fulfill the functions of the instrumentation and control system in section 12.1.1. Monitored variables include those necessary for protection and control and also those for performance analysis and logging.

Failed-Fuel Monitor — This instrument monitors the primary coolant for fission products released from the reactor core and initiates an alarm if the measured activity exceeds a designated level. The presence of excessive fission product activity in the primary coolant implies the existence of defects in the fuel pins.

Radiation Monitor — This instrument monitors radiation levels throughout the reactor plant, at its radioactive discharge paths, and in its environs. It also monitors the temperature of spent fuel outside the core. In conjunction with the protection system, the radiation monitoring system initiates closure of the stack discharge path, containment isolation, or Na/NaK heat exchanger isolation valves if excessive radioactivity is being discharged. Other monitored parameters initiate alarms that require operator attention.

12.5.3.2. Summary Description

Process Instrumentation — Process instruments measure temperatures, flows, differential pressures, pressures, levels, leakage, valve positions, and several other parameters necessary for plant operation. These measurements provide much of the data required by the protection system, and most of the data required for plant control and performance logging. In general, each process instrument consists of a detector and the minimum amount of circuitry needed to condition the detector signal for acceptance by the digital computer input interfaces. The computer then performs logic operations on the data and performs protection, control, and annunciation functions as directed by its programming and by the operator.

Failed-Fuel Monitor — Fuel cladding defects permit fission products to escape from the fuel into the primary sodium. The escaping fission products include short-lived delayed neutron precursors, which will emit neutrons in the coolant downstream from the core. The failed fuel monitor passes a coolant sample past neutron detectors, which monitor for these delayed neutrons as evidence of fuel defects. The failed fuel monitor employs circuitry similar to the nuclear instrument source range, which feeds a count rate signal to the computer. If the count rate exceeds an alarm setpoint, the computer initiates an alarm. No automatic protection or control action derives from the failed fuel monitor.

Radiation Monitor — This is a group of radiation-monitoring instruments which measure radiation levels in and around the reactor buildings in a conventional manner. This monitor also measures the temperature of spent fuel between the core and the shipping cask. The circuitry is self-contained and conventional. Data and alarms are displayed locally and are supplied to the computer for central display and annunciation. The stack discharge monitor is a protection instrument; in conjunction with the computer, it can initiate automatic closure of the stack discharge path. The containment atmosphere discharge monitor is also a protection instrument and can initiate automatic isolation of the containment (except that emergency cooling paths are not closed). Finally, the NaK activity monitor is a protection instrument

which can initiate automatic isolation of the Na/NaK heat exchangers. The other instruments perform monitoring and alarm functions only.

12.5.3.3. General Design Requirements

1. The general design requirements will be as specified in 12.1.3 for the instrumentation and control system.
2. The stack discharge monitor, containment atmosphere discharge monitor, NaK activity monitor, and those process instruments listed in Table 12-5 as performing protective functions will meet the design requirements of the protection system, section 12.3.1.3, and will consist of triplicate, independent instruments for each measurement location.
3. Those process instruments listed in Table 12-5 as performing control functions will meet the design requirements of the control system, section 12.2.1.3.
4. Replaceable detectors will be designed for a 5-year life. Non-replaceable detectors will be designed for a 30-year minimum life.

12.5.3.4. Specific Design Requirements - Detectors

Process Instrumentation — The specific design requirements for the process instruments are listed in Table 12-5 under the following headings: Operating range, Accuracy, Response time, and Detector environment (temperature and material).

Failed-Fuel Monitor — The specific design requirements for the failed-fuel monitor are listed in Table 12-6.

Radiation Monitor — The specific design requirements for the radiation monitor are listed in Table 12-7.

Table 12-5.
Process Instrumentation

Parameter	Location and quantity	Detector type	Operating range	Accuracy	Response time, sec	Detector environment		Use
						Temp, F	Material	
For Protection (Safety Grade, 2 of 3 Coincidence for Each Location - see section 12.3)								
Primary Na temp	Core outlet plenum (several)	Chromel-alumel thermocouple	300 - 1100 F	±10 F	10	0 - 1100	Inert gas	High core outlet temp scram.
Inter. Na temp	Cold leg (3)	Chromel-alumel thermocouple	300 - 800 F	±10 F	10	0 - 800	Inter. Na	Loss of heat sink scram.
Internal gas pressure	SGs and RHs (6)	Bellows	0 - 100 psia	±1%	1	0 - 1100	Argon	Na-H ₂ O reaction scram.
Na detector	Interspace around reactor vessel (several)	"Spark-plug" type	--	--	--	0 - 800	Argon	Na in interspace rundown.
Cover gas pressure	Cover gas (several)	Bellows	0 - 100 psia	±1%	1	0 - 800	Argon	Actuate containment isolation.
Earthquake	Environs (several)	Seismograph system	--	--	--	0 - 150	Air	Seismograph scram.
Na temp	Reactor vessel, bulk (several)	Chromel-alumel thermocouple	300 - 800 F	±10 F	10	0 - 800	Primary Na	Actuate emergency cooling system, energize heater in NaK-air HX.
Primary pump power monitor	Pump power supply (6)	Power monitor	1340 hp	±0.5%	0.1	0 - 150	Air	Start pony motor.
Inter. pump power monitor	Pump power supply (6)	Power monitor	4100 hp	±0.5%	0.1	0 - 150	Air	Loss of inter. flow setback & scram.
FW pump speed	FW pumps (2)	Tachometer	0 - 5000 rpm	±1%	1	0 - 150	Air	Loss of FW flow setback & scram.
TG trip	Generator output (1)	Power monitor	1200 MWe	±0.5%	1	0 - 150	Air	TG trip & loss of elec load setbacks
For Control (see section 12.2)								
Primary Na temp	Core outlet plenum - mean (several)	Chromel-alumel thermocouple	300 - 1100 F	±0.375%	10	0 - 1100	Inert gas	Core power demand, heat balance.
Primary Na temp	Core inlet plenum (1)	Chromel-alumel thermocouple	300 - 900 F	±0.375%	10	0 - 1100	Inert gas	Primary flow control, heat balance.
Inter. Na temp	IHX outlet (6)	Chromel-alumel thermocouple	300 - 800 F	±0.375%	10	0 - 1100	Inert gas	Inter. flow control, heat balance.
Inter. Na temp	SG outlet (3)	Resistance temp detector	300 - 700 F	±0.25%	10	0 - 800	Air	Na bypass control, heat balance.
FW temp	SG inlet (3)	Resistance temp detector	32 - 500 F	±0.25%	5	0 - 600	Air	Na bypass control, heat balance.
Steam temp	RH outlet (3)	Resistance temp detector	32 - 1000 F	±0.25%	5	0 - 1100	Air	RH flow control, heat balance.
Steam pressure	Steam chest (1)	Bourdon tube or bellows	0 - 2500 psia	±0.5%	1	0 - 1100	Steam in/air out	FW flow control, heat balance.
FW flow rate	SG inlet (3)	Flow nozzle + bellows ΔP	0 - 3 × 10 ⁶ lb/hr @ 2500 psia	±1%	1	0 - 500	H ₂ O in/air out	FW flow control, heat balance.
Differential pressure	FW control valve (1)	Bellows ΔP	0 - 50 psid	±0.5%	1	0 - 500	H ₂ O in/air out	FW flow control.
Megawatt demand	From grid auto. dispatch system (1)	--	0 - 1200 MWe	--	--	0 - 150	Air	Steam throttle valve control, monitoring.
Megawatt generation	Generator main bus (1)	Thermal convertor	0 - 1200 MWe	±0.5%	1	0 - 150	Air	Steam throttle valve control, monitoring.
Generator frequency	Generator (1)	Frequency meter	55 - 65 Hz	±0.1%	1	0 - 150	Air	Steam throttle valve control, monitoring.
FW throttle valve position	FW throttle valve (3)	Stroke displacement	--	±0.5%	1	0 - 150	Air	Flow control, monitoring.
For Monitoring & Evaluation								
Hydrogen detector	SG and RH gas (6)	Gas Chromatograph	0 - 25 ppm	--	--	0 - 1100	Argon	
Smoke detector	Containment (several)	Pyrotechnics or equivalent	--	--	--	0 - 150	Air	
Shield temp	Neutron shields (several)	Chromel-alumel thermocouple	300 - 1500 F	±0.75%	10	0 - 1500	Inert gas	
Cold trap temp	Cold traps (6)	Chromel-alumel thermocouple	300 - 1100 F	±0.75%	10	0 - 1100	Inert gas or air	
Containment pressure	Containment (several)	Bellows	0 - 100 psia	±1%	1	0 - 150	Air	
Containment temp	Containment (several)	Bimetallic strip	0 - 200 F	±2 F	10	0 - 200	Air	
Primary Na temp	IHX outlet (6)	Chromel-alumel thermocouple	300 - 900 F	±0.375%	10	0 - 1100	Inert gas	Heat balance.
Primary Na flow rate	Primary pipe elbow (6)	Remote volumetric capillary ΔP	0 - 10 psid	±0.5%	30	0 - 900	NaK in/Ar out	Heat balance, calibrate core ΔP inst.
Primary Na level	Core vessel (1)	Inductance in well	0 - 30 ft	±1 in.	1	0 - 1100	Inert gas	
Primary Na level	Reactor vessel (1)	Inductance in well	0 - 30 ft	±1 in.	1	0 - 900	Inert gas	
Primary pump head	Primary pumps (6)	Remote volumetric capillary ΔP	0 - 50 psid	±0.5%	30	0 - 900	NaK in/Ar out	Initial flow calibration.
Primary Na pressure	Primary pump outlets (6)	Remote volumetric capillary	0 - 100 psia	±0.5%	30	0 - 900	NaK in/Ar out	
Primary Na pressure	Core inlet plenum (1)	Remote volumetric capillary	0 - 100 psia	±0.5%	30	0 - 900	NaK in/Ar out	

Table 12-5. (Cont'd)

Parameter	Location and quantity	Detector type	Operating range	Accuracy	Response time, sec	Detector environment		Use
						Temp. F	Material	
For Monitoring & Evaluation (Cont'd)								
Cover gas pressure	Core vessel (1)	Bellows	0 - 100 psia	±1%	1	0 - 900	Argon	
Cover gas pressure	Reactor vessel (1)	Bellows	0 - 100 psia	±1%	1	0 - 900	Argon	
Inter. Na temp	IHX inlet (6)	Chromel-alumel thermocouple	300 - 800	±0.375%	10	0 - 900	Inert gas	Heat balance.
Inter. Na temp	SG inlet (3)	Chromel-alumel thermocouple	300 - 1100	±0.375%	10	0 - 1100	Air	Heat balance.
Inter. Na temp	RH inlet (3)	Chromel-alumel thermocouple	300 - 1100	±0.375%	10	0 - 1100	Air	Heat balance.
Inter. Na temp	RH outlet (3)	Chromel-alumel thermocouple	300 - 900	±0.375%	10	0 - 900	Air	Heat balance.
Inter. Na temp	Pump suction line (3)	Chromel-alumel thermocouple	300 - 700	±0.375%	10	0 - 800	Air	
Inter. Na flow rate	IHX outlet pipes (6)	P.M. flowmeter, 22-in.	0 - 20 × 10 ⁶ lb/hr	±2%	1	0 - 1100	Int Na in/Ar out	Heat balance.
Inter. Na flow rate	SG outlet (3)	P.M. flowmeter, 32-in.	0 - 25 × 10 ⁶ lb/hr	±2%	1	0 - 700	Int Na in/air out	Heat balance.
Inter. Na flow rate	RH outlet (3)	P.M. flowmeter, 26-in.	0 - 10 × 10 ⁶ lb/hr	±2%	1	0 - 900	Int Na in/air out	Heat balance.
Inter. Na flow rate	By-pass (3)	P.M. flowmeter, 14-in.	0 - 25 × 10 ⁶ lb/hr	±2%	1	0 - 1100	Int Na in/air out	Heat balance.
Inter. Na level	Pump sump (3)	Inductance in well	0 - 5 ft	±1 in.	1	0 - 800	Air	
Inter. Na level	SG (3)	Inductance in well	0 - 5 ft	±1 in.	1	0 - 1100	Air	
Inter. Na level	RH (3)	Inductance in well	0 - 5 ft	±1 in.	1	0 - 1100	Air	
Inter. pump head	Inter. pump (3)	Bellows ΔP	0 - 100 psid	±0.5%	1	0 - 800	Int Na in/air out	Heat balance.
Inter. Na pressure	Pump outlet (3)	Bourdon tube or bellows	0 - 150 psia	±0.5%	1	0 - 800	Int Na in/air out	
Inter. Na leakage	In-containment pipe can (several)	"Spark-plug" type	--	--	--	0 - 1100	Argon	
Rupture disc integrity	SG and RH (6)	Switch	--	--	1	0 - 1100	Argon	
Valve position	All valves in heat transfer paths (several)	Stroke displacement	Varies	±1%	1	Up to 900	Ar & air	
Main steam flow	Main steam lines (3)	Flow nozzle & bellows ΔP	0 - 3 × 10 ⁶ lb/hr @ 2500 psi	±0.5%	1	0 - 1000	Steam in/air out	Heat balance.
Steam flow	RH inlet (3)	Flow nozzle & bellows ΔP	0 - 3 × 10 ⁶ lb/hr @ 2500 psi	±0.5%	1	0 - 700	Steam in/air out	Heat balance.
Steam temp	RH inlet (3)	Resistance temp detector	0 - 900 F	±0.5%	5	0 - 1000	Air	Heat balance.
Steam pressure	H-P turbine outlet (1)	Bourdon tube or bellows	0 - 700 psia	±0.5%	1	0 - 700	Steam in/air out	
(Numerous additional conventional measurements in condenser and elsewhere in steam plant)								
Sodium Purity Measurement								
Primary Na purity	Inlet, pump disch pipes; outlet, reactor vessel (6)	Plugging meter, electrochemical oxygen meter, carbon & hydrogen diffusion meters, sampling provisions for off-line analysis	--	--	--	0 - 900	Primary Na in/Ar out	
Inter. Na purity	Cold leg (3)	Plugging meter & sampling	--	--	--	0 - 900	Int Na in/air out	
Emergency Cooling								
Main NaK flow	Pump outlet (2)	P.M. flowmeter, 6-in.	0 - 2 × 10 ⁶ lb/hr	±1%	1	0 - 900	NaK in/air out	
NaK level	Surge tank (2)	Inductance in well	0 - 5 ft	±1 in.	1	0 - 900	Air	
NaK purity	Mixed NaK cold leg (2)	Plugging meter & sampling	--	--	--	0 - 900	NaK in/air out	
NaK pressure	Pump outlet (2)	Bourdon tube or bellows	0 - 150 psia	±0.5%	1	0 - 900	NaK in/air out	
NaK temp	Na HX inlet (2)	Chromel-alumel thermocouple	100 - 900 F	±0.375%	10	0 - 900	Air	
NaK temp	Na HX outlet (2)	Chromel-alumel thermocouple	100 - 900 F	±0.375%	10	0 - 900	Air	
NaK temp	Air HX inlet (2)	Chromel-alumel thermocouple	100 - 900 F	±0.375%	10	0 - 900	Air	
NaK temp	Air HX outlet (2)	Chromel-alumel thermocouple	100 - 900 F	±0.375%	10	0 - 900	Air	
NaK temp	Mixed NaK cold leg (2)	Chromel-alumel thermocouple	100 - 900 F	±0.375%	10	0 - 900	Air	
Valve position	All valves (several)	Stroke displacement	Varies	1%	1	0 - 900	Air	
NaK level	Storage tank (1)	Inductance in well	0 - 3 ft	1 in.	1	0 - 900	Air	

Table 12-6. Detailed Design Requirements for the
Failed-Fuel Monitor

Failed Fuel Monitor

Temperature, F	0 to 900
Pressure, psia	10 to 100
Neutron flux	Negligible background
Gamma dose rate, R/hr (shield as required)	10^4
Environment	Air or inert gas
Minimum useful life, year	5 (10 desired)
Detector replacement	At shutdown
Operating range, c/sec	0.1 to 1000

Table 12-7. Detailed Design Requirements for the
Radiation Monitoring System

	<u>In vessel</u>	<u>Out of vessel</u>
Temperature (measured), F	0 - 1200	0 - 1200
Temperature (non-thermocouple), F	0 - 900	0 - 200
Neutron flux (all energies), nv	0 - 10^4	0 - 100
Gamma dose rate (withstand), R/hr	10^4	Varies up to 10^3
Gamma dose rate (measured), R/hr	--	Varies up to 10^3
Pressure, psia	10 - 100	10 - 100
Useful life, year	10	30
Detector replacement	At shutdown	Any time

12.5.3.5. Detailed Description

Process Instrumentation — Table 12-5 lists the process instrumentation and indicates the uses of the various measurements in addition to data logging. Numerous alarms are initiated by the process instrumentation; a partial list of these alarms is given in the protection system description (section 12.3.2.2). Each instrument employs a minimum of individual circuitry, and most signal conditioning is performed in the digital computer.

Temperature measurements for sodium in the reactor vessel and for NaK are made by chromel-alumel thermocouples installed in dry thimbles so that they are replaceable. In general, these are premium-grade thermocouples ($\pm 0.375\%$). Platinum-element resistance temperature detectors are used for sodium, water, and steam temperature measurements outside the containment; the elements are installed in wells to facilitate replacement. Gas and air temperatures are sensed by bi-metallic strip units. Miscellaneous temperature monitoring, such as in the neutron shield, is generally performed with standard grade chromel-alumel thermocouples for temperatures above 750 F and by less-expensive types for lower temperatures.

Flow measurements in the primary sodium are performed by measuring the pressure drop between selected points. Volumetric capillary sensors are used: NaK-filled capillaries extend from a remote differential pressure detector to NaK-filled capsules or bellows, which sense sodium pressure at the monitoring points. This type of detector permits the location of the transducer in an accessible region near the operating floor without requiring that its lines be filled with primary sodium (as in the case of the core differential pressure instrument, section 12.5.2). These advantages are obtained at the expense of long response time.

Intermediate sodium and NaK flow rates are measured with permanent magnet flow meters outside the reactor containment. Intermediate sodium differential pressures are measured with bellows-type transducers filled with intermediate sodium; these transducers are also placed outside the containment. Water and steam flow rates are measured with bellows-type differential pressure detectors connected across flow nozzles in the piping.

Pressure measurements for primary sodium are also made with remote transducers connected to a NaK-filled capillary system. NaK, intermediate sodium, water, steam, gas, and air pressures are measured with Bourdon tubes or bellows transducers.

Sodium and NaK levels are measured with inductance-type level detectors installed in dry wells so that they are replaceable.

Sodium leakage is detected with "spark plug" detectors like those used in the Fermi reactor. These detectors are installed in secondary containers which channel leakage to them. Additional information on sodium leakage is obtained from the instrumentation designed to detect sodium/water and sodium/air reactions and (crudely) from the sodium level detectors.

The purity of sodium and NaK is routinely monitored by plugging meters, with sampling provisions for periodic off-line analysis. On-line monitoring of the primary sodium is also performed with an electrochemical oxygen meter, a carbon diffusion meter, and a hydrogen diffusion meter. Several inputs are provided for the purity instruments to ensure that representative samples are obtained.

Failed-Fuel Monitor — It is difficult to predict either the failed-fuel detection requirements of an LMFBR to be constructed in 1980 or the instrument types that may be available at that time. The two principal criteria for conventional water reactors are related to human radiation exposure (which will be less critical in a sodium-cooled, vented-fuel reactor) and to coolant-fuel chemical reaction (which may be less critical with sodium coolant). It is assumed that failed-fuel monitoring will be required but that no sophisticated fault-location system could be justified in cost and complexity. It is further assumed that the delayed-neutron technique will remain among the fastest failure detection techniques and that its sensitivity will remain competitive.

The failed-fuel monitor is of the delayed-neutron type. Twelve small pipes divert samples of primary sodium from the IHX entrances to a holdup volume, which returns the sodium to the reactor vessel. Two pipes per IHX provide a reasonable sample of core outlet coolant. The pipes are electrically heated and insulated to prevent

plugging. The holdup volume is surrounded by thermal insulation, neutron detectors, and neutron-moderating material (canned graphite). Because of temperature limitations, it is expected that the neutron detectors will be fission counters.

The neutron detectors monitor the sodium in the holdup volume for fast neutrons resulting from the decay of delayed-neutron-emitting fission products. The coolant transit time from the core to the holdup volume is about 15 seconds, so that the principal activities monitored are the two longest-lived precursors: 22-second iodine-137, and 55-second bromine-87 (both of which decay to noble gases that promptly emit neutrons). The fission counters and their circuitry, possibly supplemented by local shielding, discriminate against background gamma radiation. Core-leakage neutrons are attenuated by the moderator built into the delayed-neutron monitor and the supplemental neutron moderator and capture material (boron).

Delayed neutrons from fissions of tramp fuel are the principal source of background counts, which limit system sensitivity. The alarm setpoint must be above this tramp level. Small pin-hole defects in the cladding would probably be undetectable unless they were very numerous. However, in general, this same limitation applies to any on-stream method of monitoring for failed fuel.

The delayed-neutron technique is selected primarily because its response time of perhaps 30 seconds is much faster than the time for any other known method. A second reason, which is particularly significant with vented fuel, is that the method is relatively insensitive to background gamma activity of any sort. It is noted that no delayed-neutron precursors should pass through the fuel vents, since the longest precursor half-life is about one minute, and the fuel pin holdup time will be several days, or some 400,000 half-lives. The delayed-neutron concept is about equally sensitive and no more complex than other known methods.

Failed fuel will be located on a coarse scale using the in-core instruments and possibly by introducing deliberate flux tilts by control rod motion. Identification of failure to a single fuel assembly will depend on out-of-core (or possibly shutdown) sniffing of the individual fuel elements, unless the defect is so serious as to void

an assembly, in which case the assembly outlet thermocouple will provide positive identification. The technique reported in reference 8 may also be developed to the point of real usefulness in locating failed fuel. This reference explores the feasibility of locating failed fuel by preferential deposition of certain fission products on nickel-bearing surfaces downstream from the failure. Conceivably either the core holddown structure or the tops of the individual fuel assemblies could be scanned in the reactor vessel with a gamma detector connected to a multi-channel analyzer. The need for such a device and the possibility of its successful development are not yet clear; it does not appear to be necessary now.

Radiation Monitor — Table 12-8 lists the radiation monitoring locations and the types of measurement performed at each. With the exception of the fuel storage rotating drum monitors and the primary and secondary sodium monitors, the radiation monitoring system is completely conventional. Detectors will be standard types: Geiger counters, ionization chambers, scintillation counters, air particle detectors, and chromel-alumel thermocouples. Several detectors are employed for each function, and both local and remote indication and alarms are provided.

The discharge stack monitor is of protection quality, since the stack must be closed if excessive radioactivity is being discharged from the plant. The containment atmosphere discharge monitor is also of protection grade and initiates containment isolation (except that emergency cooling paths are not closed) if excessive radiation is being released. The emergency cooling paths are closed by the protection system if excessive NaK activity is detected, because rupture of a NaK line could then cause a serious release of radioactivity. No similar provisions are made for liquid and solid radioactive effluents, because neither is routinely discharged on a continuous basis; in both cases discharge is effected manually and the effluent is monitored before discharge.

The instrumentation of the two fuel storage rotating drums in the reactor vessel provides the final assurance of direct measurement that the fuel is neither critical nor melting. The geometry of the storage drums is critically safe, and spent fuel is retained in the core until forced cooling is no longer necessary to remove decay heat.

Table 12-8. Radiation Monitoring System Detection Locations

Location	α - β - γ	Airborne	Neutron	Temperature
Fuel storage rotating drums			x	x
Refueling mechanism	x			x
Fuel transfer machine	x			x
Fuel shipping, receiving, transfer areas	x	x		x
Decontamination & waste storage areas, etc	x	x		
Primary Na	x			
Secondary Na	x			
NaK	x			
Cover gas	x	x		
Containment atmosphere	x	x	x	
Control room, computer room, offices	x	x		
Environs	x	x		
Discharge stack	x	x		
Discharge flume	x			
Containment atmosphere discharge	x	x		

Several chromel-alumel thermocouples in each storage drum monitor the temperature of the convectively circulating sodium coolant. To prevent their interfering with fuel handling, the thermocouples are anchored to the rotating storage racks, and the electrical connections are brought through gold-plated slip rings at the operating floor. Since the convective flow is upward, most thermocouples are placed slightly above the tops of the fuel assemblies. Two thermocouples are also mounted at the outside of each drum to monitor down-comer flow and to facilitate the calculation of fuel history during storage. The thermocouples are installed inside thimbles to permit their replacement from the operating floor during plant shutdown.

Stationary fission counters are located on opposite sides of each rotating drum (the thermocouples move with the drums). Fission counters are used, because more sensitive neutron detectors may be unable to withstand the temperature and gamma radiation levels; if suitable higher-sensitivity detectors become available, they will be used. These detectors serve as criticality monitors for the stored fuel. The detectors, mounted on the tension support members which support the reactor internals, are replaceable at shutdown.

Thermocouple operability is determined from on-scale response to the 800 F sodium temperature. Neutron-detector operability must be checked with a small, portable neutron test source, since on-scale readings are not expected.

12.5.3.6. Principles of Operation

Temperature Measurement — The operation of chromel-alumel thermocouples is described with the core instrumentation system in section 12.5.2. Platinum resistance temperature detectors employ an electrical bridge circuit to measure temperature-produced resistance changes in a coil of fine platinum wire in the sensing head. Bi-metallic strips consist of joined sheets of two metals having dissimilar thermal expansion coefficients, so that temperature variations change the shape of the strip.

Measurement of Pressure and Differential Pressure — The operation of bellows-type differential pressure detectors is described with the core instrumentation system in section 12.5.2.

A bellows-type pressure detector operates in the same way, except that one side of the bellows is maintained at a constant pressure (either sealed at the factory or exposed to the atmosphere). In a Bourdon tube pressure detector, the fluid is introduced inside a thin-walled, closed, curved tube; fluid pressure changes vary the tube curvature and move a mechanical linkage attached to the tube. In the case of volumetric capillary detectors, the transducer measures the pressure in a NaK-filled capillary tube extending from a capsule or bellows inserted into the primary sodium; the NaK pressure is determined by expansion or contraction of the bellows or capsule, which is governed by the primary sodium pressure surrounding it.

Flow Measurement — Most flow measurements are accomplished with differential pressure measurements across known flow obstructions. For feedwater and steam, flow nozzles are inserted in the piping for maximum accuracy. Intermediate sodium flows are measured by permanent magnet flow meters as described with the core instrumentation system in section 12.5.2.

Sodium and NaK Level Detection — Alternating current is applied to a long coil of wire in a dry-interior well immersed in sodium. The power dissipated in the coil is a function of magnetic coupling with the sodium, which in turn is a function of sodium level. Temperature compensation is obtained by a second coil completely immersed in sodium.

Sodium Leak Detection — The Fermi-type leak detectors consist of pairs of exposed electrodes which close electrical circuits when they are shorted by sodium.

Sodium and NaK Purity Measurement and Sampling — The plugging meter cools sampled sodium to determine the temperature at which soluble impurities precipitate. Provided that oxide is the principal impurity, the precipitation temperature is also a measure of oxide concentration. References 9 through 13 describe the more sophisticated purity instrumentation and techniques for obtaining samples for off-line analysis.

Failed-Fuel Monitor — The fission counters and circuitry operate as described with the nuclear instrument system in section 12.5.1.

Radiation Monitors — Several standard types of radiation monitors are used as required for various applications and ranges. These instruments are described in standard sources such as reference 14.

12.5.3.7. Safety Precautions

The process and radiation monitoring instruments encompass the entire plant, from inside the reactor vessel to outside the buildings. Penetrations of coolant system boundaries and of the containment require careful attention to avoid leaks. Detector installations must be designed to minimize their interference with plant operation. Lead routing paths must be carefully selected to minimize electrical noise pickup, damage due to accidents, and interference with and damage from personnel and equipment motion.

Automatic testing is provided for the more important instruments, including those used for protection or control; the remaining instruments are manually tested at intervals. Calibration means are provided as required.

12.5.3.8. Maintenance

Circuit maintenance will be straightforward; it is facilitated by effective testing provisions and modular construction. Detector replacement is simplified by design choices. Level and temperature detectors are generally installed in wells or thimbles, so that they can be replaced without contacting sodium. Pressure and differential pressure detectors are remotely located and fitted with sealed NaK systems for the same reason. Wherever possible, detectors are located outside the containment for easier access.

APPENDIX 12-A
Reactor Digital Control and Protection

Digital control systems have been applied to numerous conventional power plants in the 1960's. Nearly all existing and planned electrical-power-producing nuclear reactors employ digital data acquisition and/or processing systems. The same features that introduced digital equipment in these applications make it desirable for the control and protection of a large LMFBR plant:

1. Large quantities of data are handled sequentially in the same circuits instead of simultaneously in many circuits, reducing equipment costs.
2. Logical and arithmetic operations are performed faster, reducing delay times for many operations.
3. System faults are less likely to occur because circuit components, including transistors, must undergo very large performance changes before system performance is affected.
4. System faults are more easily detected because they are more decisive (either a component fails or it doesn't).
5. Since it can be performed automatically, system testing is easier and less damaging to the equipment and to its functions; equipment handling and the chance of human error are reduced.
6. The reliability of properly functioning on-line operation is increased by the combination of lower fault rate, more frequent testing, and reduced equipment handling.
7. Whole programs for uncommon or abnormal conditions (such as recovery from protection system action) can be stored without interfering with normal operation but are readily available when needed.

Experience with digital equipment now in use, combined with hardware and software improvements from the equipment manufacturers, will dictate whether digital computers can reasonably be used for reactor control and protection within the next several years. M. A. Schultz predicted a sweeping digital trend several years ago.¹⁵ APDA has recommended the application of a digital computer for core protection in the Fermi reactor and has identified a substantial increase in protection from its use.¹⁶ The digital computer in service at the Douglas Point

reactor,¹⁷ although designed in the early 1960's and perhaps already obsolete, will provide landmark experience.

Certainly it appears that digital equipment will be a very real factor in reactors built in 1980, where its application is justified on sound engineering and economic bases. The 1000-MWe LMFBR is considered to be such an application, particularly where individual fuel assemblies must be monitored for sodium voiding. In this reactor and very likely throughout the foreseeable future, analog equipment cannot and should not be completely supplanted, but there are many areas where it can and probably should be.

A full understanding of the digital computer system for the B&W reference plant requires further in-depth design work and evaluation. Some of the major areas that require further work are listed below.

1. More detailed component identification, which will permit reliability calculations based on mean time-before-failure data or estimates.
2. Analysis of the basic structure to determine, for example, whether a design with two large main frames is preferable to one with three large units or many small units. The factors to be considered in this analysis include the effect of a main frame failure, possible interactions of input or output devices, and maintenance provisions.
3. Detailed identification of automatic testing provisions, which will permit an accurate determination of the extent to which manual testing is required (if any).
4. Better identification of administrative controls and the internal priority structure.

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Section 13

ELECTRICAL SYSTEM

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13. ELECTRICAL SYSTEM

13.1. Introduction

13.1.1. System Function

The electrical system provides the total electrical requirements for the plant under all conditions of operation and shutdown. The emergency power supplies provide sufficient power to ensure a safe reactor shutdown and to maintain the plant in that condition in the event of a total loss of normal power.

In addition, the system provides adequate lighting levels to match each working area need and provides communications throughout the plant and site to ensure a high degree of coordination and communication between various locations, to alert personnel to any unusual conditions, and to monitor operations and equipment.

13.1.2. System Summary Description

The electrical auxiliary system is shown in Figure 13-1. The normal station auxiliary power of 25,000 kW is supplied by a unit auxiliary transformer with a double secondary winding. The transformer steps down the voltage from 25,000 to 4160 volts and is directly connected to the generator bus duct through an isolated phase bus duct. This is a delta-*W*-ye transformer in which the neutrals of the 4160-volt windings are grounded through neutral grounding resistors.

A reserve auxiliary transformer with a double secondary winding is available for startup or for use if the normal auxiliary power source should fail. The alternate source of power can replace the unit auxiliary transformer during maintenance. This unit is fed from the station switchyard, which in turn receives its power from the main station generator or the utility grid system itself. The reserve auxiliary unit steps down the voltage from 345 to 4.16 kV. It has four windings containing an internally connected tertiary winding, and the neutrals of its 4160 volt windings are grounded through neutral grounding resistors.

The 4160-volt station switchgear can be fed from either the unit auxiliary transformer or alternately from the reserve auxiliary transformer. A 3750-kVA diesel generator is connected to 4160-volt switchgear 1, and another 100% standby 3750-kVA diesel generator is connected to 4160-volt switchgear 3 to supply energy for the essential loads required for a safe shutdown.

The 480-volt unit substations are fed from the 4160-volt switchgear via 1000-kVA transformers. The motor control centers are connected to the 480-volt substations.

Three separate 250-volt d-c buses supply power for the noninterruptible d-c motor loads. Each bus is connected to an associated lead-acid battery of approximately 250 volts. Static chargers, one for each d-c system, maintain the proper charge on the batteries. The chargers are capable of concurrently supplying the normal d-c load of the 250-volt d-c systems. Two normally open bus tie circuits between d-c buses are provided for maintenance and service. Also, a bus disconnect feature permits an equalizing charge cycle on a battery while the d-c load on the associated bus is energized via the bus tie circuits.

Two separate noninterruptible, 125-volt d-c buses are each connected to an associated lead-acid battery of about 125 volts. A separate static charger for each d-c system is utilized to maintain the proper charge on the associated battery; each unit is capable of concurrently supplying the normal load of both d-c systems. A normally open bus tie circuit between the two d-c buses is provided for maintenance and service. Also, a bus disconnect feature permits an equalizing charge cycle on a battery while the d-c load on the associated bus is energized via the bus tie circuit.

There are four separate, independent instrument power buses, three of which (No. 1, 3, and 4) provide "noninterruptible" power for essential reactor plant instrumentation.* These buses are supplied

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*Figure 13-1 shows only two noninterruptible buses. A third bus is to be added by connecting a charger, battery, and inverter to 480-volt bus 5. Unit charger, battery, and inverter capacities remain the same, providing 50% spare capacity. The noninterruptible buses will then be tied to each other through normally-open circuit breakers. This change is not reflected in the cost estimates, but it is expected to amount to less than \$100,000.

Figure 13-1. Electrical Diagram for the 1000-MWe LMFBF

from the d-c buses. Provision for a manual transfer is included to maintain continuity of power in the event that any inverter set is taken out of service. The fourth instrument power bus (No. 2) is supplied by a 480 volt to 120/208 volt, 3-phase, 60-cycle transformer connected to the station's 480-volt substations 2 and 4 by a transfer switch.

One 120-208 volt, 3-phase bus is provided for the 25 regulating rod motors. This bus receives the power from either of the 125-volt d-c buses through two 3-phase static inverters and a static transfer switch.

The communication system contains a combination public address and intercommunication system, together with tone oscillation equipment, to sound emergency alarms over the paging system. Provision is included for the installation and connection of telephone sets to the local telephone company's system.

13.1.3. System Design Requirements

13.1.3.1. Performance

Two independent transmission lines provide power to the complex, and two independent sources of emergency power, each capable of supplying all emergency needs, will be available.

The 4160-volt switchgear 1 and 2 are fed from one secondary winding of the auxiliary transformer. One secondary winding of the reserve auxiliary transformer will also be connected to switchgear 1 and 2. Switchgear 3 and 4 are connected to the second 4160-volt winding of each auxiliary transformer. Provisions have been made for supplying electric heat to the sodium pipework.

13.1.3.2. Environment

Electrical equipment will be mounted outside of radiation or high-temperature zones wherever possible. Any equipment mounted within a radiation or high-temperature zone will be designed for easy accessibility and rapid maintenance or replacement. All control equipment that is vital to reactor operations will be installed in uncontaminated areas if possible.

13.1.3.3. Reliability/Availability

Electrical equipment will be of high commercial quality selected for good reliability and long life.

13.1.3.4. Maintenance

Quick-disconnect or plug-in connections will be used wherever maintenance may be frequent or difficult, especially in contamination zones.

13.1.3.5. Operation

The 4160-volt switchgear will automatically be transferred to the reserve auxiliary transformer if bus voltage is lost. In normal operation, the tie breakers between the 4160-volt switchgear buses will be open.

13.1.3.6. Safety

The emergency power supplies will be sized to provide enough power to ensure a safe reactor shutdown and maintain that condition in the event of a total loss of normal power. The emergency supplies will provide enough lighting in the control room to conduct operations and permit personnel access. It will also supply power for sensors, power for adequate coolant flow in the primary system, power for the decay heat removal system, and power for all other equipment required to ensure the safety of operating personnel, plant equipment, and the general public. The communication system will function during normal power-outage emergencies.

13.1.3.7. Design

The system will be designed in accordance with the criteria of Volume 1, section 2 of this report, including the codes, standards, and regulations cited therein.

13.1.3.8. Interface

The voltage, current, phasing, and interruptibility requirements of the various loads, which are established by numerous other systems, will be satisfied by the electrical system. The requirements imposed by interfaces with the protection system and with the engineered safeguards area are particularly stringent and detailed. Both of these areas affect plant safety.

13.2. Detailed System Description

13.2.1. General

The electrical system is shown in Figure 13-1, and electrical loads and functions are listed in Table 13-1. The system is described in detail in the following paragraphs.

13.2.2. Components

13.2.2.1. Unit Auxiliary Transformer

This transformer has a voltage rating of 25-4.16 kV; it is a 3-phase, 3-winding (delta-wye-wye), 60-Hz, outdoor, step-down, oil-immersed, 25/33-1/3 MVA, Type OA/FA unit. The no-load taps on the high-voltage winding are 2-1/2% above the rated voltage and 2-1/2% below the rated voltage; there are two taps of each type. Each low-voltage winding is rated at 12.5/16.7 MVA, and its neutral is grounded through a grounding resistor.

13.2.2.2. Reserve Auxiliary Transformer

This transformer has a voltage rating of 345-4.16 kV; it is a 3-phase, 4-winding (wye high voltage, delta tertiary and two wye low voltage), 60-Hz, outdoor, step-down, oil-immersed 25/33-1/3 MVA, Type OA/FA unit. Two no-load taps on the high voltage winding are 2-1/2% above the rated voltage and two are 2-1/2% below the rated voltage. Each low-voltage winding is rated at 12.5/16.7 MVA, and its neutral is grounded through a grounding resistor.

13.2.2.3. 4160-Volt Switchgear

The 4160-volt system has four indoor-type, metal-clad, 4160-volt switchgear units (see Figures 13-2 and 13-3). Each unit contains electrically operated air circuit breakers rated at 250 MVA interrupting capacity. The main and reserve feed breakers are rated at 350 MVA. These switchgear units are installed indoors, near the unit and reserve auxiliary transformer. A nonsegregated phase bus duct connects the low-voltage windings of the unit auxiliary transformer and the reserve auxiliary transformer to the incoming air circuit breaker of the appropriate switchgear unit. The feeder circuit breakers supply the 480-volt unit substations and motors with ratings of 250 hp and above.

The 4160-volt switchgear bus 2 is connected to 4160-volt switchgear buses 1 and 3 by bus tie circuit breakers which are normally open. All of the air circuit breakers are of the drawout type. The control voltage for the 4160-volt switchgear is 125 volt d-c.

The principal components receiving power from the 4160-volt switchgear are listed in Table 13-1 and shown in Figures 13-1 and 13-2.

13.2.2.4. 480-Volt Unit Substations

The auxiliary system consists of six metal-clad, indoor-type, 480-volt unit substations (see Figures 13-4, 13-5, and 13-6). Each substation includes a high-voltage air terminal chamber and a step-down transformer. The 5000-volt air terminal chamber is used for the termination of the incoming power cables and is connected directly to the high-voltage (4160-volt) side of the transformer.

The six substation transformers are indoor, oil-immersed, Type OA, and are rated at 1000 kVA, 4160-480Y/277 volts. All six transformers have four 2-1/2% full capacity taps in the high voltage winding—two above and two below the 4160-volt rated voltage. The low-voltage (480 volt) side of each transformer is connected to a main air circuit breaker in each unit to supply the low-voltage power switchgear. The 480-volt substations are connected by normally-open bus tie circuit breakers (not shown in Figure 13-1) in the same manner as the 4160-volt bus ties are connected.

Auxiliary equipment, such as potential transformers and undervoltage relays, is contained in auxiliary compartments. Each of the 480-volt unit substations is connected to a 4160-volt switchgear via its transformer. The 480-volt substations will power motors rated between 50 and 250 hp and other loads as shown in Table 13-1 and Figures 13-4, 13-5, and 13-6.

Table 13-1. Electrical Load List
and Functions

Switchgear	Load		Running loads (1)	Standby loads (2)	Essential loads (3)	Standby essential loads (4)	Shutdown non- essential loads (5)	Shutdown essential loads (6)	Standby shutdown essential loads (7)	Maintenance loads (8)	Full time nonessential loads (9)
4160 V 1	Circ water pump	# 1	1750 hp	x							
	Circ water pump	# 2	1750 hp	x							
	Cond pump	# 1	800 hp	x							
	Blr feed pump, 20% *		6000 hp				x				
	Service water pump	# 1	660 hp	x	x						
	Centrifugal chiller	# 1	300 hp	x	x						
	Pri pump	# 1	1175 hp	x							
4160 V 2	Inter pump	# 1	3200 hp	x							
	Service water pump	# 2	660 hp	x		x					
	Centrifugal chiller	# 2	300 hp	x	x						
	Centrifugal chiller	# 3	300 hp			x					
	Pri pump	# 2	1175 hp	x							
	Pri pump	# 3	1175 hp	x							
4160 V 3	Inter pump	# 2	3200 hp	x							
	Circ water pump	# 3	1750 hp	x							
	Service water pump	# 3	660 hp			x					
	Cond pump	# 2	800 hp								
	Centrifugal chiller	# 4	300 hp								
	Pri pump	# 4	1175 hp	x							
4160 V 4	Inter pump	# 3	3200 hp	x							
	Cond pump	# 3	800 hp	x							
	Centrifugal chiller	# 5	300 hp	x							
	Pri pump	# 5	1175 hp	x							
	Pri pump	# 6	1175 hp	x							

1. Running Loads — Loads that operate when plant is operating at normal condition.
2. Standby Loads — Loads that do not run at any time, but can replace a running load.
3. Essential Loads — Loads that may run at all times and are essential to safe operation or shutdown of plant.
4. Standby Essential Loads — Loads that do not run at any time, but can replace an essential load. They are essential to safe operation or shutdown of plant.
5. Shutdown Nonessential Loads — Loads that run only at low power or shutdown, but are not essential for safe shutdown of plant.
6. Shutdown Essential Loads — Loads that run only at low power or shutdown and are essential for safe shutdown of plant.
7. Standby Shutdown Essential Loads — Loads that do not run at any time, but will replace a shutdown essential load. They are essential to safe shutdown of plant.
8. Maintenance Loads — Loads that run only during maintenance which can be done when plant is operating or shut down. They are not essential for safe operation or shutdown of plant.
9. Full Time Nonessential Loads — Loads that may run at all times, but are not essential for safe operation or shutdown of plant.

*Since the boiler feed pump requirement is larger than originally allowed, redistribution of the switchgear units is required.

Table 13-1. (Cont'd)

Switchgear	Load			Running loads (1)	Standby loads (2)	Essential loads (3)	Standby essential loads (4)	Shutdown non-essential loads (5)	Shutdown essential loads (6)	Standby shutdown essential loads (7)	Maintenance loads (8)	Full time nonessential loads (9)
480 V 1	Inst air compr	# 1	100 hp	x		x						
	H ₂ seal oil pump	# 1	75 hp	x		x						
	Containment cooler fan	# 1	62 hp	x		x						
	Decay HX CWP	# 1	65 hp						x			
	NaK-air ECC fan	# 1 + 5	200 hp							x		
	MCC	# 1	200 hp			x						
	Building lighting	# 1	150 kW									x
	Battery charger	# 1	45 kW	x		x						
480 V 2	EHC pump	# 1	150 hp	x		x						
	Decay HX cond pump	# 1	55 hp						x			
	NaK-air ECC fan	# 2 + 6	200 hp							x		
	ECC EM pump	# 1 + 3	80 hp							x		
	MCC	# 2	150 hp						x			
	MCC	# 3	150 hp	x								
	Battery charger	# 3	76.6 kW	x		x						
	Instr trans	(1)	100 kVA	x		x						
480 V 3	Instr air compr	# 2	100 hp		x		x					
	H ₂ seal oil pump	# 2	75 hp		x		x					
	Containment cooler fan	# 3	62 hp	x		x						
	Decay HX CWP	# 2	65 hp							x		
	NaK-air ECC fan	# 3 + 7	200 hp							x		
	NaK-air ECC fan	# 4 + 8	200 hp							x		
	Building lighting	# 2	150 kW									x
	Battery charger	# 2	45 kW	x		x						
480 V 4	Battery charger	# 5	76.6 kW	x		x						
	EHC pump	# 2	150 hp		x		x					
	Decay HX cond pump	# 2	55 hp							x		
	MCC	# 4	200 hp						x			
	ECC EM pump	# 2 + 4	80 hp							x		
	MCC	# 5	150 hp			x						
	MCC	# 6	150 hp	x								
	Instrument trans (spare feeder only)											
480 V 5	Battery charger	# 4	76.6 kW	x		x						
	Gen stator cool. pump		100 hp	x								
	Building lighting	# 3	150 kW									x
	Containment cool. fan	# 2	62 hp		x		x					
	Decay HX CWP	# 3	65 hp						x			
	ECC EM pump	# 3	80 hp							x		
	Station air compr		100 hp									x
	Turb rm 200-ton crane		75/40/20/5 hp								x	
480 V 6	SG rm 100-ton crane		60/30/10/5 hp								x	
	Decon rm 75-ton crane		40/30/7.5/5 hp								x	
	Refueling rm 75-ton crane		40/30/7.5/5 hp								x	
	Turb rm exh fan	# 1	75 hp	x								x
	MCC	# 7	140 hp									x
	MCC	# 8	140 hp								x	
	Heater drain pump	# 1	70 hp	x								
480 V 6	Outdoor lighting	(4)	100 kW									x
	Containment cool. fan	# 4	62 hp				x					
	Heater drain pump	# 2	70 hp	x								
	Heater drain pump	# 3	70 hp	x								
	Fire pump		150 hp			x						
	200-ton gantry crane		10/15 hp								x	
	75-ton polar crane		40/30/10/5 hp								x	
	Turb rm exh fan	# 2	75 hp	x								x
480 V 6	Turb rm exh fan	# 3	75 hp		x							
	MCC	# 9	200 hp	x								
	MCC	# 10	200 hp				x					

Figure 13-2. Key Diagram for 4160-Volt
Switchgear 1 and 2

Figure 13-3. Key Diagram for 4160-Volt
Switchgear 3 and 4

Figure 13-4. Key Diagram for 480-Volt Unit
Substations 1 and 2

Figure 13-5. Key Diagram for 480-Volt Unit
Substations 5 and 3

Figure 13-6. Key Diagram for 480-Volt Unit
Substations 4 and 6

13.2.2.5. Motor Control Centers (MCC)

The system has 10 motor control centers for the reactor plant, turbine plant, water treating, and machine shop. Combination starters are of the drawout type, size 1 (NEMA) through size 3, with auxiliary contacts. The combination magnetic starter has a molded case circuit breaker and contains its own 480-115 volt control transformer. The circuit breaker serves as the disconnecting switch and provides short circuit protection when actuated by the breaker trip element. The motor control centers are fed from the 480-volt unit substations. Motors below 50 hp are connected to the motor control centers.

13.2.2.6. 120/208 Volt, 3-Phase Control Rod Bus

This bus supplies the a-c power for the 25 control rods which require 1/2 horsepower each. Under normal operating conditions, the a-c load is fed through the static transfer switch from the inverter, which is connected to 125-volt d-c bus 1. When a loss of power from this inverter occurs, the static transfer switch transfers the load to the standby inverter which is connected to 125-volt d-c bus 2. The static transfer switch operates in two cycles or less.

13.2.2.7. 125-Volt d-c Buses

Two 125-volt batteries (No. 1 and No. 2) are provided. Solid-state battery chargers normally supply the d-c loads and maintain the charge on their batteries. Battery charger 1 is connected to 480-volt substation 1 and battery charger 2 is connected to 480-volt substation 3.

The d-c distribution cabinet is used for the distribution of 125-volt d-c control power for the 4160-volt switchgear, the 480-volt substation, and the main control room panels. The d-c buses (No. 1 and No. 2) provide power for three instrument power inverters which provide noninterruptible power.

13.2.2.8. 120-Volt Instrument Power Buses

Four separate instrument buses are provided to supply the power requirements for all plant instrumentation.

Instrument buses 1, 3, and 4* are noninterruptible and are rated at 120-volt ac, single-phase, 60-Hz. The three noninterruptible buses are supplied from three instrument power inverters. Buses 1 and 4 are supplied from 125-volt d-c bus 1, and bus 3 is supplied from 125-volt d-c bus 2. Instrument bus 2 is 120/208 volt ac, 3-phase, 60-Hz and is supplied from the 480-120/208 volt, 3-phase, 60-Hz instrument power transformer connected to 480-volt substations 2 and 4.

If an inverter fails, the affected noninterruptible bus can be manually transferred to instrument bus 2. Each of the instrument buses will be physically separated and protected from the others. All power, control, and instrumentation wiring from each instrument bus will be physically separated and protected from the power, control, and instrumentation wiring of the other instrument buses.

The protection system, emergency lighting system, and certain other vital loads receive power from the three non-interruptible buses. The three buses provide separate sources for triply-redundant protection channels; thus a single inverter failure will not force a reactor scram.

13.2.2.9. Cables

Triplexed cable rated at 5000 volts is used for 4160-volt power, and 480-volt power is fed by 600-volt triplexed cables which have butyl rubber or cross-linked polyethylene insulation and polyvinyl chloride jackets. The cable used for control is rated at 600 volts, and has cross-linked polyethylene insulation and a polyvinyl chloride jacket. Size 12 AWG is minimum for control and size 10 AWG is minimum for current or potential transformer wiring.

The switchboard wiring is rated at 600 volts; No. 12 AWG is used for current circuits and No. 14 AWG for all other

- - - - -
* Figure 13-1 shows only two noninterruptible buses. A third bus is to be added by connecting a charger, battery, and inverter to 480-volt bus 5. Unit charger, battery, and inverter capacities remain the same, providing 50% spare capacity. The noninterruptible buses will then be tied to each other through normally-open circuit breakers. This change is not reflected in the cost estimates, but it is expected to amount to less than \$100,000.

circuits. The instrumentation cable for the reactor plant protection system will have different colored insulation jackets for each channel in order to simplify the required physical separation.

13.2.2.10. Cable Trays

Cable trays are provided for three distinct uses and are constructed as noted below:

1. Tray No. 1 — Solid metal tray with a solid snap-on-off metal cover. This tray will contain cables carrying low-level signals only. Examples include thermocouples, process instrumentation measurement and control signals, and analytical instrument signals. Wiring will be twisted pairs and shielded. Nuclear and radiation monitoring instrument cables may also be run in this tray. This includes both the high-voltage and signal circuits in coaxial and triaxial cables.

2. Tray No. 2 — Expanded metal tray that will contain cables carrying instrument power and on-off instrument and control circuits. Examples include regulated or unregulated a-c instrument power, on-off (switched) circuits for actuation of alarms, indicating lights, relays, solenoid valves, etc. These on-off circuits are limited to 5 amperes at 115 volt ac or 125 volt dc.

3. Tray No. 3 — Expanded metal tray that will contain cables carrying power and associated control circuits. Examples include 480-volt, 3-phase power and associated control circuits (either ac or dc), potential transformer and current transformer leads.

13.2.2.11. Motors

Most of the motors for indoor application will have open drip-proof frames, but those in the turbine lube oil room, the motor for the d-c emergency H₂ seal oil pump, and the motor for the d-c emergency bearing oil pump will have TEFC enclosures. Motors for outdoor use will have weather protected NEMA Type II or TEFC enclosures.

13.2.2.12. Main Generator Auxiliaries

The main generator isolated phase bus is forced-cooled and rated at 25,000 amperes. The surge protection equipment, consisting of station-type capacitors and lightning arresters, is connected to the isolated phase bus by a bus tap.

The generator neutral grounding equipment consists of a distribution-type transformer, 75 kVA, 25,000 to 240/120 volts, with a resistor connected across its secondary winding. This transformer's primary winding is connected by bus to the three neutral bushings of the main generator. The other end of the primary winding is connected to the station's grounding system.

13.2.3. Grounding

A low-resistance copper grounding system is installed throughout the plant, both indoors and outdoors, in accordance with the National Electrical Safety Code.

13.2.4. Controls, Alarms, and Protective Devices

The electrical system is controlled and monitored from the control room. Control power for the 4160-volt switchgear and 480-volt unit substations is furnished by the d-c system. Differential relaying protection is provided for the turbine-generator, the main power transformer, the reserve auxiliary transformer, and the unit auxiliary transformer. In addition, the generator is protected by loss of field, negative sequence current, and neutral ground overcurrent relays. All breaker trips, overcurrent, or undervoltage conditions will be alarmed on window-type annunciators in the control room. Selected bearing temperatures, transformer temperature and pressure, and turbine-generator pressure and temperature will be continuously monitored. Automatic control is provided where necessary. Ground overcurrent relay protection is provided for all 4000-volt motors.

13.2.5. Diesel Generators

In the event of failure of the a-c mains, two diesel generators are arranged for automatic starting to restore the supply. Each standby generator is rated at 3000 kW, 480-volts, 3-phase, 60-Hz, 0.8 p.f. The generators provide guaranteed backup power.

13.2.6. Lighting System

13.2.6.1. General

The lighting system consists of 277-volt mercury and fluorescent fixtures fed from the 480/277 volt buses, and 120-volt fixtures fed by transformers. Lighting transformers serve groups of incandescent lights and receptacles at 120 volts. Emergency lighting is serviced from the 120-volt battery and from the emergency diesel generator unit which supplies 3-phase, 480/277 volt.

General lighting is provided by mercury fixtures and metallic-vapor and Lucalox ceramic arc tube light fixtures with increased lamp lumen efficiencies. Lighting for special applications is provided by high-intensity quartz line lamps and 800 mA high-output and 1500 mA very-high-output fluorescent lamps. The lighting intensities and types of fixtures required for various locations are shown in Table 13-2.

Table 13-2. Plant Lighting Intensities and Fixtures

<u>Location</u>	<u>Footcandle intensity</u>	<u>Lighting type</u>
Turbine room main floor	30	Lucalox high bay
Turbine room mezzanine	20	Fluorescent industrial
Turbine room ground floor	20-30	Fluorescent industrial
Water & chemical treating	20-30	Mercury vaportight
Auxiliary switch rooms	25	Fluorescent industrial
Locker & toilet rooms	20	Fluorescent shielded
Offices	120	Fluorescent shielded
Control room	100	Fluorescent louverall
Reactor bldg main floor	30	Lucalox high bay
Decontam & maint bay	20-30	Fluorescent industrial
Fuel handling bldg	20-30	Mercury vaportight
Reactor vessel main floor	30	Lucalox high bay
Laboratory & shops	100	Fluorescent shielded
Reactor tank rooms	20	Fluorescent industrial
Reactor bldg ground floor	20	Fluorescent industrial
Shipping & storage area	25	Fluorescent industrial
Reactor vessel lower floors	20	Fluorescent industrial
Machine shop	50	Lucalox high bay
Battery rooms	20	Fluorescent vaportight

Emergency lighting is provided for low-level illumination in halls, stairwells, control room, and reactor and turbine areas to enable the movement of personnel in the plant. Additional emergency lighting in the reactor enclosure and in strategic areas such as stairways or air locks is provided by automatic battery lanterns.

Lighting cabinets containing branch circuit breakers are strategically placed for distribution of the plant lighting. Lighting fixtures are installed in accessible areas for convenience in maintenance and relamping. High-bay fixtures are maintained from traveling cranes. Lighting fixtures are installed at all instruments and gauges requiring special illumination. All through-vision gauges are illuminated for their full lengths by gauge light fixtures placed behind the gauge glasses.

13.2.6.2. Fixtures

Mercury, metallic-vapor, and Lucalox fixtures are of the integral ballasted type. Fluorescent industrial fixtures provide approximately 20% of the upward light component. Industrial fixtures are used throughout ground floor, mezzanine, and gallery areas where fixture-mounting heights are over 9 feet.

The fluorescent strip units have oversized-channel housings with spring lamp holders; they are used throughout ground-floor, mezzanine, and gallery areas where fixture-mounting heights are 9 feet or less. Commercial fluorescent diffuser luminaires, both pendant- and surface-mounted, have acrylic plastic enclosures, and fluorescent troffer luminaires have acrylic plastic lens bottoms.

All ballasts for fluorescent and mercury lamps are high-power-factor units, internally protected by a thermally activated, automatic reclosing device which is sensitive to both winding temperature and current. Fluorescent ballasts are the CBM/ETL type, and mercury ballasts are the constant-wattage type. Color-improved mercury lamps and cool-white fluorescent lamps are used.

13.2.7. Communications

The communications system includes a combination public address and intercommunications system serving both the turbine plant and the reactor plant. It consists of telephone-type handset stations and

transistorized amplifier loudspeaker assemblies. The system has two channels, one for paging and one for party-line service. The two channels may be used simultaneously without interference. Tone oscillator equipment is provided to sound emergency alarms over the paging system. These alarm signals are controlled from the main control room. Provisions are included for the installation and connection of suitably placed telephone sets to the local telephone company system. Sound-powered telephone jacks are provided at the main control boards and at strategic positions in the plant to facilitate plant servicing and operation.

13.3. Principles of Operation

13.3.1. Startup and Normal Operation

Initial plant startup uses auxiliary power furnished by the reserve auxiliary transformer, which is connected on its high-voltage side to the 345-kV switchyard and on its low-voltage side through normally-open air circuit breakers to 4160-volt switchgear buses 1-2 and 3-4.

After the main turbine-generator has been brought up to speed and synchronized with the rest of the system, the unit auxiliary transformer will be placed in service by closing its 4.16-kV air circuit breakers. These breakers are connected to 4160-volt switchgear buses 1, 2, 3, and 4. During transfer operation from the reserve auxiliary transformer to the unit auxiliary transformer, the two transformers are momentarily paralleled. The reserve auxiliary transformer is placed on standby and will remain energized. This transformer will also be used for normal shutdown power requirements in the plant. The 4160-volt bus tie air circuit breakers are normally open.

As the plant is now designed, it can be started up only with the assistance of power from the utility's grid. However, with the addition of two more 3000-kW diesel generators and an auxiliary steam boiler, a hot startup could be accomplished without outside power.

13.3.2. Abnormal Operation

The loss of the unit auxiliary transformer due to a turbine generator trip or failure of the transformer itself will initiate an automatic transfer to the reserve auxiliary transformer. In approximately

8 seconds the standby diesel generators will be started simultaneously with the trip action of the generator and will be available for connection to the 4160-volt system in case of a "blackout."

If the reserve auxiliary transformer failed while on standby, then the two standby generators would be started. The reserve feed breakers to the 4160-volt buses are normally open, so there would be no breaker trip action at that time.

If the plant should experience a complete loss of power from the simultaneous loss of all transmission lines and the tripping of the turbine-generator, then the two standby generators would be automatically started. The standby generators have the ability to pick up the full-rated electrical load within 10 seconds after a start signal is initiated. In this case, all the feeder breakers of the 4160-volt switchgear are tripped, tie breakers between 4160-volt buses 1, 2, and 3 are automatically closed, and the motor feeder breakers are tripped. The standby generator breaker will be closed and essential loads will be sequentially started. The second standby generator will be running at rated speed and be available as a 100% spare unit. Direct-current auxiliaries and uninterruptible a-c equipment will continue to operate from the two station battery systems.

13.3.3. Tests and Inspections

Normal central station practice includes a program of regular inspection and functional tests of equipment and protection devices; adherence to a program of this kind will ensure the operability of the auxiliary distribution system components. Routine testing of emergency transfers to the various emergency power sources will prove the operability of the systems involved. The inverters, battery chargers, and emergency lighting facilities will be tested periodically.

Detectors in the ungrounded d-c system will indicate a ground on either side, even though a ground on one side will not cause equipment malfunction. Logical isolation of the individual circuits connected to the faulted system will locate a ground, using necessary precautions to maintain the integrity of the vital bus supplies.

Section 14

SAFETY SYSTEM

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The following B&W drawing is placed at the end of
this section:

A20512F 1000-MWe LMFBR Reference Concept - Fault Tree Analysis
Diagram (Sheets 1 and 2)

14. SAFETY SYSTEM

14.1. Introduction

This section is an orderly discussion of the safety characteristics of the 1000-MWe reference design LMFBR. The problems were approached by defining a series of safety criteria to be applied to the plant. The general design criteria in Appendix 14-A were prepared by upgrading the present-day water reactor general design criteria to cover the special features of an LMFBR. The resulting criteria were then upgraded to the anticipated 1980 state of the art. Many of the design requirements were applied in the developing the other reactor systems described in other sections of this report; for example, the emergency core cooling mechanisms are described along with their design bases in the auxiliary system description, section 8. However, to make the assessment complete, a list of the specific criteria or principal design criteria to which the reference concept was designed is given for each general design criterion listed in Appendix 14-A. This shows how the reference design meets the general design criteria.

A fault tree was developed to aid in understanding the credible accidents and their effects on other systems and equipment in the plant. The various events have not been analyzed to determine the consequences of each path on the diagram, but the frequency with which operator error occurs as an initiating event should be noted. As developed, the fault tree must be considered to be a preliminary first effort. Many of the events should be developed and extended to incorporate more detailed paths.

There are many ways to arrive at the safety aspects of a given nuclear power plant. One could make several iterations in the plant design, thus incorporating safety features that provide the desired degree of protection with a minimum capital investment. Obviously, this was not possible during this follow-on study. Another approach would be to

incorporate a high degree of conservatism into the protection system design and then demonstrate that protection is adequate. The latter approach was used in many instances in the reference design plant. A large number of analytical safety analyses were performed to evaluate the dynamic response of the reactor and the protection system to credible accidents and to selected reactivity insertions. The reactivity insertion studies were made for cases that were above and beyond those that are possible for the reference design, in order to demonstrate that the core protection system is adequate to maintain core integrity during any credible accident and that the core protection system prevents propagation of the local failure (for those cases where local failure of the core cannot be avoided, such as a blocked subassembly).

In several cases the analyses have been extended to investigate very serious accidents that are deemed to be hypothetical because of the improbable sequence of events leading to their initiation. These hypothetical accidents, however, dictate the functional requirements for the engineered safety features that are incorporated to protect the public and the plant personnel. It is not intended that the protection system should provide protection for the core in this type of accident.

14.2. Principal Design Criteria

This section describes the principal design criteria that reflect the design intent for the 1000-MWe reference design plant in consideration of and addressed to the general design criteria for the LMFBR using 1980 technology. The general design criteria are presented in Appendix 14-A. The principal safety features that satisfy the sections of the criteria are summarized herein by stating the particular general criterion and answering it with the applicable principal design criterion.

14.2.1. Overall Plant Requirements

14.2.1.1. Quality Standards

"Those systems and components of reactor facilities which are essential to the prevention of accidents that could affect the public health and safety or to the mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Generally recognized codes or standards on design, materials, fabrication, and inspection shall be used."

Answer

1. Essential Systems and Components — The integrity of systems, structures, and components essential to the prevention of accidents or to the mitigation of their consequences has been included in the system descriptions. These systems, structures, and components are as follows:

- a. Fuel subassemblies.
- b. Reactor vessel internals.
- c. Reactor coolant system.
- d. Reactor instrumentation, controls, and protection systems.
- e. Engineered safety features.
- f. Radioactive materials handling system.
- g. Reactor vessel.
- h. Reactor biological shield.
- i. Reactor building.
- j. Electric power sources.

14.2.1.2. Performance Standards

"Those systems and components of reactor facilities which are essential to the prevention of accidents that could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public over the plant life, those forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect (1) appropriate consideration of the most severe combination of these natural phenomena that have been recorded for the site and the surrounding area and (2) an appropriate margin for withstanding forces greater than those recorded to allow for uncertainties in the historical data and their suitability as a basis for design."

Answer

The essential systems and components listed under quality standards will be designed to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena. The design bases for natural phenomena are listed below:

1. Seismology — The proposed reactor structure will utilize the Bassfield limestone strata as the foundation. This rock has good strength properties and will produce no amplification of ground motion from an earthquake. This area is not seismically active; however, the effects of earthquakes from distant sources will be experienced at the site. The

design basis earthquake has twice the acceleration of the operational earthquake, i.e., 0.2 g horizontal acting simultaneously with 0.15 g vertical.

Seismograph equipment is provided for this site, and at recording levels of 0.10 g horizontal the plant will be shut down automatically, giving an additional factor of 0.10 g for a safe shutdown before the plant design requirements are exceeded.

2. General Building — All site buildings have been designed for the most severe combinations of wind velocities of 100 mph (with shape factor $\approx 25 \text{ lb/ft}^2$) and snow loads of 30 lb/ft². The ground floor level is located not less than 6 feet above maximum river level; thus, flooding should not occur. In addition, areas that contain sodium have fire-resistant walls, which also act as flooding barriers.

3. Reactor Building and Control Room — These structures have been designed to withstand the effects of tornadoes having wind velocities of 300 mph (with appropriate shape factors) and external missiles consisting of wooden poles 12 feet long moving at 300 mph.

4. Ice Conditions — The plant contains equipment in the form of bypass pipes and valves to prevent possible loss of cooling water by icing during adverse winter conditions.

14.2.1.3. Fire Protection

"The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events on safety. Non-combustible and fire-resistant materials shall be used wherever practical throughout the facility, particularly in areas containing critical portions of the facility, such as the containment, control room, and components of engineered safety features.

"The design and arrangement of plant equipment containing metallic sodium shall be such that equipment failure and the possible subsequent sodium fire will not propagate additional failures that would further endanger the plant."

Answer

1. The reactor facility is designed to minimize the probability of fires and explosions and to minimize the potential effects of such events to safety.

2. The use of flammable materials in instrumentation and controls and in the control room is minimized as a general practice under the reference standards and codes; however, some flammable materials, such as paper charts, will be used.

3. The reactor support system is designed to minimize the probability of such events as fires and explosions. An inert-gas environment will be maintained in all areas where sodium is present. Noncombustible or fire-resistant materials are used throughout the system.

14.2.1.4. Shared Systems

"Reactor facilities shall not share systems or components unless it is shown that safety is not impaired by the sharing."

Answer

Not applicable to the single-facility, 1000-MWe LMFBR reference design.

14.2.1.5. Redundancy

"If the reactor facility has redundant systems, such as multiple primary coolant loops, then the systems may be considered independent if it can be demonstrated that the independence is not compromised in an unusual transient situation.

"Multiple failures resulting from a single event shall be treated as a single failure."

Answer

The three coolant systems, located between the reactor and the turbine inlets, are independent, and a failure in one does not precipitate a failure in another. No credit is taken for this in the design, since these systems are not necessary for decay heat removal. The redundancy of the engineered safety features is covered in the appropriate descriptions.

14.2.2. Protection by Multiple Fission Product Barriers

14.2.2.1. Reactor Core

"The reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel

and cladding damage limits. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation; appropriate margins will be allowed for uncertainties and for transient situations that can be anticipated, including the effects of the loss of power to circulation pumps, tripping out of a turbine-generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power. (The transient situation shall be assumed to involve complete loss unless redundancy can be demonstrated.) The core shall be designed to reduce to a very low level the probability of fuel subassembly blockage by regular shaped objects, i.e., by flat plates, or by segments of conically shaped objects.

"The prompt power coefficient shall be negative under all credible circumstances. Local power coefficients shall be such that the reactivity control system will be capable of ensuring the stability of the power distributions under all credible situations.

"The core design, together with reliable controls, shall ensure that power transients and/or oscillations that could cause fuel damage in excess of acceptable limits are not possible.

"The core shall be designed so that, in the event of sodium loss from the most reactive fuel element, the result is a reactivity insertion of less than one dollar."

Answer

1. The core was designed to limit the maximum cladding surface temperature to 1300 F in the hot channel at a sustained 103% of rated power to reduce the probability of cladding failure and to minimize chemical reactions between cladding, fuel, and coolant.
2. The subassembly sodium velocity was limited to 40 fps to guard against excessive corrosion and erosion.
3. The fission gases were vented to the coolant to limit the internal gas pressure buildup with lifetime.
4. The fuel temperature in the hot channel was limited to the melting temperature at 111% of rated power corresponding to the scram trip point.
5. The 85% theoretical density particulate fuel was chosen to reduce cladding stresses through (1) a more uniform circumferential stress distribution, (2) a mobile peripheral outer rim for accommodation of swelling, and (3) the elimination of the ridging effects found in pelleted fuel.

6. The decay heat of the fuel will be removed through one of the two decay heat removal systems. The decay heat cooling systems contain NaK as a heat transport medium between the reactor vessel and the atmosphere. The electromagnetic pumps and air circulators for the cooling coils operate from the guaranteed but interruptible emergency power supply.

7. The reactor primary circulators contain sufficient inertia (angular momentum) in the driver and eddy-current coupling to prevent damage to the fuel if power to the prime drivers is lost. Each primary pump eddy-current coupling and pony motor obtains its power from the guaranteed noninterruptible emergency power supply.

8. The reactor core is designed so that the Doppler coefficient is the dominant power coefficient overriding the positive sodium temperature (negative sodium density) coefficient. The thermal expansion coefficients are also negative, adding to the magnitude of the negative power coefficient. The ridged design of the subassemblies, using twisted ribbons as spacers, eliminates fuel pin bowing as a possible positive thermal input, since the spacers provide frequent support points for the pin along its length.

9. The reactivity insertion from the most reactive subassembly voiding is about 6¢ . The subassembly can is designed for normal operating pressure differentials as severe as those expected from any voiding accident in which the sodium temperature does not exceed 2000 F.

10. The plant control system is designed to regulate the plant operation in a manual or automatic mode of control in order to follow the electrical demand of the load. The integrator-type control loops provide stable operation during normal power transients.

11. The nuclear instrumentation system, the protection system, and the reactor control system ensure that power transients in excess of acceptable limits have an exceedingly small probability of occurrence.

12. The large neutron migration area and the high leakage from the core results in a stable fundamental power mode. Thus, the local power coefficients prevent power oscillations. The incore nuclear instrumentation and the exit thermocouples provide the necessary control

to account for long-term local reactivity changes and rod-produced power shifts.

13. The reference design uses a special inlet nozzle which projects below the bottom grid plate and has an irregular "castle-type" design to prevent complete subassembly blockage by regular shaped objects, i. e. , by flat plates, or by segments of conically shaped objects.

14. 2. 2. 2. Reactor and Intermediate Coolant Boundaries

"The coolant boundaries shall be designed and constructed so that they have a very low probability of gross rupture or significant uncontrolled leakage through the design lifetime of the reactor.

"The design shall provide for the detection of leakage from the reactor and intermediate coolant boundaries."

Answer

1. The reactor vessel system is designed to withstand the design temperature and pressure of 925 F and 30 psig.

2. The selection of the design pressure will include comprehensive consideration of hydrodynamic shock and excess pressure potential from external sources, such as from failure of pressure regulating or relieving devices.

3. Pressure barriers within the reactor vessel will be designed to withstand the maximum possible pressure and temperature differentials across the barrier.

4. The design stress of each structural part will be selected with due regard for the thermal or mechanical shock, coolant mixing effectiveness, and irradiation damage.

5. The reactor vessel system is designed to give a high degree of integrity, being of regular shape and containing no blisters or indentations.

6. The reactor vessel will be designed and fabricated in accordance with the requirements of Section III, Nuclear Vessels, ASME Boiler Code, and supplemented by special requirements of creep, liquid metal thermal shock, sodium corrosion, and mass transport.

7. The reactor vessel is surrounded by a gas space; the volume of this gas space is such that, in the unlikely event of gross rupture, the sodium level in the reactor vessel will remain high enough to cover the core, the primary pumps, and the emergency core cooling system heat exchangers. The gas space is hermetically sealed from the remainder of the plant and is instrumented with leak detection equipment that will initiate a reactor rundown if sodium is released to the gas space.

8. The intermediate heat exchangers (IHXs) and the steam generators are designed to reduce thermal gradients across tube sheets.

9. The intermediate coolant systems operate at a higher pressure than that of the primary system; hence, sodium leakage will be from the intermediate system into the primary system. Level instrumentation in the major components of the intermediate system, as well as that in the reactor vessel, is used to detect gross leakage of the IHXs. If leakage is in the opposite direction, it will be detected by gamma instrumentation monitoring each intermediate loop. If leakage is detected, then the reactor will be scrammed and the isolation valves in the leaking intermediate loop will be closed.

14.2.2.3. Containment

"Containment shall be provided for all radioactive materials and for plutonium in sufficient quantities to compromise the health and safety of the public. Containment structures shall be designed to sustain the initial effects of gross equipment failures without the loss of their required integrity, and, together with such other engineered safety features as may be necessary, to retain (for as long as the situation requires) the functional capability to protect the public."

Answer

The containment system (the biological shield and the reactor cover structure, which act as an energy and sodium barrier, and the reactor building, which acts as a fission product barrier) are designed to protect the public in the unlikely event of a design basis accident.

The biological shield and the reactor cover structure are designed to withstand an accident causing an internal pressure buildup of 100-psig. These same structures are designed to prevent

the formation of missiles in the reactor building and to prevent those formed inside the vessel from reaching the reactor building.

The reactor building is designed to protect the public and the plant personnel from leakage of radioactive material and to provide shielding against direct radiation. (See the system description in section 1, Land, Structures, and Conventional Systems.)

The fuel transfer machine is designed to meet the requirements of the cover structure; it forms part of that structure when it is attached for fuel transfer into and out of the reactor vessel. When the access plug is out, the interior surface forms a continuation of the cover structure in contact with the argon cover gas.

14. 2. 3. Nuclear and Radiation Controls

14. 2. 3. 1. Control Room

"The facility shall have a control room from which actions to maintain the safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to the equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without exposing personnel to radiation in excess of 10 CFR 20 limits. Adequate physical protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility. This physical protection shall be provided against plant missiles and natural phenomena such as earthquakes, tornadoes, flooding conditions, wind, ice, and other local site effects. The design bases shall reflect (1) appropriate consideration of the most severe combination of these natural phenomena that have been recorded for the site and the surrounding area and (2) an appropriate margin for withstanding forces greater than those recorded to allow for uncertainties in the historical data and their suitability as a basis for design."

Answer

1. Following proven power station design philosophy, all control stations, switches, controllers, and indicators necessary to start up, operate and shut down the nuclear unit will be located in one control room. Control functions necessary to maintain safe conditions, even after a design basis accident, will be initiated from this central control room.

2. Safe occupancy of the control room during abnormal conditions will be provided for in its design. Adequate shielding will be used to

maintain tolerable radiation levels in the control room for any accident up to and including the design basis accident. The control room ventilation system will include radiation detectors and appropriate alarms. Provisions will be made for control room air to be recirculated, with a minimum of makeup, through high-efficiency and charcoal filters. Emergency lighting will be provided.

3. For personnel physical protection required for the control room structure, refer to section 14.2.1.2.

14.2.3.2. Instrumentation and Controls

"Instrumentation and controls shall be provided as required to monitor and maintain variables within the prescribed operating ranges. The design shall provide for monitoring and maintaining control over the fission process throughout core life and for monitoring all conditions that can reasonably be anticipated to cause variations in the reactivity of the core.

"Core protection systems, together with associated equipment, shall act automatically to prevent or to suppress conditions that could result in exceeding the acceptable fuel damage limits. The protection action shall go to completion.

"Protection systems shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, or from accident conditions."

Answer

1. The control system maintains the following variables at prescribed levels: reactor flux, turbine throttle steam pressure and temperature, reheat steam temperature, reactor inlet and outlet temperatures, reactor coolant flow, IHX intermediate loop inlet and outlet temperatures, and intermediate coolant flow.

2. The protection system provides automatic reactor shutdown for the following conditions: high neutron flux, high or low neutron flux rate of change, high power-to-core flow ratio, high core outlet coolant temperature, high rate of increase of the outlet coolant temperature from two adjacent subassemblies, loss of intermediate sodium flow, sodium-water reaction, loss of feedwater flow, loss of heat sink, earthquake, high rate of increase of one fuel assembly outlet coolant temperature in conjunction with medium rate of increase in an adjacent

assembly, and sodium in the interspace around the reactor vessel. In addition, excessive core flux tilt will cause an automatic rundown to 95% of full power. Once initiated, these actions go to completion. The operator can also manually initiate scrams, rundowns, and fast setbacks.

3. The protection system also acts to limit radioactivity discharged from the reactor containment. Except for core heat removal paths, the containment is automatically isolated if high containment atmosphere discharge radioactivity or very high cover gas pressure is detected. The stack discharge path is automatically closed if discharge activity is excessive. The protection system automatically starts the emergency cooling system upon detection of high primary sodium temperature in the reactor vessel. The pony motors take over the load of the main pump motors when these motors fail.

Many other measured parameters can initiate alarms that require operator attention. These include radiation level measurements at numerous points inside and outside of the containment, including normal effluent discharge paths.

14.2.4. Reliability and Testability of Protection Systems

"Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed."

Answer

1. Redundancy and independence designed into the protection system are sufficient to ensure that no single failure, or a multiple failure resulting from a single event, or the removal from service of any component or channel, will result in the loss of a protection function. The redundancy provided includes as a minimum two channels of protection for each protection function. Different principles are used where necessary to achieve true independence of redundant components.

2. The protection system is separated from control instrumentation systems to the extent that the failure or removal from service of any control instrumentation component or channel, or of any single component or channel common to control and protection, leaves intact a system that satisfies all protection requirements.

3. The effects of adverse conditions to which redundant channels or components might be exposed in common, either under normal conditions or those of an accident, will not result in loss of the protection function.

4. In the event that all normal power is lost, sufficient alternate sources of power are provided for the required functioning of the protection system.

5. Means will be provided for testing the protection system while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

6. The protection system and its individual channels will be designed to fail into a safe state, or into a state established as tolerable on a defined basis, if such conditions as disconnection, loss of energy, or adverse environments are experienced.

14.2.5. Reactivity Control

"The reactivity control system shall be capable of sustaining any single malfunction, such as the unplanned continuous withdrawal of a control rod, without causing a reactivity transient that could result in exceeding acceptable fuel damage limits.

"Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on the rates at which reactivity can be increased. This will ensure that the potential effects of a sudden or large change of reactivity cannot disrupt the core, its support structures, or other vessel internals enough to impair the effectiveness of emergency core cooling."

Answer

1. No single malfunction will impair the shutdown capability of the reactivity control system.

2. Loss of power will produce a scram.

3. The transient produced by such control rod withdrawal at the maximum rate will not result in core damage.

4. The maximum worth of a single rod is less than \$1.00; hence, no single rod action can lead to a prompt critical condition.

5. Control logic will prevent the use of more than two groups of three rods under automatic control. If all six rods are withdrawn at the maximum rate, the resulting transient will not produce core damage.

6. Rod ejections during shutdown are not possible, since hydraulic pressures in the control rod guide tubes are not capable of lifting the rods.

7. Sufficient excess control is provided to ensure reactor shutdown with one control rod stuck out of the core.

14.2.6. Reactor Coolant Boundary

"The reactor coolant boundary shall be designed to minimize the probability of rapidly propagating failures. Consideration shall be given (1) to the state of material stress under static and transient loadings, (2) to the quality control specified for materials and component fabrication to limit flaw sizes, and (3) to the provisions for the control of service temperature and irradiation effects that may require operational restrictions.

"The reactor coolant system shall be designed to permit access to critical areas of the reactor coolant boundary for inspection, testing, and surveillance by appropriate means to assess the structural and leak-tight integrity of the boundary components during their service lifetime."

Answer

1. Many of the principal criteria associated with this general criterion are given in section 14.2.2.

2. The reactor vessel, which forms the coolant boundary, is of regular geometric configuration and has no blisters, penetrations, attachments, or stress risers located in the areas normally wetted by the reactor coolant.

3. The pressure in the secondary circuit of the IHX is higher than that of the low-pressure primary coolant, but, since this secondary system is also a low-pressure system, failure of any IHX tube will not propagate failure to other tubes.

4. The main heat transport temperatures, temperature differences, and flows will be controlled to minimize thermal stress in the heat transport systems, components, and reactor core.

5. The hot leg temperature differences in the IHXs, steam generators, and reheater will never exceed 100% of load operating conditions.

6. The cold leg temperature difference in the IHX and steam generator is maintained constant, whereas the cold leg temperature difference in the reheater increases as load decreases.

7. The annular gas space between the reactor vessel and the concrete liner is available for inspection of the reactor vessel by appropriate instruments (should it be considered necessary), since no insulation is used on the reactor vessel. Should it also be considered necessary, the development of non-destructive testing methods (which may include remote visual inspection for sodium penetration in steel, and cracks in steel) should be completed by the 1980's.

14.2.7. Engineered Safety Features

14.2.7.1. General

"The facility shall have engineered safety features to back up the safety features built into the core design, the reactor coolant boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any accident up to and including the design basis accident.

"All engineered safety features shall provide high functional reliability and ready testability. In determining the suitability of a facility, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected (where appropriate) during the life of the plant.

"An alternate power system shall be provided with adequate independency, redundancy, capacity, and testability to permit the required functioning of the engineered safety features assuming that a single component in the system fails.

"The engineered safety features shall be protected against thermal effects, dynamic effects, and missiles that might result from plant equipment failures. Engineered features, such as emergency core cooling and containment heat removal systems, shall provide sufficient performance capability to accommodate the partial loss of installed capacity and still fulfill the required safety function. As a minimum, the engineered safety features shall provide this required safety function assuming the failure of a single component, such as a pump, valve, or the like.

"The engineered safety features shall be designed to avoid any action that might accentuate the adverse after-effects of an accident.

"The performance of the engineered safety features and the condition of the reactor shall be monitored by instrumentation during the course of and after an accident."

Answer

The reactor is designed to protect the core during credible accidents and to ensure that the core is usable after the accident. In those cases in which the initial detectable event is localized fuel failure, the design features and the core protection system must be designed to maintain the local nature of the fuel failure.

It is not intended that the protection system should maintain the operable nature of the plant following an accident or malfunction of the plant and two or more malfunctions in the protection system or the engineered safety features. However, in such an event, it will be necessary to provide for the safety of the operators and the public by incorporating additional engineered safety features in the form of energy barriers and radioactive material barriers.

Two types of emergency power systems are provided with two systems in each. The guaranteed non-interruptible power supply will provide power to those systems and components of the protection systems and to those engineered safety features whose operation must be maintained on a continuous basis. For those systems and components for which continuous operation is not necessary, a guaranteed but interruptible emergency power source is provided. The guaranteed non-interruptible power supply is battery-backed. The guaranteed but interruptible power supply is backed by two diesel-driven generators.

Two emergency core cooling systems are provided; they are described in section 14.2.7.2.

The assurance of instrument power for monitoring both the reactor and the engineered safety features is given in the answer to the "Reliability and Testability of Protection" criterion in section 14.2.4. In general, the detectors will be located external to the reactor vessel system and will be structurally protected. This includes nuclear instrument detectors and non-nuclear detectors to monitor the cooling system's performance. Since numerous detectors are provided to monitor the primary coolant, it is felt that at least a few will survive. (Hot leg monitors include 288 subassembly outlet thermocouples,

several core outlet plenum monitors, and inlet monitors to all six IHXs. The cold leg monitors include three in the core vessel (core inlet plenum), several monitors at all outlets of all six IHXs, and several distributed throughout the reactor vessel). Nuclear instrumentation is provided for monitoring all effluent paths to the environment and in the reactor building.

14.2.7.2. Emergency Core Cooling Systems

"The emergency core cooling systems remove decay heat from the reactor core after an accident that results in the loss of the normal heat removal path. Loss of the normal heat removal path may occur by either loss of the heat sink or loss of access to the heat sink.

"At least two heat removal mechanisms, preferably of different design principles and each capable of accomplishing emergency core cooling, shall be provided. The heat removal mechanisms and the core shall be designed to prevent damage to the fuel and cladding that would interfere with the emergency core cooling function.

"The heat removal mechanisms shall be capable of fulfilling their functions in any situation wherein the normal path of heat removal is compromised.

"The heat removal mechanisms shall share neither active components nor other features or components unless it can be demonstrated that (1) it can be readily ascertained during reactor operation that the shared feature or component is capable of performing its required function, (2) failure of the shared feature or component will not initiate a design basis accident, and (3) the ability of the shared feature or component to perform its required function is not impaired by the same effects of the design basis accident and is not lost during the entire period during which this function is required following the accident.

"Wherever possible, the design shall provide for physical inspection of all critical parts of the heat removal mechanisms, including those parts within the reactor vessel. Any exception shall be justified.

"The design shall provide for periodic testing of the delivery capability of the heat removal mechanisms at a location as close to the core as is practical.

"Provisions shall be included for testing, under conditions as close as practical to design conditions, the full operational sequence that would bring the emergency heat removal mechanisms into action, including the transfer to alternate power sources."

Answer

1. The emergency core cooling system consists of two independent and identical sodium-to-NaK-to-air loops. Each loop contains one

sodium-to-NaK heat exchanger in the pot, coils containing NaK embedded in the biological shield, three half-capacity electromagnetic pumps, one NaK-to-air heat exchanger, four full-capacity fans and drivers, one NaK storage tank, and the appropriate instrumentation and connecting piping.

2. Each loop is capable of removing 50 MWt power from the cold leg of the reactor coolant system under design conditions. Both loops depend on the primary pump pony motors, which operate in the reactor coolant system, to remove the decay heat from the core. These motors are connected to the guaranteed non-interruptible buses. The power to operate the emergency core cooling system comes from the guaranteed but interruptible buses, which in turn are connected to the two 3000-kW standby diesel generators.

3. The system is designed so that one of the two available loops will come on the line when the core inlet temperature reaches 850 F. The design is conservative in that operation of the loops could be delayed for approximately 2 hours, and the average temperature of the reactor coolant system would only be 1200 F. At that time, each loop would be capable of removing considerably more than 2% since the logarithmic mean temperature difference (LMTD) for the heat exchangers would have increased. However, at this time (2 hours after shutdown) the heat generation rate would have dropped to approximately 14 MWt.

4. The system is designed to permit periodic operation of the system without interrupting the normal operation of the plant. Each loop is standing by hot and isothermal. Heat losses in the piping and components will cause a certain amount of natural circulation, which is desirable. Upon actuation of the system, the air dampers in the NaK-to-air heat exchangers will open as the fan motors are energized. Simultaneously, two of the three EM pumps will come on the line.

5. During standby conditions, the effectiveness of the NaK-to-sodium heat exchangers, which are located in the reactor vessel, can be tested by operation as explained above.

6. A level detector in the NaK surge tank provides leak detection from this system to the primary system. This is backed up by chemical analysis of the reactor coolant. Leakage of primary sodium from the primary system into the NaK will be detected by radiation detectors.

7. In the event that the primary system temperature reaches 300 F, the emergency core cooling system becomes a heating system by automatically firing propane heaters in the NaK-to-air cooler section.

8. Except for components located inside the reactor vessel in the primary sodium, in the biological shields, and in penetrations of shields, all components are available for visual inspection periodically.

14.2.7.3. Containment

Reactor Building

"The function of the reactor building is to contain, without failure, all of the material and energy released by an accident up to and including the design basis accident. The reactor building shall be designed to provide sufficient biological shielding and retention of radioactive materials and plutonium to meet the requirements of 10 CFR 100.

"The reactor building, including its access openings and penetrations and any necessary heat removal systems, shall be designed to accommodate any energy release, pressures, and temperatures that result from any accident (up to and including the design basis accident) without exceeding the leakage limits for radioactive material and plutonium; this provision includes a considerable margin for the effects that could occur as a consequence of failure of the emergency core cooling system.

"The reactor design shall provide sufficient inert barriers around all liquid sodium systems within the reactor building to ensure that a sodium fire resulting from any accident (up to and including the design basis accident) shall not produce temperatures and/or pressures in excess of the reactor building design limits.

"The principal load-carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures, under normal operating and testing conditions, are not less than 30 F above the nil ductility transition (NDT) temperature.

"Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

"Where active heat removal systems would be needed under accident conditions to prevent the containment pressure and temperature from exceeding the design pressure and temperature, at least two systems, preferably of different principles and each with full capacity, shall be provided.

"Water, or equipment containing water or other hydrogen compounds shall be excluded from the reactor building, unless it can be demonstrated that the material can neither (1) endanger

the integrity of the reactor building if it reacts chemically with sodium, nor (2) result in a significant reactivity insertion.

"The reactor building design shall provide for integrated leakage rate testing at design pressure after erection and installation of all penetrations is complete. The leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance. The design shall provide for the testing of penetrations having resilient seals or expansion bellows and for the demonstration of leak-tightness at design pressure at any time. The design shall provide for the testing of the functional operability of valves and associated apparatus essential to the containment function to establish that no failure has occurred and to determine that valve leakage does not exceed acceptable limits.

"The design shall provide for the testing of the functional operability of personnel locks and equipment locks or passageways with valves and associated apparatus essential to the containment function to establish that no failure has occurred and to determine that the leakage does not exceed acceptable limits."

Answer

1. The design requirements of the building are 10 psig and 200 F. All damaging effects that are capable of causing damage to structural components, such as shock waves, sodium hammer, and internal pressure and temperature effects, will not propagate beyond the primary sodium retention barrier.

2. All liquid metal lines within the reactor building are doubly contained with leak-detection instrumentation in the inerted interspace.

3. The reactor vessel and the cover structure are designed to prevent sodium release into the air space above the charge face floor at any time.

4. In the event that all heat transfer equipment inside the reactor vessel becomes inoperative, the gas interspace between the reactor vessel and the lined biological shield, will be filled with intermediate system sodium. The decay heat generated in the reactor core will then be removed from the reactor vessel by cooling coils which are located on the interface of the reactor pit liner and the biological shield. Two separate systems of coils carrying NaK and circumscribing the reactor pit liner are connected to the emergency core cooling system NaK loops described above. Each set of coils is capable of removing all of the decay heat generated in the reactor. Approximately one hour will be

available to put this system into operation before excessive reactor coolant temperatures are reached. Therefore, the mechanism whereby sodium is transferred to the interspace will be judiciously designed to prevent inadvertent transfer.

5. The top cover structure and the rotating shield plugs provide sufficient shielding, 1 mR/hr, to have man-access on the operating floor level during normal reactor conditions and during the refueling operation.

6. The cover structure also provides the reflective and thermal insulation necessary to prevent degradation of the shielding material and to ensure a minimum operating floor level temperature of 100 F ambient. The maximum temperature of the serpentine aggregate is 800 F.

7. The reactor building will be subjected to an integrated leakage rate test to measure the percent by weight of air that can leak out of the reactor building in a day. The specified design leakage of 0.10% or less per day will be measured. The duration of the test will be a minimum of 24 hours.

8. All electrical penetrations are of a double-barrier type and are equipped for periodic leakage monitoring.

9. The equipment door is of the double barrier type and is equipped for periodic leakage monitoring.

10. The personnel air lock is also equipped for periodic leakage monitoring.

11. No principal ferritic material, load-carrying components are exposed to the external environment, since the reactor building membrane is located on the inside surface of the 4-foot containment shield. The concrete shell also protects the liner from potential externally generated missiles.

Auxiliary Containment

"The function of the auxiliary containments is to prevent the undue release of fission products from equipment located outside the reactor containment. "Undue release" is defined as any release of radioactive nuclides to the environment that exceeds the limits of 10 CFR 20.

"The nature of the auxiliary containments shall be determined by the physical form and the quantity of the contained

radioactive material. For example, equipment that contains radioactive liquids may be contained in cells or containers large enough to contain all credible leakage. Penetrations of any auxiliary containment—for example, drains or vents—shall be designed so that positive isolation devices prohibit the unexpected release of any contained radioactive material. The auxiliary containments shall be cooled as necessary to prevent their exceeding the design pressure and/or temperature."

Answer

1. All gaseous waste storage is maintained inside the containment except that required for the low-level, low-pressure storage, which will be in a shielded compartment with double doors. This storage tank will serve as a surge tank for the stack discharge and will be used during adverse weather conditions.

2. The liquid wastes are stored in tanks in shielded rooms large enough to contain the entire volume of the tanks.

3. The solid wastes are stored in shielded rooms.

4. The spent fuel shipping casks contain only one subassembly per cask. The shipping cask is designed to sustain the worst accident conditions of transport, as specified by federal shipping regulations, 10 CFR 71. This accident results in the following package design requirements on the cask:

a. Double container construction. The inner container is designed to contain primary sodium at a pressure of 300 psia and a temperature of 2000 F.

b. The shielding will be such that, if melting occurs, the molten shield material cannot escape from the cask or reduce the shielding effectiveness below allowable levels.

c. The secondary container is cooled by a nitrogen-to-air heat exchanger.

d. The criticality problem is eliminated through proper control of the physical geometry of the cask.

14.2.8. Fuel and Waste Storage System

"Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations will be emphasized over procedural controls.

"Reliable heat removal systems shall be designed to prevent any damage to the fuel in the storage facilities that could result in the release of radioactivity to plant operating areas or to public environs.

"Shielding for radiation protection shall be provided in the design of the spent fuel and waste storage facilities, as required to meet the requirements of 10 CFR 20."

Answer

1. Criticality of fuel assemblies in the spent fuel storage drums is prevented by maintaining the assemblies in a critically safe geometric configuration and by the presence of neutron-absorbing material in the storage drum.

2. New fuel is maintained in a critically safe, sealed container until its insertion into the fuel transfer machine.

3. The fuel storage drums are located in the reactor vessel at an elevation that ensures that all assemblies are under sodium at all times. Thus, cooling and shielding are provided for the stored assemblies.

14.2.9. Plant Effluents

"The facility design shall provide for maintaining control over the plant radioactive effluents, whether gaseous, liquid, or solid. The appropriate holdup capacity will be provided for retaining gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (1) on the basis of 10 CFR 20 guidelines for normal operations and for any transient situation that might reasonably be anticipated, (2) on the basis of 10 CFR 100 dose level guidelines for potential reactor accidents of very low probability of occurrence, except that reduction of the recommended dose levels may be required where high population densities can be affected by the radioactive effluents. An environmental monitoring program shall be maintained to confirm that excessive effluent releases have not occurred."

Answer

1. The radioactive waste system collects, segregates, processes, and disposes of radioactive solids, liquids, and gases in such a manner as to ensure compliance with 10 CFR 20 guidelines.

2. Liquid and solid wastes are processed in a batch manner for off-site disposal. Gaseous wastes released to the environment are

monitored and discharged with suitable dilution to ensure tolerable activity levels on the site and at the site boundary.

3. The gaseous waste system stores the gas accumulated during operation. The contents of the delay tanks is sampled, and a release rate is established consistent with the prevailing environmental conditions. In-line monitoring will provide a continuous check on the release of activity.

4. Permanently installed area detectors and the plant vent detectors are used to monitor the discharge levels to the environment. In addition, portable monitors are available on site for supplemental surveys, if necessary.

5. Radiation detectors monitor the residual gases prior to their discharge to the plant vents and the environment.

14.3. Fault Tree Analysis

A preliminary fault tree analysis diagram for the 1000-MWe LMFBR reference concept was completed as part of the plant safety analysis. The symbols used in the fault tree analysis are defined with the diagram on drawing A20512F.* The fault tree, as drawn, defines the fault combinations and sequences required for an abnormal release of radioactive material to the environment. The ultimate objective of any safety program is to ensure that there is no undue risk to the health and safety of the public. No attempt has been made to establish the most probable paths to an exposure or a material release, since failure rates and other data for most of the system components are unknown at the present time.

In all cases, at least two failures are required in the protection system to permit any one malfunction to propagate through the system and release radioactive material. At least two failures in the protection system are also required to permit any one malfunction to propagate through the system and cause core melting, in all cases except those in which the malfunction is local core failure. No protection action prevents

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* B&W drawing A20512F is placed at the end of this section.

local damage in the event of complete flow blockage to a given subassembly. However, the core design and the protection systems guard against the propagation of a failure to adjacent subassemblies, and thus they maintain the local nature of the failure. The various events have not been analyzed to determine the consequences of each path on the diagram. Operator error or malfunction as an initiating event is worth noting because of the frequency with which it occurs in the fault tree. The necessity of preventing operator error is already accepted in the operating procedures for nuclear power plants. It should be noted that the use of direct digital computer control for the plant will reduce the operator error significantly, since the operator will be able to intervene at the point where he can comprehend the control requirements. The acceptance of digital-computer-evaluated safety actions will permit the protection system to function more efficiently. The digital computer can absorb more input data from plant detectors, assimilate it in a useful form, and take appropriate protection actions.

In order to simplify the fault tree, an input is indicated by a triangle, and an output is indicated by a square. The character of each of these input-output functions is given in Table 14-1.

Table 14-1. Input-Output Number Notation
for Fault Tree

-
1. Intermediate coolant to IHX abnormal.
 2. Abnormal core thermal conditions.
 3. Subassembly cladding failure.
 4. Turbine trip.
 5. Steam relief valve malfunction.
 6. Reheater tube failure.
 7. Steam generator tube failure.
 8. Dump valve malfunction.
 9. Power to flow scram.
 10. Primary pump coupling failure.
 11. Abnormal subassembly thermal conditions.
 12. Thermocouple scram failure.
 13. Plug release.

14.4. Reactor Parameters

The safety criteria presented in the introduction form the basis for a series of reactor safety calculations, the primary purpose of which was to evaluate the ability of the safety system to protect the core for large, though credible, reactivity insertions. Care has been taken to ensure an adequate degree of conservatism in the analytical models for these safety calculations; in some cases, the degree of conservatism may be extreme because of uncertainties in some parameters. Continued effort in research and development should permit a reduction in the conservatism by providing better correlations for the analytical analysis.

The geometry, composition, and physical characteristics of the reference core will not be discussed here, since they are described in great detail in the core system description (see section 4). The pertinent reactivity parameters for the reference design are listed in Table 14-2. These parameters are basically those required for the detailed analytical model used in FORE-II, and any departure from these parameters will be specifically noted in the subsequent discussions.

Table 14-2. Reactivity Coefficients and Other Parameters

Parameter		Value
Core radial expansion coefficient $\left(R \frac{\delta K}{\delta R}\right)$		-0.532
Core axial expansion coefficient $\left(H \frac{\delta K}{\delta H}\right)$		-0.158
Doppler coefficient $\left(T \frac{dK}{dT}\right)$, core full dense		-0.00470
Doppler coefficient $\left(T \frac{dK}{dT}\right)$, core void		-0.00176
Coolant density coefficient $\left(\rho \frac{\delta K}{\delta \rho}\right)$:		
Axial section	Core and axial blankets voided	
1	0.00067	
2	-0.00174	
3	-0.00736	
4	-0.00995	
5	-0.00736	
6	-0.00174	
7	0.00067	
Total		-0.02681
Linear expansion coefficient $\left(\frac{1}{L} \frac{\delta L}{\delta T}\right) \cdot F^{-1}$		0.0000104
β_{eff}		0.00364
t^* , μsec		0.290

The techniques and assumptions used to calculate the coefficients presented in Table 14-2 are described in detail in the core system description (Volume 2, section 4) with the exception of the coolant density coefficient, $\rho(\delta K/\delta \rho)$. The axial dependence of the coolant density coefficient was determined by segmenting the reactor into seven axial nodes—five for the core region and one for each axial blanket. The sectional worths were determined from the section volume average of the axial sodium point worth distribution as evaluated from perturbation theory. The methods used to arrive at the coolant point worth distribution are also presented in section 4. The agreement between the perturbation theory and diffusion theory δK calculation is quite good.

For the reactor excursion runs using the modified WEAK EXPLOSIONS code, it was assumed in the safety analysis calculations that the radial power distribution and the radial fuel self-worth distribution could be adequately described by a sixth-order polynomial of the following form:

$$y = a_1 + a_2 x + a_3 x^2 + a_4 x^3 + a_5 x^4 + a_6 x^5 + a_7 x^6$$

We can generate polynomials of this form, in fact, that closely approximate the actual distributions by using a weighted, least-squares-fitting technique.¹ For the reference design, the coefficients for the radial fuel self-worth distribution are given in Table 14-3.

Table 14-3. Constants for Self-Worth Distribution Polynomial Fit

Constant, units	Value
a_1 , constant, $\delta K/\text{cm}^3$	0.66244×10^{-8}
a_2 , coefficient of r , $\delta K/\text{cm}^4$	-0.58452×10^{-10}
a_3 , coefficient of r^2 , $\delta K/\text{cm}^5$	0.42387×10^{-11}
a_4 , coefficient of r^3 , $\delta K/\text{cm}^6$	-0.11777×10^{-12}
a_5 , coefficient of r^4 , $\delta K/\text{cm}^7$	$+0.14144 \times 10^{-14}$
a_6 , coefficient of r^5 , $\delta K/\text{cm}^8$	-0.74204×10^{-17}
a_7 , coefficient of r^6 , $\delta K/\text{cm}^9$	0.13679×10^{-19}

14.5. Credible Accident Analysis

The principal purpose of this section is to describe the dynamic response of the reactor under accident conditions and to determine that the core protection system and the overall reactor design are adequate to maintain the basic core integrity. No attempt has been made to evaluate operating incidents caused by minor equipment malfunctions or errors in judgment, since such incidents are normally implicit in the reactor design margins. The functional requirements of the core protection system are described, along with definitions of pertinent control parameters. It should be noted that the control parameters for the safety system incorporate significant degrees of conservatism and in no way represent minimum functional requirements for the core protection system.

The FORE II computer code was used to evaluate the reactor transient response to accident conditions. The input description is based on the analytical model used in FORE II, even though this analytical model does not correspond in every respect to reality. Where differences exist, the analytical model is conservative.

Three scram trips are available in the FORE II code: (1) power level, (2) high core outlet temperature, and (3) high fuel temperature. Other scrams which are produced in external systems must be included in the reactivity driving function. Only two of the internal scrams were normally set, since there are no detectors available now for the fuel; these were set at (1) a power level that exceeds 111% of full operating power and (2) a coolant outlet temperature that exceeds 1200 F. Certain departures have been made from this scram representation, and they are specifically noted in the discussions that follow.

14.5.1. Reactivity Insertion Representation

The reactivity control system includes two types of control rods—the safety rods (seven rods with a total worth of 0.015 δk) are positioned with the bottom of the rod at the upper core axial blanket interface at all times during operation. Of the remaining 18 shim-regulating rods, 9 are also normally out of the core during operation, since their function is to provide the extra worth required if the core is voided. The worth of this group is approximately 0.02 δk . The remaining rods may be partially or fully inserted; hence, they might not contribute

significantly to the scram reactivity. The reactivity control system is described in detail in section 5, Volume 2.

The description of the negative reactivity insertion following a reactor scram trip was simplified for the analytical studies performed with FORE II. The worth of the rods driven into the core was taken to be 0.030 δk , which is somewhat conservative, since it does not take into account all of the available rod worth. The insertion rate was constant at 50 cm/sec, and the insertion was started after a 300 msec time delay between production of the scram trip condition and the time the rod motion was started. This 300-msec time delay is derived from two sources: (1) 150 msec elapses from the production of a trip condition until the magnetic coupling is broken and the brake applied, and (2) 150 msec is allowed for the rod to accelerate to 50 cm/sec. Up to this point, the model has been highly conservative. The model used a linear rod worth curve, however, which is not necessarily conservative. Thus, it was necessary to determine that the negative reactivity insertion rate for the real scram was indeed more rapid than the model used in the analysis. These rates are presented for comparison in Figure 14-1a. It is clear that the model used is conservative. A comparison is also made in Figure 14-1b between the linear rod worth, which was used in Figure 14-1a as a function of position, with that calculated using both one-dimensional diffusion theory and perturbation theory.

14.5.2. Reactivity Insertion Analyses

Although the major emphasis in the safety analysis of the reference design 1000-MWe LMFBR has been placed on evaluating the results of real, foreseeable accidents, some effort has been made to predict the behavior of the reactor system for arbitrary transient conditions. The conditions are termed arbitrary because they have not been identified with specific initiating events. The study of system response to these reactivity insertions is useful in defining acceptable control parameters for the safety system, since they can be used for sensitivity-type analyses of various parameters. The study of the reactor response was also useful in defining acceptable control parameters for the safety system. In a sense, these studies provide data on transient conditions which can reasonably be assumed to represent an upper limit for credible accidents, such as rod withdrawal, etc.

Most of the parametric work done in this area was done for the preliminary reference core without BeO. Since these results have an important bearing on the final values of some control parameters, a brief description of the preliminary reference core and pertinent results is presented in Appendix 14-B.

The reactor response to a ramp insertion of $\$2/\text{sec}$ was studied for verification of the final reference core. A normal scram, as defined in the introduction to this section, was initiated upon detection of an 11% overpower condition. The maximum power of 8150 MWt occurred 0.360 second after the ramp was initiated. The maximum fuel temperature in the peak channel (maximum powered channel), 3760 F, occurred at 0.52 second, while the maximum coolant temperature of 1140 F occurred at 0.60 second. All transients peaked well below limiting conditions. The results are presented graphically in Figures 14-2 and 14-3. Other higher reactivity insertions are discussed to some extent in section 14.6.1.

Figure 14-1a. Rod Acceleration Curve - Reactivity Vs Time

Figure 14-1b. Rod Worth as a Function of Percent Insertion

Figure 14-2. \$2/sec Ramp, Normal Scram - Power Vs Time

Figure 14-3. \$2/sec Ramp, Normal Scram - Maximum Fuel and Coolant Temperatures Vs Time

14.5.3. Equipment Malfunction or Failure

Particular attention has been given to examining the consequences of equipment malfunctions or failures that could result in seriously abnormal operating conditions. It is especially important that the design and operation of the safety system be such that no single failure or malfunction can jeopardize the integrity of the reactor. In order to evaluate the effectiveness of the safety system for this broad category of accident conditions, a number of calculations have been performed. The results of these calculations are discussed in this section.

14.5.3.1. Loss of Electrical Power

The loss of electrical power to the reactor plant in no way alters the ability of the safety system to scram the reactor and remove decay heat from the core. The primary pump pony motors and their eddy-current couplings are on the guaranteed non-interruptible power supply. The emergency core cooling systems are on the diesel-backed, guaranteed but interruptible power supply.

The most obvious consequence of a loss of electrical power is the loss of primary pumping power. In evaluating the reactor response to a flow coastdown, it was assumed that a normal scram was initiated upon detection of electrical power loss. The flow decay curve for a uniform, six-pump flow coastdown is presented in Figure 14-4; it is based on the projected characteristics of the pump and includes momentum from motor, coupling, and impeller in the coastdown. The results indicate an outlet coolant temperature rise of less than 1 F, since the scram is faster than the flow coastdown rate. Thus, flow coastdown from full power leads to virtually no coolant temperature rise before fuel cooling begins.

It was shown, for the preliminary reference core without BeO, that failure to detect a loss of pumping power will not lead to coolant boiling, since a high coolant outlet temperature trip set at 1200 F will scram the reactor. The results of such a coastdown are reviewed in Appendix 14-B.

Power/Flow Ratio for Coastdown Accident

— In the event of loss or interruption of power to one or more primary

Figure 14-4. Flow Decay Curve - Flow Vs Time After Loss of Pumping Power

pumps, it is desirable to employ a fast setback to avoid a scram, since a scram may have detrimental effects on reactor lifetime. A few cases were studied to determine whether a power/flow scram can be avoided by a setback. The results of the calculations are presented in Figures 14-5 and 14-6.

The number of rods used in the fast setback and the delay time in the control system before insertion begins were varied to develop the setback requirements.

The normal rod speed, 15 inches per minute, was used in the parametric study of the delay time (the time lapse between loss of pumping power and first motion of the rods). In all cases a six-pump coastdown was used. It is interesting to note that, for a 200-msec delay after loss of pumping power followed by a fast setback, the power level reached 50% of its initial value more rapidly than did a fast setback with no flow coastdown. The principal reason for this behavior is the slower fuel cooling and consequent smaller positive Doppler feedback in the case with the flow coastdown and 200-msec delay. It appears that neither a 3-rod setback with a 200-msec delay nor an 18-rod setback will avoid a power-to-flow scram for loss of all six pumps. However, for less than a six-pump coastdown, the power-to-flow scram can be avoided. As is shown in Figure 14-6, a 25-rod setback is more than sufficient, even for the six-pump coastdown.

14.5.3.2. Control Assembly Motor Bearing Seizure

The speed and direction of motion of each control assembly is governed by the action of two synchronous motors operating through a planetary gear mechanism. In the event of complete seizure of the secondary drive mechanism between the flywheel and the planetary gear, the control assembly would be driven from the core at a maximum rate of 30 inches/second. In the calculational model, the rod was withdrawn at the maximum rate, starting with the reactor at full power for a total of 150 msec, after which the protection system disengaged the magnetic clutch, and the rod decelerated and fell at 0.8 g back into the core. This study did not investigate the possibility of eliminating the rapid withdrawal by permitting sudden acceleration to break the magnetic connection between the rod and the drive units. The analysis of

this malfunction is presented in Figure 14-7. It can be seen that the 11% overpower trip point was reached, and that a normal scram followed (300-msec delay). At the time the scram begins, the subject control assembly has returned to a full-in position. The temperature increases in the peak channel were very small; the maximum fuel temperature increase was 20 F, while the outlet coolant temperature increased only 4 F.

It is possible to conceive of a situation in which three control assemblies could be withdrawn continuously at the normal shim speed of 15 inches per minute. The transient thus produced is not nearly as severe as that produced by the seizure of the secondary motor drive just described. The ability of the protection system to safely terminate the latter transient ensures its ability to terminate the transient that is produced by continuous withdrawal of three shim rods.

14.5.3.3. Partial Voiding

Earlier LMFBR designs have been based on the criterion that there be a zero or negative reactivity change when the core and axial blankets are voided. The trade-off study comparing spoiled and unspoiled geometries (Volume 4) indicates that such a criterion and low fuel costs are mutually exclusive. By properly designing the reactor vessel, the primary coolant system and the engineered safety features, the probability of voiding the reactor becomes very small. The principal mechanisms of voiding are drainage and boiling.

Drainage is not possible with the integral pot arrangement used in the reference design. Even in the event of vessel rupture, the annular volume between the reactor tank and the concrete vessel is small enough to maintain the coolant level above the core, the primary pumps, and the IHX inlets. The core protection system and the engineered safety features prevent widespread boiling during all conceivable malfunctions. Local boiling, however, cannot be prevented following the blockage of one subassembly, but since each subassembly is canned to prevent cross propagation to other subassemblies, the void will not propagate. Loss of pumping power to the primary pumps will result in a power/flow scram which has been shown to be safely terminated. Analyses show that the fuel and sodium temperatures peak at far below the maximum allowable levels.

Figure 14-5. 25-Rod Fast Setback - Power Vs Time

Figure 14-6. Fast Setback - Power/Flow Vs Time

Figure 14-7. Rod Withdrawal at Full Power - Power Vs Time

A sudden power increase can result only from reactivity insertion. Even if all the shim rods are inadvertently withdrawn at their maximum rate, the ramp reactivity insertion rate is less than that used to demonstrate the adequacy of the core protection system, \$2/sec ramp. A second mechanism is voiding by flow blockage or gas bubbles introduced into the core. The introduction of large gas bubbles in the pool primary system is eliminated in the reference design. Since the reactor coolant system contains six independently pumped loops, the simultaneous blockage of more than one loop is not credible, because their intakes are widely spaced in the reactor vessel. Since the blockage of only one loop does not lead to an extremely large coolant temperature rise, we must examine the entrance to the individual fuel assembly.

The grid plate is designed to prevent the blockage of coolant flow to a subassembly by a flat object. However, for the purposes of the safety analysis, it is assumed that one fuel subassembly can be blocked by a properly shaped object. The simultaneous blockage of more than one subassembly is not considered credible. The coolant temperature in the blocked subassembly would rise rapidly, depending on the degree of blockage, and the rate of void formation would depend on the degree of liquid superheat before boiling begins. However, fast void formation is not anticipated, since the vapor pressure is relatively low even at 300 F superheat, about 55 psig. The subassembly can be designed so that very large degrees of superheat are required before the energy of the expanding sodium vapor bubble is great enough to damage the subassembly can. Considering the cladding materials proposed for the reference plant, it is reasonable to assume that the fuel in the blocked subassembly will be damaged by the time the void begins to form. Since the six adjacent fuel subassemblies maintain normal flow, the sodium from these subassemblies, even at the exit, will remain about 500 F subcooled. Thus, the void will not propagate radially beyond the blocked subassembly.

As an added degree of conservatism in the safety analysis, however, the void is assumed to spread immediately to the six subassemblies adjacent to the voiding assembly. The reactivity insertion associated with the voiding of seven subassemblies, rather than one, provides a more stringent test of the protection system.

Analyses were performed to determine the reactivity effect of complete voiding of the seven most reactive fuel subassemblies. The total reactivity insertion resulting from the voiding of the seven subassemblies was conservatively taken to be 0.0027 δk , which is a factor of two above the calculated values. Voiding was assumed to occur in 0.010 second, which is highly conservative, since it is about the period of time required to void a subassembly when molten fuel from the pins is injected into the coolant. The accident was followed by a normal scram initiated by detection of an 11% overpower condition. The results of the void accident are presented in Figures 14-8 and 14-9. The maximum power of 9400 MW occurs at 0.012 second. The maximum fuel temperature (center line pin, central core segment) in the peak channel, 4000 F, occurs 0.5 second after the void begins to form. (The coolant outlet temperature rise in the average channel is approximately 60 F. The voided elements are probably destroyed during this transient period. However, the voiding of the seven most reactive elements produces a reactivity insertion which does not jeopardize the integrity of the remainder of the core.)

14.5.3.4. Loss of Flow From One Primary Pump

The integral pot design for the 1000-MWe LMFBR has significant advantages in the primary coolant system. First, it contains six primary coolant pumps, which take suction from the integral pot and deposit the fluid into the core vessel where the flow is mixed. After passing through the core, the coolant returns to the pot through any one of six IHXs. Even though there is a one-to-one correspondence between pumps and IHXs, no one pump can be identified with one particular heat exchanger.

There are many situations that could result in the loss of flow from one primary pump. Two are considered in this subsection: (1) the double-ended failure of the pipe between the pump and the core vessel, and (2) a failure as if the suction to the pump were blocked. The asymptotic core flow from the five operating pumps differs considerably for the two cases—20,000 lb/sec and 24,500 lb/sec, respectively. These compare with the normal flow at full power of 25,738 lb/sec. Figure 14-10 gives a graphical illustration of the flow and its distribution.

Figure 14-8. Void Accident, Seven Assemblies Void -
Power Vs Time

Figure 14-9. Void Accident, Seven Assemblies Void -
Maximum Fuel Temperature Vs Time

Figure 14-10. Asymptotic Flow Curves - Loss of One Coolant Loop

In order to observe the complete transient, no scram setpoints are incorporated in the analysis that used FORE II. In both cases, fuel heating and radial expansion reactivity feedback turned the excursion around. The principal results are tabulated in Table 14-4 and are shown graphically in Figures 14-11 and 14-12. In neither case was the transient continued until the steady-state thermal conditions were established.

14.5.3.5. Loading Accident

The design of the 1000-MWe LMFBR is such that an accidental misloading of a fuel assembly is not credible. This is primarily due to the fact that the control assemblies must be in the core and disconnected from their drives while refueling operations are underway. Operation of the refueling mechanism is not physically possible until the drives are detached from the control assemblies. The design of the control assemblies is such that they cannot be ejected from the core by hydraulic pressure, even at flows normally associated with full power operation. Such an accident would require many simultaneous, independent, undetected errors, if it were possible at all. For this reason, loading accidents have not been analyzed.

Table 14-4. Temperatures and Flows Following a Pump Loop Malfunction

<u>Parameter</u>	<u>Double-ended failure</u>	<u>Pump blockage</u>
Average pin temp, F	130	
Surface of fuel (initial/max)	1303/1350	1303/1312
Core outlet (initial/max)	1100/1186	1100/1112
Power level, MWt (initial/peak)	2450/2453	2450/2474
Flow, lb/sec (initial/asymptotic)	25,738/20,000	25,738/24,500

Figure 14-11. Temperature Transient - Double-Ended Break

Figure 14-12. Double-Ended Break at Nozzle - Power
Vs Time

14.6. Design Basis Accident Analysis

The design for most of the plant is based on a chosen set of normal operating conditions, along with the conditions existing during shutdown and the conditions existing along the chosen path between shutdown and normal operating conditions. However, there are certain aspects of the design for which the design basis is the desire to provide for the safety of the public and the plant personnel during and following severe accidents. For these reasons, the design requirements for the reactor cavity, including the cover structure and the reactor building, are set by the requirement that the containment withstand any energy released as a result of a nuclear burst, energy transfer from fuel to coolant, and the decay of fission and activation products. The design requirements must contain significant margins for conservatism.

This section discusses in some detail the nature of a number of severe accidents and their effects on the plant and the environment. It is seldom possible to satisfactorily define a series of initiating events for accidents of this nature. In the discussions that follow, no attempt will be made to establish the events leading to the nuclear excursion; however, in some instances it will be pointed out rather clearly that the normally considered design basis accident for fast oxide reactors cannot be attained. It is included here to establish that the public and the plant personnel are protected even for this accident, since it is realized that the safety requirements are often legislated by regulatory bodies and need not be possible in nature.

This section is broken down into several subsections to consider the analysis of the energy release mechanisms which establish the design of the reactor cavity, the boundary of which forms the energy-absorbing barrier and the sodium barrier. Since there are a number of seals for components and plugs in the reactor cover structure, the reactor cavity barrier cannot be considered as the plutonium, fission product, and activation product barrier. Thus the reactor building becomes the containment barrier for these radioactive materials.

The design requirements that the reactor cavity be capable of containing the primary sodium, and that the intermediate system be contained, were set before adequate sodium fire analyses could be performed.

The requirements have not changed. However, analyses have been performed to establish the effects of a sodium release on the reactor building design.

14.6.1. Energy Production for Large Reactivity Insertions

The energy release to the coolant following the melting of a significant segment of the core is felt to be the most realistic evaluation for the requirements of the energy and sodium barriers. Since no source for a large, positive, reactivity insertion could be determined analytically, several ramps were assumed as the reactivity driving function, starting from steady state at full power. The FORE II code was used as the calculational tool for the resulting nuclear excursion and the analysis of the core thermal condition at the time of the release of molten fuel to the coolant. The analysis did not include the axial blankets, since the most detailed thermal analysis possible was desired in the core. Thus, the reactivity coefficients were different from those given in subsection 14.4. The clad failure was assumed to occur when the clad was contacted with molten fuel. During the nuclear excursion the stored energy of the core increased until the molten fuel contacted the cladding. At this point in time, the molten fuel in the core was released to the coolant associated with the pin length that contained the molten fuel.

The nuclear excursion was stopped with cladding failure, and a hydrodynamic calculation was made, assuming that all of the molten fuel was released instantaneously to the sodium in the coolant channels associated with the molten fuel. The pressure generated by the sodium coolant owing to the release of the molten fuel was relieved upward and caused the sodium above the core to be propelled toward the cover structure. The maximum pressure was approximately 100 psig on the cover structure; the pressure buildup is presented graphically in Figure 14-13.

A \$60/sec ramp was used to establish an upper limit to the fraction of the core which could be molten and released to the coolant, since the time response of the power level to \$60/sec is so fast that the thermal energy transfer from the fuel through the clad to the coolant is very small. Thus, it would appear to form an upper limit to any processes of reactivity insertion for a core filled with sodium, based on the

Figure 14-13. Molten Fuel-Sodium Interaction - Cover Gas Pressure and Sodium-Fuel Bubble Radius as a Function of Time

failure criteria indicated above. If the rate of reactivity insertion is decreased significantly, the percentage of the core which is molten at the time when pin failure occurs is reduced as the heat transfer becomes more significant. At about \$4/sec to \$5/sec, the protection system will be able to maintain the integrity of the fuel, but some central melting of the peak pins will occur. It should be emphasized that we have found no credible accident that even approaches these lower rates for a period of time long enough for the total reactivity insertion to approach one dollar. Thus, if we considered only those accidents which are credible even though of a low probability, the design requirements of our reference concept would be fixed by the normal operation design requirements for the reactor vessel. However, the design requirements of the cover structure for the reference concept design have been conservatively set to contain the energy release of 15,000 MW/sec associated with the \$60/sec ramp insertion. For this ramp, approximately 14% of the fuel was molten.

14.6.2. Nuclear Excursion for Drained Core

In the analysis of fast reactors for licensing, one type of accident has always been considered when establishing the design requirements for containment structures. This accident has its origin in the following facts: (1) the core fuel is not assembled in its most reactive arrangement, (2) criticality can be achieved without any coolant in the core, and (3) the reactor core has a negative coolant density coefficient. The design of the energy barrier requires a prediction of the magnitude of the energy released in an accident during which the core is disassembled due to the vapor pressure of the fuel.

In the following analyses and discussions of this type of accident, it is assumed that the coolant has been permanently removed from the core while the reactor is at full power.

14.6.2.1. Reactivity Addition Upon Core Collapse

A simple one-dimensional, diffusion-theory calculation was performed to determine the criticality status of the reference core before and after collapse. For this calculation, it was assumed that the fuel began melting and ran into the cooler lower axial

blanket, where, after completely filling the vacant sodium spaces, it froze. The remainder of the molten fuel collected in a pool above the lower axial blanket. When the melting was complete, the upper axial blanket collapsed as a unit onto the surface of the pool of molten fuel. It was further assumed that core melting was complete and that the radial blanket remained intact. To obtain an upper limit for the multiplication of the final configuration, zero radial buckling was used in the axial calculation. All molten fuel was compacted to 100% of theoretical density, and no control material was present in the final configuration.

Initial K	1.025
Final K	1.054 max
δK	0.029

Approximately \$8 of reactivity is added as a result of the collapse of the drained core. If it is assumed that the core materials accelerate at 1 g to final configuration, then the rate of reactivity insertion is about \$40/sec. The insertion rate does not take into account the effect of any resisting force on the spatial dependency of the melting processes, and hence, it is very conservative.

The probability of this type of accident occurring in the reference design is extremely small owing to the nature of the design, i. e., the fuel pins are supported from the top. Should failure of the cladding occur because of melting, the lower part of the core would fall away from the upper part.

14.6.2.2. Energy Release From Disassembly

In a reactor meltdown accident which involves the formation of a secondary critical assembly, we can postulate a sequence of events that leads to explosive disassembly of the core. The disassembly of the core is caused by high pressures developed in the fuel during the course of the power excursion initiated when the fuel moves from its original configuration to one of higher reactivity. It is usually assumed, in the analysis of such accidents, that the reactivity rises above prompt critical and causes a power excursion to develop. The disassembly pressures remain small until the temperature becomes

high enough to produce high saturated vapor pressures in the fuel. By the time the disassembly pressures become significant, the reactor is usually on an extremely short exponential period, and disassembly occurs with potentially large energy releases. In order to predict the magnitude of the energy generated as a result of a meltdown accident, modified Bethe-Tait calculations were performed using the WEAK EXPLOSIONS code.

This code requires that the reactor core material be homogenized into one uniform sphere. Since the densities in the zones of the reference core are essentially the same, the use of an equal volume spherical core will overestimate the energy release and will tend to give a conservative estimate of the energy release.

In the WEAK EXPLOSIONS code, the only factor (other than fuel movement) operating to limit the power at which disassembly occurs is the Doppler coefficient, hence this coefficient is of central importance in the meltdown analysis. Since the reactor was assumed to be operating at full power when the excursion was initiated, the Doppler feedback began at that point. The sensitivity of the energy release to the magnitude of the Doppler coefficient was investigated for a broad range of values. The change in the Doppler temperature coefficient was considered to be proportional to T^{-1} . The results of the calculations of energy release for a ramp reactivity insertion of \$50/sec are shown as a function of the magnitude of the Doppler coefficient in Figure 14-14. The dependence of the energy release on the magnitude of the Doppler coefficient is somewhat weaker than might have been expected. The primary cause of this behavior is the relative flatness of the power and fuel self-worth distributions. As a result of the flattened distributions, a large fraction of the core will experience energy densities sufficient to produce pressures high enough to cause disassembly. It would be desirable to have very large pressures generated in a very small volume of the core in order to produce disassembly with the smallest possible explosive energy release. Power flattening, of course, produces the opposite effect. Similar results have been reported elsewhere.²

Figure 14-14. Energy Generation - Core Collapse

14.6.3. Sodium Fires

In reactors that use sodium as a coolant, it is necessary to examine the influence of sodium on the design of containment structures. Although the energy boundary is designed to prevent primary sodium release to the reactor building, and all intermediate loops are doubly contained within the reactor building, we must examine the effects to determine whether the containment structure could be jeopardized and radioactive material could be released to the environment. Such fires constitute the greatest threat to personnel, plant, and environment. The fires examined fall into two categories—reactor vessel fires and reactor building fires. The computational model is based on the reaction rates between the surface of a stagnant pool of sodium and the available oxygen; heat transfer to the structure and the atmosphere by radiation, conduction, and convection are considered. The reaction rates are based on experimental data and are described as a function of oxygen concentration.

$$R = H_c C$$

where

R = sodium burning rate, lb/hr-ft²

C = oxygen concentration, lb/ft³

H_c = reaction rate constant
(function of flame temperature)

The geometry used in the cell fire calculations is presented in Figure 14-15.

Reactor vessel fires (cell 1 fires) were examined as a function of the area of the opening (A_H) between the reactor vessel and reactor building (cell 2). Reactor building fires were investigated for (1) expulsion of sodium from the reactor vessel, and (2) pools of stagnant sodium at the charge face. The principle results of the reactor vessel fires are tabulated in Table 14-5 as a function of the size of the opening (plug removed) between the reactor vessel and the reactor building. The air in the reactor building, the building itself, and all the structures inside it were initially at 80 F. The argon atmosphere, the sodium in the reactor vessel, the vessel itself, and all the structures inside the vessel were initially at 1100 F. The slight increase in the density of argon, as compared with air, at the same temperature was not sufficient to offset the higher argon temperature; thus, natural circulation currents are set up fairly rapidly.

The two types of reactor building fires were started using the same initial conditions. Both fires assumed that the entire cover structure was covered with burning sodium. They differed by considering a spray ejection into the reactor building of 500 pounds of sodium at 1100 F. It should be pointed out that, in the case of the sodium expulsion, the effects of the spray fire are calculated first and used as initial conditions for the pool fire. This has the effect of raising the initial temperature and lowering the initial oxygen concentration for the pool fire. The temperature and pressure transients for a reactor building fire with sodium expulsion are plotted in Figure 14-16. The results of these two analyses are tabulated in Table 14-6.

These analyses show that, even though some small sodium fires may be permitted in the reactor building, the design

Figure 14-15. Geometry for Sodium Fire Calculation

requirement that the plugs be retained was a good choice. When removed, even the smallest plug in the cover structure produced a fire which raised the containment temperature approximately 100 degrees. In addition, the analyses showed that, even though the system is designed to contain all the sodium, reasonable quantities could be ejected into the reactor building without exceeding the design conditions.

Table 14-5. Maximum Temperatures and Pressure Buildup During a Reactor Vessel Fire

A_H , ft ²	$(T_1)_{\max}$, F	$(T_2)_{\max}$, F	$(\Delta P_2)_{\max}$, psig
1.069 (rod drive removed)	1100	163.7	3.2
23.34 (pump plug removed)	1138	295.4	5.5
52.38 (IHX plug removed)	1100	376	7.6
2124.0	1100	548.6	11.9

where

$(T_1)_{\max}$ = maximum temperature in reactor vessel, F
 $(T_2)_{\max}$ = maximum temperature in reactor building, F
 $(\Delta P_2)_{\max}$ = maximum increase in pressure in the reactor building, psig

Table 14-6. Reactor Building Maximum Temperature and Pressure During a Sodium Fire

Case	T_o , F	T_s , F	$(T)_{\max}$, F	$(\Delta P)_{\max}$, psig
No sodium expulsion	80.0	80.0	554.7	12.3
500-lb sodium expulsion	80.0	183.9	553.0	14.1

where

T_o = initial containment temperature
 T_s = containment vessel temperature when Na pool begins burning (after Na expulsion), F
 $T_{\max}$ = maximum containment vessel temperature, F
 $P_{\max}$ = maximum containment vessel pressure increase, psig

Figure 14-16. Spray Fire in Reactor Building -
Cell 2 Temperature and Pressure
Increase Vs Time

14.6.4. Design Basis for Environmental Requirements on Reactor Building

In general, federal restrictions on radiation dose levels resulting from major accidents are not to be regarded as acceptable dose levels but as a guide in comparing one plant site with another. The acceptability of the calculated dose, as compared with the 10 CFR 100 dose level guidelines, appears to be a function of the probability of the accident. It is also observed that the AEC restrictions on permissible dose levels are becoming more stringent all the time. Two permissible dose levels are selected for comparison, considering protection of the general public. The first is a site evaluation condition taken from 10 CFR 100, and the second is based on guideline for fuel restrictions that might be imposed for accidents that are considered exceedingly improbable. The guidelines for direct dose level restriction are more in line with those set down in 10 CFR 20 for exposure of individuals in restricted areas per calendar quarter. The permissible thyroid dose selected is chosen primarily for comparison. The permissible dose criteria for the general public are as follows:

1. Site acceptability — At the exclusion radius (taken to be 600 meters, same as for DUKE site) whole body direct dose shall be limited to 25 Rem or thyroid dose to 300 Rem integrated over the first 2 hours following the accident.

2. Credible accident — At the exclusion radius whole body direct dose shall not exceed 3 Rem or thyroid dose 25 Rem integrated over the first 2 hours following the accident.

Two additional sets of criteria were selected considering protection of plant personnel in a shielded control room (outside the containment building) and protection of workers performing maintenance operations during days 5 through 30 following the accident. Limited- and extended-exposure criteria were chosen. The limited-exposure permissible direct dose level is the once-in-a-lifetime emergency dose. The extended-exposure permissible direct dose level is the maximum permissible quarterly dose for workers in restricted areas. (The plant personnel and maintenance workers would be protected from inhalation of radiation; therefore, no inhalation criteria have been set.)

The time periods chosen are meant to represent situations which are the most conservative for limited and extended access. The 8-hour level was chosen for plant personnel, since operators in the control room would have to remain in the control room immediately after the accident to perform operations that could be mandatory to ensure no further propagation of the accident or to mitigate the consequences of the accident. For postaccident maintenance, the period from day 5 through day 30 following the accident was selected on the basis of the levels of exposure to individuals performing cleanup operations after it has been determined that such operations are possible.

The criteria are as follows:

Plant Personnel

1. Limited access — Whole body direct dose inside the control room must be limited to 25 Rem integrated over the first 8 hours following the accident.
2. Extended access — Whole body direct dose inside the control room must be limited to 3 Rem integrated over a 30-day period following the accident.

Postaccident Maintenance

1. Limited access — Whole body direct dose must be limited to 25 Rem integrated over days 5 through 30 following the accident immediately outside the containment building.
2. Extended access — Whole body direct dose must be limited to 3 Rem integrated over days 5 through 30 following the accident immediately outside the containment building.

The analyses that determined the reactor building shielding requirements are given in Volume 2, section 3, along with the methods used in the evaluation.

14.6.5. Emergency Core Cooling Requirements

The emergency core cooling system and the backup cooling system are discussed in other sections of this report (see Volume III, section 8). The basic heat removal requirements are given here, based on a simplified model. This model permits the reactor

coolant and internals to come to a temperature of 1300 F. Thus the heat removal requirements from the core have this base level, which permits a delay before the backup system must be operational. The energy stored and produced up to any given time can be found by integrating the production rate and adding to this the stored energy, i.e.,

$$q = \text{fission product decay heat} + \text{sodium-24 decay heat} \\ + \text{stored energy} + \text{excursion energy}$$

The stored and excursion energies are approximately 5×10^6 and 16×10^6 Btu, respectively. The fission product decay heat comes from two types of operation: (1) long-term, steady-state operation (assumed to be design power level), and (2) the excursion, which is short-lived.

$$q_{FP} = \text{energy generated by fission product decay, Btu} \\ = [0.18t^{0.8} - 0.07t^{-0.2}] \times 10^6 \\ t = \text{time, sec}$$

The sodium-24 decay heat generation is obtained from the following expression:

$$q_{^{24}\text{Na}} = 369 \times 10^5 (1 - e^{-1.283 \times 10^{-5} t}), \text{ Btu}$$

where $q_{^{24}\text{Na}}$ is the energy generated by ^{24}Na decay. The combined energy release becomes

$$q = [0.18t^{0.8} - 0.07t^{-0.2} + 36.9(1 - e^{-1.283 \times 10^{-5} t}) + 21] \times 10^6 \text{ Btu}$$

APPENDIX 14-A
Design Criteria

1. Introduction

The rapid growth of nuclear power stations in the past few years has resulted in the need for criteria by which different plants could be compared. Such criteria have been prepared by the U.S. Atomic Energy Commission ("General Design Criteria for Nuclear Power Plant Construction Permits" issued November 22, 1965, and July 10, 1967).

As the sodium-cooled fast breeder reactor economy develops, similar criteria will be required by the reactor manufacturers and the licensing agencies to ensure uniformity in reactor quality requirements. A set of general design criteria has been prepared to serve as a design guide for the 1000-MWe LMFBR design study. These design criteria, presented in section 2, were prepared by first upgrading present-day water reactor criteria ("General Design Criteria for Nuclear Power Plant Construction Permits," July 10, 1967) to cover the features of an LMFBR and, second, upgrading these LMFBR criteria to the anticipated 1980 state of the art.

In the interval from 1967 to 1980, changes in LMFBR technology are anticipated to be primarily in the area of engineering development; basic concept changes are not anticipated. Areas of development considered are methods, codes and standards, and hardware.

The methods development is expected to permit much more sophisticated analyses of reactor designs, especially the transient response. While these methods will permit predictions of reactor response to be made with much more confidence than is possible today, the methods are not expected to result in design criteria changes.

By 1980, the codes and standards used for design, fabrication, inspection, and materials specification are expected to be highly developed. Experience gained in the preparation of codes for water reactors and the operating experience gained in the next 10 years should permit those codes applicable to an LMFBR to be more highly developed in 1980 than the present-day codes for water reactors.

2. General Design Criteria

2.1. Overall Plant Requirements

2.1.1. Quality Standards

Those systems and components of reactor facilities which are essential to the prevention of accidents that could affect the

public health and safety or to the mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Generally recognized codes or standards on design, materials, fabrication, and inspection shall be used.

2.1.2. Performance Standards

Those systems and components of reactor facilities which are essential to the prevention of accidents that could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public over the plant life, those forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect (1) appropriate consideration of the most severe combination of these natural phenomena that have been recorded for the site and the surrounding area and (2) an appropriate margin for withstanding forces greater than those recorded to allow for uncertainties in the historical data and their suitability as a basis for design.

2.1.3. Fire Protection

The reactor facility shall be designed (1) to minimize the probability of such events as fires and explosions and (2) to minimize the potential effects of such events on safety. Noncombustible and fire-resistant materials shall be used wherever practical throughout the facility, particularly in areas containing critical portions of the facility such as the containment, control room, and components of engineered safety features.

The design and arrangement of plant equipment containing metallic sodium shall be such that equipment failure and the possible subsequent sodium fire will not propagate additional failures that would further endanger the plant.

2.1.4. Shared Systems

Reactor facilities shall not share systems or components unless it is shown that safety is not impaired by the sharing.

2.1.5. Redundancy

If the reactor facility has redundant systems, such as multiple primary coolant loops, then the systems may be considered independent if it can be demonstrated that the independence is not compromised in an unusual transient situation.

Multiple failures resulting from a single event shall be treated as a single failure.

2.2. Protection by Multiple Fission-Product Barriers

2.2.1. Reactor Core

The reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel and cladding damage limits. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation; appropriate margins will be allowed for uncertainties and for transient situations that can be anticipated, including the effects of the loss of power to circulation pumps, tripping out of a turbine-generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power. (The transient situation shall be assumed to involve complete loss unless redundancy can be demonstrated.) The core shall be designed to reduce to a very low level the probability of fuel subassembly blockage by regular shaped objects, i.e., by flat plates or by segments of conically shaped objects.

The prompt power coefficient shall be negative under all credible circumstances. Local power coefficients shall be such that the reactivity control system will be capable of ensuring the stability of the power distributions under all credible situations.

The core design, together with reliable controls, shall ensure that power transients and/or oscillations that could cause fuel damage in excess of acceptable limits are not possible.

The core shall be designed so that, in the event of sodium loss from the most reactive fuel element, the result is a reactivity insertion of less than one dollar.

2.2.2. Reactor and Intermediate Coolant Boundaries

The coolant boundaries shall be designed and constructed so that they have a very low probability of gross rupture or significant uncontrolled leakage through the design lifetime of the reactor.

The design shall provide for the detection of leakage from the reactor and intermediate coolant boundaries.

2.2.3. Containment

Containment shall be provided for all radioactive materials and for plutonium in sufficient quantities to compromise the health and safety of the public. Containment structures shall be designed to sustain the initial effects of gross equipment failures without the loss of their required integrity, and, together with such other engineered safety features as may be necessary, to retain, for as long as the situation requires, the functional capability to protect the public.

2.3. Nuclear and Radiation Controls

2.3.1. Control Room

The facility shall have a control room from which actions to maintain the safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to the equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without exposing personnel to radiation in excess of 10 CFR 20 limits. Adequate physical protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility. This physical protection shall be provided against plant missiles and natural phenomena such as earthquakes, tornadoes, flooding conditions, wind, ice, and other local site effects. The design bases shall reflect (1) appropriate consideration of the most severe combination of these natural phenomena that have been recorded for the site and the surrounding area and (2) an appropriate margin for withstanding forces greater than those recorded to allow for uncertainties in the historical data and their suitability as a basis for design.

2.3.2. Instrumentation and Controls

Instrumentation and controls shall be provided as required to monitor and maintain variables within the prescribed operating ranges. The design shall provide for monitoring and maintaining control over the fission process throughout core life and for monitoring all conditions that can reasonably be anticipated to cause variations in the reactivity of the core.

Core protection systems, together with associated equipment, shall act automatically to prevent or to suppress conditions that could result in exceeding the acceptable fuel damage limits. The protection action shall go to completion.

Protection systems shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, or from accident conditions.

2.4. Reliability and Testability of Protection Systems

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

2.5. Reactivity Control

The reactivity control system shall be capable of sustaining any single malfunction, such as the unplanned continuous withdrawal of a control rod, without causing a reactivity transient that could result in exceeding acceptable fuel damage limits.

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on the rates at which reactivity can be increased. This will ensure that the potential effects of a sudden or large change of reactivity cannot disrupt the core, its support structures, or other vessel internals enough to impair the effectiveness of emergency core cooling.

2.6. Reactor Coolant Boundary

The reactor coolant boundary shall be designed to minimize the probability of rapidly propagating failures. Consideration shall be given

(1) to the state of material stress under static and transient loadings, (2) to the quality control specified for materials and component fabrication to limit flaw sizes, and (3) to the provisions for the control of service temperature and irradiation effects that may require operational restrictions.

The reactor coolant system shall be designed to permit access to critical areas of the reactor coolant boundary for inspection, testing, and surveillance by appropriate means to assess the structural and leak-tight integrity of the boundary components during their service lifetime.

2.7. Engineered Safety Features

2.7.1. General

The facility shall have engineered safety features to back up the safety features built into the core design, the reactor coolant boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any accident up to and including the design basis accident.

All engineered safety features shall provide high functional reliability and ready testability. In determining the suitability of a facility, the degree of reliance on and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected (where appropriate) during the life of the plant.

An alternate power system shall be provided with adequate independence, redundancy, capacity, and testability to permit the required functioning of the engineered safety features, assuming that a single component in the system fails.

The engineered safety features shall be protected against thermal effects, dynamic effects, and missiles that might result from plant equipment failures. Engineered features, such as emergency core cooling and containment heat removal systems, shall provide sufficient performance capability to accommodate the partial loss of installed capacity and still fulfill the required safety function. As a minimum, the engineered safety features shall provide this required safety function

assuming the failure of a single component, such as a pump, valve, or the like.

The engineered safety features shall be designed to avoid any action that might accentuate the adverse after-effects of an accident.

The performance of the engineered safety features and the condition of the reactor shall be monitored by instrumentation during and after an accident.

2.7.2. Emergency Core Cooling Systems

The emergency core cooling systems remove decay heat from the reactor core after an accident that results in the loss of the normal heat removal path. Loss of the normal heat removal path may occur by either loss of the heat sink or loss of access to the heat sink.

At least two heat removal mechanisms, preferably of different design principles and each capable of accomplishing emergency core cooling, shall be provided. The heat removal mechanisms and the core shall be designed to prevent damage to the fuel and cladding that would interfere with the emergency core cooling function.

The heat removal mechanisms shall be capable of fulfilling their functions in any situation wherein the normal path of heat removal is compromised.

The heat removal mechanisms shall share neither active components nor other features or components unless it can be demonstrated that (1) it can be readily ascertained during reactor operation that the shared feature or component is capable of performing its required function, (2) failure of the shared feature or component will not initiate a design basis accident, and (3) the ability of the shared feature or component to perform its required function is not impaired by the same effects of the design basis accident and is not lost during the entire period during which this function is required following the accident.

Wherever possible, the design shall provide for physical inspection of all critical parts of the heat removal mechanisms, including those parts within the reactor vessel. Any exception shall be justified.

The design shall provide for periodic testing of the delivery capability of the heat removal mechanisms at a location as close to the core as is practical.

Provisions shall be included for testing, under conditions as close as practical to design condition, the full operational sequence that would bring the emergency heat removal mechanisms into action, including the transfer to alternate power sources.

2.7.3. Containment

2.7.3.1. Reactor Building

The function of the reactor building is to contain, without failure, all of the material and the energy released by any accident up to and including the design basis accident. The reactor building shall be designed to provide sufficient biological shielding and retention of radioactive materials and plutonium to meet the requirements of 10 CFR 100.

The reactor building, including its access openings and penetrations and any necessary heat removal systems, shall be designed to accommodate any energy release, pressures, and temperatures that result from any accident (up to and including the design basis accident) without exceeding the leakage limits for radioactive material and plutonium; this provision includes a considerable margin for the effects that could occur as a consequence of failure of the emergency core cooling system.

The reactor design shall provide sufficient inert barriers around all liquid sodium systems within the reactor building to ensure that a sodium fire resulting from any accident (up to and including the design basis accident) will not produce temperatures and/or pressures in excess of the reactor building design limits.

The principal load-carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures, under normal operating and testing conditions, are not less than 30 F above the nil ductility transition (NDT) temperature.

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Where active heat removal systems would be needed under accident conditions to prevent the containment pressure and temperature from exceeding the design pressure and temperature, at least two systems, preferably of different principles and each with full capacity, shall be provided.

Water or equipment containing water or other hydrogen compounds shall be excluded from the reactor building, unless it can be demonstrated that the material can neither (1) endanger the integrity of the reactor building if it reacts chemically with sodium, nor (2) result in a significant reactivity insertion.

The reactor building design shall provide for integrated leakage rate testing at design pressure after erection and installation of all penetrations is complete. The leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance. The design shall provide for the testing of penetrations having resilient seals or expansion bellows and for the demonstration of leak-tightness at design pressure at any time. The design shall provide for the testing of the functional operability of valves and associated apparatus essential to the containment function to establish that no failure has occurred and to determine that valve leakage does not exceed acceptable limits.

The design shall provide for testing the functional operability of personnel locks and equipment locks or passageways with valves and associated apparatus essential to the containment function to establish that no failure has occurred and to determine that the leakage does not exceed acceptable limits.

2.7.3.2. Auxiliary Containment

The function of the auxiliary containments is to prevent the undue release of fission products from equipment located outside the reactor containment. "Undue release" is defined as any release of radioactive nuclides to the environment that exceeds the limits of 10 CFR 20.

The nature of the auxiliary containments shall be determined by the physical form and the quantity of the contained radioactive material. For example, equipment that contains radioactive

liquids may be contained in cells or containers large enough to contain all credible leakage. Penetrations of any auxiliary containment—for example, drains or vents—shall be designed so that positive isolation devices prohibit the unexpected release of any contained radioactive material. The auxiliary containments shall be cooled as necessary to prevent their exceeding the design pressure and/or temperature.

2.8. Fuel and Waste Storage System

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations will be emphasized over procedural controls.

Reliable heat removal systems shall be designed to prevent any damage to the fuel in the storage facilities that could result in the release of radioactivity to plant operating areas or to public environs.

Shielding for radiation protection shall be provided in the design of the spent fuel and waste storage facilities, as required to meet the requirements of 10 CFR 20.

2.9. Plant Effluents

The facility design shall provide for maintaining control over the plant radioactive effluents, whether gaseous, liquid, or solid. The appropriate holdup capacity will be provided for retaining gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (1) on the basis of 10 CFR 20 guidelines for normal operations and for any transient situation that might reasonably be anticipated, (2) on the basis of 10 CFR 100 dose level guidelines for potential reactor accidents of very low probability of occurrence, except that reduction of the recommended dose levels may be required where high population densities can be affected by the radioactive effluents. An environmental monitoring program shall be maintained to confirm that excessive effluent releases have not occurred.

3. Discussion

The criteria presented in section 2 are briefly discussed here to illustrate how the 1980 criteria compare with the July 10, 1967, criteria.

Even though the 1967 criteria have been subjected to considerable criticism, they do represent several years of development, and thus, they were considered to be the best available starting point for the development of the criteria for an LMFBR. Only those criteria which represent significant changes are discussed herein.

3.1. Overall Plant Requirements

The quality and performance standards required of the 1980 LMFBR will be much more strict than those now applied to water reactors. These requirements will reflect a significant period of operating experience with large water reactors.

Fire protection in a reactor cooled with liquid sodium presents a new design restraint. Features must be built into the LMFBR to make the probability of a sodium fire extremely low, and in the unlikely event of a sodium fire, to limit the extent of the fire and the affected equipment. The use of fire breaks and/or fire extinguishing systems is indicated.

3.2. Protection by Multiple Fission Product Barriers

The major departure from present-day thinking for the 1980 criteria is the amount of reactivity associated with sodium voiding—it may be much greater than is now practical. By 1980, calculational models and methods which have experimental verification should be available to predict with great certainty the sodium void reactivity and its effect on core operation. Also, models and methods which accurately describe the movement of fuel in solid or molten states should be available. These tools should permit the design and analysis of the transient response of design features which will limit the consequences of a sodium voiding accident. Research and development programs in the area of instrumentation should produce instruments that give adequate warning of the approach to sodium boiling conditions so that remedial action may be taken before the onset of boiling. Thus, it was concluded that the reactivity problems associated with sodium voiding can be controlled in the 1980 criteria, provided the reactor does not become prompt critical.

3.3. Nuclear and Radiation Controls

These criteria are essentially unchanged from those of July 10, 1967.

3.4. Reliability and Testability of Protection Systems

These criteria are essentially unchanged from those of July 10, 1967.

3.5. Reactivity Control

These criteria specify that only one reactivity control and safety system is required in the 1980 LMFBR. This is a departure from present-day criteria, which require two systems. The requirement for a second shutdown system results from consideration of (1) shutdown reactivity requirements, and (2) the time response of the shutdown reactivity.

The reactivity requirements for an LMFBR are quite different than those for a thermal reactor. The reactivity requirements for PWR and LMFBR plants are contrasted below.

	<u>$\Delta K/K\%$</u>	
	<u>LMFBR</u>	<u>PWR</u>
Cold-hot	<1	3 - 4
0 - 100% power	<1	4 - 5
Burnup effects	<3	16
Total	<5	23 - 25

In addition to these, xenon and samarium reactivity effects are not present in an LMFBR because of the high energy spectrum; thus, the delayed reactivity insertion requirements do not exist.

If a second shutdown system were required for the LMFBR, the requirements for reliability, the time response of reactivity insertion, and the position of safety control elements would be the same. Thus, any event that could make one system unavailable in the core would have the same effect on both systems. The results obtained from the use of two different systems can be obtained by providing multiple sources for driving the rods into the core. For instance, control rod drives which provide powered and inertial follow-up to a gravity drop have been proposed for the follow-on reference design. The inertial

follow is provided, since the motors are continuously operating, and it is not a matter of trying to start a motor.

To account for a local fault which would prevent insertion under any circumstances, such as a stuck rod, the large number of units in the shutdown system and the excess shutdown reactivity would guarantee adequate shutdown. In the event of a local fault, such as a stuck rod, the large number of units in the shutdown system and the stuck rod criteria would permit adequate shutdown.

3.6. Reactor Coolant Boundary

These criteria are essentially the same as those of July 10, 1967, except for changes to reflect the use of austenitic materials and low pressure, instead of ferritic materials and high pressure.

3.7. Engineered Safety Features

The functional requirements of the 1980 criteria are essentially identical to those of the 1967 criteria. The containment requirements have been changed to reflect the presence of liquid sodium and the possibility of a sudden energy release during the design basis accident. The 1980 criteria require sufficient inert barriers between liquid sodium and the atmosphere to prevent a sodium fire.

System information in the control room to ensure proper operation of the engineered safeguards during and after any accident, up to and including the design basis accident, is required.

3.8. Fuel and Waste Storage Systems

No changes from the criteria of July 10, 1967.

3.9. Plant Effluents

The 1967 criteria have been augmented by the requirement for an environmental monitoring program to confirm the adequacy of effluent control features.

APPENDIX 14-B
Analyses for Preliminary Reference Core
Without BeO

1. Introduction

The characteristics of several preliminary core designs were studied during the evolution of the final reference design. Among the cores studied before the reference design was finalized was the preliminary reference core in which BeO was used as a fixed moderator. The results of these preliminary studies indicated that a fixed moderator was not necessary to ensure the integrity of the core during credible accident conditions; for that reason, it was decided that the final reference design would not contain a fixed moderator. The inclusion of the fixed moderator did not significantly affect the design requirements of the reactor protection system. Since it became necessary to complete a parametric survey of control parameters before the reactivity coefficients for the final reference designs were available, reactivity coefficients were estimated for a core having the geometry of the preliminary reference core but without BeO fixed moderator. The estimates were based on the conclusions of the "Spoiled Vs Unspoiled Core Trade-Off Studies." The resulting core was referred to as the preliminary reference core without BeO.

The major effort in the parametric studies was made with the reactivity coefficients of the preliminary reference core without BeO moderator; the results of this parameter study had a major influence on the optimization of control parameters for the reference design safety system. Once the reactivity coefficients were known for the reference design, certain pertinent calculations were performed to compare and adjust control parameters. Because of its importance in indicating control parameter trends, the preliminary reference core is described in the following tables.²⁰

Table 14-B-1. LMFBR Preliminary Reference Core Description

	Core	Radial blanket	Axial blanket	Radial reflector	Axial reflector
Outer diameter, in.	120.03	144.08	120.03	167.70	120.03
Height, in.	37.42	37.42	12	37.42	11.81
Volume, liters	6938.7	--	4450.4	4380	4380.2
No. of fuel assemblies	288	138	288	--	--
Fuel pin OD, in.	0.290	0.510	0.290	--	--
Fuel pin pitch, in.	0.340	0.542	0.340	--	--
Clad thickness, in.	0.010	0.018	0.010	--	--
Assembly pitch, in.	6.461	6.461	6.461	--	--
Area/assembly, in. ²	36.152	36.152	36.152	--	--
Fuel, % dense	85	90	85	--	--
BeO, % dense	80	--	--	--	--
B ₄ C, % dense	80	--	80	--	--
Steel, type	304 SS	304 SS	304 SS	304 SS	304 SS
Avg temp, F					
Fuel	2500	2000	2000	--	--
Sodium	975	975	975	975	975

Volume Fractions

	Fuel	Na	Steel	BeO	B ₄ C
Core zone 1	0.4157	0.3752	0.1519	0.0461	0.0111
Core zone 2	0.4405	0.3439	0.1506	0.0488	0.0062
Core zone 3	0.4472	0.3482	0.1502	0.0495	0.0049
Radial blanket	0.6199	0.2141	0.1660	--	--
Axial blanket (avg)	0.4824	0.3356	0.1597	--	0.0223
Radial reflector	--	0.5000	0.5000	--	--
Axial reflector	--	0.3356	0.1597	--	--

Type of loading - 3-cycle reload: 96 elements in zone 1
84 elements in zone 2
108 elements in zone 3

1/3 of integral control rods assumed in core.

Enrichments^(a) and Burnup

	BOL	EOL (≈790 days)
Radial core zone 1	0.10746	0.12288
Radial core zone 2	0.11602	0.12617
Radial core zone 3	0.14264	0.14079
Core average	0.12317	0.13064

BOL: E₁:E₂:E₃ = 1:1.08:1.33

EOL: E₁:E₂:E₃ = 1:1.03:1.15

Burnup

Lifetime, days	≈790
MWD/T	69,300
gm fissile/MWD	1.05 (assumed)

$$^{(a)} E = \left[\frac{\text{mass } ({}^{239}\text{Pu} + {}^{241}\text{Pu})}{\text{mass (total U + Pu)}} \right]_{\text{core}}$$

Table 14-B-2. Reactivity Parameters^(a)

Parameter	Value
Core radial expansion coefficient, $R(dK/dR)$	-0.4487
Core axial expansion coefficient, $H(dK/dH)$	-0.1214
Doppler coefficient, full dense core, $T(dK/dT)$	-0.0050
Coolant density coefficient, $\rho(dK/d\rho)$	
Axial sections 1 and 7	0.000765
Axial sections 2 and 6	-0.000545
Axial sections 3 and 5	-0.00768
Axial section 4	-0.01343
Linear expansion coefficient/F	0.0000104
Scram delay, msec	Variable

(a) It is evident, upon comparing the table above with the table of reactivity parameters in section 13.4, that the estimated values agree quite well with the calculated values.

2. Parametric Analysis Results

The dynamic response of the reactor was investigated for parameters of rod velocity, scram delay time, and reactivity insertion rates. In all cases a scram was initiated by detection of a 25% overpower condition; all ramps are assumed to terminate when the rods enter the core. The tabulation below lists the variable parameters for the 12 cases studied.

<u>Rod speed,</u> <u>cm/sec</u>	<u>Delay time,</u> <u>msec</u>	<u>Reactivity ramp,</u> <u>\$/sec</u>
50	100	
	250	0.75
	400	2.00
100	100	
	250	0.75
	400	2.00

The results of interest for these 12 cases are presented below:

Rod speed, cm/sec	Delay time, msec	$P_{\max}$, MWt	t_1 , sec	$(\Delta T_f)_{\max}$, F	$(\Delta T_c)_{\max}$, F
Reactivity ramp - \$2/sec, 25% overpower scram					
50	100	3,239	0.360	21	5
50	250	8,149	0.360	135	15
50	400	30,540	0.510	609	179
100	100	3,998	0.208	25	5
100	250	8,142	0.359	109	31
100	400	30,380	0.511	464	143

Reactivity ramp - \$0.75/sec, 25% overpower scram					
50	100	3,239	0.360	21	5
50	250	3,905	0.511	68	15
50	400	4,884	0.662	129	36
100	100	3,239	0.360	21	5
100	250	3,905	0.511	68	15
100	400	4,881	0.663	94	26

where

$P_{\max}$ = maximum power

t_1 = time peak power is reached

$(\Delta T_f)_{\max}$ = maximum change in center line fuel temperature based on average channel

$(\Delta T_c)_{\max}$ = maximum change in outlet coolant temperature based on average channel

It can easily be seen from the tabulated results that the variable that has the maximum effect is scram delay time. For a given reactivity insertion rate, rod speed has little effect. In view of these preliminary results, the delay time and rod velocity for the reference core analysis were fixed at 300 msec and 50 cm/sec, respectively. The overpower margin was 11% in the reference design which is considerably less than the 25% used in this section.

Calculations were also performed to determine the transient coolant temperature behavior during a coolant coastdown from full flow at power. It was conservatively assumed that a scram was initiated by coolant outlet temperature in excess of a fixed limit. Results were obtained for the peak and average channel for scram initiation temperatures of 1200 to 1250 F. In all cases the scram delay time was 250 msec. The flow decay curve used in this analysis was shown in Figure 14-4; the results of this study are presented graphically in Figure 14-B-1. It can be

seen that failure to scram until high outlet temperature is reached in no way jeopardizes the core, since sodium boiling temperature is never approached and the change in the fuel temperature is even less than that of the coolant.

Figure 14-B-1. Maximum Coolant Temperature Vs Time After Loss of Pumping Power

APPENDIX 14-C
Calculational Methods and Models

The computer codes described below are those used in the safety analysis.

1. FIRE

This is a modified version of the Atomics International SOFIRE code. Like SOFIRE, it is a Fortran IV program that calculates the pressure and temperature conditions within a building that contains a stagnant pool of burning sodium. The burning rate of sodium is described as a function of oxygen concentration. Heat released by the burning sodium is transferred to the sodium and the reactor vessel or reactor building structures by radiation, conduction, and convection. The FIRE code has an option to calculate the effects of sodium fires initiated by a pressurized spray; this segment of the code uses available experimental data.

The most common reaction encountered in using liquid sodium is its interaction with air and possibly with water. The potential hazards of gross sodium leakage from a high-temperature sodium system into air results from the liberation of the combustion energy and reaction products from the resulting fire. Although the probability is remote that a system failure would lead to a sodium spill, such accidents must be considered in the safety analysis of sodium-cooled nuclear power plants. Knowledge of the combustion characteristics of liquid sodium is utilized to predict the consequences of sodium spills and leakage, and to determine the engineered safety features required to eliminate the cause or control the effects of such occurrences.

Sodium may be released from a system as a result of a spill, leak, or rupture. The reaction rate and effects will depend on the mode of release, prevailing environmental conditions, and physical characteristics of the area into which the release occurs. For the purpose of defining the potential air reactions and effects, it is convenient to consider two conditions: (1) reaction at the surface of a pool of sodium, and (2) reaction associated with a spray-type release. In an enclosed system, the energy release rate by a burning pool to the surrounding atmosphere would cause a very slow increase in pressure. However, in the case of a spray-type release, pressure rises and peaks at a considerably higher rate. Under given temperature conditions, the rate of reaction is a function of the sodium surface, which is a function of spray particle size.

1.1. Pool Fires

When a pool of liquid sodium burns in air, the combustion reaction is characterized by a very low flame region and a dense cloud of white oxide smoke. At temperatures above 1200 to 1300 F the sodium pool appears reddish in color, with wispy flames on the surface. The minimum temperature at which a pool of sodium will ignite depends on the characteristics of the particular system involved. The ignition temperature is a function of such variables as the surface mass ratio, surface conditions, metal purity, absolute humidity, temperature of the surrounding atmosphere, and the velocity of the air over the surface.

A SOFIRE code is under development at AI to calculate the sodium burning rate and the energy and oxygen transport in a reactor building. A preliminary issue of this code, obtained from AI and modified slightly for use on the Philco 2000-211, was renamed FIRE. The pool-burning segment of FIRE calculates the temperatures, pressures, flows, and oxygen concentration in the reactor building during a sodium fire. A large number of sodium fire tests have been conducted at various installations to determine approximate burning rates.³⁻⁹ The quantities of sodium burned have ranged from grams to several thousand pounds, with large variations in pool diameter and depth, initial sodium temperature, and amount of container insulation. In a majority of the experiments, a systematic variation of parameters was not attempted, the technique for evaluating burning rate was limited, and a detailed temperature measurement was not obtained. It was not surprising to find that the sodium burning rates reported range from 1 to 17 lbm/ft²-hr.

The energy release per unit of sodium consumed depends on the relative abundance of the sodium and oxygen. For excess sodium the reaction is as follows:

$$\Delta H_{25^\circ\text{C}} = -4100 \text{ Btu/lb Na} \quad (2)$$

For excess oxygen, the reaction is

$$\Delta H_{25^\circ\text{C}} = -4850 \text{ Btu/lb Na} \quad (4)$$

Since the burning of a pool occurs very near its surface, the oxidation for pool burning is assumed to be an excess sodium reaction.¹⁰ The exothermic heat of reaction of 4100 Btu/lbm of sodium consumed is built into the model.

The rate of the sodium-air reaction is controlled by the concentration of oxygen in the fire zone. The model used in the code is as follows:

$$R = H_c C \quad (5)$$

where
 R = reaction rate, lbm/ft²-hr
 C = oxygen concentration, lbm/ft³
 H_c = proportionality constant, ft/hr

There are some indications from earlier data that the sodium-air reaction is controlled by the diffusion mechanism in the gas that transports the oxygen to the flame. If this is the case, then the model should incorporate the effects on molecular diffusion, such as temperature of the gas above the flame in the reactor building. Yevick suggested the following equation as a model.¹¹

$$R = H_c C \sqrt{T_g} \quad (6)$$

where T_g is the gas temperature. The constant of proportionality must be evaluated from experimental data. The suggested range for the reaction for pool burning in air is 5 to 10 lbm/hr-ft². These results were observed in a series of experiments conducted in insulated containers with areas greater than 1 square foot.^{5,6} The experimental studies have shown that the reaction will continue until almost all of the oxygen is depleted from the gas (0.1 to 0.2 vol %), even though ignition may not occur unless the oxygen concentration is above 4 to 5 vol %.¹²

1.2. Spray Fires

Several investigations have been reported involving combustion of sodium discharged in the form of a spray into a chamber.^{1, 11-13} The peak pressure change for the spray fire in air was 38 psig for Mangold and Tidball¹³ and 78 to 85 psig for Humphreys.¹⁴ Pressure changes recorded as a function of time indicate that the sodium ignited as soon as it entered the container.

Calculations have been carried out for equilibrium conditions in sodium-air mixtures by minimizing the free energy.^{15, 16} A comparison of these results with the experimental pressures observed by Humphreys shows that all the calculated values are high. A subroutine incorporated into the FIRE program evaluates the pressure, temperature, and oxygen concentration based on Humphreys' experimental results. This subroutine predicts the maximum pressure and temperature measured in Humphreys' experiments, and exceeds the measured values for other sodium-to-oxygen ratios. The effect of the spray fire appears as a change in the initial conditions for the pool burning that may follow.

2. FORE II

FORE II is a coupled, thermal-hydraulics, point-kinetics digital computer code designed to calculate reactor parameters under steady-state conditions or as functions of time during transients.¹⁷ The transients may result from a programmed reactivity insertion or a power change. Variable inlet coolant flow rates and temperatures are considered. The code calculates the reactor power, the individual reactivity feedbacks and temperature of coolant, cladding, fuel, structure, and additional material for up to seven axial positions in three channel types which represent radial reactor zones. The heat of fission accompanying fuel melting, the liquid metal void reactivity, and the spatial and time variations of the fuel-cladding gap coefficient due to changes in gap size are considered. The programming language is Fortran IV.

Considerable difficulties have been experienced with the FORE II computer programs received from the Argonne Code Center. Early in the safety analysis of the reference core, it became evident that FORE II was calculating erroneous fuel temperatures. This was corrected in the original issue from the ANL Code Center, and a second release was

requested from ANL when it became known that a new version had been released to the code center by General Electric for distribution, since there were other corrections that improved the feedback by using the most recent temperatures. Unfortunately, the second release contains errors in the reactivity calculations which have not been located. The second release incorporated a new subroutine, REAC, and somewhere in the shuffle an error was introduced into the reactivity calculation. It is anticipated that General Electric will release a third version within a few months.

In the meantime, in order to fulfill the requirements for the safety analysis, corrections are being made to the first version of FORE II (released by the ANL Code Center). The feedback calculation now uses the most recent temperature and density values. This was accomplished by relocating a segment of the subroutine FEEB so that the total reactivity calculation now follows the feedback calculations. The FEEB subroutine was modified so that it will calculate the density feedback for the coolant and structure when there is no additional material present. At least three problems that still exist with our modified first release have not been located or corrected:

1. The calculation of the fuel conductivity for the steady-state condition.
2. The Doppler feedback calculation when only the average channel temperatures are used.
3. The restart from a wrapup.

The first problem can cause an unstable steady-state transient. No scheme has been derived to eliminate this; thus, we have used fuel conductivities that were constant with respect to temperature. The second problem was faced by incorporating the peak pin in the calculations by choosing the weight functions properly to eliminate any effects. The third problem causes no error, but it does increase the required computer time considerably. This problem essentially eliminated the study of accidents that start from low power.

As a result of the errors in the released versions of FORE II, the results must be interpreted with extreme caution. It is anticipated that more problem areas will be uncovered as the code is used and the results compared with those produced using other methods.

3. BANGO

BANGO is a preliminary version of a calculational technique to be developed for studying the effect of molten fuel release to the sodium coolant. The computational model assumes that (1) the released fuel is intimately mixed with the coolant in the same region of the core, (2) the pressure is relieved by upward motion of sodium, (3) heat is transferred from the sodium bubble to the surrounding coolant at a rate governed by $h(\text{Btu/hr-ft}^2\text{-}^\circ\text{F}) = 22,000 - 36,000/\Delta T$, (4) the gas above the sodium pool is compressed and exerts a counterpressure, and (5) the cone of sodium above the bubble is accelerated upward.

4. WEAK EXPLOSIONS

The WEAK EXPLOSIONS program performs a coupled neutronics-hydrodynamics calculation for a spherical homogeneous reactor core with a given composition to obtain the time-dependent energy release that results from the insertion of reactivity at a given rate.¹⁸ The neutronics calculation uses a point model having a power distribution and material worth fixed by polynomials in the radius. The kinetics and hydrodynamics equations are coupled and solved by the Milne method at specified time intervals. The equation of state of the core material is in the exponential form, and the Doppler effect is taken into account by using the E^{-n} law model. The power excursion is initiated by a linear reactivity increase.

Several other codes used in the course of this analysis are described elsewhere.¹⁹ Included are the DHRG3-SIZZLE one-dimensional diffusion-theory package and PERT, a perturbation-theory code.

The fuel self-worth and reactor power distributions were described as sixth-order polynomials in the radius. These polynomials are weighted, least-square fits of the distributions determined from one-dimensional diffusion and perturbation calculations. The polynomial fits are given below:

Power Distribution

$$P(r) = 0.3888 \times 10^3 - 2.1674r + 0.19235r^2 - 0.62716 \times 10^{-2}r^3 \\ + 0.87048 \times 10^{-4}r^4 - 0.52621 \times 10^{-6}r^5 + 0.11311 \times 10^{-8}r^6$$

Fuel Worth Distribution

$$\begin{aligned} W(r) = & 0.66244 \times 10^{-8} - 0.58452 \times 10^{-10}r + 0.42387 \times 10^{-11}r^2 \\ & - 0.11777 \times 10^{-12}r^3 + 0.14144 \times 10^{-14}r^4 - 0.74204 \times 10^{-17}r^5 \\ & + 0.13679 \times 10^{-19}r^6 \end{aligned}$$

In the analyses presented in this report, δK Doppler is assumed to be proportional to T^{-1} . The Doppler feedback was started at equilibrium power in all cases.

The equation of state used in the calculations is based on a core fuel density of 3.5 gm/cm³. A weighted, least-square fit of the curve shown in Figure 14-C-1²¹ was made and used to relate inverse energy to pressure.

Figure 14-C-1. High-Range Pressure Vs Internal Energy,
Using AHT Critical Constants

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