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NPR HAZARDS REVIEW - VOLUME I

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1. INTRODUCTION

1.1 Purpose and Scope

A preliminary hazards review of the N Reactor was published in January 1958. At that time, the design concept had not been fully developed and detailed design had not been started. The preliminary review, therefore, was general in nature and served mainly to isolate characteristic nuclear hazards associated with reactors of the general type of the New Production Reactor (N-Reactor) being built at Hanford.

At the present time (October 1961), the scope design is essentially complete, most of the detailed design has been done and construction is about 25 per cent complete. Therefore, it is possible to present a much more detailed treatment of the nuclear hazards involved. On the other hand, not all the hazards analyses and experimental programs to define safety limits have been completed. Some systems, particularly instrumentation and control systems have not been completely designed. Detailed operating procedures have not been defined, nor have operating limits been set. Therefore, this review cannot be complete in all respects. It is believed, however, that significant nuclear safety problems have been identified and, if specific solutions to the problems have not been worked out in detail, the probable approaches to solutions have been determined.

This document is designed to present as complete a review of the hazards associated with the operation of the N Reactor as is feasible at this time. Where completeness is not possible the problems are

indicated and probable paths of solution are suggested. Background material is provided to put the hazards in context. Supplements to this report will be issued when the additional data and information required for completeness are obtained.

1.2 Organization of Review

This review is organized into two parts. The first part issued as Volume I is in the nature of an expanded summary. The second part of the review will be issued in a succeeding volume or volumes and will consist of results of evaluations obtained following the publishing of this volume plus more detailed supporting information.

2. SUMMARY AND CONCLUSIONS

2.1 Nature of Hazards

Release of radioactive fission products in damaging quantities to the areas outside the Hanford reservation is the primary hazard of concern in this review. A more demanding objective of the General Electric Company in operating the NPR will be to prevent any releases of fission products which could endanger personnel or interfere with operations within the Hanford reservation or within the N Reactor buildings.

The best confinement for fission products is the clad, cooled, fuel element itself. Thus, the first line of defense against release of fission products is the maintenance of adequate cooling to fuel elements during all conditions of operation and shutdown.

Inadequate cooling can result from either of two general situations:

(1) A power level excursion could increase the heat generation rate of the fuel charge to a level beyond the capacity of the cooling system, and (2) Reduction of the heat removal rate to a level below the heat generation rate of the fuel. Either situation could result in extensive release of fission products from the fuel. Since irradiated fuel elements continue to generate heat after the fission reaction has been terminated, adequate cooling must be provided for an extended period after reactor shutdown.

Failure of the cladding of a single fuel element would have a very small potential for spreading fission product contamination. Resulting contamination can be readily controlled.

2.2 Philosophy of Operation

The philosophy of operation of the NPR will parallel that for the operation

of the existing Hanford production reactors. It will be the objective of the General Electric Company to operate the NPR efficiently for the production of plutonium within the overriding requirement that the potential for the release of fission products shall be maintained at acceptably low levels. The most rigorous attention will be devoted to control of the nuclear process, and to maintenance of adequate cooling to the fuel elements during all phases of operation, shutdown and discharge operations by all organizations associated with the operation of the NPR.

Prime responsibility for production is organizationally separated from prime responsibility for nuclear safety. Prime responsibility for nuclear safety is vested in the Research and Engineering Section of IPD which does not have production responsibility. This organization establishes process limits designed to assure nuclear safety and conservation of the reactor plant life. After managerial review and acceptance, these process limits become mandatory upon operational personnel.

These process limits are inherently conservative. They are designed to assure operation within known-safe limits. Process limits are based on experimental data and interpreted by analytical techniques, using, where necessary, conservative interpolations and modest, always conservative, extrapolations. The limits will be constantly reviewed and updated as additional information becomes available from operational experience and aggressive research and development programs.

Qualified reactor operators will be relied upon to operate the reactor within these mandatory limits. Conformance will be monitored. The reactor has been designed, and limits established, for manual operation of the

reactor plant. However, multiple automatic instrument systems have been provided to automatically terminate the nuclear reaction and shut the plant down in the event process limits are exceeded.

The incident rate of fuel cladding failures can be controlled to a degree by control of power levels and exposure limits. It is anticipated that the acceptable rate of fuel cladding failures will be determined by process economics.

2.3 Special Precautions Against Identified Hazards

2.3.1 Inadequate Cooling

Two independent electrical power systems are provided to power critical pumps and components in the heat removal system. Equipment is divided between the two independent power sources and the equipment on each power source is sufficient to shutdown and maintain the shutdown reactor indefinitely without fuel damage. Thus, each system acts as an operating backup to the other system. These two electrical supplies are supplemented by an electrical intertie to the D and H Reactor area substations which was provided for construction.

The primary loop pumps will normally be powered during operation by process generated steam in the secondary loop. During initial phases of startup and after shutdown they will be powered by steam generated by the local oil-fired boiler. In the event of failure of the local boiler the process can be shut down and maintained in the shutdown condition without fuel damage by the use of BPA electrically powered motors provided for the purpose. Any two of the five primary loop process pumps have the capability of providing adequate

flow for the shutdown transient.

In the very unlikely event of simultaneously failure of all electrical power systems, a diesel-powered once-through coolant supply system is provided for the reactor.

Multiple sensing and control systems, each consisting of multiple, fail-safe components, are provided to shut the reactor plant down automatically without fuel damage in the event that process limits are exceeded. The term "fail-safe" means that loss of capability of the component to perform its design function will generate a signal which will shut down the reactor complex.

2.3.2 Nuclear Control

The nuclear flux level and rate of change of flux is monitored in the sub-critical range, in the intermediate power level range, and in the operating range by multiple sensing and control systems, each consisting of multiple fail-safe components. Any one of these systems, except for the subcritical instruments, will automatically shut down the reactor complex in the event that process limits are exceeded.

In addition to the instrumentation in the safety circuitry, additional instrumentation systems are provided for operational control of the reactor complex. These additional instruments are designed to permit reactor operators to manually maintain control of the system or, in the event of approach to process limits, to shut the reactor down independently of the automatic safety circuitry. The safety circuitry will over-ride manual controls.

2.3.3 Equipment Failure

The NPR has been designed to accommodate, without fuel element damage, equipment failures which would normally be expected to occur during operation in spite of rigorous maintenance and inspection. Critical power supplies, pumps, controllers, dump valves, and similar equipment have been provided at least in duplicate and usually in multiple units. Rigorous inspection and maintenance will minimize the occurrence of operational failures.

2.3.4 Operational Error

The potential for operational error will be minimized to the extent that careful selection and rigorous training of operating personnel can accomplish. Operating personnel participated in the design of the plant. In the event that process limits are exceeded during operation, automatic sensing and control instrumentation will override the operators' and shut the plant down.

2.3.5 Confinement

A confinement system is provided for the NPR to minimize and control the release of airborne fission products in the event of significant release of fission products into the reactor building. The most severe confinement problem would be associated with rupture of the primary loop. Vents are provided in the primary confinement zone to release the steam and reseal the building before fuel element melting could occur. Fog sprays are provided to maintain building pressures within tolerable values after the building has been resealed. When appropriate, building exhaust ducts leading to the filter system may be opened. The filter system consists of particu-

late filters designed to remove more than 99.9 per cent of the particulate matter and greater than 50 per cent of the halogens from the effluent air. Tests indicate that we may expect greater than 90 per cent retention of halogens.

The exhaust air from areas of the reactor building, which have a potential for becoming contaminated, is normally directed through the filter system.

2.4 Operational Safety

It is concluded that the NPR can be operated with an extremely low probability of significant release of fission products to the environs. However, this probability will not be zero. It is believed that the hazards associated with operation of the NPR will be acceptably low as evaluated in the public interest by the highest standards of the nuclear industry.

This conclusion reflects scope and detailed design, insofar as now completed, of the NPR. Operational safety will, of course, be contingent upon appropriate completion of the detailed design, procurement, fabrication, and erection of the plant. No areas have been identified where this cannot be accomplished.

2.5 Fission Product Release

No order of magnitude differences have been identified in either the potential for or consequences of a major fission product release from the NPR compared with the existing production reactors at Hanford. The location and environment are essentially the same. The fission product inventory of the N Reactor does not differ in a major way from that of one of the K Reactors. The advanced design of the confinement system tends to be counterbalanced by the necessity for venting steam and resealing the building, and the improved monitoring and control systems tend to be counterbalanced by the increased complexity of the total plant.

3. DESCRIPTION OF FACILITY

3.1 Location

The new plant is located within the existing Hanford reservation, midway between the 100-K and 100-D reactor areas, immediately adjacent to the Columbia River (Figure 3-1). An architect's sketch of the 100-N Area is shown in Figure 3-2.

3.2 General Description

The reactor is of the graphite moderated, light-water cooled, plutonium production type, fueled with zirconium alloy clad metallic uranium. The fuel elements are contained in horizontal zirconium alloy process tubes which penetrate the graphite core. The high-purity cooling water is pumped through these tubes, where it absorbs the heat which is released from the uranium fuel in the course of the plutonium production process. From the reactor the coolant goes to ten heat exchangers in which the process heat is dissipated through evaporation of pressurized water on the shell side. A small fraction of the steam so generated in the secondary loop will be used to drive the primary-coolant pumps and as a backup source of power for a plant service turbo-generator. The remainder is passed through dump condensers, cooled by circulation of low temperature Columbia River water. Part of this dumped steam would be used for the generation of electric power in the event of conversion to by-product power recovery.

Back-up cooling systems and standby power sources are provided to assure safety and continuity of operation. A brief description of the major components of the reactor plant is included below. A more de-

tailed treatment of systems having critical safety roles is included in later sections of this volume and in appendixes. The reactor and associated coolant piping and the heat dissipation system is shown in Figures 3-3, 3-4, 3-5, 3-6, 3-7, and 3-8. The layout of the reactor and heat dissipation systems (105, 109 Buildings) is shown in Figure 3-9.

3.3 Major Reactor Components

3.3.1 Graphite Core

The reactor core is a structure of interlocking graphite bars, with over-all stack dimensions of 39 feet, 5 inches long, 33 feet, 0-5/8 inches wide, and 33 feet, 4-5/8 inches high. The stack is penetrated from front to rear by 1072 horizontal channels on an eight-inch horizontal and nine-inch vertical lattice spacing; however, only 1004 channels will initially be provided with zirconium alloy process tubes and loaded with fuel.

Venting provisions are included in the stacking pattern to permit the escape of coolant which would be released if a process tube ruptured. These vent channels which allow dissipation of the steam-water mixture in the longitudinal, transverse, and vertical directions, have sufficient cross-sectional area to limit any potential damage to the particular bar of graphite in which the tube rupture occurs.

The density of the bulk graphite from which the finished bars are machined is about 1.7 gms/cc. However, in the central

zone of the reactor where the vent channels are provided, the effective stack density is about 1.29 gms/cc.

Gas plenums are located between the active zone and the reflector graphite at the inlet and outlet ends of the stack as well as at both sides. The stack atmosphere is helium, normally circulated through the active zone of the reactor from front to rear. Provisions are made for the storage, recirculating and drying of the helium, the function of which is to dry the moderator when necessary, serve as a medium for detection of water and air leaks, prevent in-leakage of air by maintaining a slight positive pressure, and to control moderator temperature through heat conduction.

3.3.2 Control and Cooling Channels

The graphite stack also is penetrated by 87 horizontal side-to-side channels for the control-rod system, in a 32-inch horizontal by 36-inch vertical spacing, and by 108 vertical channels in a 32-inch square pattern for the ball safety system. In addition, there are 640 horizontal side-to-side penetrations for the moderator cooling system provided to limit long-term dimensional change in the graphite. The coolant tubes are located to provide cooled support piers through the stack in an effort to maintain correct positioning of the fuel tube pattern and other stack penetrations. The 0.75 inch outside diameter zirconium alloy cooling tubes are located between each tube row vertically (nine-inch spac-

ing) and on 16-inch centers horizontally. They are arranged to that every other cooling tube in a horizontal row is directly under a tube-support point in the tube bearing layer above. The graphite cooling tubes are equipped with end fittings which permit their use as detector locations for in-reactor neutron flux instrumentation. Cooling water is normally supplied by a 5000 gpm recirculating system. Alternate coolant sources also are available so that the system can serve the emergency function of providing a reliable heat sink to limit the extent of fuel melting in case of a major catastrophe which disrupts both the normal and backup cooling water supply to the process tubes.

3.3.3 Shielding

A thermal shield surrounds the graphite stack and intercepts most of the radiation and heat energy which escapes from the reactor core, thus reducing the heat load on the biological shielding. One-inch thick boron steel is used as the thermal shield material except at the inlet and outlet ends of the reactor where eight-inch thick cast iron blocks are used. Cooling tubes are welded to the boron steel plate to permit removal of absorbed heat. The cast iron inlet and outlet thermal shield is cooled by contact with the exterior surface of the process tubes through which the primary cooling water flows.

Biological shielding prevents activation of equipment and components immediately adjacent to the reactor which require

personnel access. High density concrete is used on all sides and on the top of the reactor, and ordinary concrete is used on the bottom. Thicknesses range from 40 to 72 inches. Personnel entry to locations immediately adjacent to the primary shields is normally made under continuous radiation protection monitoring and only while the reactor is shut down.

The primary shield supports the thermal shield, fuel tube assemblies, control rods and ball hoppers. It also restrains leakage of the reactor gas atmosphere.

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The areas adjacent to the reactor are, in turn, surrounded by secondary shielding walls. These help maintain radiation levels below 0.1 mr/hr. in continuously occupied areas of the reactor building, and less than one mr/hr. in less frequently occupied areas which may be entered or passed through without radiation zone procedures.

3.3.4 Control Rods

The 87 horizontal control rods are hydraulically driven and can be extended through the active zone of the reactor. Approximately half of them enter from each side of the reactor block. The rod system provides both operating control and rapid shutdown functions of the reactor. This is accomplished by having all rods water cooled and by having the individual rod drives capable of rapid insertion from any position as well as slower drive speeds for fine control of reactor flux

distribution and power level. Individual rods are set up for control use by means of selector switches in the reactor control room. However, a specified number of rods must always be reserved for the fast-scram function. Interlocks prevent rod withdrawal unless the specified number of safety rods are withdrawn and ready to scram. Each hydraulic cylinder rod drive unit has its own independent accumulator. Control equipment is fail-safe in that loss of electric power will de-energize the control solenoid valves and accumulator pressure will scram the rods.

3.3.5 Ball Safety System

The ball safety channels provide a completely independent nuclear control system to back-up the shutdown function of the horizontal rods. The control elements for this system are 3/8 inch diameter ceramic balls containing neutron absorbers. Ball hoppers are embedded in the top biological shield over each of the 108 vertical channels. The hopper gates, designed to open in case of an electric power failure, as well as upon appropriate safety-circuit signal, allow the balls to enter the channels by gravity.

3.3.6 Fuel Elements

The reactor will be fueled with metallic uranium fuel elements coextruded in a zircaloy-2 jacket. The fuel geometry selected is that of two concentric fuel tubes. Figure 3-10 shows the fuel and process tube assembly.

3.3.7 Process Tubes and Fittings

The fuel elements are contained in zircaloy-2 process tubes which penetrate the graphite stack, extend through the thermal and biological shielding, and connect with the inlet and outlet cooling water manifolding system. The tubes have an inside diameter of 2.7 inches, and a wall thickness of 0.250 inches.

The inlet end of each process tube is attached to a carbon steel nozzle assembly by means of a rolled joint. The nozzle assembly is free to move axially to accommodate thermal expansion and contraction of the process tube. The outlet end of the process tube is attached by means of a rolled joint to a similar steel nozzle, which is fixed to the surface of the biological shield. There are gas seals at each reactor face, between the nozzle and the biological shield, to prevent leakage of the helium atmosphere. Each nozzle also is equipped with a removable end-closure to permit access for charging and discharging of fuel elements during reactor outages as shown in Figure 3-11.

Each inlet nozzle has cooling water fed to it through an individual carbon-steel header-to-nozzle connector of 2-5/8 inch outside diameter and 0.260 inch wall thickness. Average length of these connectors is about 52 feet. Each inlet connector is equipped with a butterfly valve for flow control and a venturi meter for monitoring the flow rate.

The signal from any flow monitor will scram the reactor if the flow rate deviates from an acceptable range. The flow monitors are designed to have response times sufficiently short to ensure safe reactor shutdown.

The outlet nozzle-to-header connectors are of the same size and material as the inlet connectors. The outlet assembly for each process channel includes equipment for monitoring outlet water temperature and for detecting fuel element cladding failures. Each outlet connector has a remotely operated three-way valve which permits diversion of the outlet water from a particular process channel to a contaminated water disposal system to minimize spread of radioactive materials through the primary cooling loop in event of a fuel element failure.

Individual process channel outlet temperatures are indicated and recorded for operational purposes and for backing up the nuclear instrumentation in detecting an excessive local power density in a particular region of the reactor.

3.3.8 Nuclear Instrumentation

Nuclear instrumentation measures the level, the distribution, and the rate of change of neutron flux density in the reactor. Monitoring data and safety circuit trip signals are provided, as appropriate, over the full range of neutron flux levels from those existing during sub-critical shutdown conditions to those characteristic of full production level operation. The NPR lattice is designed so that reactivity is decreased by loss of coolant from the process channels.

3.3.9 Radiation Protection

Secondary shielding walls and other essential building structural components are utilized to provide (a) zoning of the ventilation system to minimize spread of radioactive contamination during normal operation, and (b) an effective method of controlling the release of large-scale airborne radioactive contamination in case of a major operating accident. The design of the confinement system which fulfills this latter function is based on consideration of the entire spectrum of credible accidents. They range from the partial burning of a single fuel element to the melt-down of a significant proportion of the total fuel load. Advantage is taken of the fact that should a major line fail, the resulting initial surge of steam of large volume would contain relatively insignificant amounts of radioactive contamination. This material would be vented outside the reactor building. The vents would then be closed and the smaller volume of contaminated gases, which would result if emergency cooling system failed to operate, would be confined within the reactor structure or filtered before release to the atmosphere.

The primary confinement zone which surrounds the entire primary cooling system is designed to maintain structural integrity with internal pressures ranging from +5 psig to -2 psig. Vents are designed with adequate area to dissipate the maximum rate of steam evolution without exceeding these internal pressure limits and with closures capable of re-

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sealing the vent openings within the time interval between dissipation of the steam surge and initiation of gross melting of the reactor fuel. In addition, a fog-spray system would reduce the pressure within the primary confinement zone and remove a significant fraction of the contamination from the confined air. Controlled release of the confined gases is through a filter system designed to remove more than 99 per cent of the particulate matter as well as more than 50 per cent of the radioactive halogens.

3.4 Major Heat Removal Systems

3.4.1 The Primary Coolant System

The primary coolant system provides for recirculation of purified water to remove the heat generated in the reactor and to transfer it to the secondary coolant system.

The water is recirculated by five, parallel, steam driven pumps. The water flows from the pumps to the reactor and from the reactor back to the heat dissipation (100-N) Building through carbon steel connecting lines. At the 100-N Building, the water gives up its heat to the secondary system in ten parallel steam generators and then flows back to the suction of the circulating pump.

The primary coolant design flow rate is 76×10^6 lbs./hour. Reactor bulk inlet and outlet temperatures will be about 369 degrees F and 527 degrees F, respectively.

3.4.2 Secondary Coolant System

The primary coolant transfers the heat it absorbed in the reactor to the boiling secondary coolant in ten parallel steam generators. About 94 per cent of this steam is condensed in 16 parallel condensers and the heat "dumped" to raw Columbia River water. The remaining steam is used for driving the five primary coolant system pumps, a 15 MW turbine generator and for supplying process heat and steam requirements.

The capacity of the secondary system is matched to the reactor power by controlling steam pressure in the steam generators, and by controlling the condensate level in the dump condensers. Changing the steam pressure changes the temperature of the boiling secondary coolant and, hence, the rate at which heat is transferred from the primary coolant. Varying the condensate level in the dump condensers varies the amount of tubing surface available as condensation surface.

3.4.3 Circulating Water System

Untreated water from the Columbia River is supplied to the dump condensers as well as to the condensers on the primary-drive turbines and on the plant-service turbine-generator. This condenser cooling water is then returned to the river. Untreated water is also supplied to a water treatment facility to provide plant requirements for filtered, sanitary and demineralized water. The total untreated water flow is

supplied by four deepwell pumps, each of 100,000 gpm capacity and driven by a 3500 hp synchronous electric motor. Two of these pump drives are supplied with BPA power and two with locally generated power. The water is conveyed from the river pump house to points of process use via two separate headers. The output of any one of the four pumps is sufficient for all essential process requirements during a reactor scram and subsequent shutdown.

3.4.4 Emergency Coolant Supply

A separate, independent untreated water system is provided for emergency once-through cooling of the reactor. Diesel driven pumps located in a separate section of the river pump house supply this water to a 200,000 gallon storage tank where it is heated to 150°F. Three diesel driven high-lift pumps are provided to deliver the "tempered" cooling water to the bottom of the front face risers where it may enter the primary cooling system, through check valves, when the system pressure falls below the shutoff head of the diesel pumps. This emergency water supply is intended for use only in case the normal recirculating treated water heat removal system is disabled. The 200,000 gallon emergency water storage tank serves as the water source for the confiner fog spray system as well as for emergency coolant supply for the moderator cooling system. It will be augmented by a continuous flow of raw water upon demand.

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3.4.5 Demineralized Water Supply

The demineralizer plant consists of four primary cation-exchange units in parallel, a single vacuum degassifier, and four parallel anion-exchange units, followed by four secondary exchange units in parallel. The effluent water, degassed and having a resistivity greater than two megohms per cubic centimeter, is stored in a special 1,000,000 gallon demineralized water storage tank from which it is drawn for injection into the primary cooling system and for secondary system make-up. It is also possible to recirculate the stored water back through the demineralizer plant if the quality of the stored water falls below required standards.

A high pressure water make-up system, consisting of four 1350 gpm pumps which take their suction from the demineralized water storage tank, maintains the supply of treated water in the primary loop under all but the emergency conditions. The basic continuous make-up requirement is replacement of the rupture monitor system bleed which it is estimated will be 410 gpm. This serves as the normal bleed and feed necessary to maintain water quality. An additional continuous requirement which may vary with time is the primary pump shaft seal water supply. Also, up to 1000 gpm of high purity water must be available to the primary loop for diversion to waste of the water from a number of process tubes. The capacity of the water treatment plant

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is such that only one or two tubes can be diverted continuously but the make-up pumping and storage capacity is such that as many as five tubes can be diverted for several hours while still retaining 2000 gpm make-up capacity in reserve to cover potential rupture of a process tube connector. Such a rupture would, of course, scram the reactor, but it would be possible to keep the primary cooling system pressurized until an orderly plant shutdown and initiation of repair could be effected. All normal make-up water is warmed to reduce thermal shock at the point of injection.

Low head pumps, taking suction from the same source of high purity water as the high pressure injection pumps mentioned above, are provided to supply water for fuel element charging and for primary system filling.

3.4.6 Electric Power System

With the exception of the steam-turbine-driven primary loop pumps, most equipment for normal operation of the reactor and its cooling system is electrically powered. This power is supplied by two independent systems, the 230 kilovolt Hanford Utility Power Loop and the 13.8 KV turbine generator in the local power house. The 230 kilovolt supply is transformed to 13.8 kilovolts for the largest loads such as the river pump motors, then further stepped down to 4160 volts and lower for other loads. Local power is generated at the 13.8 kilovolt level.

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The two systems are isolated from each other, both physically and electrically. Both sources are required for normal operation but either system can support the necessary loads during a scram and for shutdown conditions. Normally, no connection between the systems will exist during reactor operation. In the event of failure of one system, interconnection is possible by manual operation. This arrangement assures independence and precludes failure in one system inducing failure in the other or a single fault causing failure of both systems.

3.4.7 Loop Decontamination Facility

Facilities are provided for chemical decontamination of the entire primary coolant system, or each heat exchanger cell on an individual basis. This includes equipment for storage and preparation of the necessary chemicals and piping for injection at appropriate points.

Circulation of the chemicals will be by means of the primary pumps and facilities for heating, where required, will be provided. The spent chemicals as well as contaminated flushing water will be sent to waste storage facilities for treatment prior to final disposal.

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TABLE 3-1

SUMMARY OF NPR DESIGN PARAMETERS

Power Level

4000 MW thermal

Lattice

Spacing
Pattern

8" horizontal x 9" vertical

Tubes per horizontal row:
Tubes per vertical row:
Omitted from each corner:
Number of tubes:

34
32 installed, 34 total
21 as installed, 21 total
1004 installed, 1072 total

Graphite

Over-all dimensions

Length:
Width:
Height:
Weight of graphite:
Density of graphite:
Maximum temperature:
Bulk average temperature:

39 feet, 5 inches
33 feet, 0-5/8 inches
33 feet, 4-5/8 inches
3300 tons raw, 2000 tons finished
1.7 g/cc material, 1.29 g/cc as stacked
1460 degrees F
970 degrees F

Shield

Thermal, front and rear:
Rest of thermal shield:
Biological Shield:
Front and rear:
Sides:
Top:
Bottom:

8" cast iron blocks
1" boron steel plate
40" serpentine heavy aggregate concrete
40" iron-limonite heavy aggregate concrete
65" iron-limonite heavy aggregate concrete
102" regular concrete

Safety Systems

87 horizontal, water-cooled, hydraulically driven controls rods.
41 entering from left side; 46 entering from right side.
32" horizontal by 36" vertical spacing, 29.5 foot travel.
11 horizontal by 9 vertical pattern, 3 omitted from each corner.
B4C absorber in regular rod, LiAl in isotope producers.
Titanium sheath material for both types of rods.

108 vertical ball channels; 4" x 4" square opening.
32" x 32" spacing.
10 wide by 12 long pattern; 3 omitted from each corner.
Ceramic balls of Alumina; Samarium Oxide as neutron absorber.

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Table 3-1 - Continued

Graphite Cooling System

Side to side orientation, 640 tubes.
Every filler layer vertically and on 16" centers, horizontally.
Zircaloy-2 tubes, 3/4" o.d.; carbon steel end fittings, pigtail connectors.
Normally 5000 gpm recirculating process water.
Once-through raw water back-up of maximum reliability.
Inlet temperature: 100 degrees F.
Outlet temperature: 180 degrees F.

Process Tube and Fittings

Zircaloy-2; 2.7" i.d. and 0.25" wall; 53' long, ribless
End connection to carbon steel nozzle assembly
Inlet end; rolled joint to short nozzle; dynamic gas seal
Outlet end; rolled joint to long nozzle; static gas seal

Reactor Manifolding

Individual tube nozzle-to-riser connectors on both inlet and outlet
Carbon steel; 2.105" i.d., 2.625" o.d.; average length of 52'
Piping to and from 109-N:
16 parallel lines, 8 on each side
Carbon steel, nominal 18" diameter

Primary Cooling System

Five parallel cells fed from reactor outlet manifold piping.
Each cell contains two heat exchangers in parallel, a primary pump and valving
Pumps are horizontal, centrifugal, injection shaft seal type.
Pump drives are 9000 hp steam turbines with 350 hp electric pony motors
Heat exchangers are horizontal, U-tube type of 16,000 ft² surface
Shell design pressure is 800 psi
Surge vessel of 1200 ft³ capacity, with 2350 KW heater capacity
Piping is carbon steel except for heat exchanger tubes and valve trim.
Flow is 76 x 10⁶ pounds per hour
Bulk inlet temperature: 369 degrees F.
Bulk outlet temperature: 527 degrees F.
Design pressure: 1825 psig.
Pump Inlet Pressure: 1218 psig.
Pump Outlet Pressure: 1544 psig.

Secondary Steam Loop

Design pressure: 600 psig.
Operating Pressure: 72.8 psig.
Operating Load: 14.9 x 10⁶ pounds per hour.

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Table 3-1 - Continued

River System

Condenser cooling supplied by 4 - 100,000 gpm electric driven pumps.

Backup by 2 - 35,000 gpm direct diesel driven pumps.

Emergency cooling from 200,000 gallon storage tank.

Supplied by 3 - 4500 gpm direct diesel driven booster pumps.

Last ditch supply to graphite cooling system.

Supplied by 2 - 5000 gpm direct diesel driven pumps.

Power Generated at Plant

Steam supply from 575,000 pounds per hour, 400 psi, oil-fired boiler.

Local station electric power from 15,625 KW turbine-generator set.

Water Treatment

Filter plant with 700,000 gallon storage

Demineralized water plant with 1,000,000 gallon storage

Confinement

Primary zone design for +5 psig, -2 psig.

Sufficient vent area to dispose of steam surge.

Fog-spray system to control confiner pressure to near atmospheric.

Filter system to remove 99 per cent of particulates and 50 per cent
radio-halogens.

FIGURE 3-1

APPROXIMATE VIEW

100-N PLANT

Unclassified

~~Figure 3~~
Figure 3-2

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N - REACTOR INLET PIPING

THERMAL BARRIERS, AND PIPING FOR ROD AND SHIELD COOLING SYSTEMS OMITTED

3-4
Figure 3-4

AS-25 (REV. 10-68)

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N-REACTOR OUTLET PIPING

THERMAL BARRIERS, AND PIPING FOR ROD AND SHIELD COOLING SYSTEMS OMITTED

Figure 3-5

Unclassified

NPR COOLING SYSTEM - FLOW DIAGRAM

- PRIMARY COOLANT
- STEAM
- RIVER WATER
- LAST DITCH RIVER WATER
- MODERATOR COOLANT
- CONDENSATE
- ⊗ ELECTRIC MOTOR
- ⊙ DIESEL ENGINE

Figure 3-6

Figure 2

BLOCK DIAGRAM OF NPR-PLANT

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7000X

Figure 3-8
Figure 3-8

Unclassified

100EX

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105-109-N
PLAN VIEW
0'-0" LEVEL

GENERAL ELECTRIC
ZONE I
ZONE II
ZONE III
ZONE IV
ZONE V

BURNS & ROE
ZONE I
ZONE II

Figure 3-1
Figure 3-2

UNCLASSIFIED

FIGURE 8-9
THE FUEL PROCESS TUBE ASSEMBLY

105-N
PROCESS TUBE ASSEMBLY
LONG TUBE

4. REACTOR HAZARDS

4.1 Introduction and Summary

4.1.1 Introduction

The release of fission products to the plant environs will require a sequence of events in which (1) some portion of the fuel is overheated, releasing fission products (either as a result of inadequate cooling or excessive heat generation rate), (2) the primary loop is ruptured to admit fission products into the building spaces, and (3) the fission products escape from the building to the atmosphere.

Features designed to minimize the likelihood of these events are discussed in the following sections of this report.

Section 5 discusses the control of nuclear reactivity and the safety circuitry provided for shutting the reactor down, Section 6 the heat dissipation plant, and Section 7 the system provided to confine fission products within the reactor building. This part of the document will, therefore, not dwell on these systems but, rather, with the events following various accidents and possible mechanisms leading to various accidents.

Definition of Terms

Special, restrictive meanings have been given to certain terms used in the following discussion of reactor hazards. Definitions are given below:

1. Incredible Accidents

Incredible accidents are those whose probability of occurrence is so low that specific protection is not provided. Accidents

resulting from the simultaneous failure of three reliable, independent components or accidents resulting from an event which in itself would be catastrophic, are considered incredible accidents. Examples of these accidents include earthquake of greater severity than have ever been recorded in the Western United States and the falling of large meteor.

2. Credible Accidents

Accidents resulting from two simultaneous, independent system failures are considered credible accidents. For example, a major primary loop rupture combined with failure of the emergency once-through cooling system to supply coolant, is the maximum credible accident.

3. Improbable Accidents

Improbable accidents are those considered to have a probability of less than about 0.01 of occurring during the reactor lifetime. The failure of a major line in the system is considered an improbable accident.

4. Probable Accidents

Accidents which have a probability of occurrence during the reactor life in the 0.5 to 0.01 range are termed probable accidents. Failure of a process tube or of a front connector is considered to be a probable accident.

5. Expected Accidents

Expected accidents are those which probably will occur frequently throughout the reactor life. Examples of expected accidents are fuel failures and minor leaks in instrument lines. Such accidents have little importance from a reactor hazards viewpoint and are

more properly considered operational problems.

4.1.2 Summary

Typical accidents considered in this review are shown in Table 4-1.

TABLE 4-1

REACTOR ACCIDENTS

<u>Accident</u>	<u>Type</u>	<u>Steam Vented From Bldg.</u>	(1) <u>Maximum Extent of Fuel Melting</u>	(2) <u>Fission Product Release Potential</u>	(3) <u>Applicable Remaining Back-up Systems</u>
Simultaneous Failure of Major Line and Emergency Once-Through Coolant System	Maximum Credible	Yes	Extensive	Environ	4,5,6,7
Inadvertent Opening of Dump Valves With Simultaneous Failure of Emergency Once-Through Cooling	Credible	No	Extensive	Building	4,5,6,7
Failure of Inlet Connector With Simultaneous Failure of Flow Monitor	Credible	No	Single Tube	Building	1,2,3,4,5,6,7
Loss of all 13.8 KV Electrical Power	Credible	No	None	None	3,5,6,7
Partial Failure of Inlet Connector Yielding Zero Flow	Improbable	No	Several	Contained	1,2,3,4,5,6,7
Failure of Major Line	Improbable	Yes	None	None	3,4,5,6,7
Failure of Tube Nozzle Closure	Improbable	No	Several Elements	Building	1,2,3,4,5,6,7
Inadvertent Discharge of Fuel Element Onto Elevator	Probable	No	Several Elements	Building	6
Complete Failure of Inlet Connector	Probable	No	None	None	1,2,3,4,5,6,7
Failure of Process Tube	Probable	No	None	None	1,2,3,4,5,6,7
Inadvertent Opening of Dump Valve	Probable	No	None	None	3,4,5,6,7
Loss of One Power System	Probable	No	None	None	2,3,4,5,6,7
Fuel Failures	Expected	No	None	Contained	1,2,3,4,5,6,7
Small Leaks in Auxiliary Systems, e.g. Instrument Lines	Expected	No	None	None	1,2,3,4,5,6,7
Rising Nuclear Level of Shorter Than Normal Period	Expected	No	None	None	1,2,3,4,5,6,7

Code words and numbers (Columns 1,2, and 3) are defined on the following page.

(1)

Extensive	-	Fuel Melting Extensive But Not Complete
Several Tubes	-	Meltdown of Several Tubes
Single Tube	-	Meltdown of Single Tube
Several Elements		Meltdown of Several Elements
None	-	No Fuel Melting Would Be Expected

(2)

Environ	-	Significant Release to Building Spaces; Limited, Controlled Release to Environ
Building	-	Limited Release to Building Spaces, Negligible Release to Environ
Contained	-	Contained Within Stack and Primary Loop, Very Limited Release to Building Spaces or Environ
None	-	No Fission Product Release From Fuel Would Be Expected

(3)

1	-	Primary Coolant System, Normal Operation
2	-	Primary Coolant System on First Back-up
3	-	Emergency Once-Through System
4	-	Regular Graphite Moderator Coolant System
5	-	Emergency Once-Through Graphite Coolant System
6	-	Fog Spray System
7	-	Shield Cooling Control Rod Cooling and Reactor Gas System

4.2 Inadequate Cooling

4.2.1 General Considerations

Adequate process cooling is defined, as a minimum, as that required to maintain all components of all fuel elements in the reactor below the melting point. In practice cooling is adequate when damage to any reactor component is prevented. Credible accidents which could jeopardize adequate cooling of a major portion of the reactor include rupture of a major line in the primary coolant system combined with coincidental failure of the emergency once-through, raw-water cooling system. Other failures could result in loss of adequate cooling to one or more process tubes.

Since inadequate cooling is a major reactor hazard, the subject is covered in some detail in the following pages. Various classes of credible accidents are analyzed from the standpoints of the potential for inadequate cooling and the expected consequences of loss of adequate cooling.

In summary, three classes of ruptures are considered:

1. Major line ruptures in which the system depressurizes rapidly through the break without need for reliance upon operation of the dump valves. For such ruptures the steam can be vented and the building resealed before fuel can melt.
2. Minor ruptures (up to about 10 square inches in effective area) for which the injection system can provide adequate makeup water. For such ruptures the building will not be vented. No fuel melting would be expected.

3. Intermediate sized ruptures which are beyond the capacity of the injection system, and which would not, by themselves, depressurize the system within the minimum time for fuel melting (~ 260 sec.). Dump valves are provided to rapidly depressurize the system under these circumstances. Only in the event of multiple failures of dump valves to open combined with specific locations of particular sized ruptures would the potential exist for releasing any fission products from the confinement zone before the vents could be resealed. This potential is believed to be low.

4.2.2 The Meltdown Incident

If all heat removal facilities are lost from the reactor during operation or for an extended period after shutdown, the fuel elements must melt with consequent release of fission products into the primary loop piping and probably into the 105-109 Building confinement zone.

The major facilities for removal of process heat from the reactor are:

1. The pressurized primary loop on recirculation.
2. Portions of the primary loop on low pressure using once-through raw-water coolant.
3. The moderator cooling system on normal recirculation operation, and
4. The moderator cooling system on emergency once-through coolant operation.

Minor heat removal facilities include thermal shield, biological shield, and control rod cooling systems. These systems are not usually con-

sidered in hazards analyses, but may be significant if the reactor has been shutdown before major heat removal facilities have been lost.

It is not considered credible that all four major facilities for heat removal could be simultaneously lost, unless the loss was caused by an event such as a very severe earthquake which would in itself be catastrophic.

4.2.3 The Maximum Credible Accident

The maximum credible accident is defined as the failure of a large line in the primary coolant system with a coincidental failure of the emergency once-through, raw-water reactor cooling system. It is postulated that the accident could occur during reactor operation at the full design power level and that the line failure may be circumferential, occur instantaneously, and result in a condition where flow out both ends of the break is unrestricted.

Analyses show that should such an accident occur part of the fuel load would melt. However, the steam released would not overpressurize the building and the steam vents can be resealed before fuel melting starts.

Plant design has been on the basis that in case of an accident of this nature, the following sequence of events will occur:

1. A major line rupture and the emergency once-through raw-water cooling system fails (the accident assumed).
2. Pressure in Zone I area increase to 1/4 psig and the fog spray turns on automatically.

3. Pressure further increases to 1.5 psig and the cross-vents between the 105 and 109-N Buildings and the weather caps on the roof steam vents blow open.
4. All openings into Zone I except for steam vents automatically close.
5. The pressure in Zone I peaks at a level equal to or lower than the maximum design value, i.e., 8 psig in the steam generator cells, 5 psig elsewhere in the zone.
6. Pressure in primary coolant system and in Zone I decays.
7. Steam vents automatically reseal when steam release rates drop to near the fog spray capacity.
8. Backup closures are manually actuated from the control room.
9. Fuel melting starts.
10. Pressure in Zone I reaches second peak below 5 psig and decays as steam remaining in the building is condensed by the fog spray.
11. Vents to filters opened by manual control (optional).

In the preceding sequence, it is tacitly assumed that the fog spray system functions as designed, that all openings into Zone I are, or can be closed, and that the steam vents can be resealed. It is believed that these assumptions are all reasonable.

Assuming that the fog spray system would also fail would be tantamount to assuming:

- a. Three independent systems, the primary coolant system, the reactor last-ditch system and the fog spray system would fail at the same time by coincidence, or

b. A single incident could cause the failures.

Neither assumption is credible. It is not credible that three highly reliable systems would by coincidence fail simultaneously. Secondly, considering the strength, integrity and independence of the systems, it is not believed credible to assume that a single incident could cause the multiple failure, unless, like a very severe earthquake, the causitive event alone would be catastrophic.

Most penetrations into Zone I will be closed and sealed during reactor operation. Those that are not, e.g., ventilation ducts, are each fitted with two independent closure devices. The closure devices are designed to be highly reliable (details are given in Volume II). Should for some reason the normal closure fail it is reasonable to assume that the secondary closure would operate. The steam vents are likewise fitted with double closures. The primary closures are butterfly valves. Should these not completely seal the vents from the control room inflatable seal type back-up closures are provided which can be inflated to provide the seal.

It is critical that the pressure within Zone I not increase above the design value. Otherwise, confiner leak tightness would be compromised. The vent area provided, a total of 364 square feet from 13 vents, is ample to maintain the initial surge within bounds. One analysis considered breaks occurring at a number of different locations in the primary coolant system. This analysis showed that the maximum initial surge would not exceed 3.6 psig. This maximum would be obtained from a failure of the inlet manifold that supplies coolant to the sixteen inlet vertical headers.

The above quoted steam release rate calculations were based on the assumption that the steam generators would continue to function during the transient. This assumption is reasonable because even if all pumping power for the secondary loop were lost there is enough water in the shell side of the steam generators to provide for cooling by boiling during the time required for the initial pressure buildup in the building. Further, the feed-water lines to each of the ten steam generators are protected by check valves. Therefore, even if the cause of the maximum credible accident were a severe blast or shock, the continued operation of most of the steam generators during the early part of the transient seems assured. Other assumptions used in the analysis are conservative, i.e., they tend to overstate steam release rates and, hence, the initial pressure surge.

Should a major accident occur, it is necessary that the steam vents reseal before fission products are released to environs. This has been simplified to a requirement that the building reseal before fuel elements melting begins. This simplification is probably highly conservative. There will probably be a significant interval between initial fuel melting and the time when fission products could reach an open building vent. This requirement leads to the question: "If the vents are resealed before fuel melting starts, will the secondary pressure buildup exceed the design limit?" The answer to this depends both on the time when fuel melting does start and the relationship between the magnitude of the secondary pressure peak and the time the vents reseal.

For the major sized line breaks studied, the steam vents can be closed

within 86 seconds or less after the occurrence of the failure without a subsequent over-pressurization. For the case of the inlet manifold failure, the vents can be resealed about ten seconds after steam release starts.

The minimum time for fuel melting to occur has been calculated to be 260 seconds. This calculation considered that the fuel were cooled only with the water in the tube at the time of the incident. This water is boiled out in about two seconds. In most line failures, cooling will be continued for an additional period of time as water from the inlet piping flows through the tubes. Furthermore, the melting point of uranium, but not zircaloy-2 has been reached in 260 seconds. Escape of fission products from the fuel will not occur until some time after uranium melting starts. Hence, for major pipe failures, the steam vents may be resealed after a maximum elapsed time of 86 seconds. Under the most severe case, uranium melting will not occur for an additional 174 seconds. Therefore, it is concluded that the building can be vented, and resealed after a maximum credible accident without over-pressurizing the building and without permitting release of fission products out the steam vents

It should be noted that the melted uranium will be resolidified as a consequence of heat removal by the graphite cooling system within a short time, perhaps as short as five minutes. This will terminate melting, tend to stabilize the system, and minimize release of fission products from the fuel.

4.2.4 Major Line Failure Followed by Proper Operation of The Last-Ditch System

The damage resulting from a major line rupture is greatly minimized if the reactor last-ditch system functions as intended. A typical sequence of events which would occur in this instance would be:

1. Line rupture. (The accident assumed).
2. System pressure drops, reactor automatically scrams and emergency once-through system diesel pumps automatically start.
3. Dump valves and inlet emergency supply valves automatically open.
4. Building pressure reaches 1/4 psig and fog spray system is automatically actuated.
5. Openings into Zone I automatically seal.
6. Building pressure increases to 1.5 psig and the cross-vents between the 105 and 109 Buildings and the weather caps on the roof steam vent are blown open.
7. Primary water pressure decays to about 335 psig and emergency water enters the front manifolding.
8. Reactor cooled on once-through basis by last-ditch system.
9. Steam vents reseal.
10. Reactor maintenance initiated.

In this instance, inadequate cooling would not be encountered except possibly, in the special case where the failure occurred on one of the sixteen inlet supply headers at a point downstream of the check valve. A failure at such a place might result in inadequate cooling to parts of the two vertical columns of process tubes supplied. This case has not yet been examined in detail. The probable consequences would be

limited to fuel damage - with the possibility of some fuel melting, in a maximum of about 55 process tubes. The potential for fuel melting would be reduced by back-flow from the pressurized outlet risers and by heat transfer to adjacent cooled process and graphite cooling tubes.

4.2.5 Inadvertent Operation of Dump Valves

A spurious primary coolant system dump during equilibrium power operation is a credible, but unlikely incident. The primary loop has been designed to withstand the thermal and vibrational shock of at least six dump transients. The incident is chosen for discussion because it represents the most severe of a class of incidents involving rapid depressurization of the system and reliance on the emergency once-through raw water system without a preceding major line failure.

The opening of some or all the dump valves will be followed immediately by an automatic reactor scram.

The diesel-driven pumps which circulate the emergency water are started by the reactor scram signal. Then the cause of the scram has been identified the engines will usually be turned off manually since practically all scrams will be caused by incidents not requiring emergency once-through cooling. In the event of an inadvertent dump, however, the pumps will remain on and when the primary coolant system pressure decays to about 335 psig, raw water will enter the inlet manifold.

For a period of time, fuel in some tubes will be cooled by two phase

flow or, in the most severe case by superheated steam. As the fuel power decays and as the system is further depressurized, liquid coolant flow will be re-established. Transient calculations indicate that fuel element temperatures will not exceed 1580 degrees F, about 480 degrees F (267 degrees C) below the melting point of uranium. Hence, in the incident postulated, adequate cooling is maintained and no fission product release will occur. At least two of the eight dump valves can fail to open without resulting in fuel melting. There are two compounding events, neither of which are likely to occur, which could cause inadequate cooling. First, it is conceivable, but hardly credible, that the dump valves would open but that the emergency cooling system would not provide coolant. Should this occur, fuel melting would probably result; but fission products would be confined to the primary coolant loop and the quench tank.

Escape of any significant quantity of fission products to the atmosphere would be highly unlikely.

Second, it is possible that fewer than six of the eight dump valves would open. However, even in the most severe situation, the resultant limited fuel melting would not result in major release of fission products to the environs. The fission products would largely be confined to the primary coolant system and the quench tank. Some noble gases and vapor entrained fission products would probably issue through the quench tank vents.

4.2.6 Line Failures of Intermediate Size

The analysis of events following line failures of intermediate size is still in progress. It is probable that with a certain limited range of failure sizes:

1. The leak-rate will be in excess of the capacity of the primary coolant injection pumps.
2. The leak rate will be too low to cause a rapid system depressurization with steam venting and zone resealing.

Dump valves are provided to rapidly depressurize the system to assure building resealing before fuel melting.

An incident of this nature will probably not result in inadequate cooling. The probable sequence of events is:

1. A line failure of intermediate size occurs (the accident assumed).
2. Pressurizer level drops, injection pumps accelerate to maximum capacity.
3. Pressurizer level continues to fall causing a reactor power setback to occur.
4. At a still lower liquid level in the pressurizer, the reactor scrams.
5. The fog spray system turns on automatically when Zone I pressure reaches 1/4 psig.
6. Cross vents between the 105-N and 109-N Buildings and the roof vents are blown open.
7. Dump valves open as loop pressure decays and coolant flow rate drops.
8. Reactor cooling with emergency once-through water started.
9. Steam vents reseal.

10. Maintenance is initiated.

No fuel melting would be expected unless the incident should be compounded by failure of the emergency water supply. Even if emergency cooling should fail, the building would be resealed before fuel melting could occur if five or more dump valves opened on signal. While fission products would not be confined entirely within the primary coolant system, the line ruptures would by definition be relatively small in size and escape of fission products to the building atmosphere would be limited.

It is conceivable, but hardly credible, that the number of dump valves opening would be insufficient to permit depressurization, steam venting and vent resealing before fuel melting initiated. Should this occur, escape of some fission products out the open steam vents could occur. While this incident is highly unlikely, it will be examined in more detail in the future.

4.2.7 Minor Sized Line Failure

The last type of line failure to be considered are minor sized failures. These failures are, by definition, so small that the primary coolant injection pumps can accelerate and prevent system depressurization. Thus, the reactor can be shutdown in an orderly fashion, cooled and depressurized without dependence on the emergency once-through cooling system.

The steam released by the leak may pressure the Zone I area to 1/4 psig turning on the fog spray system. Most leaks of the size that can be accommodated by the injection system will result in steam release rates

within the fog spray capacity. That is, when the fog spray turns on the building pressure will cease increasing and the steam vents will not open.

Leaks of the size considered could occur anywhere in the system but will affect cooling adequacy only if they should occur in process tubes or in the individual connectors supplying coolant to the connectors. Fuel melting will still probably not occur even with these special cases.

Should a front connector fail completely, the reactor would automatically scram and back-flow through the process tubes would be adequate to prevent fuel melting. With partial failures of the connectors, the leak rate would probably be low enough so that flow down the connector in the normal direction will provide adequate cooling or the leak will be large enough so that again adequate "backflow" will occur. There is a special case where for a range of certain sized holes front and rear pressures would be nearly equal, and flow through the tube would be nearly stopped. In this unlikely case, fuel melting could occur, however fission products would be confined, at least to a major extent, within the process tube (or the graphite stack) and significant release of fission products outside the reactor building would be highly unlikely.

Breaks in the process tubes could cause fuel overheating. However, fuel melting resulting from such a break is highly unlikely. If fuel melting should occur fission products would largely be confined within the graphite stack.

4.2.8 Power Loss

The electrical power supply system has been designed to high standards of reliability. There are two independent electric power sources either of which is adequate for shutting the reactor down and providing adequate cooling for indefinitely long periods of time. (See Section 6.3). The plant will not be operated unless both these power sources are on line. That is, should either power source be lost, the reactor will immediately be shutdown.

In the unlikely event that both power sources should be lost simultaneously, the reactor will be scrammed and when the coolant flow and temperature drops will be automatically cooled with emergency once-through cooling water. This transient will not be as severe as that discussed in Section 4.2.4, "Spurious Operation of the Dump Valves".

Electric power is required to keep the safety balls and control rods out of the reactor. Upon electric power failure, the safety balls would fall into the reactor by gravity, and the rods would be driven into the the reactor by compressed nitrogen.

4.3 Power Excursions

The potential for power excursions in the NPR, their consequences and the effectiveness of remedial measures are discussed in this section. Power excursions are directly related to nuclear excursions. They are induced by an upset in the reactivity balance in the reactor which can be brought about by either errors or accidents involving the control systems or possibly by other accidents which can alter the neutron balance.

4.3.1 Accidents Involving Control Systems

The purpose of the horizontal control and safety rods is to maintain control over the nuclear reaction in the reactor at all times. Accidents involving these systems, therefore, have direct consequences on nuclear safety.

The design of the control and safety rods precludes accidents such as rod fall-out or shoot-out. To be considered then, are the potentials for withdrawing rods at the maximum rate and the protection afforded by the safety systems for such an accident.

The drive system proposed for the NPR rods is powered by low pressure oil pumps. There will be two pumps on each side of the reactor, having capacities of 3.25 and 5.25 gpm. There will also be two high pressure pumps per side to feed safety circuit accumulators, which, if called upon, will scram the rods in the event of a safety circuit trip. The scram speed will be sufficient to attain 90 per cent control

strength ($3/4$ rod travel through the active zone) in 1.25 seconds after the scram valve has been de-energized. It will be physically impossible to scram the rods out of the pile.

As designed, use of both low pressure pumps simultaneously will be prohibited by a switch which will allow power to only one pump per side at a time. Use of the 5.25 gpm pump has been calculated to yield a maximum withdrawal rate of 1.44 per cent k per minute.

Fast Startup

Prior to any reactor startup, a reactivity status evaluation must be made to determine the amount of control rod to be withdrawn on the initial rod pull. The minimum shutdown margin will be ~ 3.5 per cent k (in the xenon-free, ripe pile), thus a minimum of 40 per cent of the horizontal rod system can be withdrawn without achieving a critical state.

Based upon past Hanford experience and taking into consideration additional factors unique to N Reactor, an upper limit for error in a reactivity prediction prior to startup is 1 per cent k. If rods are withdrawn at the maximum rate possible (1.4 per cent k/min) beyond the point of criticality, approximately 30 seconds of rod withdrawal would be required to reach prompt critical and approximately 40 seconds to attain an excess reactivity equal to the prediction error.

During this entire range, however, the low-level nuclear instrumentation is effective and both level and period trips will take corrective action. If the postulated error went undetected by the nuclear instrumentation, the reactor by virtue of its negative Doppler and coolant temperature coefficients of reactivity would limit the excursion. Heating both fuel and water to 200 C (which is below the saturation temperature for the water at the 300 psia loop pressure required before startup) would introduce over 1 per cent k negative reactivity. The relaxation time of the graphite is too long to permit any significant graphite heating during the excursion, thus any positive reactivity effect is precluded. The excursion would therefore be a mild one and the total energy required to heat the fuel and coolant sufficiently to stop the excursion would not be large enough to physically damage the reactor beyond thermal stressing of the fuel elements.

Pulling the entire control system at the maximum rate possible until all rods were out of the reactor would result in an intolerable situation if not detected and the action reversed. The transient conditions and ultimate reactor state for this accident have not been studied in detail but the following qualitative statements can be made:

1. Six minutes are required to pull all control and safety rods out of the reactor at the maximum speed.
2. Simultaneous failure of all protective instrumentation

and all power level measuring devices must occur for this accident to proceed. The reactor would not reach prompt critical after initial criticality in less than 30 seconds.

3. The maximum excess reactivity is 6.2 per cent k - rods out, cold, clean, ripe reactor.
4. The negative reactivity contributions of the coolant temperature coefficient up to 210 C (saturation temperature at the 300 psia startup loop pressure) will be 2.2 per cent k.
5. The negative reactivity contribution of the Doppler coefficient will be 0.5 per cent k up to 210 C.

Control Rod Accidents During Operation

A rod-swap may be made at the discretion of the reactor operating crew to correct an unbalanced flux distribution. In this operation, a rod is inserted in small increments (at the low rod speed) while another is alternately withdrawn in increments to maintain a nearly-constant pile power level and reactivity. It is unlikely, but possible, that simultaneous, full-speed insertion and withdrawal of two rods would be attempted. A limiting case would be one where one rod could be withdrawn at full speed while the other was not inserted.

A fully inserted control rod may be fully withdrawn, using the high speed oil pump, in not less than 60 seconds (fast speed is 30 feet/min). The maximum single rod strength is

_____ per cent k. If it is assumed that all the rods could be withdrawn before a flux level trip point is exceeded, the reactor would attain a rising period of _____ seconds. If it is further assumed that a full three seconds are required to insert the scram rods, the reactor would exceed the equilibrium power level by _____ per cent. The same accident under the same assumptions in a present Hanford reactor would result in exceeding the equilibrium power level by 35 per cent. This simple approximation ignores the negative prompt temperature coefficient and the actual response of the safety system. The combination of these events would greatly reduce the excursion.

4.3.2 Other Accidents Leading to Possible Excursions

Accidents other than those involving errors or malfunctions in the control systems which can lead to power excursions in the NPR are the cold water accident and the flooding accident. Loss of coolant will not lead to a power excursion (See Section 5).

Cold Water Accident

The NPR lattice gains reactivity when the density of the coolant water is increased. Thus any decrease in inlet water temperature must be investigated to assure that any such occurrence is within the reactor safety criteria.

Since the primary coolant is under high pressure, there are only limited sources of cold water available to reduce the temperature of the reactor coolant. The secondary coolant system and the last-ditch coolant supply is at a lower pressure and therefore cannot be accidentally injected into the pressurized system.

Thus, for the case of large cold water injection the system must be at a low pressure period of the operation cycle. When the system is at operating pressure, either at low power or full power, there exists only two sources of water below loop temperature. The makeup water supply is injected by high pressure pumps and preheated before entering the loop. There may also exist source of cold water in piping not in the flow stream. Such would be the case for a heat exchanger loop removed from service.

Normal Makeup Water

The makeup water at 100 F will be available from three pumps with a total available flow of approximately 5000 gallons per minute (2.58×10^6 lb/hr). This water passes through a preheater before entering the primary loop. The maximum injection rate, not preheated, would result in lowering the reactor inlet temperature 10 F. This produces a reactivity variation of approximately +1.2 mk for fuel with an average exposure of 700 MWD/T. Such a nominal reactivity increase would present no safety problem.

Cold Water From Within Loop

During reactor operation, normally all five heat exchangers will be in use and the reactor will never be operated with fewer than four in use. If one assumes operation with four cells, the reactivity effects of suddenly "cutting in" the fifth cell can be calculated. Actually, this will not be attempted during reactor operation and the probability of it occurring accidentally without a reactor scram is vanishingly small. However, even if the cold water in the "down" cell were to be suddenly injected

into the stream at a rate equal to the flows from each of the other cells, then for a period of about 15 seconds, the reactor would be obtaining about 20 per cent of its flow from this low temperature source. If it is assumed that this cold water is at 100 F, (a conservatively low number) and the reactor inlet is at its design level of 367 F, the average bulk inlet would drop to 314 F. This would result in a six mk increase in reactivity.

Due to different lengths of flow paths, this cold water would not arrive at each process tube at the same time. If we assume five seconds for the fifth heat exchanger loop to open and an additional three seconds of varying flow paths, the time of decreasing temperature and increasing reactivity would be spread over a minimum of eight seconds. The resulting pile reactivity transient would cause a process variable to exceed trip limits and result in a reactor scram. Under some operating conditions, the transient might be mild and simply result in a mild power increase. Due to the slow application of the cold water to the active core, no nuclear event beyond a mild power increase and reactor scram would occur in any case.

Emergency Coolant Injection

The injection of the emergency coolant supply (cold, raw river water) into the primary coolant system can occur only on major failures within the coolant system, such as major pipe failure or on opening of the rear dump valves. Even before startup, the primary loop will be pressurized so in no case could emergency

water flow into the reactor without major component failure. Any such failure would scram the reactor. Only after the system pressure had dropped to 350 psi, could emergency coolant enter. The time required for such a series of events would assure that the reactor would be scrammed and be subcritical before the cold water could enter.

Such a failure would result in a rapid depressurization of the system. Any lowering of water density within the process tubes will result in a reactivity loss. Thus, the immediate effect would be a reduction in reactivity. The later addition of cold water to the lattice will result in a reactivity increase from 3.1 to 3.6 per cent k, depending on reactor conditions. In all cases, the incremental control held in safety rods and the ball 3X system are sufficient to control the excess reactivity. See Table 4-2.

TABLE 4-2
MAXIMUM COLD WATER REACTIVITY
(Green Fuel)

<u>Reactor Condition</u>	<u>k Cold Water</u>	<u>k Safety Rods*</u>	<u>k Ball 3X*</u>
Equilibrium Power	+3.6%	-7.1%	-8.0% k
Zero Power (Cold Graphite, Hot U and Water)	+3.1%	-5.1%	-8.0% k
Just Critical (Xenon Free-Cold)	+3.1%	-3.3% k	-8.0% k

*Assumes only one system in pile. Combined strength of both systems will be less than algebraic sum due to shadowing effects.

Flooding Accident

The NPR has an undermoderated lattice, a water loss results in a reactivity loss, and therefore the addition of more moderator

within the lattice may result in an increase in reactivity. The mechanism of introducing pressurized high temperature water to the low pressure within the graphite voids is difficult to formulate. For the case of high temperature graphite, the situation approaches the incredible. In the following analysis, the lattice has been assumed to be progressively flooded with water in a favored (low thermal flux) position and the maximum reactivity increases calculated. Thus, such an investigation establishes an upper bound for all such accidents. Since even with these near incredible assumptions, the reactivity increase is within the strength of the control system, the "flooding" accident is not considered serious. The worst conceivable case of an optimum amount of water, located in an optimum position, in every lattice could never be reached. A realistic analysis of expected water distribution would show a change in pile reactivity considerably lower than the maximum local change.

Two sources of water are available within the reactor; the primary coolant and the graphite cooling systems. A tube failure in the graphite cooling system could not result in sufficient quantities of water to directly result in an event of any magnitude due to the low pressure and flow rate. The process tubes which carry the primary coolant are at high pressure and their large flow capacity presents the only source of large quantities of water. Due to the open nature of the graphite structure, the stack will not retain any large amounts of water. An optimum amount would only be available during a transient per-

iod, in the form of steam, or under steady flow conditions.

The rupture of any process tube in a manner to introduce large quantities of water will result in flow and pressure transients resulting in a reactor scram. The time required to flood a large fraction of the reactor core with water from one process tube would be sufficient to allow all reactor control systems to be fully inserted. In the remotely possible case of multiple tube rupture, the flooding time for a significant fraction of the pile may be short.

The reactivity transient resulting from flooding will depend on the reactor conditions. In all cases the transients for green unirradiated fuel will be the most severe. During reactor startup, the most severe conditions are calculated to be with the coolant at operating temperature. Figure 4- 1 depicts the transient for the case of green fuel, hot coolant and fuel, and cold graphite. The maximum reactivity increase is less than .25 per cent k.

During the process of heating the coolant and fuel to operating coolant temperature, it is necessary to withdraw 5.1 per cent k of control rods. Thus, sufficient horizontal control rods are available to counteract the reactivity increase due to flooding. Flow monitoring instruments on each process tube would sense any gross water loss to the reactor core and would scram the horizontal control rods in 1.2 seconds. For a leakage rate equal to the normal flow to one process tube, 150 seconds would be required

to insert the volume of water necessary for total optimum flooding (assuming no leakage).

The release of high pressure, high temperature water into the graphite voids with the graphite at high temperature presents a most difficult time sequence of events to predict. The calculations were made assuming saturated water at low pressure had flooded the entire reactor core an optimum amount and in an optimum location. The assumption was also made that the graphite remained at operating temperature. Figure 4- 1 also shows the pile reactivity effects of flooding the lattice under operating conditions. These present more severe transients than the previous case. The maximum reactivity gain is 2.2 per cent k, which is well within the scrammable control at equilibrium. (See Table 4- 2).

If an accident should occur that would flood the reactor, there exists the necessity of maintaining the reactor subcritical if the flooding cannot be stopped before the xenon decays. Flooding of the cold reactor results in a maximum reactivity gain of seven mk above the cold, clean reactivity. The highest reactivity state that can be obtained in the cold reactor is less than the reactivity state during the combined cold water and flooding accident at operational power (See next section).

Combined Cold Water Accident Plus Flooding

In the case of an extreme process tube rupture, the reactor core may be flooded and the backup coolant supply injected into the

primary coolant system. These combined occurrences present the greatest possible positive reactivity transients. The flooding accident would present the same problem presented in the previous section. At a later time the backup water would reach the reactor core. The time lapse is sufficient to assure all available reactor control to be fully inserted previous to the cold water arriving within the reactor core. Figure 4-2 shows the reactivity as a function of lattice flooding for both cold and operational temperature water at zero power (cold graphite). The same accident occurring at full power is also described in Figure 4-3..

The maximum conceivable reactivity increase is 5.0 per cent k. The calculated upper limit assumes the most severe conditions and can never be experienced. Any of the following variations from the most severe condition would lower the reactivity increase: (1) cooling of the graphite during the time required to inject cold water, (2) flooding of only a part of the graphite stack with water, and (3) flooding with more or less than an optimum amount of water.

In both cases considered the maximum conceivable k due to the combined accident (5.0 per cent k in the reactor at equilibrium, and 3.7 per cent k in the zero power reactor) is well within the control held in either the safety rod system or the Ball 3X system alone.

4.4 Earthquakes

No damaging earthquake has ever been recorded in the immediate vicinity of the Hanford Plant. However, earthquakes causing moderate to severe damage have occurred in the Pacific Northwest. Moreover, earthquake damage could result in a severe accident if suitable resistance were not provided. Therefore, prudence required that the critical systems in the N Reactor complex be built to withstand severe earthquake shocks.

A detailed account of the seismic activity in the state of Washington will be included in Appendix I of Volume II. Briefly, tremors recorded at Hanford since 1949 have apparently approached an intensity of six on the Modified Mercalli Scale of 1931. This corresponds to a ground acceleration of about 0.05 g and is sufficient to cause the "falling of plaster and chimneys, (with) damage small as is shown in Table 4- 3 . In keeping with experience, the area around Hanford has been included in Zone 2 in the seismic probability map adopted as part of the Uniform Building Code by the International Conference of Building Officials.

The Puget Sound area to the west of Hanford, a part of Montana near Helena to the east, and coastal California to the southwest is classed as Zone III areas (areas where destructive earthquakes have occurred, and might reasonably be expected at any future time). Figure 4- 4 shows the seismic zones in contiguous, continental United States.

A consideration of this past seismic activity and the potential seriousness of earthquake damage led to the issuance of a "Seismic Resistance Criteria" in 1960 which was later included in a revised issue of the Technical Criteria for N Reactor. These criteria recommend that all

UNCLASSIFIED

MODIFIED MERCALLI INTENSITY SCALE (1931) WOOD & NEUMANN		a - GROUND ACCELERATION	BUILDING CODE ZONE
I	Detected only by sensitive instruments.	$\frac{\text{cm}}{\text{sec}^2}$ $\frac{a}{g}$	
II	Felt by a few persons at rest, especially on upper floors; delicate suspended objects may swing.	2 3	0
III	Felt noticeably indoors, but not always recognized as a quake; standing autos rock slightly, vibration like passing truck.	4 5 6 7	
IV	Felt indoors by many, outdoors by a few; at night some awaken; dishes, windows, doors disturbed; motor cars rock noticeably.	8 9 10	
V	Felt by most people, some breakage of dishes, windows and plaster, disturbance of tall objects.	20 30	1
VI	Felt by all, many frightened and run outdoors; falling plaster and chimneys, damage small.	40 50 60	
VII	Everybody runs outdoors, damage to buildings varies, depending on quality of construction; noticed by drivers of autos.	80 100	2
VIII	Panel walls thrown out of frames; fall of walls, monuments, chimneys; sand and mud ejected; drivers of autos disturbed.	200 300	
IX	Buildings shifted off foundations, cracked, thrown out of plumb; ground cracked, underground pipes broken.	400 500 600	3
X	Most masonry and frame structures destroyed; ground cracked, rails bent, landslides.	800 1000	
XI	New structures remain standing; bridges destroyed; fissures in ground, pipes broken; landslides, rails bent.	2000 3000	
XII	Damage total; waves seen on ground surface, lines of sight and level distorted; objects thrown up into air.	4000 5000 6000	

TABLE 4-3
APPROXIMATE RELATIONSHIP AMONG EARTHQUAKE INTENSITY,
GROUND ACCELERATION AND ZONES

structures and buildings critical to reactor safety be designed utilizing the requirements of Zone III of the Uniform Building Code. Also the criteria recommends that components and items of equipment critical to reactor safety be designed to withstand a horizontal ground acceleration of 0.20 g using the dynamic response calculation methods or, where that calculational approach is not practical, be designed to withstand a static lateral force coefficient of 0.5 g.^(a)

Zone II requirements are recommended for buildings not critical to reactor safety and a basic horizontal ground acceleration of 0.1 g for non-critical structures, buildings, components, etc., which are not specifically defined by the building code.

The piping and components required for cooling the reactor on a once-through basis with raw river water have been designed on the basis of a 0.5 g static force which is in accordance with the criteria. This includes the last-ditch raw water system, the reactor inlet piping and pipe supports starting from the common wall between the 105-N and 109-N Buildings, and the rear face connectors and supports. The critical buildings, the river pumphouse, the "high-lift" pumphouse, the reactor building, and the heat dissipation building are designed according to Zone III Building Code requirements.

- (a) In static method of analysis, it is assumed that the earthquake produces forces proportional to the structural weight and the severity of the earthquake. The dynamic method relates the structural response to the natural period and damping of the building. The dynamic response method is considered to be the preferred method wherever it is practical to apply.

The balance of the primary coolant system piping, the secondary coolant system, the circulating water system, and the "fog-spray" system are designed on the basis of a 0.2 g static force. While this static force is not necessarily in conformance with the criteria, the use of 0.2 g does give a structure with considerable earthquake resistance and in some cases this may be equivalent to the 0.2 g dynamic response load.

Further, the systems could probably be upgraded to meet the criteria without great difficulty should these systems prove, on further analysis, to be at an undesirably great variance from the criteria.

As one further precaution, a seismoscope has been included in the reactor scram circuit. Should an earthquake of intensity of MM 5 or greater occur, the reactor will be scrambled and the Ball 3X poison system activated to ensure that the reactor will be sub-critical should the shock be sufficient to cause damage to any of the cooling systems.

Thus, the reactor has been designed to prevent the melting of fuel and the consequent release of fission products during an earthquake comparable to the "El Centro" earthquake in California (19___), and for exceeding the intensity of any earthquakes experienced at the Hanford Site.

4.5 Other Natural Phenomena

4.5.1 Fire

Fires are not considered to be a serious hazard to the N Reactor. The plant buildings, other than the administrative building, and their contents are largely fireproof.

A diesel pump powered water system and portable and fixed chemical extinguishers have been provided to control any localized fire that would occur in electrical equipment or in grease, oil or solvents stored in the buildings. Further, a continually manned central fire station is located near the plant site.

4.5.2 Storm

Buildings and structures were designed in accordance with the provisions of the Uniform Building Code to secure protection against storm damage.

4.5.3 Flood

The highest recorded flood stage at the plant site was 415 feet MSL and occurred in 1894. The area building with the lowest elevation is the river pumphouse. The pedestals for the normal and emergency pumps and the grade level at the pumphouse are at an elevation of 421 feet MSL. Hence, this vital equipment is six feet higher than the highest recorded flood.

The other buildings and structures in the area are at an elevation of 433 feet or higher.

The probability of a "naturally occurring" flood of the proportion of the 1894 flood is quite small. The four dams on the Columbia River upstream of the Hanford site represent a considerable storage capacity; however, the quantitative effects of this series of dams on reducing flood stages has not been determined.

No special measures were taken to protect the reactor against the flood resulting from the breaching of the large dams on the river upstream of the reactor. Such a flood could have devastating results and would likely cause severe damage to the N Reactor as well as the eight older production reactors.

4.6 Strikes and Riots

The possibility of strikes or riots posing a real threat to reactor safety is small. Due to the remote location of the plant and the presence of a plant security force, the occurrence of violence at the plant site is improbable.

4.7 Fuel Handling and Storage

The enrichment level of NPR fuel elements is 0.947 w/o U^{235} . After irradiation, the net effect of U^{235} burnout and Pu^{239} buildup has been calculated to yield a maximum effective enrichment level of 1.0 per cent U^{235} . Light-water-moderated assemblies of uranium of these enrichment levels can be made critical.⁽¹⁾ Therefore, it is necessary that all shipping, handling, and storage procedures of NPR fuel throughout the entire Hanford plant be restricted so as to preclude inadvertent attainment of a critical mass. This section discusses the procedures and safeguards which have been incorporated into the design of the NPR fuel handling system to meet this requirement.

Criteria

Use of 0.947 w/o U^{235} , enriched uranium will not be new to the Hanford plant. Fuel of this enrichment level is presently used in amounts not exceeding ten per cent of the total plant throughput for pile flattening purposes and reactivity compensation.

The procedures and safeguards which have been successful in preventing the attainment of inadvertent critical masses with enriched fuel are based on the following criteria.

1. The handling, shipping and storage of unirradiated, enriched uranium shall be such that both an error or accident in the physical arrangement or mass of fuel and water inundation is required to achieve a critical mass.
2. The handling, shipping and storage of irradiated fuel (where handling, shipping and storage under water is required for cooling and

(1) Criticality is not possible without a moderator present.

radiation protection) shall be such that a critical arrangement is prohibited by physical means wherever possible, and by procedures in only those remaining situations where accidental critical configurations are not subject to prevent by physical means.

3. In cases where control over configuration or shape is not possible, the allowed mass or volume of enriched uranium will be $2/3$ of the value for the uranium-to-water volume ratio which yields the minimum critical mass or volume and will be calculated for the optimum shape (sphere).
4. In cases where control over configuration or shape is possible, the same criterion as 3 will apply except the allowed mass or volume will be calculated for the actual shape or configuration.

Basic Data

Criticality data on 1.0 w/o U^{235} enriched NPR fuel elements in various water-moderated lattices have been obtained from exponential measurements on prototype NPR fuel elements.⁽²⁾ These data have been translated into the critical mass, volume, and fuel densities shown in Table I for both the unirradiated enrichment level (0.947 w/o U^{235}) and for the effective discharge enrichment level (1.0 w/o U^{235}).

Procedures and Safeguards

The procedures and safeguards for the handling, shipping, and storage of NPR fuel are based on the criteria as stated. Assessment of the potential for inadvertent criticality is first made for normal procedures, equipment and facilities, and then for potential procedure-break-

(2) HW-69399, "Exponential Measurements and Critical Parameter Calculations - NPR Fuel Elements in Light Water", R. C. Lloyd and C. L. Brown, July, 1961 1961 (Confidential)

downs or malfunctions.

Unirradiated Fuel Storage

Shipment of NFR fuel from the 300 Area Fuels Preparation Department, will be made by truck. The fuel elements, both inner and outer fuel tubes assembled together, will be shipped in pallets which are designed to hold 36 two foot long fuel elements in a $2 \frac{3}{4}$ " x $4 \frac{3}{4}$ " horizontal array. When a shipment of loaded pallets arrives at the unloading dock of N Reactor, the pallets will be taken from the truck with a fork lift and will be stored in a storage area. The height of the fork lift will limit the stacks of pallets to eight feet, or four pallets high. There is sufficient space in the storage area to store a minimum of 50 per cent more than the number of fuel elements normally required for one month's throughput in the reactor. This will be approximately 90 tons of stored uranium or almost eleven minimum critical masses if the uranium were completely inundated by water and arranged in the optimum geometry.

The mass of uranium per square foot which is physically possible, i.e., by stacking pallets four high, is 708 pounds per square foot of floor area. This is not less than $\frac{2}{3}$ of the value of 740 pounds per square foot shown in Table IA, so procedures will be necessary to limit the stacking to two pallets high which gives a mass of 354 pounds per square foot of floor area.

There are no large water pipes in this storage area and drainage is provided in the floor of the room. Thus, water flooding is not probable.

If overstacking is done (even with pallets loaded with only outer tubes

of the NPR fuel element, for which the maximum material buckling is greater), the maximum fuel densities in the storage area will still be below the minimum densities given in Table 4-4A. Thus, two highly improvable events; (1) severe overstacking which would require equipment beyond that normally available, and (2) complete water inundation to a height of eight feet would both be required to achieve a critical mass in the storage area.

Unirradiated Metal Preparation and Magazine Loading and Storage

From the storage room, the pallets are moved to the vicinity of the depalletizer. The remarks of the preceding section apply here as well -- the same stacking requirements will prevail in the new location. In this area, an entire column of fuel elements and dummies will be laid out in the required charging order and pushed into a tubular magazine preparatory to reactor charging. The loaded magazine will then be moved onto the front work elevator into one of two charging machines. Five charged magazines at a time will be elevated into one of several storage racks on the walls of the charging area. There are sufficient storage racks on each wall to support approximately 65 tons of uranium. The racks are spaced vertically by nine inches and are wide enough to hold five magazines.

The fuel elements while in storage on the walls of the charging area can be represented if flooded as isolated slabs, approximately one fuel element diameter thick and five fuel element diameters wide. The geometrical buckling for such a slab is so large that it is a certainty that a critical mass is precluded as long as the fuel remains in this

storage configuration.

The front work elevator is designed to adequately clear the fuel elements stored on the walls, and is provided with interlocks which will stop the elevator if it brushes against any obstruction. However, an accidental toppling of fuel from the walls cannot be ruled out as a possibility. To achieve a critical mass, however, the fuel must drop into water and be arranged in an optimum geometry. It would require a minimum of 22 magazines of fuel (380 two foot fuel elements) to achieve a critical mass if the fuel fell into a pool of water on or below the front work elevator. In spite of the presence of high pressure piping in the vicinity of the work area, it is very improbable, however, that large amounts of water could collect either on or under the work elevator since drainage holes are provided both on the elevator and in the space below the elevator. Thus, a highly improbable rearrangement of fuel coincident with flooding would be required to achieve a critical mass.

Irradiated Metal Handling and Storage

The discharge of irradiated fuel from the reactor will be affected by pushing the fuel out of the rear of the elevator. The free fall will be broken by a submerged chain net and the fuel elements will be directed towards a conveyor system. This all occurs under water. The conveyor will transport the fuel to an underwater separation area where the fuel elements will be automatically segregated by type, length, and whether broken or not broken.

The maximum fuel discharge rate for small numbers of tubes is estimated to be one column every two minutes or about nine two-foot elements per

minute. Procedures will limit the load on the conveyor at any one time to that of two fuel columns, or approximately 35 two-foot elements. The conveyor speed will be regulated to meet this requirement. The load limit of the conveyor will be about 4500 pounds which is equal to the weight of six columns of fuel. The width of the conveyor is slightly larger than the diameter of a fuel element; thus, the geometry of the fuel during the conveying process is that of essentially an infinite cylinder, three inches in diameter, which is well under the minimum critical diameter of an infinite cylinder. (See Table 4-4B). Thus, under normal discharge conditions, there is no possibility of a critical mass forming on the conveyor.

A stoppage of the conveyor from malfunction so that fuel can pile up at the entry point must be considered. At the maximum discharging rate of a column of fuel every two minutes, which is about 9 fuel elements per minute, it would take 16 minutes of discharging onto a stopped conveyor to build up a mass $2/3$ of the minimum critical mass. It is extremely unlikely that fuel discharging would continue at all under such circumstances. An interlock will immobilize the charging machine when the conveyor fails. Also remote viewing of the conveying mechanism can be done by closed circuit television if necessary, and the conveyor operator who is regulating the speed will also have visual access to a portion of the conveyor.

Up to 1000 gpm of water will be continuously introduced into and skimmed from the discharge and storage basins to keep the water clean, clear and at a temperature below 80 F. Long-life underwater lights will also aid in providing good visibility into the basin.

As the fuel elements leave the conveyor and segregation system, they will be bundled and placed in storage containers called cannisters. These cannisters will hold six bundles of 24 inch long fuel elements, the bundles stacked three high. Each bundle will contain seven fuel elements banded together. Each cannister, therefore, will hold 42 fuel elements or approximately 1750 pounds of irradiated uranium. A 15-ton, overhead, travelling bridge crane with two-one-ton carriages and hooks will be used to move the cannisters from the cannister's loading and banding area to the storage basin proper. All this operation will be under water.

The storage basin will cover an area of 100 feet x 50 feet and will contain water to a depth of 24 feet. This will permit a minimum of 18 feet of water over the fuel elements in the storage positions and 11 feet of water over the fuel elements being transported, since cannisters will be stored in a rectangular array of 24 inches center-to-center spacing in one direction and 33 inches center-to-center spacing in the other direction. They will be physically separated by outriggers so that the minimum distance between cannister edges will be eight inches.

As stored, the fuel mass will be 340 pounds per square foot of floor area which is less than $2/3$ of the 610 pounds per square foot given in Table IB as the minimum critical mass per unit area for the irradiated fuel element.

The possibility of a criticality incident in the irradiated uranium

storage area is higher than in the unirradiated uranium storage area since one factor required for a critical mass is already present, namely the water moderator. Thus, only one accident would be required to achieve a critical state.

Procedures will, therefore, be an important factor in the safe handling and storage of the irradiated fuel. Standards will be provided to prevent double- or triple-stacking of cannisters and to prevent leaving loose fuel elements on the bottom of the storage basin or in the vicinity of the conveyor and fuel segregation system.

The overhead crane can carry two loaded storage cannisters at one time, one on each of two one-ton carriages and hooks. Should two cannisters drop and break apart, spewing fuel elements into toehr loaded cannisters below, it would be possible to build up as much as 11,000 pounds of uranium in one area (two cannisters dropping into the area between four stored cannisters). This exceeds the minimum critical mass of the irradiated fuel elements by about 1000 pounds. One cannister dropping alone would not exceed the minimum critical mass. The dropping of two cannisters is not highly probable since each hook is attached to an independent carriage and one breaking would not affect the other. Furthermore, the probability of fuel dropping (if dropped from two cannisters) into the optimum geometry and spacing to achieve a critical mass is also low.

Transfer Area and Irradiated Fuel Shipment

The transfer area provides the facilities for transferring individual cannisters of fuel from the storage area into the shipment casks pre-

paratory to shipment to the 200 Area Chemical Processing Department. The transfer pits will be about 10 feet x 12 feet in area and approximately 26 feet deep. A 75-ton crane will be used to lift the empty and loaded casks. The 15-ton crane will be used for loading the cannisters into the cask. Each cask will hold four cannisters of fuel (168 fuel elements). This is 7400 pounds of uranium which is 70 per cent of the minimum critical spherical mass and 46 per cent of the minimum mass for the cask geometry itself. These figures are based on the optimum water-to-fuel ratio, which is given by an edge-to-edge spacing between all fuel elements of 0.6 inches. If the fuel bundles do not come apart, the optimum moderator ratio cannot be reached and a significant additional degree of safety is added.

The fuel will be in water in the casks. The casks will be shipped in special railroad flat cars, two casks per car. The casks will have twelve inches of lead in all six sides, hence the minimum cask-to-cask spacing on the flat car would be 24 inches. This amount of lead will limit any neutron interaction between casks so that each cask can be considered isolated.⁽³⁾

Any accident such as derailment or train crash during interplant shipping cannot result in a critical mass. To bring the fuel in the two casks into a critical assembly requires breaking the casks in some manner. This event would release the water and thus remove the moderator.

(3) Broder et. al., J. Nucl. Energy, Part A, Reactor Science 13, 65 (1960)

There are no bodies of water along side the rail track between the N Plant and the separation areas. Hence, there is no chance of the train derailing, the cask breaking, bringing the fuel into optimum geometry, and the resulting mass being dumped into water.

Radiation Alarms

In addition to the physical and procedural safeguards in the handling of enriched uranium, a system of radiation detectors and alarms is to be provided in strategic points in the irradiated-fuel handling areas. These detectors will only be able to signal a criticality situation after the fact.

The consequences of a critical mass forming in any portion of the under-water storage and handling areas are not expected to be of disaster magnitude. Quenching of the reaction is very probable if the chain reaction becomes violent enough to boil water or disrupt the critical assembly. Any radiation emitted, to be damaging, must pass through several feet of water shielding, and harm to personnel should be confined to those who may be directly over the reaction.

The contamination problem will probably be more severe since melting may occur and fission products released into the water. The radiation alarm will be important in evaluating the radiation level of the areas in this event and should warn personnel against unsafe entry.

TABLE 4-4

CRITICALITY DATA - NPR FUEL ELEMENTS IN LIGHT WATERA. UNIRRADIATED FUEL - 0.95% U²³⁵(All buckling values are to an accuracy of $\pm 3 \times 10^{-4} \text{ cm}^{-2}$)

	<u>Tube-in-Tube</u>	<u>Outer Tube Only</u>	<u>Inner Tube Only</u>
Maximum Material Buckling	.001825 cm^{-2}	.002280 cm^{-2}	.002090 cm^{-2}
Minimum Critical* Mass	7620 kg U (380 elements)	4300 kg U (310 tubes)	5820 kg U (870 tubes)
Minimum Critical* Volume	1180 liters	825 liters	945 liters
Minimum Critical p-Cylinder Diameter	38.0 inches	33.5 inches	35.5 inches
Minimum Critical** Mass Per Unit Area	740 lbs U/ft ² (17 elements/ft ²)	520 lbs U/ft ² (17 tubes/ft ²)	656 lbs U/ft ² (44 tubes/ft ²)
Minimum Critical oo-Slab Thickness	22.6 inches	19.9 inches	20.8 inches

B. IRRADIATED FUEL - 1.0% U²³⁵

Maximum Material Buckling	.002480 cm^{-2}	.002900 cm^{-2}	.002680 cm^{-2}
Minimum Critical* Mass	4500 kg U (225 elements)	2800kg U (211 tubes)	3820 kg U (570 tubes)
Minimum Critical* Volume	705 liters	546 liters	620 liters
Minimum Critical oo-Cylinder Diameter	31.7 inches	28.1 inches	30.4 inches
Minimum Critical** Mass Per Unit Area	610 lbs U/ft ² (14 elements/ft ²)	430 lbs U/ft ² (14 elements/ft ²)	545 lbs U/ft ² (36 elements/ft ²)
Minimum Critical oo-Slab Thickness	18.5 inches	16.9 inches	17.8 inches

*Spherical Geometry

**oo Slab Geometry

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4.8 System Stability

The dynamic performance of the N Reactor system has been analyzed in considerable detail.⁽¹⁻³⁾

The most significant aspect of the reactor is that it is relatively fast compared to the total loop. The time constants of the fuel element and coolant transport time through the reactor are on the order of one and one-half seconds, whereas the total coolant transport time is 60 seconds and the effective time constant of the heat exchanger is 20 seconds. Furthermore, there is a very tight loop in the reactor kinetics in the form of the two prompt, negative, temperature-coefficient feedbacks from the coolant water and fuel. These two coefficients produce a power coefficient of $-.00025\%$ k/MW.

The relatively slow driving functions which can be applied to the reactor through primary coolant temperature changes are followed almost exactly by the reactor. In responding to changes in reactor coolant inlet temperature, the combination of the fuel and coolant temperature coefficients of reactivity is such that changes in inlet temperature are attenuated to approximately one-third their value, and appear at the reactor outlet with the opposite sign.

The significant feature of the heat exchanger is that, at rated power operation, any changes in inlet temperature are attenuated by approx-

- (1) HW-66665, "Development and Performance of NPR Reactor Analog Model", D. W. Leiby
- (2) HW-65544, "NPR Heat Exchanger and Load Control Analysis", D. W. Leiby.
- (3) HW-68168 (GEL-100), "NPR Primary Loop Performance Study", D. W. Leiby, October 25, 1960.. (Confidential)

imately a factor of four when they appear at the primary coolant outlet of the heat exchanger. The transport time through the primary side of the heat exchanger is three seconds. Therefore, with a heat exchanger time constant of 20 seconds, it is found that changes in the inlet temperature to the heat exchanger appear much more rapidly at the outlet than do changes in secondary steam flow. In the control of the heat exchanger, it is found that the time constant of the steam flow valve will be an important parameter in the control loop.

The analysis also showed that there can be changes in reactor power level (due to various system disturbances) some of which will be of a transient nature and others which will maintain a steady state deviation from the nominal value. The analysis indicated that operator action to bring the power level back to its original level will be relatively unimportant in the over-all performance of the primary loop. The main feature of operation noted is that if there is a power disturbance, the action of the operator would tend to bring the power back to the nominal value immediately, whereas, in the case of no operator action, there will be a delay for the temperature effects to get around the loop and bring the power level back to the operating value. From the limited amount of work done to date on the effect of operator action on over-all loop performance, it appears that there are no detrimental effects of normal operator action on the over-all loop performance.

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4.9 Xenon Oscillations

The combination of a large core (in comparison to the neutron migration distance) and the relatively high power density in N Reactor results in a complication in the dynamic behavior of the reactor which is associated with the formation and destruction of xenon.

Xenon instability can persist only above a threshold flux which depends both on the size of the reactor and the flux distribution in the reactor. The over-all temperature coefficient also influences the oscillatory tendency; a negative over-all coefficient raises the threshold flux whereas a positive coefficient has the opposite effect - thus acting as an additional driving agent for the oscillation.

A comparison of parameters and characteristics which are important in the assessment of xenon instability is given in Table 4-5 for both N Reactor and the K Reactors. The information in the table is for an axial oscillation of the first mode which is the most prevalent mode in the Hanford reactors - i.e., has the lowest threshold. The threshold fluxes are given for two degrees of axial flattening; the 0% flattening efficiency corresponds to the normal cosine distribution along the axis of the reactor whereas the 30% flattening efficiency includes some depression of the flux in the center of the reactor which tends to make the reactor appear larger.

The parameter "A" listed in the table is a measure of the severity of the disturbance. It is the ratio of the peak local power obtained in a free oscillation to the minimum local power. Amplitude ratios of the magnitudes indicated have never been obtained in any of the Hanford reactors

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chiefly because the cycling periods are long (on the order of 25-30 hours) thereby allowing adequate time to either damp out the oscillation by proper control action or to shut down the reactor if control action is not successful. There have been no shutdowns forced by xenon oscillations in the past three years in any Hanford reactor.

The information in Table 4- can be summarized by saying that N R actor should experience a considerably smaller tendency towards xenon-instability than do the K Reactors. Furthermore, the much more flexible control system in N R actor will make oscillation-control much easier and N Reactor, by virtue of its net negative over-all temperature coefficient, will tend to self-damp oscillations if they start. The threshold fluxes given in Table 4- are computed for a zero temperature coefficient. The net positive temperature coefficient in the K Reactors will, therefore, lower the threshold quoted in Table 4- for these reactors.

The chief factors then contributing to N Reactor's better stability against xenon oscillation are as follows:

1. The ratio of equilibrium flux to threshold flux is low compared to the K Reactors.
2. The net temperature coefficient is negative.
3. Greater control flexibility exists: 87 control rods in N Reactor compared to 20 in the K Reactors.

The only factor acting to increase the cycling tendency is the greater length of N Reactor as compared to the K Reactors.

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TABLE 4-5

PARAMETERS IMPORTANT TO XENON INSTABILITY
A COMPARISON BETWEEN N AND K REACTORS

<u>Parameter</u>	<u>K</u>	<u>N</u>
Average Thermal Flux At 4000 MW	4.3×10^{13}	3.2×10^{13}
Equilibrium Xenon	3.0 % k	3.0 % k
Threshold Flux (0% Flatt)	3.3×10^{13}	2.7×10^{13}
Threshold Flux (30% Flatt)	2.3×10^{13}	2.0×10^{13}
A (0% Flatt)	5	3
A (30% Flatt)	9	7
Migration Area, M^2	600 cm^2	660 cm^2
Reactor Length H	946 cm	1190 cm
H^2/M^2	1490	2140

4.10 Miscellaneous Reactor Hazards

4.10.1 Failure of a Tube Nozzle Cap

Should a front nozzle cap fail and come off the tube nozzle, the fuel load would be ejected from the tube with great force against the thermal barrier and drop into the elevator pit. Possibly fuel elements rebounding from the thermal barrier would become lodged in the front face piping. The steam release would turn on the fog spray but the roof steam vents would not be required to open. It is possible that some of the fuel would melt or ruptured fuel oxidize releasing fission products into the Zone I confinement spaces.

The incident, while serious from an operations" standpoint, does not present a critical hazard to areas adjoining the plant, due to the efficiency of the fog spray in sweeping out fission products, the opportunity for fission products to plate out on ducts and walls, and the filter system provided for the effluent air.

A rear cap failure would be probably much less significant. The fuel elements would be expected to fall into the pickup chute and thence into the storage basin where they would rest under about 20 feet of water. If fuel elements should lodge in the rear face piping they would present a problem similar in magnitude to fuel elements lodged in front face piping.

4.10.2 Rupture of a Process Tube

Cooling of fuel elements inside a ruptured process tube was covered in Section 4.2 above. It is also of interest to consider the effects of a ruptured tube on the reactor core and adjacent tubes.

Steam vents have been provided within the graphite stack to vent the

steam resulting from a ruptured process tube. These vents would serve to minimize the extent of a pressure build-up within the stack after such an incident. It has been found that even for the top rows of tubes that a tube failure will not result in a large enough pressure build-up to cause lifting of the stack sufficient to result in failure of adjacent tubes. This statement holds true even for tubes embrittled beyond the expected range of loss of ductility due to irradiation.

Process tubes will be monitored by non-destructive tests in place, and removed for destructive testing to assure that tube ductility does not decrease beyond allowable limits. If tests indicate significant reduction in ductility, tube replacement might become necessary.

4.10.3 Inadvertent Discharge of Fuel Elements

It is possible that due to an operating error or equipment failure fuel elements may be inadvertently discharged on the front or rear elevators or becomes lodged in the front or rear face piping.

A special fog spray system has been provided to cope with such situations and it is highly unlikely that the fuel will reach a melting temperature. However, should one or a few fuel elements overheat and ignite, fission product release to the building could occur. Significant release of fission products (other than noble gases) outside the building is improbable, however, due to the efficiency of the fog spray, the opportunity to plate out on walls and ducts and the exhaust air filtration provided.

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4.10.4 Fuel Element Failure

Fuel element failures are considered "expected accidents", i.e., it is anticipated that fuel element failures will occur frequently during the reactor life. Hence, it is necessary that the consequences of a fuel failure not be severe. It is concluded that by fuel, reactor and instrument design, this objective has been achieved.

A sample of coolant is collected from each process tube for monitoring for fuel failures. It is highly probable that most fuel failures will be detected soon after the failure occurs and before more than a small amount of uranium has reacted with the coolant. This will limit the amount of rupture debris available to contaminate the primary coolant system and will limit the extent of physical distortion of the fuel which could tend to block the flow stream and, perhaps, damage the zircaloy process tubes.

As soon as a fuel failure is detected, the coolant from that tube may be diverted directly to a crib and, hence, contamination of the system is limited to the process tube and its rear connector.

The fuel element geometry selected, two concentric fuel tubes represents a further guarantee against complete blockage of coolant flow by the rupture products. Total blockage of only one of the three coolant passages (the hole of the inner fuel tube) is credible. A total plug in this channel would not result in uranium melting.

A failure on the outer surface of the outer fuel tube could result

in a situation where the reaction products are pressed against the zircaloy-2 process tubes. Severe hydriding of a tube in an experimental facility was experienced as a result of a situation similar to this. (Ref) A tube failure resulting from such an incident is considered to be an improbable accident. However, as pointed out in the previous sections this would not be a serious accident.

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K_{eff} Relative to Pre-Accident Reactivity

FIGURE 4-1
EFFECTS OF FLOODING ON REACTIVITY (HOT COOLANT - ZERO EXPOSURE)

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FIGURE 4-2

FLOODING AND COLD WATER ACCIDENT - (ZERO EXPOSURE - ZERO POWER - COLD GRAPHITE)

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K_{eff} Related to Pre-Accident Reactivity

FIGURE 4-3

FLOODING AND COLD WATER ACCIDENT (FULL POWER - ZERO EXPOSURE - HOT GRAPHITE)

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FIGURE 4-4

UNIFORM BUILDING CODE ZONING MAP

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5. CONTROL OF REACTIVITY

5.1 General Considerations

In contrast to previous Hanford production reactors, N Reactor will have only one rod system, combining safety and control functions, in which all the rods are horizontal. Because of higher power levels, higher graphite temperatures, and design flexibility characteristics, all requiring additional control strength and adjustability, more rods are provided at N than at the older reactors and, because of space limits the rods will enter the reactor from both sides. Also, the recirculation of primary coolant, at high temperatures and pressures, makes the use of temporary poison columns highly inconvenient thereby creating an additional demand on control rod strength over previous reactors.

The backup control system in the N Reactor is a vertical ball system. This system is independent of the horizontal rod system and provides capacity to control any credible operating or disaster reactivity situation.

5.2 Reactivity Transients

The sources of the reactivity variations over a normal operating cycle, including shutdown and startup, in N Reactor are similar to those in the other Hanford reactors. However, higher power levels, higher temperatures, higher fuel exposures and an undermoderated lattice serve to increase the normal reactivity variations and to introduce sources of reactivity variations which are not present in the old Hanford reactors.

5.2.1 Temperature Coefficients

The temperature-dependent sources of reactivity variations in N Reactor include, in addition to the usual fuel and graphite heat-

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ing effects, one associated with coolant heating. The fuel temperature and coolant temperature coefficients of reactivity are negative; the graphite temperature coefficient of reactivity is positive.

The temperature coefficients of reactivity averaged over the operating temperature ranges are given in Table 5-1. Additional information on the temperature coefficients will be given in Appendix III, Volume II.

TABLE 5-1

N REACTOR TEMPERATURE COEFFICIENTS

	<u>Range</u>	<u>Unirradiated</u>	<u>Irradiated^d</u>
Doppler ^a	20 - 380 C	$-2.3 \times 10^{-5}/C$	$-2.3 \times 10^{-5}/C$
Moderator ^a	20 - 550 C	$+5.3 \times 10^{-5}/C$	$+5.7 \times 10^{-5}/C$
Coolant Density ^b	1 - .82 gm/cm ³	$-4.6 \times 10^{-2}/g\text{-cm}^3$	$-3.7 \times 10^{-2}/g\text{-cm}^3$
Coolant Temperature ^c	-20 - 230 C	$-9.5 \times 10^{-5}/C$	$-7.8 \times 10^{-5}/C$

^aHot coolant conditions

^bCold graphite, cold water

^cCold graphite, equilibrium coolant density

^d700 MWD/T average pile exposure

5.2.2 Fission Product Poisoning

Xenon poisoning reaches its equilibrium value at 30 MWD/T (2.5 days) and is 2.8% k for an average pile exposure of 700 MWD/T. Xenon-135 is unstable and its reactivity effect is not permanent. After reactor shutdown, the xenon poisoning initially increases,

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due to the decay of the precursor iodine-135, and eventually decays to zero. The complete xenon transient will be given in Appendix III, Volume II.

Samarium-149 also contributes significantly to the fission product poisoning. This isotope is stable and is a product of the radioactive decay of promethium-149. Samarium poisoning reaches an equilibrium value at about 100 MWD/T exposure. The poisoning at equilibrium is 0.65% k. After reactor shutdown, the samarium poisoning will increase due to the decay of the precursor promethium-149.

The complete samarium transient will be given in Appendix III, Volume II. Other fission product poisonings are included in the long term effects.

5.2.3 Neptunium Holdup

Calculations of the reactivity gains in a reactor due to plutonium buildup assume instantaneous formation of plutonium. The fact that plutonium-239 formation is delayed through the beta-decay of 2.33 day neptunium-239, makes it necessary to correct the calculated reactivity gains. This is most conveniently done by assigning a reactivity "loss" for the neptunium holdup effect. The equilibrium neptunium holdup defect is .41% k.

5.2.4 Long Term Gain (Cold)

The effects of plutonium formation, uranium-235 burnout and fission product formation (other than xenon and samarium) are partially included in the exposure-dependence of the graphite and coolant temperature coefficients (See Table 5-1). However, the reactivity of the cold reactor is also dependent on these effects.

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Based on observation of long-term reactivity effects in the present reactors, the cold long term effect is positive in natural uranium at exposures up to 550 MWD/T. In the NPR, the higher enrichment will tend to decrease the long term gain but the harder neutron spectrum should be a compensating factor. The cold long term reactivity transient is shown in Appendix III, Volume II. The gain expected for an average exposure of 700 MWD/T is 0.2% k.

5.2.5 Reactivity Breakdown

The preceding discussion enables a complete breakdown of the reactivity requirements and transients. The estimated breakdowns for the green and ripe piles, i.e., the reactor with the fuel load unirradiated and irradiated to goal exposure, are shown in Table 5-2.

5.2.6 Maximum Excess Reactivity (Non-Accident)

The maximum excess reactivity which can be obtained during any time during normal reactor shutdown or operation depends not only on the magnitudes of the various reactivity effects but on their time-dependences.

To indicate the maximum excess reactivity, typical startup reactivity (reactor with unirradiated fuel, and fuel irradiated to goal exposure) transients are shown in Figure 5-1 for the green and ripe piles. The green piles curve should represent the most extensive case insofar as excess reactivity is concerned during a startup. The curves are for startups from an infinite outage (zero xenon).

The startup procedure assumes operation at ten per cent power (400 MW) for the first two hours in order to bring the coolant inlet temperature up to the design value of 185 C. Thus, the initial negative transient is due to water and uranium heating. The rise to full power is accomplished at a rate of 100 MW/min. Further uranium and coolant heating plus

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TABLE 5-2

REACTIVITY BREAKDOWN

	<u>Fuel Exposure (Ave)</u>	
	<u>0 MWD/T</u>	<u>700 MWD/T</u>
Xenon	-3.1% k	-2.8% k
Samarium (cold)	0	- -0.6% K
Neptunium (cold)	0	-0.4% k
Long Term Gain (cold)	0	+0.2% k
Temperature Coefficients		
Doppler	-.8% k	-.8% k
Coolant	-2.8% K	-2.3% k
Moderator	+2.8% k	+3.0% k
k, Cold, clean	1.059	1.061
k, Cold to hot	-.8%k	-.1%k
k, equilibrium	1.020	1.024

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graphite heating and xenon buildup contribute to the transient during this period-peak, called turnaround, is reached approximately five hours after startup. At this time the full temperature effect has been attained, and the reactivity loss beyond turnaround is due to further xenon buildup.

The startup transient curves indicate the the maximum excess reactivity during startup and operation is the cold, clean reactivity. Turn-around reactivity level is below startup level. This differs from the present Hanford reactors where the turnaround level for an infinite-outage startup is higher than the initial cold clean reactivity level.

Even if the reactor were scrammed at turnaround, the immediate reactivity effect associated with cooling down of the fuel and coolant would not result in an excess reactivity beyond that for the cold reactor.

5.2.7 Loss-of-Coolant Reactivity Effect

The N-Reactor lattice is fail-safe upon loss of coolant. That is, the dry lattice is less reactive than the wet lattice. The coolant-loss reactivity effects for the green and ripe pile are given in Table 5.3. The values for the cold green reactor are experimental values (See Appendix III, Volume II).

TABLE 5-3

LOSS OF COOLANT REACTIVITY DEFECT

<u>Condition</u>	<u>Zero Exposure</u>	<u>700 MWD/T</u>
Cold Lattice	-6% k	
Hot Lattice		

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5.2.8 Lattice Flooding

The NPR has an undermoderated lattice, and as just discussed loss of coolant results in a reactivity loss. Conversely, addition of more moderator to the lattice may result in an increase in reactivity. The effect of core-flooding has been analyzed theoretically.⁽¹⁾ In the analysis, the lattice was progressively flooded with water in a favored (low thermal flux) position and the maximum reactivity increase determined. Such an investigation should establish an upper bound for the effect. The calculated maximum Δk for flooding under various lattice conditions are given in Table 5-4.

TABLE 5-4

FLOODING REACTIVITY EFFECT (MAXIMUM)

<u>Condition (All for Green Fuel)</u>	<u>$\Delta k/k$</u>
Cold Graphite, Hot U and Water	.25%
Equilibrium Power	2.2%

5.2.9 Cold Water Reactivity Effect

The N lattice will gain reactivity when the density of the coolant water is increased. This reactivity effect has been calculated to be a maximum of 3.6% k in the hot reactor lattice at equilibrium power and for green fuel.⁽¹⁾ Values for other reactor conditions are given in Table 5-5.

TABLE 5-5

COLD WATER REACTIVITY EFFECTS (MAXIMUM)

<u>Condition (All for Green Fuel)</u>	<u>$\Delta k/k$</u>
Zero Power (Cold Graphite, Hot U and Water)	+ 3.1%
Equilibrium Power	+ 3.6%

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5.2.10 Maximum Reactivity Added by Increasing Temperature

The maximum reactivity which can be added by increasing temperature can only be estimated since reactivity effects occurring during extreme reactor conditions are very uncertain. The items which must be considered are temperature coefficients, loss of coolant, fuel melting, removal of control poisons by melting or vaporization and chemical and physical changes.

Temperature Coefficients

The temperature coefficients in the wet reactor have already been discussed. Any unlimited increase in reactor temperature in the reactor must be associated with an interruption in reactor cooling. Thus, the dry pile temperature coefficients are important in the following discussion.

Unlike the graphite temperature coefficient in the wet pile, the coefficient in the dry reactor is negative at low exposures. However, as plutonium is built up with exposure, the graphite coefficient becomes positive. Calculations indicate an upper limit of 3.4% k to the reactivity gain from graphite heating to the melting point of zirconium (1900 C).⁽²⁾

The dry pile Doppler coefficient has been measured over a temperature range of 20 - 968 C (See Appendix III, Volume II). The total Doppler defect to uranium melting based on these measurements is 3.7 % k; extrapolated to zirconium melting the defect is 5.3% k.

- (1) Reference: HW-64755 - "N-Reactor Hazards Review of Cold Water, Flooding and Startup Accidents", C. W. Allen, April 25, 1960 (Secret)
- (2) HW-61628 - "Interim Hazards Review New Production Reactor", R. Nilson and G. J. Rogers, August 21, 1959 (Secret)

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Physical Rearrangement of Uranium

In the present Hanford reactors, once the melting of uranium is accomplished, the uranium will be redistributed in the lattice in such a fashion that a large loss of reactivity will result. The presence of high-melting-point, zirconium alloy process tubes in the NPR will delay the redistribution of uranium, and thus it is conceivable that physical effects unique to a zirconium tube reactor will occur. One such effect is slumping of the fuel element into a more favorable shape as far as neutron economy is concerned.

The change in reactivity when the fuel elements melt and puddle in the bottom of the process tube has been assessed theoretically to be no greater than a 2% k gain. (2)

A second physical change which may occur during the high temperature portion of the meltdown is some alloying of the uranium and zirconium. Tests have been made at temperatures up to 1580 C to determine the behavior of molten uranium in a zirconium tube. (3)

There was no positive, visual indication of any penetration of the melt through the zirconium during the tests. Metallographic examination after the highest temperature tests revealed that zirconium alloy was destroyed in the area of contact with the molten uranium and that some zirconium was dispersed through the uranium mass.

Free flow of uranium following partial zirconium alloy destruction did not occur, however. It seems safe to conclude that the zirconium

(3) HW-64573 - "Observations on a Zircalloy-Uranium-Aluminum System at High Temperatures", A. J. Scott, May 22, 1960 (Unclassified)

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process tube in the NPR will contain the uranium melt for some time beyond the time the assembly has reached the melting point of uranium.

At the eventual temperature when the uranium is redistributed throughout the graphite stack, a very large negative change in reactivity will occur. As discussed in Reference 4, the effect has been experimentally verified for natural uranium.

Melting of Controls

Only poison which melts above 1900 C can be counted on for control beyond zirconium melting. The Ball 3X ball will be made of a ceramic material with a melting point in the range of 1800-2000 C. Thus, sufficient control should remain intact in the reactor until uranium melting and redistribution can occur and render the reactor permanently subcritical.

The expected reactivity status of the reactor up to the temperature of zirconium melting (1900 C) is shown in Table 5-6. Water is assumed lost in the cold, just-critical, green, xenon-free reactor.

This represents a conservative assessment for the following reasons:

- (1) Horizontal rods out of the reactor are not counted scrammable.
(3.6% k additional rod control is therefore ignored).
- (2) Ball 3X strength of 7.0% k is minimum. Experimental determinations indicate additional strength.
- (3) Horizontal rods in the reactor are assumed removable at the melting

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temperature of steel.

(4) Uranium redistribution will undoubtedly occur before 1900 C.

TABLE 5-6

REACTIVITY VARIATIONS AND FINAL STATUS
PRIOR TO URANIUM REDISTRIBUTION

<u>Variation</u>	<u>Reactivity</u>
Excess Reactivity	5.9 % k
<u>Horizontal Control Rods in Reactor</u>	<u>-5.9 % k</u>
Loss of Coolant	-6.0 % k
Ball 3X	-7.0 % k
Doppler Defect to 1900 C	-5.3 % k
Graphite Temperature Contribution to 1900 C	+3.4 % k
Fuel Slumping	+2.0 % k
<u>HCR Removal</u>	<u>+5.9 % k</u>
Status Prior to Uranium Redistribution	-7.0 % k

5.3 Reactivity Control Capacity

5.3.1 Horizontal Control Rods

The functions of the horizontal rods are (1) to control the excess reactivity in the reactors during all times of normal shutdown and operating periods, (2) adjust the flux distribution within the reactor, (3) automatically lower the power level for certain abnormal conditions, and (4) quickly shut down the reactor when required for hazardous conditions.

The design characteristics of the horizontal rods are given in Table 5-2. The control strengths given in the table are based on experimental determination. (See Appendix III, Volume II).

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TABLE 5-7

N-REACTOR NUCLEAR CONTROL SYSTEMS

A. Horizontal Rods

Number of Rods	87
Arrangement	9 high x 11 wide, 3 omitted from each corner, 41 enter from left side, 46 enter from right side.
Spacing	36" vertical x 32" horizontal
Rod Type	2-3/4" Od titanium sheath surrounding an internally-and-externally cooled aluminum extrusion containing boron carbide.
Rod Travel	29.5 feet
Rod Drive	Combination hydraulic cylinder and chain drive.
Rod Speed	
Operating-fast	30 ft/min.
slow	6 ft/min.
Sethack	6 ft/min in 8" steps, adjustable interrupter provides effective speed of 2-6 ft/min.
Scram	75% insertion through active zone in 1.2 sec. Approximately 20 ft/sec.
Rod Strength	
Full system (1004 tube pile) ^a	9.5 - 10.9% k
Central Rod	0.16% k, max.
Maximum Number of Rods Under One Actuator	87
Minimum Shutdown Margin-operating ^b	3.4% k
Minimum Shutdown Margin-accident ^c	0.4% k
Maximum Positive Ramp-Automatic Control	0
Maximum Positive Ramp - Manual Control	1.5% k/min.

a. Range listed for cold, green to hot ripe pile conditions.

b. Cold pile - average fuel exposure 700 MWD/T

c. Cold-water accident plus complete core flooding at turnaround level during a xenon-free startup. Green pile.

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TABLE 5-7 (Cont'd)

N-REACTOR NUCLEAR CONTROL SYSTEMS

B. Ball 3X System

Number of Channels	108
Channel Spacing	32" x 32"
Channel Diameter	3.0"
Ball Type	3/8" Ceramic ball - 0.380 gm/cc natural samarium oxide
Drive	Gravity
Ball Insertion Rate	
Total Reactivity Worth ^a	6.8 - 7.5% k
Minimum Total Reactivity Worth ^b	6.1% k
Minimum Shutdown Margin	
Maximum Cold Water Accident	2.5% k
Maximum Core Flooding	3.9% k
Maximum Cold Water Plus Core Flooding	1.1% k
Maximum Accidental Positive Ramp ^d	.007% k/min.

a. Range listed is for cold, green to hot ripe pile conditions.

b. For cold water in hot reactor and complete core flooding.

c. All postulated at turnaround level during xenon-free startup in the green reactor 4.1% in horizontal rods are already in reactor at time of accident. Shutdown margin assumes no shadowing of ball 3X system by rods.

d. Assumes maximum channel emptying rate of 1 channel every 11 minutes.

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5.3.2 Ball 3X System

The function of the Ball 3X system is to provide backup control to the rod system in the event of an accident which would disable the rod system. The design characteristics of the Ball 3X system are given in Table 5-7. The control strengths are based on experimental determinations (See Appendix III, Volume II).

5.4 Instrumentation For Reactivity Control

5.4.1 Rod Operating and Safety Circuits

The horizontal rod system in the NPR provides the following functions:

- a. Normal reactor control (1C circuit)
- b. Automatic slow setback of power level (2C circuit)
- c. Automatic (or manual) fast scram (1X circuit)

Up to six gangs of rods will be possible. Any rod can be controlled by any one of the six control switches, but by not more than one switch at a time. It is possible to control all 87 rods with one switch, but the capacity of the drive system will limit the total rod movement so that the withdrawal rate is limited to a maximum value of 1.5% k/min.

Each rod will have a five-position, service-selector switch. The five positions are (1) assigned safety rod, (2) assigned power setback, (3) manual, (4) withdrawal, and (5) off. All positions will require the insertion of a key for movement of the switch.

Interlocks are provided to restrict control of rods by the operator, depending on the level of nuclear flux. No restrictions are applied to prevent the insertion of any rods into the reactor at any time. However, withdrawal of rods shall be restricted by interlocks as follows:

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- a. Fast withdrawal (30 ft/min) of rods is precluded if rising period is between 20 and 60 seconds.
- b. Fast rod withdrawal is prevented if subcritical monitor count rate is less than ten counts per second.
- c. Fast rod withdrawal is prevented if two out of the three intermediate power level instruments exceed trip points.

Operating procedures rather than automatic interlocks will be relied upon to preclude unsafe or faulty operation in the following instances:

- a. To ensure adequate fast shutdown at all times up to 20 rods must remain fully withdrawn and in the safety assignment. Preselection of these rods shall be made with lockable switches.
- b. It is possible, practically, to circumvent the interlocks which restrict rod withdrawal speed to the slow operating speed (6 ft/min). If the operator were to preselect all rods for manual control, withdrawal of all 87 rods at the slow speed approximates the withdrawal of 16 rods at the fast speed as far as reactivity insertion rate is concerned. Operating procedures must be relied on to limit the number of rods which can be tied to one control switch at a time.
- c. It is possible to put all 87 rods on the off assignment during operation. Again, procedures must be relied on to preclude this from happening.

Setback and Scram Conditions

The conditions which apply with respect to the scram (1X) and setback (2C) functions of the horizontal rods are tabulated in Table 5-9.

Auxiliary 1X Safety Circuit Functions

Whenever the 1X safety circuit is tripped, certain other circuits must

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also be actuated or tripped. The functions that will be performed by these circuits will include the following:

- a. Start diesel-driven, emergency-cooling pumps.
- b. Reset setpoint for primary coolant flow to 25 per cent and close steam valves to primary-coolant turbines. Valves will open automatically to control 25 per cent flow.
- c. Open or adjust bypass valves around heat exchangers in operating loops.
- d. Start all available injection pumps.
- e. Start all available makeup and fill pumps in the secondary cooling system.
- f. Initiate programmed startup of standby boiler if it is banked at time of scram.
- g. Switch in water-level controllers to the heat-exchanger.

Scram Circuit Reset Circuit

The main scram circuit breakers must be reset by energizing an auxiliary relay. Certain conditions must be met before the auxiliary relay can be energized.

- a. The necessary primary loop valves must be properly positioned.
- b. All ball drain valves must be closed.
- c. All three emergency cooling system bypass switches must be open.
- d. The intermediate-level flux monitors must be unbypassed.
- e. The rate-of-power rise circuit must be unbypassed.
- f. No more than four ball hoppers may be locked out.
- g. The switch which bypasses the emergency cooling system trip in the ball safety system must be unbypassed.

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5.4.2 Ball System Safety Circuit

The ball safety system is actuated by the 3X safety circuit. The conditions which apply to a 3X scram are tabulated in Table 5-10.

5.4.3 Auxiliary and Operational Control Instrumentation

Much of the nuclear, temperature-monitoring, flow-monitoring and other instrumentation which are in the safety circuits will have useful functions for process control. Auxiliary instrumentation systems which will be provided for N Reactor which are not in the safety circuit are described below.

In-Core Flux Monitors

The NPR will be provided with at least 25 miniature, in-core, flux detectors, and a traveling wire monitor for calibration purposes. The initial design calls for the detection to be inserted in five moderator, cross-coolant channels, five monitors per channel.

Galvanometers

N Reactor will have two galvanometers mounted on the control console to indicate total level and deviation from a reference level. The signal for the galvanometers will come from twelve ion chambers located on the top and bottom of the reactor.

Power Calculation

A power calculating system, based on the measurement of flow and ΔT of the various coolant systems will be furnished to determine the total thermal output of the reactor.

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TABLE 5-8

N-REACTOR ROD SERVICE CATEGORIES

<u>Service Assignment</u>	<u>Purpose</u>	<u>Consequence 2C Trip</u>	<u>Consequence 1X Trip</u>	<u>Physical Limitations</u>
Safety	To provide full-out rods reserved for scram use solely.	None	Scram Rods	These rods must be fully withdrawn before any other rods can be withdrawn during startup.
Setback	To reserve certain rods for power setback.	Setback	Scram Rods	Rods in this assignment cannot be manually controlled.
Manual	To permit rod to be operated manually for normal control.	Setback unless setback function of particular rod is bypassed.	Scram Rods	
Withdrawal	To permit withdrawing up to three rods from reactor after a scram by bypassing 1X scram circuit.	None	Scram Rods (See right)	If more than three rods are withdrawn in this manner, bypassed 1X circuit is tripped driving all rods into reactor.
Off	To remove rod electrically from all control circuits in event rod must be removed from service.	None	None	Rods in off assignment are not removed from a rod-out counting circuit. If more than three rods are out (off position and withdrawn or withdrawn on withdrawal assignment) 1X circuit is tripped driving rods into reactor.

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TABLE 5-9
SETBACK AND SCRAM INSTRUMENTATION AND CONDITIONS

<u>Variable</u>	<u>Source</u>	<u>Logic</u>	<u>Setback Conditions</u>	<u>Scram Conditions</u>
<u>Neutron Flux</u>				
Subcritical Range	Two subcritical flux monitors in active zone.	1/2	Fast Period (20-40 Seconds)	
Intermediate Range	Three channels of intermediate level nuclear instrumentation.	2/3	Approaching Fast Period	Fast Period, Excessive Flux
High Level	Twelve channels of high level nuclear instrumentation.	2/12	Approaching excessive flux top or bottom	Excessive flux top or bottom.
Power Rate-Of-Rise	Three channels of power rate-of-rise instrumentation receiving input signals from the 12 channels of high level nuclear instrumentation.	2/3	Approaching excessive power rate-of-rise.	Excessive power rate-of-rise.
Total Power	Average flux level transducer which receives input signals from 12 channels of high level nuclear instrumentation.	2/3	High total power level	Very high total power level.
<u>Temperature</u>				
Outlet bulk temperature	Four "bulk boiling point" monitors located on rear risers.	2/4	Approaching outlet bulk boiling	Bulk boiling in risers
Process Outlet Tube Temperature	About 100 tube outlet temperature monitors chosen on a zone basis	2/100 3/100	Excessive tube outlet temperature for 2 zones.	Excessive tube outlet temperature from 3 or more zones.
Heat Exchanger Effluent Temperature	Temperature monitors located at outlet of each heat exchanger	2/10	Excessively high or low temperature of influent water.	

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Table 5-9 (Continued)

<u>Variable</u>	<u>Source</u>	<u>Logic</u>	<u>Setback Conditions</u>	<u>Scram Conditions</u>
<u>Flow</u>				
Process Tube	Fow transducer mea- suring flow through each of 1004 process tubes.	1/1004		High or low flow in any process tube.
Rod	Flow Monitors on out- let of all rod cooling tubes.	1/87		Low flow on any rod not in full- out position. 3-20 sec time de- lay in 1X circuit
<u>Pressure</u>				
Main Steam Header	Four pressure switches on main steam header	2/4	High Pressure	
Primary Loop	Four pressure switches located on rear risers.	2/4	Low pressure in rear risers. High pressure in rear risers.	Extremely high pressure in rear risers.
Condenser Circu- lating Water	Three pressure monitors on each of two headers	2/3 on either header		Low pressure on either header
<u>Miscellaneous</u>				
Pressurizer Level	Three liquid level instruments in pres- surizer vessel	2/3	High water level	Low water level
Surge Tank Level	Three level instruments in surge tank	2/3	Very high conden- sate level	Extremely low condensate level
Primary Pump Steam Supply	Pressure monitors lo- cated on supply lines to the five process pumps	2/5		Loss of steam to two or more pumps.
Heat Exchanger secondary side water level	Level indicators on each of ten heat exchangers	2/10		Loss of steam to two or more pumps.
Ball 3X System				Lower drain valve open

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Table 5-9 (Continued)

<u>Variable</u>	<u>Source</u>	<u>Logic</u>	<u>Setback Conditions</u>	<u>Scram Condition</u>
Emergency Cooling				Signal actuating emergency coolant.
Electrical Power	Monitors on each critical bus			Electrical outage longer than three seconds or BPA 4160 volt bus or on emergency turbo-generator 4160 volt bus.
Manual	Button on console			Manual trip
Battery Failure				Loss of battery power for rod solenoid.
Equipment Failure		1/_		Failure of critical components in LX system.

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TABLE 5-10
BALL SAFETY (3X) CIRCUIT

<u>Signal</u>	<u>Source</u>	<u>Coincidence Required</u>	<u>Purpose</u>
1. Failure of rods to scram.	Monitors measuring travel speed of rods and limit switches determining when rod has been 75% inserted.	a. Any two rods in a zone fail to scram. b. Any three rods fail to scram.	a. Drops balls in that zone. b. Drops a row of balls on each side of each failed rod.
2. Failure of neutron flux to decay following trip of LX safety circuit.	Two safety circuit controllers operating with signal from the average flux transducer of the high level nuclear instrumentation system.	1 out of 2	Provides a ball drop if the neutron flux has not decayed by a defined amount in 1 to 10 seconds following a scram.
3. Reactor supercritical while rods are fully inserted.	Three channels of intermediate level nuclear instrumentation.	1 out of 2 or 2 out of 3	Provides a ball drop if for any reason reactor becomes supercritical while safety rods are fully inserted.
4. Earthquake	Three seismoscopes located in 105-N Building.	2 out of 3	Provides a ball drop whenever an earthquake considered dangerous to core structure occurs.
5. Operation of Emergency cooling system	Activation of emergency cooling system master relay.	-	Indicates an extremely hazardous event - prevents power surge caused by cold water from emergency cooling system
6. Manual trip.	Button located in control room	-	Permits manual initiation of ball drop.
7. Power failure of two battery sources	Loss of operating power causes hoppers to open	2 out of 2	Provides fail-safe operation.

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6. HEAT REMOVAL AND DISSIPATION

6.1 General Considerations

Heat removal and dissipation rank with nuclear control in importance to reactor safety. Hence, the design provides for:

1. Adequate cooling at design power level, with ample reserve heat removal capacity.
2. A highly reliable heat removal system.
3. Continuous monitoring of process variables so heat generation rates can be reduced if inadequate cooling conditions are approached. These reductions can range in speed and magnitude from a manually controlled manipulation of control rods to an automatic reactor scram.
4. Back-up systems for all critical cooling systems.

By far the most (about 98 per cent) of the heat generated in the reactor is removed from the reactor by the primary coolant system, transferred to the boiling water secondary system and dissipated to the Columbia River via condenser cooling water streams. Failure of the primary coolant system to remove heat from the reactor^(a) at design power level will result in certain fuel melting unless the last-ditch backup cooling system functions.

Other heat removal systems include:

1. The graphite moderator cooling system.
2. The control rod cooling system.
3. The shield cooling system
4. The reactor gas system

These heat removal systems are not as critical to reactor safety as the

(a) This could be caused by failure of the secondary system or condenser cooling system as well as by a direct failure of the primary system.

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primary coolant system. Fuel melting would not result from the failure of any or all these systems. On the other hand, the proper functioning of these systems - particularly the graphite moderator cooling system - would markedly reduce the extent of fuel melting should the primary coolant system and its backup systems fail.

6.2 Heat Generation Rate and Limiting Factors

The design reactor power level is 4000 MW (th). About 94 per cent of the power or 3760 MW is generated in the fuel elements, about 5 per cent or 200 MW is generated in the moderator and the remaining one per cent or 40 MW in the zircaloy-2 process tubes.

It is not believed that there are any factors which are likely to limit the reactor power below the design level of 4000 MW. Some of the more critical factors which at some point would limit heat generation rate are discussed below:

6.2.1 Heat Dissipation System

About 98 per cent of the heat generated in the reactor is removed by the primary coolant system, transferred to the secondary system through ten parallel steam generators, and then transferred from the secondary system to river water through 16 parallel dump condensers plus six turbine exhaust condensers.

The capacity of these systems should normally be ample to support design power levels. The steam generator and condenser designs include fouling factors to accommodate reasonable degrees of scaling.

The weak point in the system from the operational continuity standpoint is the lack of operational spares. Malfunction of critical

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items of equipment, i.e., a primary coolant pump, would sharply reduce the capacity of the heat dissipation system. The loss of some items, e.g., the plant turbine generator condenser, would require a reactor shutdown. Power reductions, or reactor shutdown can be accomplished without fuel element damage.

6.2.2 Process Tube Thermal Hydraulics

At a given tube flow rate, inlet temperature and system pressure the amount of power generated in a process tube must be limited to avoid heat transfer "burnout". With heat generation rates near the burnout limit, a small reduction of tube flow or pressure or a small increase in tube power would result in film boiling at the heat transfer surface which causes the surface to be insulated by a layer of steam. This condition causes a rapid increase in fuel temperatures which, if unchecked, would result in fuel melting.

Available information shows that the limit which will be imposed to avoid the "burnout" phenomena will not restrict reactor powers below design levels.

Figure 6-1 presents results of experimental studies of burnout with the NPR tube and fuel geometry. At a process tube power of 5000 KW (the maximum expected at a 4000 MW reactor power) tube flows could be reduced about 45 per cent below the design value before burnout would occur. This is an ample margin to accommodate normal transients.

Sizable power surges can also be accommodated before burnout is reached. However, the exact size of the surge that can be

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accommodated is not yet known. A large power surge will cause boiling in the tube before burnout is reached. This boiling will reduce tube flow, thus aggravating the situation. The extent of the reduction of flows resulting from power increases has not yet been studied. This data as well as data describing various transient conditions are needed before limits can be firmly established and before the situation can be completely described. A full-scale, electrically-heated mock-up assembly is available for obtaining these data and will be put in operation as soon as minor modifications are made. The results will be available long before reactor operation commences.

6.2.3 Fuel Performance

A fuel failure in the NPR will necessitate a speedy reactor shutdown. Should failures become frequent, operating efficiency would be drastically reduced and alleviation of the problem required. The frequency of fuel failure will depend on fuel powers as well as on other variables such as fuel exposure and metal quality. Therefore, fuel performance could be a practical limit to the reactor power although reactor safety need not be directly involved.

The use in NPR of metallic uranium fuel elements in high temperature water coolant represents a departure from past practice. Freedom from all fuel problems is not anticipated. However, it is not anticipated that fuel performance will limit reactor power below the design level.

The N Reactor fuel consists of two concentric fuel tubes, each co-extruded in zircaloy-2 jackets. The fuel will operate with a maximum uranium temperature of 850 degrees F, about 375 degrees F below the alpha-beta transformation point where the uranium density decreases markedly. At this temperature, core stresses are expected to be tolerable and frequent cleavage-type core failures are not anticipated.

Operating conditions should likewise be mild from a fuel cladding corrosion standpoint. Partly as the result of the symmetry of the coolant passages, partly because of the extended surface area of the fuel and partly because of careful sizing of fuel components, maximum cladding surface temperatures are expected to be less than 600 degrees F. This is well within the range ordinarily considered appropriate for the use of the zircaloy-2 alloy.

Some problems are expected to result from the irradiation induced distortion of the fuel cores. This distortion produces tensile stresses in the cladding and, if severe enough, can result in mechanical failure of the cladding. The extent of distortion increases with increasing irradiation temperature and exposure time and decreases with increasing coolant pressure. For a given degree of distortion, the probability of fuel failure is minimized by obtaining a cladding of uniform thickness. The tensile failures occur preferentially at weak places in the cladding. It is believed that the cladding for NPR fuel can

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can be made uniform enough to accommodate the expected core distortion without an excessive failure rate. Testing of production type N Reactor fuel under conditions which meet or exceed design operating conditions is expected to begin in January 1962 in two modified KER loops.

6.2.4 The Graphite Moderator

The temperature of the graphite moderator will vary nearly linearly with the reactor power. The temperature of the graphite has a profound effect on reaction rates involving graphite and impurities in the gas atmosphere. Temperatures may also affect the irradiation induced distortion rates to some extent. Both of these factors are important. However, neither is likely to restrict reactor powers below 4000 MW.

With the conditions that will exist at design power levels, graphite gas reactions will be controlled within tolerable bounds by controlling the trace constituents in the helium gas atmosphere. The reactor gas must also be compatible with the exterior surfaces of the Zircaloy-2 process tubes. The range of tolerable compositions considering both the graphite and the tubes has not yet been determined. It is probable, however, that the amounts of oxygen, hydrogen, and water vapor will each be limited to well below 1 MM of partial pressure. Trace concentrations of water vapor or oxygen will probably be required at the surface of the zircaloy tubes to inhibit hydriding of the exterior tube surfaces.

The physical distortion of graphite by a neutron flux is strongly temperature sensitive. At low temperatures (below 200 degrees C), the net effect of neutron radiation is to cause graphite to expand; at higher temperatures contraction occurs. When scope reactor design was in progress, graphite contraction rates at quite high temperatures (say over 700 degrees C) were only poorly known. There was some fear that at these high temperatures contraction rates might increase rapidly with increasing temperatures. To avoid a potentially catastrophic graphite contraction problem, a special graphite cooling system was provided. When this cooling system, the maximum graphite temperature was reduced to 1200 degrees C and the temperature load-bearing graphite components were reduced to 800 degrees C maximum.

It now appears that the fears of excessive contraction rates at the higher temperatures were unfounded. However, the heat removed by the graphite moderator coolant system has been factored into the total heat removal system to limit process tube outer surface temperatures to 350 degrees C maximum. This is the maximum tube temperature believed allowable to avoid gas-side tube hydriding. The system also performs an important role in the limiting the extent of fuel melting during serious accident as described in Section 7.

6.2.5 Zircaloy Process Tubes

At a given coolant temperature, increases in the reactor power will increase the magnitude of the temperature gradient across

the process tube wall. This increases the level of thermal stresses within the tube. It also results in higher outer surface temperatures which aggravate the gas-side chemical reaction problem. However, conditions existing at reactor power of 4000 MW are believed to be tolerable. Process tube stresses will not be excessive so long as hydriding does not cause an excessive deterioration of the tubes' physical properties. The hydriding problem has not been completely defined or solved. Hydriding could indeed limit process tube life. However, the maximum expected outer skin temperature (350 degrees C) at the design reactor power level is not beyond the range where prolonged service life could be expected.

An extensive program is now in progress to define and solve the hydriding problem. This will be covered in more detail in Section 10 of this volume.

6.3 Heat Removal System

6.3.1 The Primary Coolant System

The primary coolant system removes more than 98 per cent of the heat generated in the reactor. This system is divided functionally into two portions: a heat removal section in the 105 N Building and a heat dissipation system in the 109-N Building.

The system is comprised of five nearly identical, parallel loops, which are in the 109-N Building. Each loop contains

one primary pump, drawing coolant from two steam generators arranged in parallel. The pumps are normally powered by steam turbines. Electric "pony" motors on the same shaft are provided for emergency backup and for pumping power during reactor outages if steam is not available. These motors are supplied with BPA power. The water from each loop is pumped through a 24-inch line to a common manifold still in the 109-N Building. From this manifold, the coolant flows through 16 parallel lines of 18 inch pipe, eight of which are on each side of the reactor. These lines extend up and over the reactor and down beside the inlet face. The vertical sections by the front side of the reactor are called "inlet headers". The bottoms of the inlet headers are manifolded together for admission of raw river water in case of emergency (Emergency Coolant System). Filtered water may also be supplied to this manifold during reactor outages for once-through cooling or for flushing.

Each inlet header is connected to 54 to 66 in-reactor process tubes by horizontal connectors. Each connector is fitted with a flow-monitoring venturi and an adjustable two-position butterfly valve. When in the "open" position, these valves provide high flow orificing to distribute the flow to the tubes in quantities roughly proportional to the tube powers. In the "throttled" position the valves limit coolant flow for charge-discharge operations during reactor outages. The valves are remotely operated.

The water flows from the 1004 connectors to the corresponding process tubes. These tubes are made of Zircaloy-2, have an I. D. of 2.70 inches with a 0.250 inch wall thickness. Connectors at the rear of the reactor conduct the coolant to 16 outlet headers in a manner similar to the front face arrangement. Each of the rear connectors is fitted with a three-way valve so the coolant can be passed to one of 16 pressurized diversion headers instead of the regular header. This diversion system will be used in case of a fuel failure to prevent contamination of the main system. The diversion system is pressurized to maintain constant flow and to prevent boiling in the tubes when they are routed to diversion. Water discharged to the diversion system is cooled and sent to the crib, the quench tank, or the chemical waste tanks depending on particular circumstances.

The 16 normal outlet headers extend up the sides of the rear face, back, and down to near the bottom of the reactor and thence back to the outlet mixing manifold.

The bottom of the outlet risers are joined to a flushing manifold which can be used during reactor outages for once-through cooling of the reactor, for reactor decontamination, or for flushes after decontamination. Water from the flushing manifold may be sent to the same three points as water from the diversion system.

The 16 lines are also manifolded to provide for "dumping" the

primary loop should the normal cooling mode fail.

Water from the dump system is conducted uncooled to the quench tank or diverted directly to the river.

The mixing manifold consists of seven parallel lines. Each line is fed by all 16 outlet risers and each of five of these lines conducts water back to one of the five parallel loops. The remaining two lines are spares provided for the eventual addition of a sixth and seventh loop should this be authorized at some future time.

Primary coolant pressure is maintained sufficiently high to prevent boiling at any point in the system by means of a pressurizer vessel. This vessel is maintained about 60 per cent full of water with a steam blanket. Pressure is regulated by a combination of electric heaters in the water filled and spray cooling in the steam filled portions of the vessel. The pressurizer also serves as a surge tank to accommodate part of the water volume fluctuations resulting from density variations due to temperature transients. This is augmented by a water injection system and a spill cooler for the more extreme transients.

Water quality is maintained by a feed-and-bleed system. There are four bleed streams:

1. A sample of about 0.45 gal/min is withdrawn from each process tube, cooled, and routed to a monitor for detection of fuel failures. This water is then sent to the crib.

2. The diversion system already mentioned above, will represent an additional, approximately 160 gal/min per tube on diversion. Ordinarily, however, there will be no tubes on diversion and the diversion flow will be about 100 gal/min.
3. The spill cooler will accommodate flows up to about 1300 gal/min during temperature transients. Normal flow through the spill cooler will be about 40 gal/min.
4. The remaining required bleed to maintain water quality is obtained from the flush system. This water will be cooled in a regenerative heat exchanger and sent to crib.

There are two makeup streams to balance the system bleed and to accommodate water volume transients. Approximately 375 gal/min total will be injected into the primary system through injection seals on each primary pump. The remaining required feed is routed through the regenerative heat exchanger and into the primary system at points just upstream of the suction of the five primary pumps.

Should a pressure transient occur beyond the ability of the normal control equipment, the loop is protected by a system of mechanical safety and relief valves. Four pressure switches attached to the rear risers operate to cause a reactor power setback in case of high or low pressure, and another set of four switches scram the reactor in the event of extremely high pressures.

6.3.2 The Secondary Coolant System

The secondary coolant system removes heat from the primary

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coolant system and transfers it to the circulating water system.

The secondary system is shown schematically in Figure 6-2.

The main components of the system are:

1. Ten steam generators.
2. One diesel fired boiler.
3. Five pump drive turbines with their condensers and condensate return pumps.
4. One turbo-electric generator with its condenser and condensate return pump.
5. One deaerator.
6. Sixteen dump condensers and their condensate return pumps.
7. A surge tank system and an afterheat removal tank.

The heat picked up in the reactor by the primary coolant system is transferred to the secondary system via the ten, parallel, steam generators. At design reactor power levels, this will result in the generation of a total of 13,900,000 pounds per hour of steam at 87.5 psia.

During reactor outages the steam generators will be flooded with water stored in the afterheat removal tank. Heat will then be transferred from the primary to the secondary coolant by water-to-water operation. Similarly the heat will be transferred to the river loop by water-to-water operation of the dump condensers.

During reactor operation the steam from all ten generators will enter a common steam header. This header has isolation valves

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so it can be divided into four parts. During normal operation, however, the valves will be open. The distribution of steam from the common steam header is shown in Table 6-1.

Steam may also be supplied to the common manifold by a oil-fired boiler included in the secondary system. During normal reactor operation when the steam generators are carrying the load, the boiler will be banked, producing about 1200 pounds per hour of steam. On demand, the boiler can pick up to its full capacity of 575,000 pounds per hour in eight minutes. The "thermal flywheel" effect will permit the steam generators to drive the turbines after a scram until the boiler assumes the load. The boiler capacity is sufficient to meet steam requirements during reactor outages and during startup periods when the steam generators cannot furnish the steam.

The pump drive turbines power the primary loop circulating pumps. Ordinarily these drive turbines received their steam from the steam generators. During outages enough steam to provide up to one-half of normal primary loop flow will be obtained from the boiler. The condensers on the pump drive turbines will operate at less than atmospheric pressure. The design pressure is 2 - 3 inches of mercury, absolute. Any leakage in the condensate system will be inward and some air leakage into the system is expected. Therefore, the condensate from these units is routed to the deaerator before returned to the common condensate steam header.

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The turbine generator plays a key role in reactor safety. It supplies electrical power to various services around the plant and serves as a back up to the BPA electrical system. Should the BPA system fail, power from this local source will permit an orderly, safe shutdown and will provide the necessary "shutdown" power for an indefinite period.

The sixteen dump condensers normally condense most of the steam generated. These units operate at about 74 psia. Any leakage at these units will be out from the secondary system and will not effect coolant purity. Therefore, condensate from these units is pumped directly back to the steam generators.

The surge tank system and afterheat removal tank are provided to accommodate changes in the water volume in the system. The surge tank is designed to operate partially filled with water with a steam blanket. Events which would tend to change the water volume in the system will first be sensed in the surge tank. For instance, if the steam generators "call" for extra water, the level in the surge tank will drop. This will actuate a pump which will inject water into the loop from the afterheat removal tank.

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TABLE 6-1

STEAM DISTRIBUTION FROM SECONDARY SYSTEM

<u>Destination</u>	<u>Pounds Per Hour</u>	<u>Percent of Total</u>
Sixteen Dump Condensers	12,720,000	92
Five Pump Drive Turbines	600,000	4
Turbine Generator	290,000	2
Process and Building Heat	100,000	1
Deaerator	<u>150,000</u>	<u>1</u>
Total	13,860,000	100

6.3.3 The Circulating Water System

The Circulating Water System supplies raw Columbia River water to various services in the plant. Figure 6-3 is a simplified schematic diagram of the system. Table 6-2 shows the major loads on the system.

One unique feature of the system is that the two main return lines to the river will provide a syphon effect. This reduces pumping power requirements and the size of the pumps required. Limits will be imposed on the condenser cooling water temperatures to ensure that vapor formation will not occur breaking the vacuum and causing a sharp reduction in flow. This consideration is of practical importance only on the dump condenser effluent. The effluent from the pump drive and turbine generator condensers will be at a temperature far below that where such vapor formation would occur.

The water is pumped from the river by four, parallel, electrically driven pumps. Two of these pumps are supplied with power from BPA and two from the local turbine generator. Ordinarily all four pumps will be required when the reactor is operating at design power. During parts of the year when river temperatures are low, three pumps will handle the load. Any one pump will furnish sufficient cooling to meet needs during a reactor scram and during reactor outages.

It is planned that the water will ordinarily receive no

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TABLE 6-2

DESIGN LOADS ON THE CIRCULATING WATER SYSTEM

<u>LOAD</u>	<u>DEMAND GAL/MIN.</u>
Dump Condensers	238,000
Pump Drive Condensers	68,000
Filter Plant	10,000
Graphite Cooling Heat Exchanger	17,100
Rupture Monitor Heat Exchanger	1,500
Turbogenerator Condenser	37,000

treatment except straining by coarse traveling screens at the river pumphouse. However, provisions for chlorine injection have been provided should algae growth become troublesome.

6.3.4 The Emergency Raw Water Cooling System

The emergency raw water cooling system will provide cooling should the normal cooling systems fail. Figure 6-4 is a simplified flow diagram of the system.

Two, low-lift, diesel powered pumps at the river pumphouse supply raw river water directly to a 200,000 gallon storage tank and to two diesel driven "fog spray" pumps.

Three other, high-lift, diesel driven pumps take their suction from the storage tank and pump water directly to a manifold connecting the bottoms of the sixteen inlet supply headers of the primary loop. In case of demand, the water will enter the primary loop through the inlet supply headers, flow through the reactor and out dump valves. Check valves at the top of the inlet supply headers prevent the last ditch water from bypassing the reactor.

The two "fog spray pumps" are also diesel driven. These pumps supply water to various services through the same line that normally conducts circulating water to the rupture monitor heat exchangers. The "fog-spray" pumps supply backup cooling water to the following cooling systems in addition to supply-

ing fog spray systems:

1. The graphite moderator cooling system.
2. The horizontal control and safety rod cooling systems.
3. The shield cooling system.
4. Reactor gas cooling system.

The emergency system is believed to be both adequate and reliable. The three high-lift and the two river diesel pumps will be automatically started every time the reactor scrams. This ensures that cooling will be available should the scram be caused by a major incident. It also provides for functional testing of the pumping plant.

The diesel engines are started by compressed air from a reliable compressed air system. Each engine is provided with an accumulator fitted with check valves with sufficient air to give two "starts" in the event of failure of the air compressors or lines from the main air accumulator to the individual engines.

In the unlikely event that neither of the two lift pumps start on demand, the high lift and the fog spray pumps can be supplied with water from a one million gallon demineralized water storage tank and the 850,000 gallon filtered water storage tank as well as the raw water storage tank. The total storage of about two million gallons will supply requirements of both sets of pumps for more than an hour which should allow significant time for starting the low

lift pumps. Also, raw water can be obtained from an export water line from the old reactors to help extend the storage capacity of the tanks. Finally, water for all needs for indefinite periods of time can be supplied to the last-ditch and fog spray systems from the circulating water system if either of the BPA powered circulation system pumps are operating.

6.3.5 The Graphite Moderator Cooling System

The graphite moderator cooling system limits graphite temperatures protecting against excessive chemical reaction rates and contraction rates; it limits the outer skin temperature of the Zircaloy-2 process tubes, protecting against excessive metal-gas reaction rates and limiting thermal stresses, and it will minimize the extent of fuel melting should all other reactor coolant systems be lost. Thus the graphite moderator cooling systems performs a vital role during normal operation and is a critical part of the reactor safety system.

Figure 6-5 is a simplified flow diagram of the system.

Water is recirculated by three pumps in the auxiliary equipment cell in the 109-N Building. One of these pumps is supplied with power from BPA, one from the local generator, and one with power from both sources. Water from all three pumps enters a common 12 inch line and is transported to the 105-N Building.

In the 105-N Building, the water is transported from the

twelve inch line by two ten-inch lines to the two bottom, rear corners of the reactor. At this point the ten-inch lines reduce to eight-inch headers which extend along the sides of the reactor to the front face. Each of these eight-inch headers supplies ten 2-1/2 inch risers which extend vertically up the sides of the reactors. These risers in turn transfer the water to the 640 side to side cooling tubes. The tubes are 0.75" O. D. with a .025" wall thickness and are made of zircaloy-2.

The outlet arrangement is nearly identical to the inlet arrangement. The water is gathered into vertical outlet risers and transferred to two ten-inch outlet headers. These join at a twelve-inch return line which extends back to the cell in the 109-N Building. There the water passes through four parallel heat exchangers and back to the suction of the three pumps. The heat exchangers are cooled with raw river water from the circulating water system.

System pressures will be controlled by a surge tank pressurized with helium, floating on the line just upstream of the heat exchangers. The surge tank will also accommodate small variations in water volumes.

The total system flow will be about 5300 gallons per minute. The inlet temperature to the reactor will be about 100 degrees F and the outlet about 180 degrees F.

In the event of failure of the normal pumping system, raw river water from the "fog-spray" system will be supplied to the system through a twelve inch line joining the system at a point in the 109-N cell downstream of the circulating pumps. During this "once-through" cooling operation, water will be removed through a twelve inch drain valve in the 105-N Building.

6.3.6 Miscellaneous Cooling Systems

In addition to the five cooling systems previously discussed there are a number of miscellaneous systems. Proper functioning of these cooling systems is essential to reactor operation and may be essential for obtaining the planned reactor life. However, these systems are not highly critical to reactor safety. Hence, these systems are covered very briefly:

TABLE 6-3

MISCELLANEOUS COOLING SYSTEMS

<u>System</u>	<u>Normal Source Of Coolant</u>	<u>Normal Pumping Power</u>	<u>Back-up Systems</u>	<u>Power For Back-up Systems</u>
Horizontal Control & Safety Rods	High Pressure Filtered Water System	Three Pumps Both Electric Power Supplies	1. Low Pressure Filtered Water System 2. "Fog-Spray Raw Water System	1. Three pumps, bot electric power supplies. 2. Two Diesel Pumps (See 6.3.4)
Thermal Shield & Corners of Biological Shield	Low Pressure Filtered Water	Three Pumps Both Electric Power Supplies	Fog-Spray System	Two diesel pumps (See 6.3.4)
Main Part Of Biological Shield	Graphite Moderator Cooling System	Three Pumps Both Electric Power Supplies (See 6.3.5)	Fog-Spray System	Two diesel pumps (See 6.3.4)
Spill Cooler & Rupture Monitor Coolant	Circulating Water System	Four River Pumps Both Electric Power Supplies (See 6.3.3)	Fog-Spray System	Two diesel pumps (See 6.3.4)
Reactor Gas System	Gas Circulated by Two Blowers	One blower powered by BPA; one powered by local turbine generator	None	-
	Gas Cooled With Low Pressure Filtered Water	Three Pumps Both Electric Power Supplies	None	-

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6.3.7 The Electrical Systems

Though not strictly a part of the heat removal and dissipation system, the electrical system is critical to reactor safety and loss of all electrical power would throw the reactor on last-ditch cooling.

The system consists of two essentially independent sub-systems, each supplied by different power sources. One sub-system obtains power at 230 kilovolts from the Hanford BPA loop. The second sub-system obtains power at the 13.8 KV level from the local turbine generator at the N Plant. These two systems may be augmented by a 13.8 KV connection to the D-H Area Substations during reactor outages. The system is shown in Figure 6-6.

One boiler feed-water pump, a boiler air fan and other equipment connected with the boiler and turbine generator set are powered from the "A" bus connected to the BPA system. This "inter-connection" was provided because:

1. It is important that it be possible to operate the boiler even if the turbine generator is out of service.
2. BPA power will be required to start-up the boiler and turbine generator after periods when these units have been down. This can conveniently be done by the scheme provided.

All critical auxiliary equipment on these units is duplicated and also powered off the turbine generator. Therefore, during reactor operation the turbine generator will be operating and BPA power

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will not be needed for operation of either the turbine generator or the boiler. Hence, during reactor operation, independence of the two units is preserved.

No power will ordinarily be drawn from the D-H intertie line. However, this power source can be manually switched to either bus at the 13.8 KV level. This intertie line can be used to supply power to the area during outages for maintenance reasons, e.g., when the boiler is shutdown, equipment on the "B" sub-system can be powered from the D-H intertie line if desired for system maintenance or testing.

The D-H intertie line may also be used infrequently to prevent the reactor from being thrown on the last-ditch cooling system from shut-down operation. If all electrical power to the plant is lost, the reactor will be cooled by the last-ditch system. The probability that both power sources will be lost simultaneously is quite low. Hence, the occasion where power loss is most likely to cause reliance on the last-ditch system is when only one of the regular sources is available and, hence, when the reactor is already shut down. Under such occasions the D-H intertie will still probably not be switched into service; however, it will be available should failure threaten the remaining regular power source.

It is believed that the intertie will be adequate to supply all shutdown power required for indefinitely long periods of time. However, this point is not absolutely certain, and the capabilities

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of the system probably will not be known in detail until shortly before reactor startup.

While this intertie may have an important back-up function, this is not assumed in the balance of this review because of remaining uncertainties.

6.4 Instrumentation For Control of Heat Removal

6.4.1 General

Instrumentation has been provided to measure and, where necessary, to control pertinent variables in the heat removal and heat dissipation systems. The pertinent variables and mode of control are shown in Table 6- 4.

Most of the heat generated by the reactor is removed by the primary loop cooling water. In order to ensure efficient, safe, heat removal, reliable instrumentation for measurement and control of the flow, temperature and pressure of the coolant have been provided.

Most of the heat from the primary loop is transferred to the secondary loop. Instrumentation is provided to measure and control the water and steam flow and pressure.

Heat from the secondary loop is transferred to river water and finally to the Columbia River. Instrumentation is provided to measure cooling water flow, temperature and pressure to guide manual control of these variables.

In order to keep the graphite temperature from exceeding the chosen limits a separate cooling system has been provided. Flow, pressure and temperature of the circulating coolant are measured and controlled by reliable instrumentation.

If the normal cooling systems fail, emergency raw water is used for cooling. Reliable instrumentation has been provided to

TABLE 6- 4INSTRUMENTATION IN HEAT DISSIPATION SYSTEM

<u>System</u>	<u>Variable</u>	<u>Mode of Control (Automatic or Manual)</u>
Primary Loop	Water Flow	Automatic
	Temperature	Automatic and Manual
	Pressure	Automatic
Secondary Loop	Water Flow	Automatic
	Steam Flow	Automatic
	Steam Pressure	Automatic
Circulating Water System	Water Flow	Manual
	Temperature	Manual
	Pressure	Manual
Graphite Cooling System	Flow	Manual
	Pressure	Automatic
	Temperature	Manual
Emergency System	Flow	Automatic
	Pressure	
	Temperature	Automatic

measure the pertinent variables and to initiate and control emergency raw water flow when required.

6.4.2 Specific Safety Features

Failure or partial failure of the river cooling water system, secondary loop system or primary loop system will lead to fuel melting unless the reactor power level is reduced. Instrumentation has been provided to either reduce reactor power level automatically or scram the reactor when cooling capacity is insufficient.

The normal cooling system is designed to provide adequate cooling after all accidents, except a major pipeline rupture or loss of both sources providing the reactor is shutdown promptly after the accident. Emergency once-through cooling water is provided by a separate system with diesel engines in case the normal piping is ruptured or both electric power sources are lost. Table 5-9 shows the instrumentation which is provided to cause a power set back or reactor scram when cooling is inadequate. Table 6-5 shows the instrumentation which actuates the emergency cooling water system.

If the normal cooling system and the emergency cooling system fail, the emergency graphite cooling system will prevent a full reactor meltdown. Reliable instrumentation is provided to actuate the emergency graphite cooling system.

6.4.3 Primary Loop Flow Instrumentation

Flow from each pump is measured by a flow nozzle. The flow from all pumps is controlled at a preset value based on comparison of

actual flow to set point. Flow is varied automatically to give a constant preset value by changing pump speed by means of steam turbines.

Flow to each process tube is measured by a flow nozzle. (Venturi) If the flow to any tube falls outside preset limits an automatic reactor scram will be initiated.

6.4.4 Primary Loop Temperature Instrumentation

The primary loop water temperature is measured in each of the five loops upstream and downstream of the steam generators. The water temperature downstream of the steam generators (reactor inlet temperature) is controlled at a constant value during equilibrium operation. This is done by changing the temperature of the boiling water on the secondary side of the heat exchangers. If the reactor inlet temperature falls outside preset limits a reactor setback will occur.

The temperature of each process tube is measured. Selected tubes comprise the zone monitor system. The system is designed to cause an automatic power setback when the temperature approaches an unsafe value and to cause a scram when the temperature reaches an unsafe value.

6.4.5 Primary Loop Pressure Instrumentation

The pressure of the coolant in the primary loop is measured and controlled by means of a pressurizer. The pressurizer is a cylindrical vessel filled with water and steam. Steam pressure in the pressurizer is held at a constant value by maintaining

the water in the pressurizer at constant temperature with electric heaters, and cooling water sprays. As a safety measure the primary loop pressure is held substantially above the value required to prevent boiling in any process tube. An automatic power setback occurs if the pressure temperature relationship approaches an unsafe condition. An automatic scram occurs if the pressure-temperature relationship reaches an unsafe value.

6.4.6 Secondary Loop Pressure Instrumentation

During reactor operation the basic control of the secondary coolant system is steam pressure control. The steam pressure is changed automatically to maintain a constant temperature in the primary water leaving the steam generators. The steam pressure determines the water temperature on the secondary side of the steam generators which in turn determines the water temperature on the primary side. Steam pressure is controlled by opening or closing valves between the steam generators and the dump condensers.

6.4.7 Secondary Loop Flow Control

Steam flow is controlled indirectly by steam pressure control. The amount of steam generated is determined by reactor power level. As power level increases water temperature in the steam generators decreases and steam flow increases. That is, the rise in power level opens the steam valves which increases steam flow and decreases steam pressure.

Condensate flow control is accomplished by means of control valves which maintain the system volume essentially constant. This is,

condensate flows away from the condensers at the same rate that steam is condensed. Temperature of steam and condensate are controlled essentially by steam pressure control.

6.4.8 River Water Instrumentation

Flow of river water is measured at each dump condenser and controlled manually by starting and stopping pumps and by throttling valves.

The temperature of the water leaving the dump condensers is measured. Temperature is controlled by changing raw water flow manually as required.

6.4.9 Graphite Cooling Loop Instrumentation

Flow of the recirculating water for graphite cooling is reasonably constant but flow is not controlled except by starting and stopping pumps. Recirculating coolant temperature is measured and is controlled by varying the fraction of the recirculating coolant flow which bypasses the raw water cooled heat exchangers.

Pressure of the recirculating coolant is controlled by means of a helium pressurizer. When graphite cooling recirculating coolant temperature exceeds a predetermined limit, emergency once-through untreated cooling water enters automatically.

6.4.10 The Last Ditch Reactor Cooling System Control Instrumentation

For the last ditch reactor cooling system to function the following must occur:

1. Two sets of pumps must start (See Section 6.3.4).
2. Valves admitting water to the reactor (V-4) and releasing water from reactor (V-5) must open. (The latter may not be required if the event causing demand for last ditch cooling was a break in a large effluent line).

The pumps are started everytime the reactor scrams. They may also be started manually.

The V-4 and V-5 valves will be opened should any two of the following signals be obtained during reactor operation:

1. Extremely low coolant flow at the inlet of the reactor as measured in representative connections.
2. Extremely high or low primary loop pressures at the rear of the reactor.
3. Extremely high coolant temperature in reactor effluent piping downstream of the rear tube connectors.
4. Loss of primary coolant system pumps.

Each individual trip has coincidence trips in itself as illustrated by Table 6-5.

Hence, based on the 15 individual measurements, the coincidence required for calling on the last-ditch system ranges from four of fifteen to seven of fifteen depending on the identity of the particular individual trips.

TABLE 6-5COINCIDENCE TO GIVE INDIVIDUAL SIGNAL - LAST DITCH COOLING SYSTEM

<u>Circuit</u>	<u>No. of Instruments</u>	<u>Required Coincidence</u>
Low Flow	4	2 of 4
Low or High Pressure	3	2 of 3
High Temperature	3	2 of 3
Low Pump Speed	5	3 of 5

6.5 Codes and Specifications Governing the Design, Procurement, Fabrication, Erection and Testing of The Primary Coolant System

6.5.1 General

The importance of the primary coolant system to reactor safety led to the adoption of certain minimum standards for design, procurement, fabrication, erection and testing of components. Adequate mechanical design and adequate quality of materials of construction require something in addition to the minimum standards which would be acceptable for ordinary commercial or defense construction. Therefore, to ensure the integrity of the system, supplementary requirements have been added to the basic standards to provide control of quality to a degree commensurate with the hazards of failure and to provide extra conservatism in those areas where the behavior of materials of construction under the intended service conditions is not clearly defined or where strict adherence to the basic standards would actually introduce a hazard.

Design pressure and temperature for the primary system are 1825 psig and 600 F, respectively, except as limited by the pressure rating for 900-pound class carbon steel valves at temperatures above 525 F.

6.5.2 Standards of Design

All primary coolant system piping and piping attached thereto and capable of being subjected to primary coolant conditions of temperature and pressure is designed to comply, as a minimum, with the latest edition of the American Standard Code for Pressure Piping, ASA B31.1, Section I, for Class I Power Piping Systems including revisions, addenda and applicable interpretations, except as provided below.

Design of the Zircaloy process tubes is also in accordance with the additional requirements of Section I of the ASME Boiler and Pressure Vessel Code.

All unfired pressure vessels and steam generators are designed, as a minimum, in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code, including revisions, addenda, and applicable Case Interpretations, specifically including Case 1234 and the subsequent N-Series Cases.

Pressure relief is provided in accordance with applicable provisions of the cited codes except as provided below.

All valves and fittings in the primary system or associated piping are specified to be in accordance with the 900-pound Class American Standard Steel Pipe Flanges and Flanged Fittings, Including Applicable Requirements Relative to Flanged End and Butt Welding End Valves, ASA B16.5 - 1957.

6.5.3 Exceptions to Standards of Design

Some situations were encountered in NPR design where it was not safe, practical or feasible to comply with the stipulations of the cited standards. Instances of departure are few in number, and these have been documented in the form of Code Variances () with supporting justification. In brief, these variances are as follows:

1. Zirconium alloy used for process tubes is not a code-approved material for which allowable stress values have been assigned.

Allowable stress values for design purposes have been obtained

from laboratory tests conservatively interpreted, but the ratio of tensile strength to working stress will be 2.8 instead of 4 based upon short-term tensile tests of annealed Zircaloy-2 strip.

2. The pressure-relief system was not designed with sufficient capacity to relieve the maximum possible thermal energy output of the reactor without allowing the pressure to rise more than 10 per cent above the design pressure allowed by the ASME UFPV Code.

The pressure relief system was not provided primarily to meet code requirements but rather is an integral part of the over-all protection system which includes the reactor power setback and scram systems ()(). Strict adherence to the code is impractical because the maximum thermal energy output of the reactor cannot be defined. Further, at the higher conceivable powers, fuel melting would probably occur in any event. Prevention of a fuel melting accident with consequent release of fission products depends primarily on the nuclear control and coolant supply system and only incidentally on the pressure relief systems.

3. Shutoff valves are not provided at the end of each instrument take-off connection, and the minimum size of instrument take-off connections is not necessarily 1/2 or 3/4 inch as required by the ASA Code.

Shutoff valves at the end of instrument take-off connections would serve no useful purpose as they would be inaccessible during reactor operation. The takeoff connections to the first valve

cannot reasonably be sized in accordance with the code because of space limitations. In any event, failure of the small line will not constitute a reactor hazard.

6.5.4 Standards for Procurement, Fabrication, Erection and Testing

In addition to the standards for procurement, fabrication, erection and testing set forth in the design codes previously cited, a number of supplementary requirements in the form of Standard N Sections prepared for the reactor project were used to govern the quality of specific components, materials, and joints, and to ensure that the methods of testing would be sufficiently definitive to provide the necessary degree of quality control.

The Standard N-Sections embody specific portions of the ASA and ASME Codes and the ASTM Standards pertinent to control of the quality of the material, product, or operation concerned. In general, the Standard N-Sections set forth specific requirements where these primary standards permit the exercise of discretionary powers on the part of the supplier in regard to alternate materials or methods or in the interpretation of test results. In addition, certain supplementary requirements are included in instances warranted by HAPO experience or the recognized best practice within the Atomic Energy Industry.

Eighteen Standard N-Sections controlling the quality of the primary coolant system and connected auxiliary system piping and components are listed below:

Standard Section N1 for Manufacture of Electric-Fusion-Welded Corrosion-Resisting Steel Pipe

Standard Section N2 for Centrifugally Cast Austenitic Chromium - Nickel Corrosion-Resisting Steel Pipe

Standard Section N3 for Seamless and Welded Corrosion Resisting Chromium Nickel Steel Tube and Pipe

Standard Section N4 for Corrosion-Resisting Steel Bars and Forgings for Radioactive System Service

Standard Section N5 for Corrosion-Resisting Steel Plate, Sheet, and Strip for Radioactive System Service

Standard Section N6 for Carbon-Molybdenum and Manganese-Molybdenum Alloy Steel Plate Hot Rolled

Standard Section N7 for Welding Specification for Radioactive System Service

Standard Section N8 for Cleaning Requirements for Nuclear Primary Coolant Equipment Including Piping Systems

Standard Section N12 for Inspection of Material

Standard Section N13 for Preparation for Shipment and Storage of Components and Spare Parts for Radioactive Service

Standard Section N14 for Non-Destructive Testing Requirements for Metals

Standard Section N15 for Austenitic, Corrosion-Resisting Chromium - Nickel Steel Castings for Radioactive System Service

Standard Section N16 for Electric - Fusion - Welded Carbon Steel Pipe for Radioactive System Service

Standard Section N17 for Seamless Carbon Steel Pipe for Radioactive System Service

Standard Section N18 for Wrought Carbon Steel Welding Fittings for Radioactive System Service

Standard Section N19 for Carbon Steel Castings for Radioactive System Service

Standard Section N20 for Hangers, Anchors and Supports

Standard Section N21 for Carbon and Low-Alloy Steel Forgings for Radioactive System Service

6.6 Assurance of Continuity of Cooling

6.6.1 General

The reliability of critical reactor cooling system is one of the most critical areas of reactor design. Inadequate cooling could lead to extremely high levels of damage including loss of life, impairment of health, loss of crops and long term evacuation of land. Therefore, normal engineering standards routinely applied to "low risk" engineering are not considered acceptable as bases for judgement of reactor cooling system design and operation. On the other hand, generally accepted engineering standards specifically covering nuclear cooling systems have not yet been evolved by the industry. Therefore, a set of coolant continuity criteria have been developed at Hanford and were used to guide the design of critical cooling systems.

Conformance of the design with the criteria was required and achieved except for specific exceptions allowed after careful evaluations by responsible authority. Exceptions authorized will not adversely affect nuclear safety. The exceptions probably will result in a reduction of the potential maximum operating efficiency of the plant. Additional reactor shutdowns will be required in some instances to ensure reactor safety.

In the following sections the coolant continuity criteria are listed and conformance of critical systems with the criteria discussed. Exceptions to the criteria are pointed out and evaluated.

6.6.2 Reactor Cooling System Reliability Criteria

The provisions of the criteria, condensed somewhat, are as follows:

1. Three reliable, independent sources of cooling will be provided.
2. The normal cooling source will provide adequate cooling during all phases of operation and shutdown.
3. Reactor cooling by the normal cooling source is required during all reactor operation.
4. The first backup to the normal cooling source shall be independent of the normal source except for common piping. It shall be capable of providing adequate shutdown cooling indefinitely assuming instantaneous loss of power to the normal cooling source.
5. The last ditch cooling system shall be independent of the normal and first back-up cooling sources. It must be capable of providing shutdown flow indefinitely assuming instantaneous loss of power to the normal system and concurrent failure of the first back-up system.
6. Design and operation should be such that neither the first back-up or the last ditch system is expected to be the sole cooling supply system during the life of the plant.
7. Failure of a single component in a system shall not be capable of rendering that system inadequate. In practice certain piping and valves may be qualified exceptions.
8. All engineering, construction, operation and maintenance shall meet or exceed requirements specified by applicable codes and standards.

The following definitions apply to terms used above:

1. An independent system is defined as one with power sources and pumping system such that failure or cause of failure in one cannot induce failure of the alternate system.
2. Adequate cooling is defined, as a minimum, as that required to maintain reactor components including fuel elements below the melting point. In practice the objective of requiring levels of cooling capable of protecting against all reactor or fuel damage will be prudent to adopt.

6.6.3 The Main Reactor Cooling Complex

Most of the heat generated within the reactor during both operating and shutdown periods is removed directly by water flowing through the zircaloy process tubes. Ordinarily the heat is transferred to the secondary system and thence to condenser cooling water. During outages, the reactor may be cooled on a once-through basis with filtered water.

The conformance with the criteria of this main cooling complex is covered below:

1. Provision For Three Reliable Independent Cooling Sources

There are three independent cooling sources as is required by the criteria. Normal cooling is obtained when:

- a. Recirculation of the primary coolant is accomplished with the primary system pumps powered by steam driven turbines. The steam may be generated by process heat in the steam generators or by the standby boiler.

- b. Duplicate, operating, secondary system components vital to cooling continuity are powered from the BPA and the locally generated power system. Examples of the "vital" equipment include the condensate return pumps on the dump condenser system, condensate return pumps on each of the five primary pump drive turbines and the airflow supplying the standby boiler.
- c. At least one river pump on the circulating water system is powered by each electrical source.
- d. Vital auxiliary system, e.g., primary system injection pumps, pressurizer heat, etc., are duplicated and both sources of power being used to power operating components.

The first backup system is, in large part, an operating system. This feature enhances its inherent reliability. Except for the main primary coolant system pumps, the first backup system consists of reactor cooling with some (or all) vital functions obtaining power from only one of the two sources. As required by the criteria, the reactor will never be operated during such periods.

The first backup system for the primary coolant system operation is operation with the primary pumps powered by electric pony motors instead of the drive turbines. These motors are on the same shaft as the turbines. Only BPA power is supplied to the pony motors. There is no point in supplying local power to these units. If steam is available to power the turbine generator set, it could be used directly to power the drive turbines.

The last-ditch backup involves the use of raw water for once-through reactor cooling. This system will operate even if the circulating water system, the secondary coolant system, and parts of the primary system (except for inlet piping) are out of service.

There are three general factors influencing reliability of the last-ditch system: The reliability of the pumps starting and operating for prolonged periods of time; the reliability of the connecting piping, and the reliability of the instrumentation and valving required for actuating the system.

The pumps are considered highly reliable, The diesel drive engines will be tested frequently (see Section 6.3.4) and the required number of pumps will almost undoubtedly start. These engines are started by compressed air. The compressed air is obtained by two compressors powered from the two power sources. The compressors feed into a common accumulator so a reservoir is available should both the compressors fail. Further, each individual diesel engine has its own accumulator guarded by check valves with capacity for at least two starts should the connecting line to the main accumulator fail.

Fuel sufficient for about ten days of operation of the pumps will be stored in the area and will flow by gravity to the pump engines. The diesel oil will flow by gravity to both sets of motors, until the level in the tank falls to the half-full mark. Beyond this point, the fuel will continue to flow by gravity to the river pumps but must be pumped to the high lift pumps. The pumps for this job are powered by the shafts of the river diesels and, hence, there is no dependence on electrical power.

The line conducting the last-ditch raw water to the reactor has been designed to unusually high levels for reliability. For instance, this line is designed to withstand earthquakes more severe than experienced during the recorded history of the western states. (See Section 4.4).

Positive opening of inlet and outlet valves is required for the last-ditch system to operate. Though the probability is low that the system will be called on more than once or twice during the entire reactor life, efforts are being made to ensure that this valving will operate reliably on demand. Individual valve failure will not prevent the complex from performing its designed function. Even if the automatic feature should fail, the valves can be operated manually from the control room.

6.6.4 Graphite Moderator Cooling System and Fog Spray System

In a sense, the building fog spray system and the graphite moderator cooling system are a back-up to the reactor last-ditch system. Should the last-ditch system fail, the graphite moderator cooling system will provide for reactor cooling to a degree sufficient to limit the extent of fuel melting and the fog spray will dissipate heat released from the reactor to the building atmosphere. The reliability of the graphite moderator and fog spray systems is on a par with the main reactor cooling complex. Normally, BPA and local power are used to drive coolant circulating pumps. Either source alone will provide for adequate cooling should the other fail. On failure of both electrical sources, the graphite will be cooled with raw water from an independent diesel powered emergency water supply.

6.6.5 Horizontal Rod Cooling System

The horizontal control and safety rods are normally cooled by plant filtered water. This system is the same as that used for once-through reactor cooling (discussed in Section 6.5.3), except that a different set of pumps supply water from the filtered water storage tank to the rod systems.

On failure of both electrical power sources, the rods are supplied with cooling water from the same system as used for the last-ditch graphite moderator backup cooling and the fog spray systems.

FIGURE 6-2
SECONDARY SYSTEM

SHEET NO. OF

JOB NO.

River
Shoreline

River
Mid-Channel

SUBJECT

DATE

DATE

BY

CHKD. BY

River Pumps

FIGURE 6-3

CIRCULATING WATER SYSTEM

SHEET NO. OF
JOB NO.

SUBJECT

BY DATE
CHKD. BY DATE

FIGURE 6-4

EMERGENCY RAW WATER SYSTEM

UNCLASSIFIED

FIGURE 6-
GRAPHITE MODERATOR COOLING SYSTEM

UNCLASSIFIED

FIGURE 6-6

ELECTRICAL DIAGRAM - N REACTOR - NORMAL LOADINGS

UNCLASSIFIED

7. SYSTEMS FOR CONTROL OF RADIOACTIVE CONTAMINATION

7.1 Airborne Fission Product Confinement System

7.1.1 General

The function of the fission product confinement system is to minimize the release of fission products to the environs during all credible accidents. These credible accidents range from the partial oxidation of a single fuel element to meltdown of the significant part of the reactor fuel inventory.

The confinement system for NPR is based on separation of energy and controlled release concepts. The system is designed so the thermal energy in the primary loop would be released from the building as steam and the building resealed before fission products could be released from the irradiated uranium fuel in the event of a rupture of primary loop piping. The heat from fission product decay and oxidation of materials after the building re seals would be dissipated within the building by a fog spray system. The remaining building atmosphere contaminated with fission products would ultimately be released in a controlled fashion through filters and halogen collectors. These concepts of energy separation and controlled release are especially applicable to a large reactor such as the NPR. A maximum pressure containment vessel would be difficult to design and construct and would be prohibitively expensive.

The parts of the plant buildings housing contaminated or potentially contaminated areas are sealed off from other parts of the build-

ings and from the outside except at controlled openings. Should airborne contamination occur without an accompanying pressure surge, all openings into these areas can be sealed off and the contaminated air filtered before release.

Should an incident occur in which there is a major line break with pressure buildup, all openings will automatically close. When the pressure builds up to about 2 psig, seal-caps on roof steam vents will blow off releasing the steam directly to the atmosphere. When the steam venting has been largely completed and the building pressure approaches atmospheric pressure, these steam vent passages will automatically reseal. The remaining heat from steam energy and energy from fission product decay and potential oxidation of material is dissipated by a fog spray system. When desired, ventilation ducts can be opened allowing the contaminated atmosphere to be released through filters and out the stack.

During the steam venting transient, sufficient reactor cooling will occur from flow through the reactor and out the primary loop to prevent fuel melting.

Three ranges of piping ruptures are identified and considered:

1. Small ruptures (up to about 10 square inches effective area).
injection pumps will provide adequate makeup. No system dump will be required. No venting of the 105-109 Buildings will be required.
2. Major line ruptures. The loop will assuredly be depressurized, the steam vented, and the building resealed before any fuel

element melting occurs, even without reliance on dump valves.

3. Intermediate sized pipe ruptures. Eight dump valves are provided to assure that the system would depressurize with sufficient speed that the steam can be vented and the building resealed before any fuel melting occurs.

Thus, multiple failures of instrumentation and equipment concurrent with an intermediate sized pipe break would be a necessary (but not necessarily a sufficient) condition for fuel melting before the steam had been vented and the building resealed.

The energy separation concept is unique to the NPR fission product confinement system. Other functions and concepts are similar to those included in maximum pressure confinement system.

The confinement system consists essentially of three zones. Zone I surrounds the primary loop where pressurized steam and radioactive materials may be released. Zone II surrounds portions of the reactor where the potential for contamination with radioactive materials is relatively high (such as the inner rod room), but the potential for pressure buildup is low. Zone III surrounds Zones I and II. Normal building atmospheric pressures are arranged so air leakage will be inward from Zone III to Zone II or Zone I, and from Zone II to Zone I. The confinement zones are shown in Figure 7-1.

Other areas in the 105-109 Buildings are Zone IV, unrestricted

TABLE 7-1
SUMMARY OF 105-N AND 109-N BUILDING ZONES

<u>Name</u>	<u>Location</u>	<u>Confinement Function</u>	<u>Design Pressure</u>	<u>Operating Pressure</u>	<u>Precautions</u>
Primary Confinement Zone (Zone I)	Encloses the Reactor and the Primary Coolant System	Confine contamination in areas where there is a potential for pressure build-up from steam release.	Negative 2.0 psig to a positive 5.0 psig(1)	Negative 0.45 inches of water, gage	All penetrations sealed or will seal automatically. No personnel entry normally possible during reactor operation.(2)
Intermediate Confinement Zone (Zone II)	Encloses all portions of the Reactor Block not in Zone I, plus other areas which have high probability of contamination.	Confine contamination in areas where there is a potential for only a low pressure build-up.	Negative 0.1 psig to a positive 0.5 psig.	Negative 0.30 inches of water, gage	All penetration fitted with seals. Not normally sealed off. Personnel entry possible during normal operations.
Secondary Confinement Zone (Zone III) (B & R Zone II)(3)	Includes all areas surrounding Zones I and II	Provide a buffer around Zone I and II	Negative 0.1 psig to a positive 0.2 psig.	Negative 0.15 inches of water, gage.	None
Administration Area (Zone IV) (B & R Zone III)(3)	Balance of Buildings except for Zone V in 105-N and Zone VI in 109-N	None	Atmospheric	Atmospheric	None
Warranted Access Zone (Zone V)	105-N Control Room	Provide warranted access to control room for at least limited periods after any credible incident.	Atmospheric	Atmospheric	None
Reactor Decontamination Facility Zone VI (B & R Zone IV) (3)	Encloses Decontamination Solution Preparation Area	Control airborne contamination escaping from open tanks in decontamination Facility.	Negative 0.1 psig. to positive 0.2 psig.	Negative 0.70 inches water gage	

- (1) Cells in 109-N Building are designed to withstand a positive pressure of 8.0 psig.
 (2) Access to individual cells in 109-N Building is possible during reactor operation if cells are bypassed
 (3) B & R Zone Nos. refer to zone numbers given by Burns and Roe to areas in the 109-N Building

administrative and work areas. Zone V, 105 Control Room warranted access area, and Zone VI, 109-N Building decontamination solution makeup area.

The major confinement system components are:

1. Parts of the 105-N and 109-N Buildings.
2. Steam vents.
3. The ventilation and filter system.
4. The primary coolant system and the moderator coolant system.
5. The fog spray system, and
6. Drains and vacuum relief.

These components and their functions in the confinement system are briefly described below. Further detail will be included in an appendix to this volume.

7.1.2 The 105-N and 109-N Buildings

The 105-N and 109-N Buildings house the reactor, the heat dissipation system, reactor controls and other auxiliary systems and functions. These buildings are divided into zone which provide for confinement and control of radioactive contamination.

Figure 7-1 shows the arrangement of the different zones at ground level. The function, extent, and characteristics of these zones are outlined in Table 7-1.

7.1.3 Steam Vents

Thirteen six-foot diameter roof vents from Zone I to the atmosphere are provided for the venting of steam in the event of a pipe rupture. Seven of these vents are located at the east end and

four at the west end of the 109-N Building pipe gallery. Two vents are located in the ceiling of the 105-N pipe space. Figure 7-2 shows a typical vent. Each vent consists of a butterfly valve, a pipe extension, an inflatable-seal type back-up closure and a pressure relief cap. During normal operation the butterfly valve is open, the inflatable-seal back-up closure is stored in a container and the vent cap is sealed maintaining confiner integrity.

During a pressure transient the cap will be blown off when the pressure reaches about 2 psig. After the steam has escaped, the butterfly valve will close. Should the butterfly valve fail to seal completely the inflatable seal can be manually released from the control room to effect the seal.

Steam vents from both buildings may be required in case of a line rupture in either the 105-N or 109-N Building. Eight openings in the common wall between these buildings will permit steam flow to all the vents from either building. Normally, these openings are blocked off by sealed barriers to maintain proper ventilation and control contamination spread. These barriers will be blown open by a 2 psig pressure differential in either direction. One of these vents will also be operated manually to permit filtering 109-N Building air with the normal 105-N exhaust. Figure 7-3 shows the operation of the barriers.

7.1.4 The Ventilation and Air Filter System

Each confinement zone is provided with a separate forced air ven-

tilation system tailored to the characteristics of the particular zone. Air pressures in the various zones are maintained so any leakage of air will be from a less contaminated zone to a more contaminated zone. This is shown in Table 7-1. Other features of the various systems are shown in Table 7-2.

Contaminated air will be filtered and released to a 200 foot high stack to reduce contamination of the environs. Hence, the air from zones normally contaminated will be filtered at all times. Air from zones unlikely to ever be contaminated, e.g. Zone IV in the 105-N Building, is never filtered and air from some zones, e. g. Zone I in the 109-N Building, will be filtered only during and following an incident.

The reliability of the ventilation systems differs from zone to zone according to the potential seriousness of contamination. The ventilation system for Zone I in the 105-N Building where there is potential for high air contamination levels is highly reliable. Normal ventilation of this zone could be maintained even though there was a failure of the normal supply fan, one of two normal exhaust fans and one source of power. At the other extreme, ventilation systems for Zones which serve no confinement function have no spare fans and are supplied with only one source of power.

Closures have been installed on inlet and outlet ducts penetrating confinement zones so that the ventilation systems do not compromise confiner integrity. The closures on duct pene-

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TABLE 7-2
VENTILATION SYSTEMS

Zone No.	Building No.	Air Supply	Air Exhaust	Exhaust Air Treatment	Comments
I	105-N	One fan powered from one electrical source. Spare shared with Zones II & III powered from the other electrical source.	Three fans. Each rated at 50% of load. One supplied with BPA power, one with local generator power and one with power available from either source ..	100 Per cent of air filtered and then sent to stack.	Inlet and outlet ducts close automatically when pressure exceeds limit. This protects equipment and seals confiner in event of line rupture.
I	109-N	Two fans. Each rated at 100% of load. One powered from BPA and one from local generator.	Two fans. Each rated at 100% of load. One powered from BPA and one from local generator.	Normally air discharged through roof vents to atmosphere. Air can be combined with Zone I air from 105 Building, filtered and sent to stack.	Inlet and outlet ducts close automatically when pressure exceeds limit. Radiation detector in exhaust line. Excessive ^(a) air contamination will sound alarm permitting manual closing of roof vents from 105-N control room.
II	105-N	One fan powered from one electrical source shared with Zone III, 105-N. Spare for Zones II & III shared with Zone I, 105-N. Spare powered from other electrical source.	Three fans. Two each rated at 50% of load powered from one source. One rated at 100% of load powered from other source.	100% of air filtered and then sent to stack.	
III	105-N	See Zone II, 105-N	Three fans. Each rated at 50% of load. One supplied with BPA power, one with local generator power, and one with power available from both sources.	Normally part of air routed to stack and part to roof exhausts. However, all Zone III air may be filtered if necessary and sent to stack.	
III (B & R Zone II)	109-N	Two fans. Each rated at 100% of load. One supplied with power from BPA and one from local generator.	Two fans. Each rated at 100% of load. One supplied with BPA power and one with local power.	None. Air discharged through roof exhausts to atmosphere.	Excessive ^(a) air contamination will result in automatic closure of exhaust lines.

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Table 7-2 - Continued

<u>Zone No.</u>	<u>Building No.</u>	<u>Air Supply</u>	<u>Air Exhaust</u>	<u>Exhaust Air Treatment</u>	<u>Comments</u>
IV (B & R Zone III)	109-C	One fan supplied with power from both sources.	One fan supplied with power from both sources	None	
V	105-N	One Fan	One Fan	None	Air may be recirculated to prevent drawing in any outside contamination.
VI (B & R Zone IV)	109-N	None	Two fans for normal operation. Each rated at 100% of load. One supplied with power from BPA and other supplied with power from local generator. Third fan provided for occasions when exhaust air is filtered.	Normally none. However, may be sent through filter provided for purpose.	Should excessive ^(a) air contamination occur, inlet and outlet dampers will be manually closed and air exhausted through the special filter.

(a) Exact limits on air contamination have not yet been set.

trating Zone I in the 105 N and 109 N Buildings will close automatically within one or two seconds after a major line failure. This will prevent damage to ventilation and filter equipment and control of contamination spread. Closures into zones where only a small pressure build-up can occur will be manually closed in case of pressure build-up and will operate more slowly.

The air filter system and its relationship to the 105 N ventilation systems is shown in Figure 7-4. The four filter banks are essentially identical. Each contains three sections: two for removal of particulate matter and the third for removal of halogens. These filters will remove 99.9 per cent of the particulate and more than 50 per cent of the halogens. (In practice about 95 per cent removal of halogens is expected).

7.1.5 The Primary Coolant System and the Moderator Coolant System

Both the primary coolant loop and the moderator coolant system have a role in the confinement system. Check valves have been installed in the primary loop on the inlet supply lines and on the return lines just downstream of the dump valves, as shown in Figure 7-5. In case of a line rupture and failure of the last-ditch coolant systems, these valves will force a portion of the water in the loop to flow through the reactor before leaving the break. This point is critical. It is necessary that interim cooling be provided to prevent fuel melting until steam venting is completed.

The moderator coolant system will remove some heat from the reactor in the event of the complete failure of the primary coolant loop. While this will probably not prevent some fuel melting from occurring, it will minimize the extent of the melting.

Because of the potential importance of the moderator cooling system a high degree of reliability has been provided. Circulation pumps are powered both with BPA and local electric power. In the event of failure of both power systems, cooling will be supplied by raw river water on a once through basis.

7.1.6 The Fog Spray System

A fog spray system has been provided in Zone I of the 105-N - 109 N Buildings to serve four functions:

1. To condense part (120 lbs/sec) of the steam released from a break in the primary coolant loop during steam venting. This decreases the time that the steam vents must remain open.
2. To condense steam left in the building after the steam vents reseal.
3. To cool the building after the steam vents reseal. The energy released by fission product decay and any oxidation of reactor materials will be dissipated by this system.
4. To settle particulate matter and reduce the effluent contamination from the filters.

The fog spray system is composed of two parts. The main part of the system has nozzles that will cover the floor area within Zone I. This part is sectionalized into several sub-systems so the

failure of one part will not result in failure of a major portion of the total capacity. This part of the system is automatically actuated by a signal from the building air pressure sensors.

A manually operated secondary fog spray system is attached to the lower parts of the charge and discharge elevators and will spray water on the reactor front and rear face piping. Here it will serve to cool fuel elements should they become lodged in the reactor piping. It will also settle particulate matter should a uranium fire occur.

The fog spray is supplied with coolant from a highly reliable source of raw river water.

7.1.7 Drains and Vacuum Relief

In case of primary loop pipe rupture, the water in the loop which does not flash to steam, the steam condensed, and the water from the fog spray systems must be drained from the building to prevent flooding and rupture of the walls. Drains have been provided for this purpose. These drains are fitted with valves so confiner integrity is maintained.

If the fog spray system does not shut-off on signal, the building pressure would eventually drop to less than the design pressure of a negative 2.0 psig. Vacuum relief lines are provided and will open if the pressure drops to below a negative 1.7 psig. These lines will reseal after the vacuum has been relieved.

7.2 Liquid Waste Disposal

7.2.1 General

Liquid wastes from the NPR Plant are controlled to prevent:

1. Excessive pollution of the environs, and
2. Deleterious effects to the downstream reactors.

The wastes are collected into five streams:

1. High contaminated wastes and chemical wastes that should not be released to the river are routed to storage tanks.
2. Wastes of "medium" contamination are routed to a crib at the NPR Plant site.
3. Heated wastes, slightly radioactive contaminated wastes, and acceptable chemical wastes are routed to a mid-channel river disposal line.
4. Cool, non-contaminated waste water is routed to the river shoreline.
5. Sanitary wastes are routed to a tile field.

The last two streams present no waste disposal problems and are not considered further.

Figure 7-6 is a simplified flow diagram of the remaining three waste streams.

The various systems are described below:

7.2.2 Storage Tanks

Two storage tanks each of 900,000 gallon capacity have been provided as a part of a waste disposal system for handling reactor decontamination wastes. These wastes will be temporarily stored

in the tanks, treated, and then pumped out to a crib. The details of the treatment process and the location of the crib have not yet been specified. However, the final details will provide for:

- a. Retention of the bulk of the radioactivity in the wastes.

The exact amount of radioactive contamination in the wastes cannot be determined before hand. It is expected that a full reactor decontamination will remove on the order of 5,000 curies.

- b. Prevention or control of uptake of spent solutions by downstream production reactors. The decontamination solutions may contain elements, e.g. P^{31} , whose uptake and irradiation by downstream reactors would aggravate the river pollution problem.

7.2.3 Crib At The NPR Site

Some NPR wastes will, or may, contain fission products or activated corrosion products contamination of a degree to high for river disposal, but ordinarily too low to merit the special treatment afforded decontamination wastes. Such waste streams include:

1. Normal bleedwater from the primary loop.
2. Effluent from the process tube diversion system
3. Effluent from the spill cooler.
4. Shield cooling water.
5. Control rod cooling water.
6. 105-N Building drains.
7. Effluent from the reactor "dump" system.

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These wastes may be routed to a crib at the NPR Plant site. On particular occasions these wastes can also, if necessary, be sent to the storage tanks. The crib will remove particulate matter and release the water to the river.

7.2.4 Mid-Channel Disposal

The final waste stream enters the river through a line extending to the middle of the river channel. These wastes will be mixed with the bulk stream before reaching the downstream reactors.

The waste in this stream will be thermally hot, very slightly contaminated or contain "harmless" chemicals. These wastes do not present a serious pollution problem. They require only a thorough dilution and mixing before or during release.

FIGURE X-1

105 - 109 N PLAN VIEW --- 0' - 0" Level

G-132-468
AS-AS INSTRUCT, 1968.

7-2
FIGURE 82
STEAM VENT

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FIGURE 7-3
COMMON WALL VENTILATION BARRIER

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Refrigerated Air Conditioning
Equipment

Zone

FIGURE 8-4
105-H AIR VENTILATION AND FILTER

SHEET NO. OF

JOB NO.

SUBJECT

DATE

DATE

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CHKD. BY

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Page

Note: Check Valves Force Part
of Coolant Through Reactor
No Matter Where Line Breaks

FIGURE 8-5
PRIMARY LOOP - 105-N

CHKD BY _____
DATE _____

SUBJECT _____

JOB NO. _____
SHEET NO. _____ OF _____

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BY _____ DATE _____ SUBJECT _____ SHEET Page _____ OF _____
 CHECKED BY _____ DATE _____ JOB NO. _____

FIGURE 8-6

MANAGEMENT OF WASTE STREAMSUNCLASSIFIED

8. REACTOR OPERATION

8.1 General

Within the Irradiation Processing Department at HAPO, the responsibility for nuclear safety is delegated in writing to the appropriate levels of the line organization as is responsibility for every other major aspect of plant operation. The NPR will be included under the existing administrative framework in substantially the same manner as the existing production reactors.

The policy of the Irradiation Processing Department at HAPO is to operate the reactors in such a manner as to maintain the probability of hazardous incidents at very low values and to conserve reactor life while efficiently producing plutonium. This policy is implemented through the line organization. Particular responsibilities for reactor safety are clearly designated according to General Electric organization policy.

8.2 Philosophy of Operation

8.2.1 General

The existing Hanford production reactors are designed and process limits are established in such a fashion that the reactors can be controlled manually by the operators.

The operators are relied upon to maintain sensory and control instrumentation in proper operating conditions, with trip limits set as specified by documented "Process Standards".

The operators are further relied upon to comply with written "Operating Procedures" while controlling the nuclear process.

Automatic control instrumentation and equipment is provided to rapidly terminate the nuclear chain reaction in the event that critical process limits are exceeded due to equipment failure or operator error.

Production responsibility and nuclear safety responsibility are organizationally separated. Prime responsibility for nuclear safety is vested in the Research and Engineering Section of IPD. Prime production responsibility rests with the Manufacturing Section of IPD.

Research and Engineering Section produces two sets of documents designed to establish process limits consistent with safe operation - from the nuclear hazards point of view - and extended service life of the facility. The first of these is a set of technical bases for standards couched in engineering terms and mathematical formulae which establish criteria and means for measuring compliance with these criteria. These technical bases are generated within the Process and Reactor Development Subsection of Research and Engineering, and are approved by the manager of that subsection. They are used in generating the second set of documents, the Process Standards. The process standards are generated within the Operational Physics and Process Technology Subsections of Research and Engineering. They must comply with the requirements of the technical bases, are much more detailed, and are expressed in terms which are meaningful for direct application to a

particular reactor.

After approval by the manager of Research and Engineering and acceptance by the reactor plant managers, these process standards become mandatory.

Members of Research and Engineering as well as Manufacturing personnel monitor conformance with these standards.

The "operating procedures" are generated within the Manufacturing organization. They are consistent with the requirements of the Process Standards, and are sufficiently detailed to provide adequate guidance to control room operators.

8.2.2 NPR Operation

The N-Reactor (NPR) plant is more complex than existing production reactor plants, primarily in the manner in which heat is removed from the reactor. This will necessitate some deviations from the manual control philosophy used at the older reactors. For example, a reactor scram at the older reactors may be effected by pressing the "scram" button. No further action is required. At N-Reactor, a scram must be accompanied by a pattern of operations designed to prevent the contraction of the primary loop coolant from resulting in depressurization sufficient to cause gross boiling in the process tubes. This involves reducing the flow rate in the loop and bypassing steam generators. Similar actions are required in the secondary loop. It is planned, therefore, that necessary actions to minimize system damage upon scram will be

performed automatically. The operator's role will be to monitor the process and to take corrective action in the event of equipment or instrument failure.

8.3 Process Standards

Of primary importance in assuring reactor safety are the Process Standards. The Process Standards are written instructions, approved and concurred in by management, which specify for the reactor process and vital auxiliary plants, limits, materials, procedures, and equipment where such specification is necessary to safeguard against nuclear hazards and to conserve reactor life.

The Research and Engineering Operation is responsible for issuance of Process Standards. The Process Standards are concurred in by plant managers and each plant manager is responsible for assuring full and complete compliance with the Process Standards with his Operation. Once authorized, the Process Standards are binding and mandatory except for:

1. Operation under temporary waiver of the Process Standards which may be given in special cases on the authority of the Manager, Process Technology. All such temporary change authorizations are described and justified by him in writing to the Manager, Research and Engineering so that the decisions can be reviewed.
2. Operation within the conditions set forth in documents formally authorizing an experiment which permits operation outside of established limits with a controlled degree of risk for the duration of the experiment.

Such documents describe the risks and hazards involved in the test

so that approvals from Research and Engineering and plant managers signify that the risks are understood, reasonable and compatible with the gains expected from the experiment. Experiments are terminated immediately if it appears that unexpected or potentially unsafe behavior is being experienced.

Unusual incidents are investigated by members of Plant Operations organizations. Causes are determined and remedial action is specified. Depending upon the seriousness of the incident a formal report or reports may be made.

8.4 Selection and Training of Personnel

The competence of operating personnel is recognized as a vital element of safe reactor operation, for despite all the standards, procedures, and equipment which can be provided, the ultimate responsibility for reactor safety rests with the individuals who actually operate the process.

Although it is recognized that NPR operation will differ in many important respects from that of existing HAPO reactors, the preponderance of remaining similarities clearly indicate that people experienced in the operation of the existing reactor plants, represent the most plausible candidates for staffing the NPR. Accordingly, in 1959, steps to increase the pool of HAPO reactor plant trained people were initiated.

In early 1959, three "operations representatives" were drawn from the reactor plants and assigned to work with the General Electric NPR project forces. These people were used largely as design consultants

to inject an element of operational experience into design efforts. They formed the nucleus of a growing reactor-plant-experienced group which will expand into the start-up operating crew. Collectively, these people have continued to function as design consultants though gradually the emphasis of their efforts is shifting to operational planning, and gaining an intimate familiarity with N Plant. This work includes the preparation of 700 systems descriptions -- documents wherein subsystem design concepts are preserved, as base source documents for the subsequent preparation of process standards, and operating procedures. Maintenance manuals are being generated, spare parts set up, and a preventive maintenance system developed.

Four of these people were assigned to the development of a training program for the remainder of the operating organization. A seventeen week curriculum was developed covering the equipment, systems, control, and operational aspects. During the final weeks of this training, preliminary operating procedures are prepared.

The operations' personnel will assist the contractor and project forces in running acceptance tests on the plant. This will afford an opportunity to check the validity of preliminary operating procedures. The initial calibration of much of the reactor instrumentation will be done by the operating organization.

The total time for this growth of an operating organization will be approximately four years. The supervisory personnel will have been assigned to N Plant for at least two and one-half years prior to scheduled startup. Control operators will have approximately a

one year assignment prior to startup. Pre-startup assignment times range down to about one month for some maintenance craftsmen. However, these latter groups will be seeded with men who have been indoctrinated through one to two year assignments as fieldmen during plant construction.

This operating organization is to be backed up by an engineering group comprised of project-oriented men who will have two to four years of plant design and construction experience.

This approach to the selection and training of personnel is calculated to produce a functioning, knowledgeable operating organization at start up who will have had the opportunity to develop both operating and maintenance procedures and, further, to check them for adequacy in most cases.

8.5 Methods of Minimizing Operational Mishaps

An operational incident can be defined as any event which results in reactor operation outside of established limits, or the overexposure of personnel to radiation. Two approaches are used at HAPO to minimize such events. The first is to reduce the opportunity for incidents to occur. This is done by means of Process Standards, detailed operating procedures, the use of check and data sheets, and by providing trained and experienced personnel. The second approach is to provide instrumentation to shut the reactor down automatically upon indication that abnormal operating conditions exist. Examples of check and data sheets, operating procedures and process standards will be included in Volume II of this review.

The NPR will be equipped with instrumentation which is designed to provide automatic control limitations to a degree as yet unmatched at HAPO. Several factors have dictated this relatively greater degree of automation. First, N Plant being a more complex, recirculating process, logically requires that more controls be provided. Secondly, the design of N Plant takes advantage of the extensive development in the control field over the past eight years since the last HAPO plants were designed.

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9. CONSEQUENCES OF RELEASE OF FISSION PRODUCTS

9.1 The Reactor Inventory of Fission Products

The designed goal exposure for Phase I operation of N Reactor (plutonium production only) is 1400 MWD/T. Assuming a non-block type, fuel charging-and-discharging process, there will be at any time fuel exposures ranging from zero to 1400 MWD/T, or an average exposure of 700 MWD/T. The full-reactor inventory of uranium is approximately 350 tons. Thus, the average irradiation time is 60 days or 1440 hours.

The reactor inventory of fission products has been estimated from data which are now available.¹⁻⁴ The total gamma and beta activities in curies are given in Table 9-1 for 1000-hour reactor operation at 4000 MW.

It should not be assumed that the total reactor inventory of fission products will be available for release to environs. It has been calculated

1. J. F. Perkins and R. W. King, NSE, 3, 726 (1958).
2. APEX-448, "The Activity of the Fission Products of U²³⁵", W. E. Knabe and G. E. Putman, October 31, 1958 (Unclassified)
3. USNR-DL-TR-187 "Gamma Decay of Fission Products From the Slow-Neutron Fission of U²³⁵". C. F. Miller, July 11, 1957 (Unclassified)
4. ORNL-2081, Part I -"Measurement of the Spectrum of Short-Lived Fission Product Decay Gamma Rays Emitted From a Rotating Fuel Belt", R. W. Peele, W. Zobel, and T. A. Lover, September 10, 1956 (Unclassified); Part II "Time and Energy Spectra of Fission Product Gamma Rays Measured at Short Times After Uranium Sample Irradiation", W. Zobel and T. A. Love, September 10, 1956 (Unclassified)

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TABLE 9- 1
NPR FISSION PRODUCT INVENTORY

<u>TIME AFTER SHUTDOWN</u>		<u>INVENTORY, CURIES⁽³⁾</u>		
<u>Hours</u>	<u>Days</u>	<u>Beta</u>	<u>Gamma</u>	<u>Total</u>
.028	.00115	1.3×10^{10}	1.0×10^{10}	2.3×10^{10}
.28	.0115	1.0×10^{10}	5.6×10^{10}	6.6×10^{10}
2.8	.115	6.0×10^9	3.4×10^9	9.4×10^9
28	1.15	2.9×10^9	1.8×10^9	4.7×10^9
280	11.5	1.2×10^9	5.6×10^8	1.8×10^9
2800	115	1.9×10^8	8.0×10^7	2.7×10^8
28,000	1150	6.8×10^6	7.0×10^5	7.5×10^6

(1) Assuming 1000 hours of operation at 4000 MW.

A more specific breakdown into known noble gas, halogen and particulate fractions is given in Table 9-2 . The breakdown is for 1400-hour operation at 4000 MW and 24-hour cooling.

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TABLE 9-2

FISSION PRODUCT INVENTORY BREAKDOWN
(1000-HR OPERATION - 4000 MW - 24 HR COOLING)

<u>Fraction</u>	<u>Amount Present (Curies)</u>
<u>Noble Gases</u>	← 3.24×10^8
Krypton - 88	3.3×10^5
Krypton - 85	1.1×10^6
Krypton - 83	6.8×10^4
Xenon - 135	1.06×10^8
Xenon - 133	2.17×10^8
<u>Halogens</u>	← 3.00×10^8
Bromine - 83	1.5×10^4
Iodine - 135	1.8×10^7
Iodine - 133	1.0×10^8
Iodine - 132	1.2×10^8
Iodine - 131	6.3×10^7
<u>Particulates</u>	← 4.27×10^9
Sr ⁹⁰	3.4×10^6
Cs ¹³⁷	2.1×10^6
Others	<u>4.265×10^9</u>
<u>Total</u>	← 4.9×10^9

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that during the maximum credible accident only 25 - 30 per cent of the fuel elements will melt and these elements will remain in the molten form for only about five minutes, due to continued heat removal by the graphite cooling system.

The fraction of various groups of fission products which it is assumed will be released from molten fuel is shown in Table 9-3:

TABLE 9-3

FRACTION OF FISSION PRODUCTS RELEASED FROM MOLTEN FUEL

<u>Fission Product Group</u>	<u>Fraction Release</u>
Noble Gases (Kr, Xe)	1.0
Halogens (Br, I)	0.5
Volatile Solids (Se, Ru, Te, Cs)	0.5
Non-Volatile Solids (Other Fission Products)	0.01

9.2 Fraction of Fission Products Released To Environs

Not all fission products released from the molten fuel will be released to the reactor environs. If the cause of the meltdown incident is a major line failure, the fission products released from the molten fuel would have to pass down the process tube, along the length of the individual process tube connectors and out the larger line to the site of the break. The external reactor piping will be cooled by the fog spray system which should promote some condensation of fission products within the piping. Since significant amounts of melting will not occur until most of the coolant is swept out of the process tubes, leaving a semi-static atmosphere of superheated steam it appears that there will not be a large

driving force expelling the fission products out the line break. It appears that the gas flow may even be towards the molten uranium as steam is consumed by oxidation of the fuel.

Once outside the primary system piping, an additional fraction of the fission products would be "rained-out" by the fog spray system or plated out on piping and structural surfaces. Fission products escaping the building would then have to either pass through the high efficiency filters provided for such an eventuality or leak out through penetrations in the confinement zone walls. Since the zone has been designed to be nominally leak-tight the fraction not passed through the filters should be minimized. Further, the walls are about five feet thick, "cracks" to the outside would be expected to be quite narrow so some "filtering" would occur even in this event.

Table 9-4 tabulates the assumptions made to determine the per cent of fission products released from the molten uranium which will gain eventual release to the atmosphere. It is believed that these assumptions overstate the fraction which will be released to the environs.

TABLE 9-4

ASSUMPTIONS MADE TO DETERMINE FRACTION OF FISSION
PRODUCTS RELEASED FROM FUEL THAT REACH ATMOSPHERE

<u>Fission Product</u> <u>Group</u>	<u>"Plate-Out"</u> <u>Efficiency</u> <u>Per Cent</u>	<u>Fog-Spray</u> <u>Efficiency</u> <u>Per Cent</u>	<u>Filter</u> <u>Efficiency</u> <u>Per Cent</u>	<u>Combined</u> <u>Efficiency</u>
Noble Gases	0	0	0	0
Halogens	50	50	95	98.7
Particulates	70	50	99.95	99.99+

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9.3 Damage Criteria

The damage caused by the atmospheric dispersion of radioactive material may be the result of injury of people or the contamination of property. The injury type of damage may be further categorized as those injuries which result from direct exposure to the cloud of contaminants (direct effects) and those which result from material deposited on the ground (indirect or residual effects).

Injury to people ranging from minor to fatal may be caused by exposure to ionizing radiations. However, lethal effects are generally the result of a large dose of external radiation administered over a short period of time.

Table 9-5 lists the damage criteria for injury to people from external radiation. Table 9-6 lists the damage criteria selected for ingestion of contaminated crops.

9.4 Areas and Distances Involved

Calculations have been made of the areas and distances involved on the release of fission products resulting from the maximum credible accidents using the assumptions already tabulated.

The inventory of material released to atmosphere as functions of time after the shutdown is shown in Table 9-7.

The consequences of the release, assuming a one day decay time, is shown in Table 9-8 for wind speeds of 1, 5, and 10 meters per second.

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TABLE 9-5

DAMAGE CRITERIA FOR INJURY TO PEOPLE

Direct Effects (Dose Resulting During Cloud Passage)

<u>Zone</u>	<u>Dose</u>	<u>Effects or Action Required</u>
A	$\geq$ 450 rems (whole body)	Possibly lethal
B	100 - 450 rems (whole body)	Injury requiring hospitalization to varying degrees
C	5 - 100 rems (whole body)	Require medical attention and examination.

Indirect Effects (Dose Resulting From Ground Deposition)

D	$\geq$ 25 rems (whole body) ⁽¹⁾	Long term evacuation Extensive decontamination.
E	0.5 rems - 25 rems (whole body)	Temporary evacuation slight to moderate decontamination.

(1) Lifetime does resulting from external exposure from material deposited on ground.

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TABLE 9-6

DAMAGE CRITERIA FOR INGESTION OF CONTAMINATED CROPS

<u>Zone</u>	<u>Radionuclide</u>	<u>Dose to Critical Organ</u>	<u>Action Required</u>
H ₁	Cs ¹³⁷	≥ 50 rems (whole body)	Confiscation of several crops including milk supplies.
H ₂	Sr ⁹⁰	≥ 50 rems (bone)	Confiscation of several crops including milk supplies.
I ₁	Cs ¹³⁷	5 - 50 rems (whole body)	Confiscation of current crops plus milk supplies.
I ₂	Sr ⁹⁰	5 - 50 rems (bone)	Confiscation of current crops plus milk supplies.
J	I ¹³¹	≥ 250 rems (2 g thyroid)	Confiscation of milk supplies and some leafy vegetables.
K	I ¹³¹	25 to 250 rems (2 g thyroid)	Restricted use of milk supplies and some leafy vegetables.

If one assumes a constant wind speed, the maximum distances involved can be calculated and are shown in Table 9-9.

Another set of calculations were made assuming 100 per cent meltdown of the reactor core but with the same plate out, fog spray and filter efficiencies and the same percentage escape of fission products from the molten uranium. Consequences are shown in Table 9-10.

TABLE 9-7

INVENTORY OF MATERIAL RELEASED TO ATMOSPHERE AFTER MAXIMUM CREDIBLE ACCIDENT⁽¹⁾

	TIME AFTER SHUTDOWN								
	0	10 Min.	30 Min.	1 Hour	3 Hours	10 Hours	1 Day	10 Days	30 Days
Noble Gases	4.48×10^8	2.31×10^8	1.96×10^8	1.75×10^8	1.51×10^8	1.17×10^8	8.92×10^7	2.31×10^7	1.78×10^7
Halogens	2.58×10^6	1.44×10^6	1.37×10^6	1.29×10^6	1.08×10^6	7.92×10^5	5.58×10^5	9.65×10^4	1.36×10^4
Vol. Solids	1.68×10^4	9.25×10^3	7.20×10^3	5.68×10^3	3.35×10^3	2.70×10^3	2.44×10^3	1.17×10^3	772
Non Vol. Solids	159.8	118.5	98.0	85.0	69.5	54	44	25.8	16.5
Fission Product	4.50×10^8	2.32×10^8	1.97×10^8	1.76×10^8	1.52×10^8	1.18×10^8	8.98×10^7	2.32×10^7	1.78×10^7
Np ²³⁹	607.5	605	600	580	535	450	31	0.16	
Pw ²³⁹	0.0060								
Pu ²⁴⁰	0.0030								
Pw ²⁴¹	0.245								

(1) Using assumptions tabulated in Sections 9.1 and 9.2, and assuming meltdown of 25 per cent of fuel loading.

TABLE 9-8

(2)

MAXIMUM AREAS AFFECTED BY ZONES AFTER MAXIMUM CREDIBLE ACCIDENT (PROVISIONAL)

	⁽¹⁾ Strong Inversion ($>T - 4$)			⁽¹⁾ Moderate Inversion ($>T - 1$)			⁽¹⁾ Neutral			⁽¹⁾ Unstable		
	<u>1</u>	<u>5</u>	<u>10</u>	<u>1</u>	<u>5</u>	<u>10</u>	<u>1</u>	<u>5</u>	<u>10</u>	<u>1</u>	<u>5</u>	<u>10</u>
A	-	-	-	-	-	-	-	-	-	-	-	-
B	-	-	-	2.3	-	-	.38	-	-	0.056	-	-
C	1640	-	-	420	58	-	12.5	3.3	1.8	1.8	0.4	0.215
F	-	-	-	6.0	-	-	0.7	.088	-	0.10	0.0088	-
G	15,000	-	-	3000	760	320	56	15.8	8.6	7.4	1.9	1.0

Height of Release - 70 Meters

Decay Time - 1 Day

(1) Columns refer to wind speeds of 1, 5, and 10 meters per second.

(2) Areas in square miles.

TABLE 9-9

DISTANCES AFFECTED FOLLOWING MAXIMUM CREDIBLE ACCIDENT ASSUMING
CONSTANT WIND VELOCITY OF FIVE METERS PER SECOND

<u>Zone</u>	(1) Stability Parameter <u>(>T₃₀₀₋₁₀₀)</u>		<u>Maximum Distance (Miles)</u>
	1	4	
A	1	4	-
B	1	4	93 200
C	1	4	170 2600
F	1	4	19 -
G	1	4	480 1200

(1) Refers to temperature differences, degrees F.

TABLE 9-10

CONSEQUENCES OF 100% FUEL MELTDOWN⁽¹⁾
(Using Same Plate Out, Fog Spray and Release Factors)

Zone	Strong Inversion		Moderate Inversion		Neutral		Unstable	
	Area (Mi ²)	Distance (Miles)	Area (Mi ²)	Distance (Miles)	Area (Mi ²)	Distance (Miles)	Area (Mi ²)	Distance (Miles)
A	14	26	2.6	9.8	0.086	1.34	0.018	0.58
B	130	86	22	31	0.42	3.11	0.088	1.3
C	8800	890	1,500	330	10	17.0	2.0	7.0
F	260	120	42	45	0.69	4.04	0.14	1.7
G	54,000	2500	8,600	910	43	36.0	8.2	15

(1) Assuming a wind-speed of five meters per second and release at ground level.

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10. DEVELOPMENT PROGRAMS

10.1 Thermal Hydraulics

Because of the importance of adequate cooling and over-all thermal design to the successful operation of the N Reactor, and its ability to safely respond to non-standard or accident conditions, an aggressive analytical and experimental development program is being pursued. The objective of this program is to confirm the core thermal design, to establish safe operating limits, and to investigate the consequences of non-standard or accident circumstances.

Typical of the items to be investigated in this program will be the following:

1. Boiling burnout data have been obtained in the laboratory with test sections representing each of the three flow channels in the NPR tube-in-tube fuel elements. These test sections include ones for the center hole, the middle annulus with two heated surfaces, and the outer annulus with one heated surface.

The application of these data to a full-length charge of NPR fuel elements will be verified by use of a 17-foot long test section containing all three flow channels. This full size section represents the downstream half of a charge of fuel elements and has a cosine heat generating distribution.

2. The hydraulic stability characteristics of single NPR coolant channels will be determined. Experience with reactor channels at low pressures has shown that when flow is decreased at constant tube power, the

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pressure drop increases due to boiling and the flow conditions become unstable. This situation is known to be less severe at higher pressures, but the exact conditions must be determined experimentally.

Pressure drop-flow relations will be investigated first for the fuel elements at constant tube power using an electrically heated test section in the laboratory. Second, two-phase pressure drop through NPR connectors and hydraulic fittings will be determined installing these components in the heat transfer apparatus. Next, these data will be combined and treated using analytical techniques for application to all possible reactor conditions.

3. The program to investigate the effects on heat transfer conditions of the non-coaxial positioning of NPR tube-in-tube fuel element components will be continued
4. The effects on fuel element temperatures of gross pressure and flow changes will be investigated in a series of transient heat transfer experiments.

Conditions to be simulated will include a complete rupture of a hydraulic jumper with a process channel operating at an initial power of 6000 KW.

5. The flow requirements to provide adequate cooling of the NPR fuel elements during reactor shutdown conditions will be determined.
6. During the investigation of the hydraulic stability characteristics of NPR coolant channels, considerable data will be obtained concerning the pressure drop of steam-water mixtures through pipe fittings.

Two-phase pressure drop is difficult to predict with accuracy from theoretical considerations and the validity of existing correlations for pipe fittings is not known. Therefore, these data will be analyzed in detail to extend the present knowledge of the pressure drop of two-phase flow.

10.2 Zircaloy Process Tubes

The use of Zircaloy-2 pressure tubing in the N Reactor represents an application of zirconium on a scale that is without precedent anywhere in the world. Within the limitations of existing experience, the safe and economical operation of the reactor is, first of all, protected by the relatively conservative design conditions imposed upon the tubes in terms of operating stress. Continued protection will be provided by the results of materials testing programs and key in-reactor monitoring of the pressure tubes.

The materials testing program, initiated in part prior to the design of the reactor, includes pre- and post-irradiation measurement and in-reactor testing of those material properties that may impose operating and service limits upon the pressure tubes, and the manner in which they may be affected by the operating environment. Pre-irradiation testing was undertaken to distinguish those material properties that define design and application limits, such as ultimate, impact, creep, and stress rupture strengths, ductility and total deformation to rupture, and corrosion resistance. In-reactor testing and post-irradiation measurements are being undertaken to determine the direction and magnitude in which these material properties are affected by reactor operating environment, such as temperature and pressure cycles and gradients, neutron irradiation and hydrogen pickup.

A substantial portion of this work will be performed upon actual N Reactor tube sections that have been removed from the reactor. The application of these data will be to provide service limits for the tubes.

While the materials testing program will provide information of the over-all effect of cumulative operating environments upon design and service limits of the tubes, it will not provide the necessary information regarding the rate of approach to these limits. The necessary data to determine the rate of approach to material limits will be obtained through an in-reactor monitoring program. Development has been initiated for methods for remote visual examination, detection and measurement of process tube deformation, and unusual corrosion and hydrogen pickup. With the exception of a direct, nondestructive method for the measurement of hydrogen pickup, development work is well advanced, and some prototypical equipment has been constructed and tested in the Plutonium Recycle Test Reactor. Closed circuit television has been adapted for visual examination; internal diameter and wall thickness measuring instruments have been developed for measuring process tube deformation; and unusual corrosion can be detected by effluent coolant monitoring the Zr-95 activity. Until a direct method has been developed, hydride detection will depend upon a combination of visual inspection, Zr-95 monitoring, and post-irradiation examination of monitor tubes selectively removed from the reactor.

10.3 Graphite

Irradiations of N Reactor core graphite are planned for the GETR. These irradiations will provide data not yet available on graphite contraction rates for core graphite after long exposure to N Reactor service conditions. The neutron flux spectrum and intensity at the GETR is such that the data so obtained will allow prediction of effects at N Reactor years in advance.

Further studies will be made of chemical reaction rates of graphite and CO, CO₂, O₂, and water vapor to determine reactor gas compositions compatible with long service life for both the zircaloy process tubes and the graphite moderator.

10.4 Systems Analysis

Although considerable work has been, and is being, done in the area of systems analysis and evaluation in connection with detailed design of the NPR, an extensive and continued program of integrated systems analysis and evaluation is contemplated. Currently, efforts are being directed toward the procurement of an analog simulator to augment existing analytical facilities and to facilitate integrated system analyses and operator training.

10.5 Physics

The physics research and development effort in support of N Reactor will continue to be directed toward establishing the necessary physics knowledge to assure that adequate control of the nuclear operation can be maintained at all times and to provide bases for startup procedures and process standards.

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An aggressive and continuing program of studies designed to evaluate the reactivity transients resulting from deviations from normal operating conditions and potential consequences will be pursued.

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N Reactor Process Development

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N Reactor Process Development

**DATE
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