PRELIMINARY SAFETY ANALYSIS OF THE SODIUM-DEUTERIUM REACTOR (SDR)

Compiled by
C. C. Beusman

March 6, 1959

Nuclear Development Corporation of America
White Plains, New York

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NDA 84-14

PRELIMINARY SAFETY ANALYSIS OF
THE SODIUM-DEUTERIUM REACTOR (SDR)

Project Engineer: E. Bernsohn

Compiled by: C. C. Beusman

March 6, 1959

Work Performed under Contract AT(30-3)-256
for the United States Atomic Energy Commission

NUCLEAR DEVELOPMENT CORPORATION OF AMERICA
White Plains, New York

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FOREWORD

The safety analysis presented here was performed concurrently with the preliminary design of the reactor. Only the safety of the reactor itself has been considered in this work. A complete preliminary hazards report will be presented when the design of a plant for construction at a specific site has been completed.

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1. ABSTRACT

A preliminary safety evaluation has been made of the Sodium-Deuterium Reactor (SDR), a sodium-cooled, D_2O -moderated reactor utilizing a fuel tube and calandria design concept. A study of mechanical equipment malfunction, fluid system failures, and operational accidents has shown that the reactor has sufficient inherent and design safety features to render it safe from major energy release hazards. The successful operation of an integrated sodium-water system in a mockup of the reactor has verified the feasibility of the separation of sodium and water under simulated reactor operating conditions. This report covers an examination of the reactor and its supporting systems. It does not cover the consequences of reactor incidents as they relate to the proposed site and its surroundings.

2. INTRODUCTION

This report presents a safety analysis of the 10 MW_e Sodium Deuterium Reactor (SDR), a sodium-cooled, D₂O-moderated reactor being designed by NDA for construction in Anchorage, Alaska as part of the Chugach Electric Association electrical grid network.

The SDR reactor is based on a fuel tube and calandria design concept.^{1,2,3} An aluminum calandria or tank containing the D₂O is pierced by vertical aluminum tubes regularly spaced in a lattice array. Located concentrically within each aluminum tube is a stainless steel fuel-coolant tube containing sodium and a cluster of fuel rods. Between the steel fuel-coolant tubes and the aluminum calandria tubes are aluminum barrier tubes which provide backup separation of the sodium and D₂O.

2.1 EARLY FEASIBILITY PROGRAM

Major emphasis during the early stages of the SDR program was placed on the feasibility of separation of the sodium and D₂O. Test programs, summarized in Appendix D, showed that the barrier concept does provide effective separation of the two fluids in the presence of leaks, and that the consequences of sodium-water reactions in the barrier channels were not severe. The following, broad, four-fold program was undertaken.

1. testing of the effectiveness of different barrier materials against hot sodium,
2. testing of the integrity of barrier tubes against separate and simultaneous sodium and water leaks, in addition to the effects of sodium leaks against calandria tubes, and water leaks against fuel-coolant tubes,
3. static and cyclic stressing of various welded joints proposed for the sodium system,
4. operation of a reactor mockup rig at reactor temperatures and pressures under both normal and aggravated operating conditions.

Conclusions drawn from the above test programs were as follows.

1. Aluminum in very thin sections (0.020 in.) is not penetrated by sodium at 1050°F, the maximum temperature in the reactor.⁴ Penetration of 0.060 in. coupons by a high velocity sodium jet required more than 3 hr exposure at a sodium temperature of 1100°F, and 15 min exposure at a sodium temperature of 1150°F.
2. The single aluminum barrier configuration can withstand simultaneous leaks of hot sodium and cold water on opposite sides with minimum distortion.^{7,8} An aluminum calandria tube backed by water withstood a simulated sodium leak with no distortion. A fuel tube containing hot sodium was not affected by a simulated water leak. Inadvertent sodium-water reactions in the gas spaces produced no failures of the test section or apparatus. No evidence was observed of shock waves or disruptive reactions.
3. Tube and header designs can withstand cyclic loadings up to 10,000 cycles at stresses greater than those encountered in the reactor.^{8,11}

4. The reactor mockup logged over 2700 hr of operation at reactor temperatures, indicating that sodium and water systems can be assembled and operated safely under simulated reactor conditions.¹⁰

2.2 SODIUM-D₂O DESIGN CRITERIA

Particular attention has been given to those areas in the reactor where sodium and D₂O are in close proximity. Design philosophy has been to insure safety by (1) emphasizing the integrity of the sodium and D₂O systems through conservative design, (2) providing for separation of sodium and D₂O in the event of a leak in either or both systems, (3) insuring that a single failure will not induce further failures, (4) providing reliable leak detection devices, and (5) incorporating features to contain the products of a sodium-D₂O mixing incident.⁶

2.3 SCOPE OF REPORT

The scope of the present Safety Report is given by the Contract: [AT(30-3)-256, Amendment No. 1, Appendix D, Section B] "Prepare a preliminary safety report on the 10 MW_e SDR core and that part of the system inside the primary shield in sufficient detail to provide the Hazards Evaluation Branch data to evaluate the safety aspects of the reactor core concept." In accordance with the above work statement, this report concentrates on the reactor and major systems which assure sodium-water safety, and emphasizes those design features that are considered unique in the SDR.

In the sections that follow, a description of the reactor is given, followed by detailed thermal and nuclear data for steady state full-power operation. Reactor operations are reviewed with emphasis on safety interlocks and procedures for startup, shutdown, and refueling. Safety precautions against power failure, equipment malfunction, and other emergency conditions are discussed. Finally, incidents caused by a nuclear excursion or a chemical reaction are discussed. The consequences of these incidents on the integrity of the reactor and containment of fission products are presented.

3. SUMMARY AND CONCLUSIONS

3.1 REACTOR ENGINEERING DESIGN

The SDR calandria containing the D_2O moderator is a cylindrical aluminum tank penetrated by 73 aluminum calandria tubes on a 10 in. triangular lattice array. Inside the calandria tubes are 54 fuel coolant tube-barrier tube assemblies, 13 shim-safety rod thimbles, and one regulating rod-thimble. Five calandria tubes are provided as spares for either fuel assemblies or control rods.

Each fuel position consists of two concentric tubes – an outer aluminum barrier tube and an inner stainless steel tube containing the UO_2 fuel element cluster and sodium coolant. Inert gas spaces between the fuel-coolant tube, the barrier tube, and the calandria tube provide thermal insulation between the hot sodium and the cold D_2O . The fuel element is a 19-rod bundle of stainless steel clad UO_2 pellets. (See Fig. 3.1.)

Control rods are hollow, aluminum-clad, cadmium cylinders with ball nut screw drive mechanisms which can be scrambled by releasing an electric clutch. An additional means of reactor shutdown for emergency use is provided by means of a fast drain of D_2O from the calandria.

Sodium enters the reactor at $650^\circ F$ through headers located in the lower header room, and is fed to the individual fuel-coolant tubes by small diameter pipes called "pigtailed." The coolant flows up the tubes (which penetrate the lower and upper neutron shields) and around the fuel elements in the active region of the core, where it is heated to an average exit temperature of $950^\circ F$. The sodium exits through "pigtailed" and collection headers, and flows to an intermediate heat exchanger where it transfers its heat to a secondary sodium loop before being returned to the reactor. The upper and lower "pigtailed" are designed to accommodate thermal expansions in the coolant tube and header system and to minimize sodium holdup volume.

The upper and lower neutron shields are steel shot-filled, organic-fluid cooled shields that attenuate neutron leakage from the core in order to reduce piping activation in the upper and lower header rooms. These shields also reduce the fission product gammas after shutdown to allow access to the header rooms for maintenance. Side cast-iron thermal shields protect the concrete biological shields from excessive gamma heating. Heat is removed from these shields by an organic coolant circulating through coils cast into the shields.

The vertical steel coolant tubes extend to the top of the upper header room up through a top gamma shield which covers the upper header room. These coolant tube extensions provide for refueling access from above the top gamma shield. The refueling extensions are filled with shielding plugs. During normal operation, the sodium level is maintained well below the shielding plugs by a pocket of trapped nitrogen.

The upper and lower header rooms are filled with dry nitrogen, normally stagnant during steady-state reactor operation. These two header rooms communicate through the inner barrier

gas space, and under emergency conditions, gas flow can be established between the lower header room and the upper header room. The header room gas system provides for shutdown cooling of the reactor and for preheating of sodium in the coolant tube pigtails and headers during reactor startup. The calandria (reactor) room is also flooded with nitrogen; this gas system is completely isolated from the header room gas system to prevent sodium vapor from entering the D₂O area in case of a leak.

The core contains 2.01 metric tons of dense UO₂, enriched to 3 weight % in U²³⁵. The multiplication constant in the hot, poisoned condition is 1.19, which provides enough reactivity for an average burnup of 7500 MW-d/metric ton of uranium. The reactor exhibits a fast negative fuel temperature coefficient. The overall plant coefficient is essentially zero, since the sodium exhibits a slow positive coefficient which compensates for the fast negative Doppler coefficient mentioned above.

3.2 SUMMARY OF SAFETY FEATURES

The safety of the SDR is based on inherent as well as design safety features which protect the reactor from a major uncontrolled energy release. These features are summarized briefly below.

3.2.1 Inherent Safety Features

The following features in the SDR greatly add to the safety and simplify the design problems.

1. D₂O Moderation

D₂O moderation results in a long prompt neutron lifetime and slow responses to reactivity disturbances, affording time for corrective action to be taken by the control system. In addition, an increase of 6.5% in the effective delayed neutron fraction due to photoneutron production increases the reactor time constant. Photoneutrons also provide a strong neutron source to facilitate restarts.

2. Low Pressure Systems

The SDR is a low pressure system of high thermal capacity. Low pressures in the reactor coolant and moderator minimize stored energy in the fluids, and therefore no large forces are present to magnify system failures. Sudden loss of coolant by flashing is impossible. The absence of large pressure-induced stresses allows the use of conventional materials well within design stress limits.

3.2.2 Design Safety Features

Coupled with the inherent safety features discussed above, design features have been introduced which insure safe operation of the plant during normal as well as emergency conditions. Foremost in this list are the following four design features which provide for sodium-water safety:

1. High Integrity Systems to Contain Sodium and D₂O

The choice of the calandria-fuel tube concept for the SDR allows a complete mechanical independence of the hot sodium and cold D₂O systems, with consequent reductions in thermal stress and alignment problems. The stainless steel sodium system is all-welded, with a minimum of critical welds. The design will incorporate proven sodium reactor technology (e.g., weld design, fabrication and inspection techniques, etc.) to insure leaktightness.^{12,13} The D₂O system design is based on proven technology developed by Savannah River, Chalk River, and other installations.

2. Barrier Concept to Insure Separation of Sodium and D₂O

Complete separation of sodium and D₂O results from the use of an independent metal barrier between the fluids. This mechanical barrier concept has been carried throughout the reactor core

and external systems. Failure of at least three separate, independent metal walls must occur before the fluids can mix.

3. Inert Gas Blanket to Prevent Combustion Effects

All rooms where sodium is present are flooded with dry nitrogen, maintained at less than 1% oxygen concentration, to prevent sodium combustion. In the unlikely event of a sodium-water reaction, which would generate hydrogen, the inert gas blanket also eliminates the hazard of an oxygen-hydrogen explosion. This 1% level for oxygen concentration is well below the 5% oxygen concentration required to support combustion of sodium or hydrogen.^{14,15}

4. Expansion and Venting Provisions to Avoid Pressure Surges

Adequate expansion volumes and vent devices insure that shock waves do not arise and that generated hydrogen can be relieved in case of a sodium-water reaction.^{16,17} Experience with experimental rigs during the SDR program has shown that these provisions are more than adequate. Sodium-water reactions have occurred in a simulated lattice position consisting of a calandria tube, barrier tube, and fuel-coolant tube with no damage to any components.

The following design features provide for operational safety and reduce the effect of equipment malfunctions:

5. Leak Detection to Signal Incipient System Failures

Two independent leak-detection systems incorporating liquid and vapor detectors are provided to check for sodium leakage. Each individual fuel channel is monitored with detectors for gross liquid leakage. In the event of a sodium leak, the reactor will be scrammed automatically. A similar leak-detection philosophy is used for the D₂O system, where both liquid and vapor detectors are used.

6. Spill-Return System to Return Sodium Leakage

The spill-return system, using catch pans and sump lines, can handle leakage from major sodium system failures and still maintain coolant flow past the fuel elements. By providing this backup system, the severity of any loss-of-coolant accident is reduced.

With the spill-return system, the nuclear and operational advantages of the tube-type reactor are preserved, but at the same time the safety of the tank-type reactor is achieved.

7. Shutdown Gas Cooling Systems to Remove Reactor Afterheat

An important feature of the SDR is that circulation of the blanket gases through the barrier annuli can be used as effective shutdown cooling to remove reactor afterheat.

In addition to these unique safety features in the SDR design, conventional safety rules have been followed. They include: (1) use of an emergency battery and diesel power supply to insure continuity of power supply during station and power grid outages, (2) use of two main pumps to maintain adequate cooling should either pump fail, (3) use of normal system interlocks and fail-safe electronic design to protect against instrument failures.

3.3 SAFETY ANALYSIS

An analysis of possible reactivity incidents has shown that no major reactivity excursions can occur because no mechanism exists whereby large excess reactivities can be introduced into the core.

The most severe nuclear excursion involves a cold, clean startup accident. Due to the favorable reactor kinetics of D_2O moderation, the total energy release is only 50 MW-sec. The high heat capacity of the fuel and the resultant small temperature rises will prevent damage to the reactor.

Total loss of sodium coolant could introduce about 2% excess reactivity, but no reasonable mechanism for rapid sodium loss from the core can be postulated. The spill-return system will handle very large sodium leaks and will maintain coolant in the reactor.

The maximum credible sodium-water accident involves failure of a fuel-coolant tube, subsequent failure of the barrier and calandria tube, and consequent mixing of the sodium and water in the calandria. Draining of the moderator serves to limit the amount of reaction. The total energy release would be $3. \times 10^6$ Btu, with subsequent venting of the reaction products into the containment building, producing an equilibrium pressure rise of ~4 psi. No destructive shock waves should arise within the calandria because of adequate expansion and venting provisions; therefore no propagation to other lattice positions is expected. The containment building can withstand this accident and no gross fission product release to the atmosphere would occur.

This analysis of mechanical equipment malfunction, fluid system failures, and operational accidents shows that the reactor has sufficient inherent and design safety features to render it safe from major energy release hazards.

Fig. 3.1 — Cross section of fueled lattice position through calandria

4. PLANT DESCRIPTION

4.1 REACTOR COMPONENTS

The overall arrangement of the reactor given in Fig. 4.1 shows the location of the main components in the reactor. The following sections describe each of these components in greater detail. Table 4.1 presents a summary of the reactor design data.

4.1.1 Calandria

The D₂O calandria (Fig. 4.2) is a right circular aluminum cylinder, 9.5 ft in diameter and 12 ft in height. The shell is $\frac{3}{4}$ in. thick. The top and bottom tube sheets are 3 in. thick and are welded to the cylindrical shell. The tube sheets are penetrated by 73 aluminum calandria tubes, 4.125 in. OD with 0.125 in. wall thickness. The calandria tubes are arranged on a triangular pitch with a 10 in. center-to-center spacing.

The bottom end of each tube is rolled into the bottom tube sheet, making a watertight aluminum-to-aluminum joint. The top end of each tube extends about 9 in. above the top tube sheet through a stainless steel bellows used to seal the tube to the tube sheet. Gastight joints are maintained with conventional soft metal "O" ring seals. The D₂O in the calandria is maintained at a level 3 in. below the tube sheet, allowing the bellows and seals to operate dry. The bellows provide for differential thermal expansion between the individual calandria tubes, as well as between the calandria tubes and the outer calandria shell; they are designed for the emergency condition that occurs after a fast D₂O drain with the reactor at operating temperature. Under normal steady state operation no bellows would be required, since D₂O cooling minimizes temperature differences and resulting thermal stresses.

Surrounding the cluster of calandria tubes is an internal aluminum flow baffle 8 ft in diameter. The baffle stands on the lower tube sheet. Eight cutout segments at the bottom of the baffle serve to distribute the moderator flow uniformly into the center of the core. The baffle serves as a chimney to establish natural convection, which produces moderator flow velocities four to five times greater than can be obtained by simple throughflow of the D₂O.

The D₂O flows into the calandria through four $3\frac{1}{2}$ -in. inlet nozzles fed from a "C" shaped inlet header. The nozzles discharge downward between the outer shell and baffle, forming a helical flow pattern to facilitate mixing of the cold incoming D₂O and the warm D₂O being recirculated by natural convection. The D₂O flows down the outer annulus, passes through the flow baffle cutouts near the bottom tube sheet, and flows up through the bundle of calandria tubes. About one-fifth of the total D₂O flow passes to the 4 in. outlet lines and hence to the external D₂O heat removal system. The outlet lines are welded to the inner baffle and the outer calandria shell, and serve to anchor the baffle. The remaining D₂O recirculates through the outer reflector annulus and returns to the central region.

Table 4.1 — Reactor Design Data

Core		Fuel-Coolant Tubes	
Number of lattice positions	73	Material	316 SS
Number of fuel elements	54	OD (within core), in.	2.875
Number of spare lattice positions	5	Thickness (within core), in.	0.035
Number of control rods	14	ID (within core), in.	2.805
Pitch spacing and type	10 in. triangular	Primary Coolant	
Equivalent core diameter	7 ft 6 in.	Material	sodium
Core height	9 ft 5 in.	Flow rate, gpm	~3600
Fuel Elements		Average core inlet temperature, °F	650
Type	19-rod cluster	Average core outlet temperature, °F	950
Fuel	UO ₂	Maximum core outlet temperature, °F	1050
Enrichment, %	~3	Minimum core outlet temperature, °F	884
Fuel pellet diameter, in.	0.360	Maximum allowable velocity, ft/sec	25
Density, g/cc	10.4	Calandria	
Maximum fuel temperature, °F	3800	Tube material	aluminum
Bond	helium	Tube OD, in.	4.125
Nominal bond thickness, in.	0.001	Tube thickness, in.	0.125
Cladding	304 SS	Tube ID, in.	3.875
Cladding thickness, in.	0.015	Shell material	aluminum
Overall rod diameter, in.	0.392	Shell ID	9 ft 6 in.
Pitch spacing, in.	0.540	Shell thickness, in.	0.75
Spacer method	helically wound wire	Height inside	11 ft 8 in.
Fuel length	9 ft 5 in.	Moderator depth	11 ft 5 in.
Barrier		Lower tube sheet thickness, in.	3
Gap thickness between barrier and fuel-coolant tube, in.	0.250	Upper tube sheet thickness, in.	3
Barrier material	aluminum	Thermal Shield	
Barrier OD, in.	3.505	Material	iron
Barrier thickness, in.	0.065	Thickness, in.	6.5
Gap thickness between barrier and calandria tube, in.	0.185	Cooling medium	organic
Moderator		Heat generated, Btu/hr	6.9×10^6
Material	D ₂ O	Biological Shield	
Pressure	slightly pressurized	Material of construction	reinforced concrete
Average D ₂ O temperature, °F	130	Upper shield material	
Weight of D ₂ O in reactor vessel, tons	25	Upper neutron shield thickness	4 ft 6 in.
Reflector		Lower neutron shield thickness	4 ft 6 in.
Material	D ₂ O (part of moderator)	Cooling medium	organic
Nominal thickness, in.	12	Gamma Shield	
		Gamma shield material	steel balls and organic
		Gamma shield thickness, ft	3
		Cooling medium	organic

Normal D_2O level is maintained by an overflow weir attached around one-quarter of the calandria shell. The overflow returns to the main D_2O system via a 4 in. pipe. The 3 in. gas space above the heavy water contains helium. A small bleed of helium is passed to a recombiner to keep the concentration of radiolytic D_2 at a low level. The helium lines are welded to the top tube sheet.

Provision for draining the calandria consists of a 12-in. dump line and dump valve welded to the outer calandria shell at the lower tube sheet. (The calandria will drain in about 40 sec after opening the dump valve.) A vent line from the calandria to the dump tank is provided to equalize pressure changes during a D_2O dump. As an additional precaution, the calandria is designed for 10 psi external pressure and can withstand 15 psi under emergency or short-time conditions.

The entire calandria assembly is supported by a cylindrical skirt which rests on the lower neutron shield. This support skirt also serves as a collection plenum for the calandria and control rod cooling system (discussed in Section 4.3.6).

4.1.2 Barrier Tube

The aluminum barrier tube (Fig. 4.3) is 3.505 in. OD, 0.065 in. thick, and approximately 37 ft long. It is centered in the calandria tube, with a 0.185 in. gap maintained between the calandria and barrier by sets of three spacer buttons welded to the barrier. The inner clearance between the barrier tube and the fuel-coolant tube is 0.250 in. All the barrier spaces are filled with dry nitrogen of low-oxygen content at essentially atmospheric pressure.

The aluminum barrier hangs from the top of the upper neutron shield and extends through the bottom neutron shield. The barrier tube is stepped and tapered at appropriate locations to permit insertion and removal from the top face of the reactor; two of these steps are located in the upper and lower shield to reduce radiation streaming. At the upper end, the barrier is sealed to the upper neutron shield through a metal "O" ring seal. Additional gas seals are maintained at the lower faces of both the upper and lower neutron shields to prevent communication between the nitrogen in the calandria room and in the upper and lower header rooms. Below the calandria, a circumferential splash tube, welded to each barrier tube overhangs a lip in the D_2O catch pan, which forms the top of the lower neutron shield. The end of the barrier tube terminates above a sodium leak cup welded to the lower pigtail, which contains leak detectors (see Section 4.3.7).

The barrier tube serves two functions: (1) as a flow deflector in event of either separate or simultaneous leaks of D_2O or sodium, and (2) as a radiation shield to reduce thermal losses from the fuel tube to the calandria tube. The barrier gaps provide adequate thermal insulation while minimizing heat transport via free convection effects. The present design of the barrier tube has been tested successfully by the experimental program completed during the early part of the SDR program (see Appendix D).

4.1.3 Fuel-Coolant Tube

The fuel-coolant tubes (Fig. 4.3) are made of type 316 stainless steel and are about 44 ft in length and of varying thickness and diameter. Sodium flow is upward, entering the fuel-coolant tubes through butt-welded "pigtails" in the lower header room. After traversing the reactor, the sodium exits through pigtails welded to "T" joints in the upper portion of the fuel-coolant tubes.

In the lower header room, the fuel-coolant tubes are 1.75 in. OD with a 0.125-in. thick wall. Each tube flares midway through the lower neutron shield to accommodate an internal stainless steel shielding plug which prevents fast neutron streaming through the coolant tube; the plug is approximately 3 ft in length. Sodium passes up past the plug through three helical grooves which make a single full spiral over the length of the plug.

Before entering the lower tube sheet of the calandria, the fuel tube again flares, to 2.875 in. OD with a wall thickness of 0.035; it maintains these dimensions through the calandria.

In the upper neutron shield, the tube is stepped to prevent neutron streaming from the alignment gaps around the upper neutron shield plugs. An internal shield plug similar in design to the lower shield plug hangs within the fuel tube. This plug is an integral part of the fuel element assembly, which hangs suspended from a shoulder inside the fuel-coolant tube.

The fuel tube penetrates the top gamma shield through a series of stepped eccentric alignment sleeves and is sealed by a welded cap. This cap and the mechanical closure below it prevent leakage of the nitrogen cover gas trapped in the top of the coolant tube.

The fuel-coolant tube is hung from a clamp attached to the top face of the upper neutron shield. Rotation of this clamp frees the coolant tube, which can then be withdrawn through the top gamma shield for repair or replacement.

The central 25 ft portion of the fuel-coolant tube is manufactured from an integral piece of stainless steel tubing to assure leaktightness. Two welds are permitted: one in the vertical section of the pigtail in the lower header room, and one in the fuel tube in the upper header room. (The finished tubes will be subjected to ultrasonic and dye-penetrant tests to insure soundness of the final fuel-coolant tube assembly.)

4.1.4 Fuel Elements

The fuel element is a 19-rod cluster (Fig. 4.4) of 0.360 in. diameter UO_2 pellets, enriched to approximately 3 atom % U^{235} . The pellets are assembled into a type 304 stainless steel tube, with an OD of 0.392 in., a wall thickness of 0.015 in. and a length of ~10 ft. Helium fills the gap between the fuel and fuel element cladding. (The gap is nominally 0.001 in.) Helically wound spacer wires maintain a 0.540 in. pitch spacing and alignment of the rods over the fuel element length. The fuel rod cluster is held rigidly at the top by steel spacer plates.

The lower end of the fuel assembly terminates in an end fitting with radial extensions that assure centering in the fuel-coolant tube. The fuel assembly is free to expand at the lower end. Above the fuel element bundle is a top spacer and guide, terminating in a remotely operated coupling linking the fuel assembly to the upper neutron plug and top gamma plug. This permits separation of the upper plugs from the fuel element assembly during refueling.

Above the gamma plug is the fuel tube closure device that provides a mechanical gas seal between the fuel-coolant tube wall and the closure piece. This closure device prevents contamination of the sodium in the fuel tube during refueling, when the welded top cap is removed. The entire fuel element is suspended from this closure device, which rests in a shoulder in the fuel-coolant tube.

4.1.5 Reflector

Both the radial and end reflectors are D_2O , 12 in. thick. The D_2O level at the top is maintained constant by the overflow weir in the calandria. (The D_2O level is not used for normal reactor control, but by employing a fast drain it serves as a backup shutdown.)

4.1.6 Control Rods and Drives

The control rod thimbles occupy lattice positions within the calandria, and operate at low temperatures. There are 13 shim-safety rods, and a regulating rod for fine control of reactor power. The active portions of the rods are hollow cylinders, 3 in. in diameter and 10 ft in length, made of $\frac{1}{16}$ -in. thick cadmium tubing, clad on both sides with $\frac{1}{16}$ -in. thick aluminum. They are suspended in the control rod thimbles, with the annular space between being used for cooling with nitrogen. Estimated rod life is two years at full reactor power.

The control rod drives (Fig. 4.5) are mounted inside the top gamma shield, terminating in a cap at the reactor operating floor. Radiation effects in this location are relatively minor, and low operating temperatures are maintained by cooling with nitrogen introduced at the top of the gamma shield. This coolant flow passes down the thimble, past the drive mechanism and the active control rod region, discharging below the lower calandria tube sheet into the plenum formed by the calandria skirt.

The control rod drive mechanism (Fig. 4.5) is powered by an instantaneously reversible three-phase induction motor. Torque is transmitted through a multiple gear reduction unit and electric clutch. An electric brake prevents rod reversal when the motor is off. The control rod is attached to a ball nut which is constrained from rotating, and hence moves up and down as the ball screw rotates. Position indication is obtained by means of coarse and fine synchros, with two limit switches provided to prevent rod overtravel at either end. The maximum rod withdrawal speed corresponds to changing reactivity at an average rate of about $1.6 \times 10^{-4} \Delta k/\text{sec}$ when all rods are moved simultaneously.

Scram is accomplished by de-energizing the electric clutch, which frees the screw from the gear drive. The control rod and ball nut then drop, with the screw rotating as the nut and rod fall. Because the lead angle of the screw is very large ($> 50^\circ$), it offers very little resistance to the falling nut; however, a rotary spring on the screw is also provided which compensates for the slight retarding effect of the screw on the nut. A shock absorber utilizing the nitrogen coolant as a working fluid reduces the terminal velocity of the falling rod to less than 1 ft/sec.

To retrieve the rod, the clutch is reactivated. The usual driving-down operation is then not necessary. As the position indicators are geared to the screw, continuous position indication is provided even during the scram operation.

4.2 REACTOR STRUCTURE AND SHIELDING

The general structural arrangement, covered in this section, includes the permanent concrete biological shields, the upper and lower neutron shield, the top gamma shield, the side thermal shield, and the calandria support structure (Fig. 4.6). The entire load of the reactor is carried by the permanent concrete shielding.

4.2.1 Reactor Shields

Concrete biological shields surround the reactor room and the two header rooms. The reactor room is octagonal and is lined by $\frac{1}{8}$ -in. thick steel plate and a $6\frac{1}{2}$ -in. thick cast iron thermal shield. The header rooms are also lined with steel; to allow for the high operating temperatures they are provided with a 12-in. thick layer of insulation covered by stainless steel sheet. The biological shield supports the two neutron shield plugs and the top gamma shield.

All of the shielding is cooled by circulating oragnic coolant. The coolant reduces thermal stresses in the thermal shields and protects the concrete from excessive gamma heating and subsequent dehydration by maintaining its temperature below 180°F .

As a guide to the design of the main biological shields, dose rates for accessibility have been chosen, based on the following limitations.

1. Weekly average dose per worker must not exceed 100 mrem.
2. Maximum dose rate in any area at time of entry to the area must not exceed 200 mrem/hr.
3. Emergency dose levels during a failure on an overhaul should yield no more than 200 mrem (two weeks dose) during the time of exposure.

The upper and lower neutron shields consist of a 63 to 37 volume percent mixture of steel balls and Dowtherm A contained in a steel shell. Both shields are about 4 ft thick, and are pierced by 73 steel cylindrical shield tubes. The fuel-coolant tubes and control rod thimbles extend through these shield tubes. All penetrations are stepped to prevent radiation streaming. Both shields are supported on the circular concrete shoulders mentioned previously.

Internal cooling is provided in the neutron shields by a slow flow of organic through the steel ball array. The organic is introduced through a plenum distribution system and flows upward through the shields.

The top gamma shield is similar in design to the neutron shields; it consists of an iron-organic mixture in a cylindrical steel shell. The shield is 3 ft thick, with an OD of 16 ft at the top, being stepped to 14 ft at the bottom. The upper gamma shield is penetrated by the fuel-coolant tube refueling extensions, as well as the control rod thimbles and other process and monitor channels. Support is provided by the permanent side shielding.

4.2.2 Calandria Support

A conical aluminum skirt attached to the circumference of the lower calandria tube sheet supports the calandria on the eight steel pedestals mounted on the structural concrete. A shim support arrangement is used to insure leveling of the support skirt. Four 1 in. $\times$ 8 in. steel support beams are placed under the lower tube sheet to help carry the static load of the D₂O moderator and prevent excessive deflection of the tube sheet. These beams are welded to the support skirt.

The skirt also serves as an outlet collection plenum for both the calandria and control rod cooling gas systems. Nitrogen is forced down the outer barrier spaces into the skirt under certain emergency cooling conditions. In addition, the hot nitrogen from the control rod thimbles is discharged into the skirt.

4.3 REACTOR SYSTEMS

4.3.1 Primary Sodium System

The primary system flow sheet is shown in Fig. 4.7. The sodium flows in a one-loop system, entering the reactor at 650°F and leaving at a mixed mean outlet temperature of 950°F. After leaving the reactor, the sodium flows to two main sodium pumps, each capable of 50% of full flow. The sodium then flows to an intermediate heat exchanger, where it is cooled to 650°F before it is returned to the reactor. The normal sodium flow rate is about 3600 gpm; it is maintained by variable-speed, centrifugal pumps with discharge heads of 50 psi. The main sodium lines are 10 in., schedule 10 pipe. Backflow through a pump is blocked by a check valve and an automatic block valve in series.

Sodium purity is maintained by a cold trap bypass capable of handling ~0.3% of the coolant flow. Oxide concentration is maintained at 10 to 20 ppm, and is monitored by a plugging indicator in the cold trap bypass. The cold sodium leaving the cold trap system is dumped into the expansion tank provided on the reactor outlet line. The expansion tank is sized to accommodate thermal expansion of the sodium from fill temperature of ~250°F to operating temperature. A static head of 1 psig is maintained in the expansion space above the sodium by the nitrogen being used as the sodium cover gas. The total volume of sodium in the primary system is approximately 2000 gal, and the sodium expands 137 gal during startup. Under normal operation, a small bypass flow of sodium is fed through the expansion tank to prevent freezeup and to reheat the sodium discharged from the cold trap.

The system will operate as a constant-temperature plant, with power demand changes being followed by varying the flow proportionately. Process instrumentation is discussed in Section 4.4.1.

4.3.2 Spill-Return System

The function of the spill-return system is to maintain the sodium coolant in the fuel-coolant tubes in case of a sodium leak in the primary system piping. A reserve supply of sodium is located in the primary system expansion tank to account for holdup in the spill-return system. The spill-return system relies on catch pans and flashing sheets to assure proper drainage (see Section 7.6).

Leaking sodium is fed to a sump tank located in the sodium dump room, below the lower header room. The sump pump returns the sodium to the expansion tank, maintaining sufficient level in the expansion to prevent cavitation in the main pumps. In this way, the attractiveness of the tube-type reactor is combined with the safety of the tank-type reactor.

4.3.3 Secondary Sodium System

The secondary sodium system (Fig. 4.8) serves two important functions: (1) it separates the radioactive primary coolant from the steam system, preventing turbine contamination and confining radioactive sodium to the containment building, (2) it isolates the reactor from the power plant system, thereby insuring that the low pressure primary sodium system will not be exposed to the full steam pressure.

The secondary sodium enters the intermediate heat exchanger at 500°F and exits at 900°F. Normal flow rate is about 2500 gpm, with an estimated pressure drop of 50 psi. Since the static pressure is maintained at 50 psig, any leakage developing in the intermediate heat exchanger will be from the secondary to the primary system (which operates at a lower pressure throughout the exchanger).

The secondary system is similar in design to the primary system, being basically a one-loop system with two variable-speed centrifugal pumps. Bypass lines are used for the cold trap flow, and for the reheat flow to the expansion tank. Check valves and automatic block valves on the main pump lines prevent backflow if one pump is shut down. As in the primary system, the cover gas used in the expansion tank is purified nitrogen.

The intermediate heat exchanger is of a vertical, straight-tube, baffled-shell design, with primary sodium in the tubes and secondary sodium on the shell side (Fig. 4.7). Placing the primary sodium inside the tubes minimizes retention of activity in the heat exchanger when the exchanger is drained for maintenance.

Outside of the header room and reactor room the primary and secondary systems are preheated by induction heating. All the piping is sheathed with carbon steel, insulated, and wrapped by induction-heating coils. In locations where induction heating proves impractical, Calrod heaters are used.

4.3.4 D₂O System

The D₂O flow system diagram is presented in Fig. 4.9. The main D₂O circulation system removes the heat deposited in the moderator, while the purification system and recombiner maintain moderator purity. In addition, there is a fill and drain system used during reactor startup and shutdown procedures.

The main circulation system transfers about 5.9 MW of thermal energy to the organic cooling system; the flow rate in the main loop is 1210 gpm; D₂O enters the calandria at 110°F and exits at 137°F. A constant flow of 100 gpm from the overflow weir in the calandria is collected in an overflow and make-up tank and pumped back into the main D₂O line. The overflow and make-up tank absorbs D₂O volume increases caused by temperature increases and also provides for small amounts of D₂O makeup during operation. The cover gas for the D₂O system is helium.

After leaving the calandria, the D_2O flows directly to the circulating pump suction and is discharged through the D_2O heat exchanger, finally returning to the calandria. The entire system is fabricated from 3003 aluminum alloy, with the main lines being 8 in. schedule 40 pipe.

The purification system is fed by a 10 gpm bypass stream drawn off the main circulating loop downstream from the cooler. D_2O flows into a hold-up tank, sized to give a 4 min hold-up of D_2O to permit decay of N^{16} . After leaving the hold-up tank, the D_2O flows through a pre-filter into a cation exchanger, then to a mixed-bed exchanger. Effluent D_2O returns to the main system through an after-filter to remove any resin fines or solids picked up from the beds. Included in the purification system are lines to transfer the resin to waste disposal, as well as a deuterizing system.

A 12-in. dump line is connected to the calandria, flow being controlled by a fast-acting valve. The dump line discharges into a large dump tank, and a D_2O pump is used to transfer the D_2O to the make-up tank during moderator fill procedures. The maximum dump rate is about 7500 gpm, which permits complete drainage of the calandria in approximately 40 sec. Reactor shutdown can be effected by moderator dump within a few seconds after opening the valve.

4.3.5 Cover Gas Systems

D_2O Cover Gas System

The D_2O cover gas is helium, chosen for its chemical inertness and radiation stability. The helium provides surge and expansion volume in the D_2O system; it also provides an inert gas purge to remove D_2 generated by radiolytic decomposition of the D_2O .

The helium system (Fig. 4.9), consists of a supply manifold, a blower, and a catalytic D_2 recombiner. A normal purge rate of 2 SCFM is maintained through the calandria cover gas space to maintain a low D_2 concentration. The blower delivers this helium flow at a total discharge head of 25 in. of H_2O .

In accordance with the safety rules established for the reactor design, the main helium line is connected to a vent line through a rupture disc backed up by a pressure relief valve. This affords a safe and reliable vent arrangement to accommodate pressure surges that might arise from a sodium-water reaction in the calandria. In addition, there is a rupture disc on the gas space in the calandria above the D_2O .

The helium system connects to the D_2O overflow tank and to the D_2O dump tank. The vent line to the dump tank is an 8 in. pipe communicating directly from the dump tank to the calandria. In the event of a fast drain of D_2O , the helium in the dump tank will be displaced directly to the calandria, precluding any external pressure forces on the aluminum calandria.

Sodium Cover Gas System

Nitrogen is used as the sodium cover gas for both the primary and secondary sodium systems, providing pressurization (1 psig) to prevent pump cavitation as well as an expansion cushion in the event of a pressure surge. The sodium cover gas system (Fig. 4.10) is connected to the main nitrogen supply manifold. The manifold is supplied from a high-pressure storage tank filled with nitrogen which is purified as it passes through NaK bubblers.

Cover gas lines connect to the expansion tank, the dump tank, and the gas seals above the primary and shutdown pumps. Constant pressure to these components is maintained by pressure control valves on the supply line and the off-gas vent line. Nitrogen pockets are also present at the top of each fuel-coolant tube, giving additional expansion protection to the core.

4.3.6 Gas Cooling Systems

A calandria tube and control rod cooling system, and a shutdown gas cooling system, both employing purified nitrogen gas as a coolant, are also provided. These systems use the barrier gas spaces between the fuel-coolant tube and calandria tube. The functional requirements of each are discussed below.

Calandria Tube and Control Rod Cooling System

This system performs the dual function of cooling the control rods during normal operation, and cooling the control rods and calandria tubes during and after D₂O dump. During normal operation the nitrogen enters the control rods at a flow of 13,000 lb/hr through flexible lines from a 12-in. distribution header (see Fig. 4.11). During a D₂O drain, some of the flow leading to the control rods is diverted to the calandria room by the automatic block valve, via the pressure control valve on the 10 in. line leading to the calandria room. This arrangement maintains gas flow to the control rods at a reduced rate of 1300 lb/hr and flow to the calandria room at the rate of 11,700 lb/hr. In this manner, coolant flow to the calandria can start with essentially no time delay. The gas flow for calandria tube cooling is down the outer barrier space, discharging into the calandria support skirt, which also serves as a plenum chamber for control rod nitrogen flow. The gas leaves the calandria support skirt through a 12 in. gas line at a temperature of 300°F during normal operation (control rod cooling only) and 264°F during the scram operation. The inlet temperature of the gas is nominally 100°F.

The gas is circulated by a centrifugal compressor, with an additional compressor provided for standby service. Each has a capacity of 13,000 lb of gas per hour and a discharge pressure of 3.23 psig. The gas is cooled by the organic cooling system. The gas compressors are tied to the vital bus and are supplied by the emergency power system described in Section 7.2.1.

Shutdown Gas Cooling System

The shutdown gas cooling system is manually controlled and can be put into operation to remove all of the reactor afterheat if the fuel elements have decayed for 30 min, provided sodium is present in the fuel-coolant tubes to transfer the heat from the fuel to the gas. After 7 hr, this system can remove afterheat, even if no sodium is present in the fuel-coolant tubes, without exceeding a peak fuel element cladding temperature of 1600°F. This system is also used for preheating the reactor and associated primary sodium equipment rooms and galleries to 250°F during startup operation.

Cooling of the fuel-coolant and barrier tubes is achieved by circulating nitrogen through the inner barrier space. Nitrogen enters the lower header room at a flow of 24,000 lb/hr and temperature of 208°F through a 12 in. gas line (see Fig. 4.12). The gas flows up through the inner barrier space and exits from the upper header room through three 12-in. gas lines to a 24 in. manifold leading to an organic-cooled heat exchanger. The nitrogen is recirculated through the system by two centrifugal compressors operating in parallel, with a third compressor provided for standby service. Each of the three compressors is capable of delivering 12,000 lb of gas per hour at a pressure of 2.61 psig. Startup time for this system is a few minutes.

When the system is used for preheating the core and header rooms during startup, the nitrogen is heated to ~310°F in the heat exchanger by hot organic which has been heated by an external steam supply. The rate and direction of flow are the same as described previously. The time required for preheating the core and header rooms from 65°F to 250°F, with no sodium present, is about 1 hr; if sodium is present, however, a 10 hr heating time is required.

4.3.7 Leak Detection Systems

For both sodium and D₂O, liquid and vapor systems of leak detection are specified. As a safety feature, any verified leak signal from the fuel tube section of the liquid detection system calls for an immediate reactor scram.

Sodium Leak Detection System

Sodium leak detectors are located on the fuel-coolant tubes, in the sodium equipment room and upper and lower header room catch pans, in the expansion tank, and in the shutdown gas cooling system. The locations and circuits are indicated schematically in Fig. 4.14.

The liquid detection devices on the fuel-coolant tubes and on the catch pans are J-probes. Each probe consists of an Inconel sheath filled with aluminum oxide insulation and containing twin conductors welded to the tip of the sheath. The assembly closely resembles a sheathed thermocouple. A constant current is fed through the twin conductors to ground. When the cup containing the probe is filled with sodium, the voltage drop to ground suddenly decreases as the level reaches the probe and then exhibits a further gradual decrease as the level continues to rise. The voltage drop is measured by appropriate circuitry connected to the reactor safety system. The circuit can also distinguish between a genuine leak and most types of circuit defects.

The J-probes installed on each fuel-coolant tube are mounted in triplicate in cups attached below the weld connecting the fuel-coolant tube to the lower pigtail. When the presence of sodium is detected by two of the three probes in a cup, the reactor is scrammed.

The J-probes located in the catch pans are installed in pairs at appropriate points. Reactor rundown circuits are actuated when leaks are detected in the upper and lower header room catch pans. When a leak is detected in the sodium equipment room catch pan, an alarm is actuated; reactor power is changed manually since this situation is considered to be less serious. As a backup, reactor rundown circuits are actuated when the expansion tank liquid level indicator notes low sodium level.

The presence of sodium vapor in either header room is detected by bleeding samples of header room gas through the shutdown gas cooling system ductwork to conductivity cells or other appropriate devices. If the presence of sodium vapor is detected, alarm circuitry is actuated. The reactor is not scrammed until the leak is verified by a liquid detector.

The system provided also permits identification of a particular defective fuel tube. Identification of a particular defective pigtail requires the use of either direct or remote inspection methods.

D₂O Leak Detection System

D₂O leaks are detected by devices located in the D₂O system overflow tank, on the lower tube sheet, in the calandria and D₂O equipment room catch pans, and in the calandria tube gas cooling system. (See Fig. 4.14.) Capacitance probes are used to detect liquid D₂O at the rolled tube-tube sheet joint and in the D₂O catch pans; these probes are installed in duplicate at each point. An alarm light and an annunciator are used to signal the leak. Appropriate circuitry is provided to record the progress of a leak at a particular point. Coincidence circuitry is not required. The presence of D₂O vapor is detected by dew cells located in the calandria tube gas cooling system; they give anticipatory and back-up indication of a leak, and are not connected to scram or rundown circuits.

Major leaks are detected by the liquid level gage in the calandria overflow tank, where a change in liquid level of $\frac{1}{4}$ in. corresponds to a 1 gal change in system volume. A low level alarm contact is provided, together with rundown circuitry. D₂O is dumped when a major leak is detected.

4.3.8 Organic Cooling System

The organic cooling system is used as a heat sink for the neutron shields, thermal shields, and major fluid systems. Dowtherm-A has been chosen as the working fluid because of its known technology as a low pressure high temperature coolant. In addition, it has been shown to be completely inert to sodium at SDR conditions. (See Appendix A.)

The process flow diagram for the organic cooling system is shown in Fig. 4.13. Heat is discarded to plant cooling water.

4.4 NUCLEAR AND PROCESS INSTRUMENTATION

4.4.1 Nuclear Instrumentation

The nuclear instrumentation has been divided into three ranges to obtain adequate coverage of all flux levels with maximum sensitivity for all levels. These ranges are the startup range, the period range, and the power range. The relative neutron levels corresponding to each is shown in Fig. 4.15 and a block diagram of the instrumentation is given in Fig. 4.16. In describing the nuclear instrumentation, particular emphasis has been placed on the safety aspects of the schemes chosen.

Startup Range

The startup extends from source levels to about 10^{-6} of full power (10^{-12} to $10^{-5} \phi_p$, where ϕ_p is the average thermal neutron flux at full power). Because of the low flux levels, pulse-type instrumentation is used for maximum sensitivity. While only one channel is required for a signal, two detectors are needed so that during the repositioning of one chamber, a steady signal may be obtained from the other detector. In the startup range, period is the primary signal used for interlocking. Because period signals in this range are somewhat erratic, a short period signal will not cause a scram but will prevent further rod withdrawal. Effective short period protection is obtained by initiating a scram when both recorders are at full scale.

The instrumentation in the startup range consists of two fission detectors, each with three separate positions, covering three flux ranges. As the instrument approaches the upper limit of its range, the chamber is automatically repositioned by a drive motor to a less sensitive "intermediate" position. Thus, when the flux level reaches the period range, the chamber is withdrawn to the position of minimum sensitivity.

Period Range

The period range extends from 10^{-6} to $10^{-1} \phi_p$. Neutron detection in the period range uses two log N-period channels, with compensated ionization chamber detectors. Each chamber has its own regulated power supply to provide a fixed positive voltage and an adjustable negative compensating voltage. It may be necessary to provide lead shielding around these chambers if the level of gamma radiation is found to be too high. Since the primary safety signal in this range is the period signal, duplicate channels are provided for reliability. A scram can be initiated by a short period signal from either channel or by instrument failure of either channel. Scram signals from the period recorders are fed directly to the safety amplifiers of the control rod drives.

Power Range

Above a flux of $10^{-2} \phi_p$ reactor safety is assured by signals from the flux limit channels. Uncompensated ionization chambers serve as the neutron detectors and transmit d-c level signals to the safety amplifiers. Three independent channels are used. A high level signal from any one channel will cause a scram; loss of signal from two channels at the same time will also cause a scram. To avoid spurious scrams in the power range, coincidence circuitry may be desirable, so

that two high level signals are required to initiate a scram. Period signal scrams are not required in this range because of their general unreliability in the power range.

Servo Control

One additional independent instrument channel, the servo channel, is needed to furnish neutron flux level and period information to the reactor power level controller. (The operation of the power level controller is described in Section 6.3.) The servo channel utilizes a compensated ionization chamber and circuitry similar to that used for the power level channels.

4.4.2 Process Instrumentation

The major elements of process system instrumentation are shown in Figs. 4.7 to 4.13, which cover the major fluid systems. In the normal operating range, the reactor will be operated as a fixed temperature-variable flow plant requiring reactor inlet and outlet temperature signals as well as primary and secondary flow signals. The program relies strongly on automatic control, as discussed in Section 6.

Fixed temperature operation permits the use of high temperature inlet and outlet signals as a primary safety interlock. Reactor scram is provided by an abnormal temperature surge on either the inlet side of the primary sodium system or the exit of the fuel-coolant tubes.

Level detectors are provided in the sodium expansion tank, the D₂O surge tank, and the overflow tank fed from the calandria weir. These are used primarily as backup leak detection, and indicate gross spills in the reactor.

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0 1 2 3 4 5 6
SCALE IN INCHES

Detail 2
TUBE SHEET COOLING PASSAGES

Section A-A
WEIR OUTLET AND
MAIN OUTLET HEADER

- NOTES:
1. GENERAL
VESSEL SHALL BE FABRICATED AND TESTED IN ACCORDANCE WITH THE ASME CODE FOR UNFIRED PRESSURE VESSELS, 1956, AND NDA SPECIFICATION 10189
 2. MATERIALS
SHELL-ALUMINUM ALLOY 5052.0 (ASTM B178 GR20A.0)
TUBE SHEET ALUMINUM ALLOY 5052.0 (ASTM B178 GR20A.0)
PIPE-ALUMINUM ALLOY 6061-T6 (ASTM B274 GS11A.T6)
TUBES-ALUMINUM ALLOY 5052.0 (ASTM B210 GR20A.0)
ROLTS-ALUMINUM ALLOY 2014-T6 (ASTM B211 CS41A.T6)
 3. DESIGN CONDITIONS
PRESSURE-10 PSI INTERNAL AND EXTERNAL
TEMPERATURE-200°F
CORROSION ALLOWANCE-0

4. WELDING
WELDING OF ALUMINUM SHALL BE DONE BY THE INERT GAS SHIELDED ARC PROCESS
5. STRESS RELIEVING
NOT REQUIRED
6. RADIOGRAPHING
REQUIRED
7. TESTING
THE VESSEL SHALL BE SHOP ASSEMBLED IN THE VERTICAL POSITION AND HYDROSTATICALLY TESTED TO 15 PSI
8. FIELD WELDS
WELDS MARKED F ARE TO BE MADE IN THE FIELD FOR INSTALLATION OF INLET AND OUTLET HEADERS.

Fig. 4.2 — Calandria

Fig. 4.3 — Fueled lattice position — side section. Note: parts and assembly positions shown at average operating temperatures.

Fig. 4.4 — Fuel element arrangement layout

Fig. 4.5 — Control rod drive mechanism layout

Fig. 4.6 — Lower header room mockup

NEG No 1070

Fig. 4.7 — Primary sodium system process flow diagram

Fig. 4.8 — Secondary sodium system process flow diagram

Fig. 4.9 — D₂O system

Fig. 4.10 — Nitrogen cover gas system

Fig. 4.11 — Calandria tube and control rod gas cooling system

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Fig. 4.12 — Shutdown gas cooling system

Fig. 4.13 — Organic cooling system

Fig 4.14 — Leak detection schematic
See following page for additional details

Fig. 4.14 -- (Continued)

- Type 1 – count rate channel - fission detector
 Type 2 – count rate channel - fission detector
 Type 3 – period channel - compensated ion chamber
 Type 4 – period channel - compensated ion chamber
 Type 5 – flux level - uncompensated ion chamber
 Type 6 – flux level - uncompensated ion chamber
 Type 7 – flux level - uncompensated ion chamber
 Type 8 – servo flux control - compensated ion chamber

Fig. 4.15 — Ranges of nuclear instrumentation

Fig. 4.16 — Nuclear instrumentation block diagram

5. REACTOR DESIGN DATA

5.1 NUCLEAR CHARACTERISTICS

5.1.1 Steady-State Characteristics, Criticality Calculations

Detailed criticality calculations were made to determine the enrichment required to meet a target burnup of 7500 MW-d/metric ton. The number and location of control rods, and the nuclear properties affecting kinetics and safety, were also studied.

In order to obtain properly weighted fluxes, the reactor was divided into seven regions as shown in Fig. 5.1. Region I contained the 13 control rod lattice cells arranged in a symmetrical pattern, with 39 fuel cells. Region II contained the remaining 15 fuel cells. Region III contained the six spare lattice cells. Region IV was the D₂O reflector between the spare lattice cells and the aluminum flow baffle. Region V was the baffle, and Region VI was the remaining D₂O reflector. Finally, Region VII was the aluminum calandria shell.

In the fuel regions, appropriate infinite lattice calculations were made to obtain the fast and slow flux plots with and without control rods. These plots were in turn used to obtain flux volume weighted cross sections for the criticality calculations. In the void or non-fueled positions, weighted cross sections were computed assuming a flat flux across the cell.

The criticality calculations were performed using a two-group, one-dimensional cylindrical geometry code written for the Datatron digital computer. The finite height of the reactor was taken into account by including a leakage term (DB_2^2) in the absorption. Values of τ and L^2 were corrected for the presence of gaps between the calandria tube and the coolant tube, using Behrens' formula. Pertinent nuclear parameters calculated from these results are given in Table 5.1. For the cold, clean reactor with 3% enrichment, k_{eff} was 1.19; with all rods inserted, k_{eff} was reduced to 0.93. This shutdown value for k is sufficiently small to insure that at least one dollar of negative reactivity can be inserted even if all the sodium is drained from the core or if two maximum worth rods fail to drop on a scram. The worth of all the sodium in the core is 0.02 in k , as discussed in the next section.

Worths of Control Rods, Coolant, and Moderator Control Rod Worths

From the results of the control rod calculations it was concluded that 13 control rods placed as shown in Fig. 5.1 would be sufficient to control the reactor with 3% enrichment. The control rods have a total worth of 26% and have to control 19% excess reactivity. Should measurements made during the initial testing period indicate that additional control is necessary, 3% to 5% in k can be gained by inserting additional control rods in the six spare lattice positions.

Calculations were run to determine the worth of the various control rods acting in three banks. Use of programmed control rods in banks affords operational simplicity. Bank 1 was assumed to be the central rod by itself, Bank 2 was the inner ring of six control rods, and Bank 3 was the outermost ring of six control rods. The value obtained for k_{eff} for the various positions of the rod banks

Table 5.1 — Nuclear Data

Fuel Loading		
UO ₂ in core	2.01 metric tons	
U in core (3% enriched)	1.77 metric tons	
U ²³⁵	53 kg	
Volume Fractions of Typical Fuel Cell		Thermal Utilization in Fuel Cell
UO ₂	0.0223	0.7709
SS cladding	0.0043	0.0762
Na	0.0447	0.0350
SS coolant tube	0.0036	0.0810
Void	0.0571	
Al barrier tube	0.0041	0.0052
Al calandria tube	0.0181	0.0232
D ₂ O	0.8457	0.0085
		<u>1.0000</u>
Reactor Characteristics		
ϵ	1.010	
η	1.832	
ρ	0.964	
f	0.753	
τ	150 cm ²	
L^2	200 cm ²	
B^2	0.00037	
k_{∞}	1.34	
k_{eff} , cold, clean	1.19	
$\Delta k_{\text{Xe, Sm, temp.}}$	0.04	
k_{eff} , hot poisoned	1.15	
$\bar{\phi}$ thermal	2.5×10^{13} n/cm ² -sec	
ϕ max/ ϕ avg	2.2	
$\bar{l}$	6.6×10^{-4} sec	
Average Burnup	7500 MW-d/metric tons U	

is given in Table 5.2. Using these results the average worth of individual rods, neglecting shielding effects of rods in the same bank, was obtained. These results are also given in Table 5.2.

It can be seen that the maximum worth of two rods stuck out would be about 0.051 in k (corresponding to the central rod and one rod in Bank 2 stuck). Thus the reactor can still be brought below one dollar of negative reactivity in case the two most important rods fail to move. For this case, the backup drain of moderator affords an added measure of safety.

A typical radial flux plot for the reactor at start of life with all rods inserted is shown in Fig. 5.2.

Table 5.2 — Effect of Control Rod Banks on Reactivity

Position of Rod Bank				
Bank 1	Bank 2	Bank 3	k_{eff}	Δk
out	out	out	1.191	
in	out	out	1.156	0.035
out	in	out	1.040	0.151
out	out	in	1.109	0.082
in	in	out	1.025	0.166
in	out	in	1.061	0.130
out	in	in	0.955	0.236
in	in	in	0.930	0.261

Based on the above, the average worth of individual rods under various conditions is as given below.

Position of Other Rod Banks			Average Worth of Individual Rod, Δk	
Bank 1	Bank 2	Bank 3		
	out	out	0.035	Worth of Central Rod (Bank 1)
	in	out	0.015	
	out	in	0.048	
	in	in	0.025	
out		out	0.025	Average Worth of Individual Rod in Bank 2
in		out	0.022	
out		in	0.026	
in		in	0.022	
out	out		0.014	Average Worth of Individual Rod in Bank 3
in	out		0.016	
out	in		0.014	
in	in		0.016	

Coolant Worth

The worth of the sodium in the core as a poison was calculated as 0.04 in Δk . This value accounts for the thermal utilization of the sodium in the lattice cell. A Behrens¹⁸ neutron streaming correction for voids created by draining all the sodium in the fuel-coolant tube was made on the typical fuel lattice. The void was assumed to be a cylindrical hole passing through the geometric center of the homogenized fuel lattice. The corresponding increase in migration area increased leakage, thereby reducing k_{eff} by 0.02. Thus the total worth of the sodium in the core is 0.02 Δk .

Moderator Worth

The effect of moderator level on reactivity was calculated, using the expression for vertical buckling:

$$B_z^2 = \left(\frac{\pi}{H} \right)^2$$

where H = moderator height, including reflector savings at top and bottom.

For a thin D_2O reflector, reflector savings equals reflector thickness. The vertical non-leakage probability, P_z , is given by:

$$P_z = \frac{1}{1 + M^2 B_z^2} = \frac{H^2}{H^2 + M^2 \pi^2}.$$

The change in k therefore becomes

$$\frac{dk}{k} = \frac{dP_z}{P_z} = \frac{2M^2 \pi^2 dH}{H(H^2 + M^2 \pi^2)}.$$

Using an M^2 of 350 cm^2 , the top reflector is found to be worth 0.0042 in Δk . The reflector can drain in about 3 sec.

Burnup Calculations

The average thermal flux for operation at 46 MW nuclear power was calculated to be 2.5×10^{13} $\text{n/cm}^2\text{-sec}$. The excess k required for equilibrium Xe and Sm and temperature effects was estimated by conventional methods at 0.04. At the low flux level for this reactor peak, xenon poisoning after shutdown will be about 0.015 above equilibrium poisoning.

The burnup obtainable for a batch refueling scheme was calculated using a simplified model of the core. For this calculation it was assumed that the previously discussed gross flux pattern in the core remained constant with burnup (i.e., the effect of rod motion on the flux was ignored although the change in thermal disadvantage factor for the fuel element was taken into account) and that ϵ and p remain unchanged with time. The decrease in k with buildup of fission products and higher isotopes was computed in successive time steps by computing the change in η , f , and M^2 . At the end of life, corresponding to an average value of $k_{\text{eff}} = 0.04$ (for Xe, Sm, and temperature effects), the average burnup was calculated to be 7500 MW-d/metric ton of uranium.

Other Nuclear Data

The average prompt neutron lifetime was calculated using the perturbation in reactivity of the critical reactor caused by the introduction of a uniform $1/v$ absorber. The value obtained was $\ell = 6.6 \times 10^{-4}$ sec. (See Appendix C for a more detailed discussion.)

Fig. 5.3 shows the radial flux profile through the 19-rod cluster fuel element using a one-group calculation. Thermal neutrons were assumed to be born only in the moderator, i.e., slowing down in materials other than D₂O was neglected. The results indicate that the average flux in the central rod is 70% of that in an outside rod and that the average flux in a rod in the inside ring is 78.4% of that in an outside rod. Using these values the ratio of peak to average power production within the fuel cluster was calculated to be 1.09.

5.1.2 Reactivity Coefficients

The core temperature changes which have the largest effects on reactivity are changes in the sodium coolant temperature and changes in the UO₂ fuel temperature. Changes in the D₂O temperature are less important since the D₂O does not change through as large a temperature range. The overall coefficient for the reactor is essentially zero, with a prompt negative coefficient due to Doppler broadening in the fuel. Table 5.3 summarizes the coefficients. Appendix C contains more details of the calculational methods used.

Table 5.3 — Temperature Coefficients of Reactivity

Doppler Effect in the Fuel	$-2.2 \times 10^{-6} \Delta k/^{\circ}\text{F}$
Sodium	$+5.0 \times 10^{-6} \Delta k/^{\circ}\text{F}$
D ₂ O	$-3.78 \times 10^{-5} \Delta k/^{\circ}\text{F}$
Miscellaneous Expansion Effects	
Calandria	$+1.1 \times 10^{-6} \Delta k/^{\circ}\text{F}$
Fuel hanger rod	$+3.3 \times 10^{-6} \Delta k/^{\circ}\text{F}$

Sodium Coefficient

The temperature coefficient of the sodium was calculated by computing the fractional decrease in the thermal utilization of the sodium due to a fractional change in the sodium density. The calculated value is $5.5 \times 10^{-6}/^{\circ}\text{F}$. Since the fuel-coolant tube and fuel rod cladding will essentially be at the sodium temperature, there is an additional effect due to the change in sodium flow area with temperature. The increased volume of sodium in the core leads to a negative temperature coefficient of $-0.5 \times 10^{-6}/^{\circ}\text{F}$. The effective sodium temperature coefficient is therefore reduced to $+5.0 \times 10^{-6}/^{\circ}\text{F}$. This coefficient has a relatively slow action on reactivity, based on the large heat capacity of the fuel elements, and the time required for heat to flow to the sodium from the fuel elements. For a given power change the sodium temperature change will lag the fuel temperature change and be considerably smaller in magnitude.

An increase in temperature of the sodium will also cause the fuel hanger rods to expand. Since the rods are supported from the top and hang freely, this expansion is uncompensated over about a 13 ft length. The effect of the expansion will be to move the fuel elements downward with respect to the control rods. Using an average value for the reactivity worth of control rods, the expansion effect was calculated to give a positive coefficient of $+3.3 \times 10^{-6}/^{\circ}\text{F}$. There will be a longer delay time associated with the coefficient, corresponding to the added time lag between the fuel hanger rod temperature and the sodium temperature.

Fuel Coefficient

Neglecting the effect of temperature on the density of the fuel, an increase in fuel temperature will cause a decrease in reactivity because Doppler broadening of the U²³⁸ resonances decreases the resonance escape probability. The calculated value for this effect is $-2.2 \times 10^{-6}/^{\circ}\text{F}$.

The effect of fuel expansion on leakage was calculated by considering the change in geometric buckling caused by a change in size of the core, assuming constant reflector savings. This effect gives a positive reactivity coefficient but it is negligible (order of magnitude, $10^{-7}/^{\circ}\text{F}$), compared to the Doppler temperature coefficient of the fuel.

D₂O Coefficient

The effect of an increase in D₂O temperature is to decrease D₂O density and to raise the average energy of thermal neutrons in the reactor. Two problems based on criticality calculations were run with D₂O temperatures of 68°F and 135°F; the average temperature coefficient of reactivity was $-3.8 \times 10^{-5}/^{\circ}\text{F}$. The large thermal inertia of the D₂O gives rise to a long time constant of 4 min, and this coefficient has essentially no effect on reactivity excursions.

Miscellaneous Geometric Coefficients

The effect of an expansion of the calandria diameter due to an increase in D₂O temperature was calculated from the change in the radial geometric buckling due to the increase in the radial reflector savings. This effect was found to give a positive coefficient equal to $+1.1 \times 10^{-6}/^{\circ}\text{F}$.

Another reactivity coefficient is caused by lengthening of the control rods due to increased reactor flux or power. Since coolant flow rate to the control rods is constant and the heat generation is proportional to flux, the operating temperature of the rods will increase with increasing power. The rods are supported from above the core and thus will expand into the core. The amount of expansion will also be proportional to the distance the rods are in the core. The reactivity coefficient due to this effect will have a maximum value at the beginning of a core loading and decrease to zero at the end of life when the rods are fully withdrawn. The effect of this coefficient is small and can be neglected.

5.2 THERMAL AND FLOW CHARACTERISTICS AT FULL POWER OPERATION

The following discussion covers the steady state thermal and flow characteristics of the major reactor components at full power operation, including the fuel elements, pigtail piping systems, fuel-coolant tubes, barriers, and calandria. The performance of these components under transient conditions is discussed in Section 7.3.6. Table 5.4 summarizes the pertinent hydraulic and flow data.

5.2.1 Fuel Element Thermal Design

Thermal performance of the SDR fuel elements was calculated using a mean thermal conductivity of 0.85 Btu/hr-ft-°F for the UO₂. This value is based on recent experiments¹⁹ and theoretical radiation damage considerations. The temperature distributions were based on the radial power distribution at the end of core life. This condition represents the most severe heat duty for the hottest element. The design power distribution and coolant flow distribution are shown in Fig. 5.4. The orificed flow distribution was selected to minimize the temperature difference between the hottest and coolest channels over the whole life of the core.

The radial temperature distribution at the core midplane for a centrally located fuel element is shown in Fig. 5.5. The peripheral rods in the 19-rod cluster run hotter than the central rods because of neutron shadowing effects within the cluster. Fig. 5.6 gives the corresponding axial temperature distribution along the hottest fuel rods, showing a peak temperature of ~3800°F, which is below the chosen maximum allowable temperature of 4000°F and substantially lower than the equilibrium melting temperature of 5000°F.

For the design of cladding thickness, a 30% release of insoluble fission gases was assumed during normal operation. This value is the maximum reported for high-density sintered UO₂ for

Table 5.4 — SDR Core Thermal Characteristics

Primary Sodium System

Average inlet temperature, °F	650
Average outlet temperature, °F	950
Peak outlet temperature, °F	1050
Total flow rate, lb/hr	1.49×10^6
Flow through lattice position	
Maximum, lb/hr	3.64×10^4
Average, lb/hr	2.76×10^4
Sodium velocity	($\rho = 53 \text{ lb/ft}^3$)
Maximum in core, ft/sec	7.0
Average in core, ft/sec	5.3
Inlet piping, ft/sec	13.2
Main headers, ft/sec	10.3
System pressure	
Outlet from pump, psig	44
Net positive suction pressure, psig	1

D₂O System

D ₂ O inlet temperature, °F	110
D ₂ O outlet temperature, °F	137
D ₂ O flow rate, gpm	1300
System pressure	
Outlet from pump, psig	37
Net positive suction pressure, psig	1

Reactor Power Data

Heat to D ₂ O, MW	5.9
Total heat to primary sodium, MW	40

Fuel Element

Number of fuel elements	54
Average heat flux at rod surface, Btu/hr-ft ²	137,500
Peak heat flux at surface of cladding, Btu/hr-ft ²	275,000
Peak cladding temperature, °F	1050
Peak central temperature, °F	3800
Average core heat transfer coefficient, Btu/hr-ft ² -°F	8600

Other Reactor Components

Peak barrier temperature, °F	770
Peak calandria tube temperature, °F	560
Peak coolant tube temperature, °F	1050

conditions of centerline pin melting. This release was used despite the fact that no centerline melting is anticipated in the SDR elements. By providing a free expansion volume (20% of the fueled volume) at the top of the rod, the maximum internal gas pressure at the end of life is 140 psia even for this pessimistic melting assumption. The composite effects of pressure, mechanical load, and thermal stress were estimated and a cladding thickness of 0.015 in. was specified.

5.2.2 Temperature Distribution in Major Components Header Room Piping and Fuel-Coolant Tubes

Because of the low heat losses to the surroundings and the high sodium heat transfer coefficient, the header room piping and fuel-coolant tube temperatures are essentially those of the contained sodium. The temperatures therefore increase from about 650°F below the core to an average of 950°F and a peak of 1050°F above the core.

Barrier Tube

The axial temperatures along the barrier tube corresponding to sodium temperatures in the hottest fuel tube are presented in Fig. 5.7. The most critical region is the 15 in. length extending from the D₂O level upward to the bellows flange. In this region barrier temperatures may be as high as 770°F, if pessimistic values of thermal emissivity are used. This temperature does not exceed the maximum allowable barrier temperature of 800°F. Alternate solutions to this design problem have been studied, involving reduction of barrier temperatures by reducing thermal emissivities with anodized surfaces, and consideration of a composite steel-aluminum barrier. The aluminum portion of the composite barrier would be inside the core, with the steel portion above the D₂O level.

Calandria

Under normal conditions, the highest temperature in the calandria tube sheets is 150°F. The calandria tubes are at a temperature of 130°F, except above the D₂O, where the temperature increases steadily as shown in Fig. 5.8. The tube bellows has a maximum temperature of 500°F, well within the allowable temperature for stainless steel bellows.

During normal operation, the outer shell of the calandria will be cooled by the D₂O and will be at 130°F. During a D₂O dump, the outer shell will be cooled by gas flowing through the reactor room. Similarly the bellows will be cooled by the calandria tube gas coolant and by conduction to the upper tube sheet.

5.2.3 Operating Stresses

A general analysis of the steady state operating stresses in the SDR has been carried out. The results indicate that critical components can be expected to perform satisfactorily. Transient stresses are discussed in Section 7.8.

Lower Sodium Piping System

Thermal stresses due to constrained thermal expansion were computed. In the lower pigtails, the stresses due to the expansion of the fuel-coolant tube and lower header system result in a total stress of 19,300 psi, including an allowance for maximum welded joint efficiencies. The allowable stress for type 316 SS was taken to be 24,000 psi, equal to the 0.2% offset yield strength at 650°F. The ASA code for piping allows much higher allowable stresses, taking into account the stress relaxation or "self springing" effect.

Fuel-Coolant Tube

Mechanical stresses in the fuel-coolant tube from pressure or other mechanical means are negligible. The maximum stress in the fuel-coolant tube is 1700 psi; it exists at the point of its

emergence from the lower neutron shield. Again, this is within the allowable stress for type 316 SS, namely, 24,000 psi at 650°F.

Upper Sodium Piping System

The upper pigtails were assumed to be constrained at both ends by the fuel-coolant tube and the sub-header. Thermal stresses arise from the constrained lateral expansion (room temperature to 950°F) and a normal load induced by the expansion of the header. The reduced efficiency of weld material and stress concentration factors arising from geometrical discontinuities were also considered. The maximum combined stress (maximum shear) was computed to be 74,000 psi, at the pigtail and fuel-coolant tube junction. The allowable shear stress (one-half the 0.2% offset yield strength) of type 316 SS at 950°F is 10,500 psi, giving a minimum safety factor of 1.4.

Test results from cyclic fatigue tests run during the early SDR program indicated that a design lifetime of 10,000 cycles would permit an allowable stress of 29,100 psi at 950°F, a normal stress corresponding approximately to a shear stress of 14,500 psi. Thus the calculated stress of 7400 psi would clearly be good for at least 10,000 cycles.

Barrier Tube

Thermal stresses are of importance in the barrier tube due to the relatively high operating temperatures. Longitudinal bending stresses from temperature gradients are the most important. There are smaller stresses arising from the pressure drop across the wall and the weight of the tube. The allowable stress in the tube (0.2% offset yield strength of 5050-H34 aluminum alloy) limits the steady state operating temperature of the barrier tube to 800°F. The peak temperature of 770°F noted previously lies within this limit.

Calandria

During steady state operation, thermal gradients and stresses are negligible. Differential thermal expansion between the calandria tubes and the tube sheets will be taken up in the bellows joints. The longitudinal bending stress in the calandria tubes due to the axial temperature gradient (see Fig. 5.8) is approximately 2900 psi, well below the design stress of 6000 psi. The maximum stress in the calandria shell, at the juncture of the shell and the lower tube sheet, is equivalent to 3600 psi, based on a design pressure of 10 psi. The maximum hydrostatic pressure from the moderator is approximately 5 psi. Allowable stress is 5700 psi, based on the ASME Unfired Pressure Vessel Code and for 6061-T6 aluminum alloy. The maximum stress in the lower tube sheet, including a stress concentration factor equal to 2.0, is 4400 psi. The thickness of the calandria shell is designed for an external pressure of 10 psig, based on the ASME code.

Fig. 5.1 — Schematic of reactor core as divided for reactivity calculations.

- Region I — 13 control rod-fuel lattice cells
- Region II — outer fuel cell area
- Region III — spare lattice cells
- Region IV — inner D₂O reflector
- Region V — aluminum flow baffle
- Region VI — outer D₂O reflector
- Region VII — aluminum calandria shell

Fig. 5.2 — Reactor radial flux plot — all control rods inserted

Fig. 5.3 — Relative fuel cell flux vs cell radius — 0.36-inch diameter fuel rods at 0.031 ev — 10 inch lattice spacing — 19 rod cluster

Fig. 5.4 — Radial power generation and coolant flow distribution at end of life.

— $W(r)/W_{avg}$; --- $Q(r)/Q_{avg}$.

$$\frac{Q(r)}{Q_{avg}} = \frac{\text{Power generation at radial station } r \text{ (end of life)}}{\text{Average power generation in core}}$$

$$\frac{W(r)}{W_{avg}} = \frac{\text{Coolant flow rate to radial station } r}{\text{Average coolant flow rate in core}}$$

Fig. 5.5 — Radial temperature distribution at core midplane through hottest fuel cluster (group 1 cluster). UO_2 fuel; coolant; cladding.

Fig. 5.6 — Axial steady state temperature distribution — group 1 peripheral rod with 0.001 in. He bond gap

Fig. 5.7 — Steady state axial temperature distribution along barrier tube — full power

Fig. 5.8 — SDR calandria tube — steady state axial temperature distribution above D₂O level — full power

6. NORMAL OPERATING PROCEDURES

In this section of the report, the procedures to be employed in starting up, operating, shutting down, maintaining, and refueling are related to reactor safety.

6.1 NORMAL REACTOR STARTUP AND SHUTDOWN

6.1.1 Pre-Nuclear Checks

Reactor startup sets reactor process systems into operation, and includes the actual nuclear startup. These two phases are tied together when the plant is brought to operating temperature and the turbine is loaded. Steady state operation will be discussed only in sufficient detail to illustrate the SDR operating philosophy and the general safety interlocks.

Prior to reactor startup, all auxiliary coolant and power systems will be checked for proper operation. The reactor building ventilation system will be checked, and pressure zones will be established so that air flow will only be from nonradioactive to radioactive areas in the building. The header rooms, sodium equipment rooms, sodium pipe chases, and the calandria room will be purged and filled with nitrogen containing less than 1 weight % oxygen. All shield cooling components will be checked to insure that they are in proper operating order. The organic cooling system will be filled with the coolant, Dowtherm-A, which will be circulated at its rated value. The sodium and D₂O systems will be purged and filled with the appropriate cover gas – nitrogen for the sodium systems and helium for the D₂O system. A preliminary leak check will be made.

The sodium systems will be preheated prior to the filling operation by means of induction heaters attached to the major pipe runs. The fuel-coolant tubes and pigtailed will be preheated by passing hot nitrogen gas through the inner barrier spaces and header rooms; freeze valves, plugging indicators and cold traps will be preheated by hot Dowtherm-A; and the sumps will be preheated by immersion heaters.

The sodium will be aged and filtered to minimize the amount of Na₂O introduced into the system. After preheating, sodium will be pumped into the system until the expansion tank startup level is reached. The primary sodium drain valve and freeze valve will be closed to prevent inadvertent draining of the sodium. The cover gas pressure will be adjusted to maintain the appropriate net positive suction head on the circulation pumps, preventing pump cavitation. After filling, the sodium will be circulated through the system at 10% of rated flow, and flow will be established through the cold trap purification system to reduce the Na₂O concentration to about 10 ppm. This flow rate may be reduced during the fueling operation.

The D₂O system will be filled by pumping D₂O into the system by means of the pump provided in the dump tank. The system will then be set into operation with circulation established at the rated value.

With all of the systems filled, tested, and in operation, and with all except two of the shim safety rods fully inserted, fuel elements will be inserted into the fuel-coolant tubes. The fuel

elements will be preheated in the loading machine, where an inert gas blanket will be maintained to prevent oxygen pickup in the primary sodium system. During this fueling operation subcritical multiplication measurements will be made. After the fueling operation has been completed, the two withdrawn shim safety rods are inserted into the core, a final gas leak test is made on all top caps of the fuel channels to assure a perfect seal, and the reactor is ready for startup.

6.1.2 Nuclear Startup

In the startup and period ranges, the reactor is essentially independent of the power generation system, and the rate of increase of flux depends on reactor safety. In the power range, the rate of increase of flux depends on reactor safety and on the rate at which the turbines can be loaded. The rates of increase dependent on the turbines are much lower than the rates at which reactor power and system temperatures can be raised.

Attainment of criticality in the startup range will be accomplished by the manual withdrawal of control rods to the point where stable count rate and/or log N-period signals are obtained. At the end of the period range, the flux level reaches 1% of full power, and the rods will be positioned to establish a steady-state flux level. This entire operation will require about one hour. With the reactor critical at this point, the reactor cooling systems will be essentially isothermal at about 300°F, with no useful power being generated. During the next phase of the startup, the primary coolant inlet and outlet temperatures will be raised to 650°F and 950°F respectively; the steam pressure will be raised to 850 psig and the steam temperature to 850°F; and the preheated turbine will be brought up to speed and loaded. Operating conditions will be attained by a programmed increase in power level, transferring the reactor heat load from the bypass steam system to the turbine. The rate will be established by starting and loading instructions for the plant turbine generators. The elapsed time to reach full output power from the turbine is estimated to be five hours after reaching the power range.

6.1.3 Normal Reactor Shutdown

A planned reactor shutdown will reverse the sequence of operations mentioned above. Reactor power will be reduced from the operating level to about 10% of full power at a rate determined as satisfactory by the turbine manufacturer. The turbine may then be unloaded and steam bypassed to the condenser. Reactor power may be reduced from 10% of full power in an orderly manner by rod insertion and by operating the steam system to remove more heat than is generated by the reactor. Normal shutdown will be considered completed when the reactor is subcritical and all cooling systems are at the datum conditions existing at startup. This procedure may be modified or extended, depending upon the purpose of the planned shutdown — i.e., for refueling, maintenance, inspection, etc.

After the reactor is shut down, partial sodium flow must be maintained for at least 30 min. In this time the shutdown gas systems will be started, and gas cooling will be used to cool the fuel elements. Alternatively, reduced primary sodium flow can be maintained provided that the feed-water system can remove decay heat. If inspection is scheduled for the condenser, the gas systems will be used.

6.2 OPERATIONAL CONTROL OF THE REACTOR

Operational control of the reactor is accomplished by an automatic control system shown in Fig. 6.1. There are three major reactor control loops in addition to the steam plant controls. Their functions are described below.

6.2.1 Reactor Power Level Controller

This controller adjusts reactor power by regulating rod servo-drive to maintain a constant sodium temperature at the reactor outlet. The reactor power level demand signal is established by the temperature error plus a load anticipatory signal obtained from primary sodium flow. A period signal is used in the controller as an override to prevent increasing power level too rapidly.

Provision is made for controlling the reactor power level manually instead of automatically by a "manual-auto" selector and appropriate manual controls on the reactor power level controller. Period and temperature set point adjustments are provided in the controller.

6.2.2 Primary Sodium Flow Controller

Sodium temperature at the reactor inlet is held constant by this controller, which adjusts the primary sodium flow rate by means of the variable speed pump drives. The flow rate demand signal is formed by the temperature error signal, with secondary sodium flow as an anticipatory signal. The flow demand signal is compared with the actual sodium flow rate to obtain the error signal to the primary pump speed control system. The pumps are driven by a constant speed motor through a magnetic clutch with a variable slip control feature. Manual control of the flow rate is provided for in the controller, and a temperature set point adjustment is included in the controller.

6.2.3 Secondary Sodium Flow Controller

Control of the secondary sodium flow rate is similar to control of the primary sodium with the exception that steam flow is used as an anticipatory signal. The speed of the secondary pumps will be varied to maintain constant secondary sodium cold leg temperature.

6.2.4 Feedwater Flow Control

The feedwater flow control system operates to match the flow of feedwater to the total steam flow. Cumulative effects of mismatching the flow are corrected for by a temperature detector in the boiling region of the steam generator. Any serious shifting of water level due to flow mismatch appears as a temperature error. Positioning of the feedwater flow control valve is controlled by the combined flow and temperature error signals. The system also contains provision for direct manual control of the valve.

6.2.5 Feedwater Temperature Control

The temperature of feedwater obtained from feedwater heaters operating on turbine bleed steam drops as load decreases. This condition is not compatible with the requirement for a decreasing log mean temperature difference with decreasing load at the steam generator and intermediate heat exchanger. To permit maintaining a constant feedwater inlet temperature at the steam generator, superheated steam is bypassed to heat the feedwater during part load operation.

6.2.6 Turbine Bypass Control

In the event that the turbine throttle valve is tripped, a temporary heat sink must be provided for the heat stored in the steam generator, sodium systems, and reactor fuel elements. For this purpose, a turbine bypass valve to the condenser is provided. The condenser is sized to receive the full steam output of the steam generator. The bypass valve is operated by a pressure controller which is set slightly higher than operating pressure. Thus the system acts as an overpressure relief valve. Manual control of the valve is also provided to accomplish startup of a cold turbine.

Table 6.1 — Operational Safeguards During Startup and Shutdown

Operational Failure	Safeguard
1. Filling sodium into its loop when components are not heated to proper temperature (300°F).	Indicating lights normally tell operator when individual components are at proper temperature. Should sodium freeze or fail to fill, level in dump tank will indicate how filling is progressing. (Tank capacity = loop capacity).
2. Filling sodium when cover gas is not at required purity (O ₂ , H ₂ O may be present).	No interlock provided. Plugging indicator and cold trap detect and remove impurities.
3. Component heaters in sodium loops are accidentally turned off prior to nuclear startup.	Nuclear startup would be prevented because of insufficient flow should the sodium freeze.
4. Filling D ₂ O when cover gas is not at proper purity (H ₂ O, N ₂ may be present).	No interlock provided. Manual gas analyzer is only indicator of gas purity. No chemical hazards exist.
5. Filling D ₂ O when sodium is already filled and building ventilation requirements are not satisfied.	When building pressure zones are not at proper level, fill is prevented.
6. Operator withdraws control rods prior to establishing that all essential cooling circuits are in operation.	Interlocks with other systems are provided in all operating ranges to prevent control rod withdrawal.
7. Steam bypass to condenser is closed during startup prior to turbine heating.	Interlocks will be provided to prevent accidental closure.
8. Turbine is loaded or heated at a greater rate than manufacturer specifies.	No interlocks presently provided. Procedure manual will detail the proper heat rate.
9. Simultaneous circulation of hot and cold organic to freeze valves and cold traps.	Automatic three-way block valves provided to prevent simultaneous flow.

6.2.7 Operational Safeguards

The prescribed steps for starting and operating the system have been reviewed for possible operator failures, resulting from incorrect procedures in filling, closing valves, etc. Operator failures can be minimized by reliance on the reactor operating procedures. As will be seen, automatic instrumentation and interlock control is used to preclude, as much as is practical, the element of human error.

All reactor safety instruments and process or leak detection instruments pertinent to reactor safety will not be accessible to the reactor operator. Table 6.1 summarizes some of the possible operator failures that could occur during startup and shutdown. Where the consequences of a failure are relatively insignificant, i.e., with process system startup, the safety instrumentation is correspondingly less.

Table 6.2 deals with operational safeguards during normal operations, i.e., change of load, equipment inspection and maintenance. Failures associated with refueling are also considered.

Table 6.2 — Operational Safeguards During Normal Operation

Operational Failure	Safeguards
1. Throttle valve accidentally closed.	Turbine bypass valve to condenser will automatically open.
2. D ₂ O dump valve opened.	Reactor scrammed due to D ₂ O low-level control and loss of D ₂ O from core.
3. Sodium drain valve is opened before freeze valve is closed.	A manual lock will be placed on the drain valve. Reactor startup not permitted until both valves are closed.
4. Accidental startup of spill-return or transfer pump.	Residual sodium in dump tank or spill-return sump never exceeds the incremental capacity of the primary loop. Low-level signal in tanks will stop pumps.
5. Safety instruments or other process instruments are turned off.	Reactor scram initiated by fail-safe system. (Safety instruments cannot be manually shut off.)
6. Ventilation blowers turned off.	Interlocked with reactor scram.
7. Any main pump shut off.	Reactor scram initiated.
8. Main valve on D ₂ O line closed.	Reactor scram initiated.
9. Failure to provide cover gas in refueling cask (sodium) and clean-up cells.	Alarm signals operator.
10. Failure to start element cooling system in cask before raising element from core.	Fuel element hoist interlocked with blower.
11. Operator tries to exert force when element may be stuck in coolant tube.	Automatic load relief mechanism is provided in refueling machine, preventing overstresses.

6.3 MAINTENANCE AND REFUELING

Component shields have been designed so that workers will never receive more than a daily allowable radiation dose of 20 mrem in any one day. Maintenance of nonradioactive reactor components protected by the shields will be accomplished by direct contact methods. Radioactive components may be drained and flushed, and radioactive materials will be allowed to decay. Wherever necessary, portable shields will be erected.

Refueling of the reactor will be carried out with the aid of a refueling machine similar in concept to the SRE refueling machine. The machine will be capable of removing spent fuel from the reactor and inserting fresh fuel into the reactor, fuel transport being carried out in a nitrogen-blanketed atmosphere. The spent fuel will be carried in the refueling machine to the wash cells located in the water-filled fuel storage pool where they will be washed, placed in underwater storage racks and allowed to decay. After a decay period that is long enough to permit the elements to stand in air without melting, they will be transferred through a chute to a storage pool in the process building.

6.4 SCRAM SIGNALS AND SCRAM PROCEDURES

6.4.1 Types of Scram

The scram operation is accomplished in the SDR by disconnecting all shim-safety rods from their drive motors by means of electric clutches and allowing them to fall under the action of gravity, with spring assist, to their fully inserted position. Under certain emergency conditions, the D₂O moderator will be rapidly drained from the reactor in conjunction with the rod drop. However, this is not a necessary feature of the scram operation.

The use of shim-safety rods rather than separate shim and safety rods provides the following advantages with respect to safety: the maximum amount of negative reactivity available in the control rods is always inserted during a scram; the rods are released from a position in which their incremental reactivity worth is greater than that of cocked safety rods; and the drive motors need not be absolutely fail-safe, as all rods are disconnected from the drives during a scram.

A scram represents a severe interruption of the normal operation of the reactor and power plant, giving rise to rapid reductions of sodium temperature and steam temperature and pressure. While all the components are designed to withstand these transients, they are not desirable; in addition, the reactor and power plant are not immediately available to go back on the line. To avoid as many scrams as possible and yet not compromise the safety of the plant, the following series of progressively more severe corrective actions are used: alarm, automatic power reduction, rundown, and scram.

Alarm consists of bringing the disturbance to the operator's attention by means of audible and visible signals. Automatic Power Reduction is a fast but orderly reduction of all operating variables to the conditions corresponding to 10% of full power output; the reactor is still critical at the conclusion of this process. Rundown consists of motor-driven insertion of all shim-safety rods; appropriate action is automatically taken during this operation with regard to the sodium and steam systems. Both the Rundown and Automatic Power Reduction operations are automatically terminated if the disturbing condition is corrected. Scram is the immediate dropping of all rods; scram signals override at all times the other three types of corrective action.

Each shim-safety rod has an electrically operated clutch in the transmission between the drive motor and rod, controlled by the safety amplifiers. It is the function of the safety amplifiers to de-energize all the clutch coils when a scram signal is received. Each safety amplifier can de-energize the clutch coils to drop the rods by two methods. When the scram signal is nuclear, i.e., high flux level or short period, an electronic scram is used in which the clutch is de-energized as rapidly as possible by driving the currents to zero electronically. When the scram signal is non-nuclear, a relay scram is used which consists of opening the a-c lines supplying power to the safety amplifiers. Electronic scram is automatically backed up by relay scram action. In general, relay scram is reserved for those non-nuclear signals which do not require the short delay time achieved by the electronic scram.

6.4.2 Sequence of Events and Scram Signals

The sequence of events following receipt of a scram signal is as follows. The clutches are de-energized, allowing all shim-safety rods to fall into the core. The clutches remain de-energized until (1) the signal calling for the scram has disappeared, and (2) the "scram reset" button is pushed by the operator. Signals are sent to reduce the primary and secondary sodium flow rates; this reduces the severity of the outlet sodium temperature transient. The generator and steam turbine are automatically taken off the line, feedwater flow is reduced, and any steam produced is bypassed to the condenser.

The signals causing a scram may be categorized as being nuclear or non-nuclear in origin. The nuclear signals are derived from the electronic devices contained in the nuclear instrument channels (Fig. 4.16). Multiple channels are used to accomplish complete coverage of neutron

flux from source level to above full power, and the principle of duplication is employed as a safety measure.

Nuclear scram in the startup range is obtained by initiating a scram when both recorders are at full scale. The primary signal used for interlocking in this range is period. As the period signal obtained from the pulse instrumentation is somewhat erratic, a short period signal will not be used to actuate a scram but will automatically initiate a rundown. A relay scram will also be initiated upon failure of an instrument in either channel. In the period range, scram is initiated by a short period indication from either channel. A relay scram is called for upon failure of an instrument in either channel. In the power range, high flux level from any channel will initiate a scram, the trip point being set at 150% of full power. In addition, it is required that at least two channels be operative at all times. (To provide continuity of operation in the power range in the event of instrument failure, three channels have been provided.) A relay scram is initiated upon detection of instrument malfunction in any two channels.

It may be desirable to provide period protection in the power range as a secondary safety signal. However, period channels are subject to false shutdown signals. While this is not significant below the power range, it is serious if power production is interrupted because of a spurious signal. Hence, either the period scram signals derived from the log N-period channels will be deactivated or the period scram trip value will be shortened in this range. Future analysis will decide the choice. In the event that a shortened period scram signal is retained, a relay scram will be initiated upon failure of both log N-period channels in the power range only, rather than upon failure of one channel as in the period range.

Non-nuclear relay scrams are produced by abnormal process instrument signals, such as high sodium outlet or inlet temperature, boiler feedwater supply failure, high thermal power, low control rod coolant flow rate, electrical power failure, and earthquake indication. Relay scram may also be initiated by a manual signal. A complete list of Alarm, Automatic Power Reduction, Rundown, and Scram Signals is given in Fig. 6.2.

Fig. 6.1 — Schematic block diagram of reactor and plant control system

Fig. 6.2 — (Continued)

7. SAFETY ANALYSIS OF REACTOR SYSTEM

This section covers emergency conditions that can arise during normal reactor operation as a result of equipment failures or power failures, and the safety features provided to protect the reactor. The general operational philosophy for handling emergencies is also presented.

The following events and system operations have been reviewed as part of the safety analysis.

1. Power failure
2. Primary sodium pump failure
3. Moderator pump failure and D₂O dump
4. Loss of heat sink
5. Sodium and D₂O leaks
6. Spill return system
7. Plugging of sodium flow
8. Temperature and stress transients
9. Nuclear incidents
10. Chemical incidents

Table 7.1 summarizes the major failures considered, presents the detection devices for each failure and the major operational results.

7.1 POWER FAILURE

An emergency power system is provided to permit safe shutdown of the nuclear plant should normal power be interrupted. The system will provide continuous power to vital equipment (such as reactor coolant recirculating pumps, instruments and control devices) until all components are free from danger or until normal power is restored. A one line diagram depicting the emergency power distribution system is shown in Fig. 7.1.

Normal power is supplied from the Chugach Electric Association substation, which is located on the same site as the nuclear plant. There are two independent feeders, each having its own transformer and switchgear supply equipment. They supply all normal power requirements of the plants. Each source is sized so that it can supply all the necessary power should the other one fail (e.g. transformer burnout). Power is supplied to the reactor plant via two routes, the vital bus and the normal bus. The vital bus, which powers the reactor safety equipment, will be supplied with emergency power should normal power be interrupted.

The emergency power system consists of a 240-volt battery supply, an ac-dc motor generator set and a diesel-operated a-c generator. During normal operations the motor generator set, powered from the vital bus, operates continuously to send a trickle charge to the battery. This provides for full battery charge for switch gear control, and insures that the MG set is operable should normal power be interrupted. Should a power failure occur, the undervoltage and directional relays built into the system will scram the reactor. The battery powered MG set will supply the

Table 7.1 — Summary of Equipment Failures and System Responses

Equipment Failure (excluding electronic failures)	Detection Devices	Major Operational Result
Loss of power	Undervoltage power relays	1. Reactor scram 2. Emergency power system activates vital bus components
Failure of one of two primary sodium pumps	1. Pump speed and discharge pressure indicator 2. Thermocouples on reactor coolant tube	1. Reactor scram
Failure of one of two secondary sodium pumps	1. Inlet temperature to reactor 2. Thermocouples on reactor coolant tube 3. Secondary flowmeter 4. Pressure and pump speed detectors on secondary flow	1. Reactor scram 2. Primary pump gradually cut back to 10% flow
Failure of moderator pump	1. Discharge pump pressure monitor 2. Flow detector 3. Overflow from calandria weir	1. Reactor automatic power reduction 2. Alarm
Leak in intermediate heat exchanger	1. Level detector in primary sodium 2. Level detector in secondary sodium	1. Automatic power reduction
Sodium leak from primary system	1. Catch pan detectors, pigtail cup 2. Thermocouples on reactor coolant tube 3. Vapor detectors (alarm only) 4. Expansion tank level monitor	1. Reactor scram, rundown, or alarm 2. Spill return system activated 3. Cutback to 10% flow 4. D ₂ O drained
D ₂ O leak from calandria	1. D ₂ O overflow level 2. Liquid D ₂ O detectors in catch pan 3. Vapor detectors 4. Capacitance detectors on lower tube sheet	1. Alarm 2. Controls rod rundown 3. D ₂ O drained 4. Start up calandria tube cooling system
Loss of heat sink e.g., failure of feedwater pump plugging of steam generator	1. Inlet temperature to reactor at primary heat exchanger discharge 2. Feedwater flow monitor 3. Secondary sodium thermocouples 4. Thermocouples on reactor coolant tube outlets	1. Reactor scram 2. Manual startup of shutdown gas cooling system

initial demands of the vital bus (i.e., until the diesel generator goes on the line) with essentially no time delay. The diesel generator, the main source of emergency power, is automatically started and synchronized with the vital bus power. If the diesel generator fails to start, the battery operated MG set will be capable of supplying power to the vital bus for about 45 min. Periodic, routine diesel startup checks should assure diesel operation in an emergency.

When the reactor is scrammed due to a power failure, the emergency power system will operate only one of the two main coolant pumps in each sodium system. The thermal transients induced by flow decay to half of that required at full power followed by a scram are discussed in Section 7.8. Their effects have been found to be tolerable. To prevent overheating of the primary coolant, either one of the secondary sodium cooling system pumps can be operated on emergency power.

7.2 PRIMARY SODIUM PUMP FAILURE

Primary pump failure has essentially no effect on reactivity but conceivably can lead to overheating of the fuel elements. Interlocks are provided to prevent accidental stoppage of these pumps while power is being generated. The most probable source of primary pump failure is power failure. This case was discussed in the previous section. Another source of pump failure is mechanical failure.

The consequences of mechanical failure are minimized by the use of two pumps. It is extremely unlikely that both of the primary coolant pumps will fail at the same time. If one pump fails, the system flow rate quickly falls to half of its normal value and is maintained at this level by the remaining pump. Almost coincidentally with the pump failure, the reactor will scram on a high outlet temperature signal. Fig. 7.2 gives the sodium temperature as a function of time for the pump failure case. Thermal transients similar to those occurring during a power failure will be induced: these transients, discussed in Section 7.8, were found to have no harmful effects on the primary sodium system.

The effect of natural thermal convective forces on sodium flow has been found to be small because of the small difference in elevation between the intermediate heat exchanger and the reactor core, the relatively small coefficient of expansion of liquid sodium, and the relatively large friction losses in the system. After pump failure has occurred, therefore, the remaining pump will continue in operation until essentially all of the reactor afterheat has been removed. The shutdown gas cooling system will provide additional cooling capacity should this be required.

A failure of a primary sodium pump is detected by a pressure sensor at the pump discharge, and by a high outlet sodium temperature signal. The reactor is scrammed on the signal from any one of these sensing elements. Backflow through the pump which has failed is prevented by a check valve backed up by an automatic block valve.

Simultaneously, the secondary sodium flow controller reduces flow in the intermediate loop in accordance with power level and primary sodium flow. Flow of feedwater to the steam generator is also controlled with respect to power level and sodium flow rate.

7.3 MODERATOR PUMP FAILURE AND D₂O DRAIN

Failure of the moderator pump will result in rapid flow decay. The drop in pressure and flow will sound an alarm mounted on the control panel. Loss of moderator flow will cause an automatic reduction of the reactor power level.

Loss of moderator flow will result in the overheating of the moderator. To obtain a conservative estimate of the time delay between the pump failure and bulk boiling of the moderator, the following assumptions were made.

1. Flow decay is instantaneous following a pump failure.
2. No significant external thermal circulation occurs so that only the volume of D_2O in the calandria will be heated.
3. No heat is transferred across the boundaries of the calandria shell.
4. The rate of heat deposition in the moderator at any power is linearly proportional to the heat deposition at the rated power of the reactor.

Based on the above assumptions, the time necessary to get nucleate boiling in the D_2O is ~11 min at continuous full power operation and ~110 min for continuous operation at 10% of full power. Thus, the large thermal capacity of the D_2O in the calandria prevents any serious consequences arising from moderator pump failure. The calandria itself is vented through an 8 in. line to a dump tank which is protected by both a relief valve and rupture disc. Boiling of the D_2O in the calandria will eventually reduce the liquid level in the system to the point where the D_2O will be dumped automatically.

The moderator dump system provides a means of rapidly removing D_2O from the calandria in the event of (1) a major sodium leak, (2) overheating of the moderator, (3) moderator leakage which could result in the excessive loss of D_2O . The moderator is dumped from the calandria by gravity flow through a 12-in. dump line. Two fast-acting dump valves in series are provided on this line. The valves automatically close on signal of high level in the dump tank to isolate the calandria from the dump tank after a fast drain. The total time to drain the entire contents of the calandria is ~40 sec. The dump valve is designed to fail-safe, since it requires hydraulic pressure to keep it closed. Loss of hydraulic pressure will dump the moderator and will cause a shutdown of the nuclear system (Section 8.1). A vent line (Fig. 4.9) between the calandria and the D_2O dump tank prevents collapsing of the calandria during the dump. As an added safety feature, the calandria is designed to withstand an external pressure of 15 psi.

7.4 LOSS OF HEAT SINK

The loss of primary heat sink results in a positive temperature surge in the primary sodium system. Failure of the feedwater pump or of the secondary sodium pump can cause this emergency condition. A preliminary analysis of this problem was made assuming a step increase from 650 to 950°F in the primary sodium outlet temperature at the intermediate heat exchanger. After a time delay of 6 sec, the temperature rise enters the core as a damped front with an average gradient of +50°F/sec. Temperature-induced stresses in the lower pigtailed headers have been evaluated and are within the allowable limits. No failure is anticipated from thermal shock. (See Section 7.8.)

As the wave front passes through the reactor a temperature overshoot in the sodium can occur. Fig. 7.3 shows the temperature front, assuming full flow and a reactor scram just as the temperature front hits the inlet to the reactor. The inlet temperature profile reflects the attenuation of the step function as it passes from the intermediate heat exchanger to the reactor inlet. Since thermocouples will be located at the primary outlet from the exchanger, an anticipatory scram signal should be available about 6 sec before the front enters the core. The assumption of a simultaneous scram and temperature rise is therefore quite conservative.

The average outlet temperature will peak at around 1000°F, and the average temperature of the central portion of the fuel element will drop from 2615°F to ~2000°F during the transient, as a direct result of the scram. The conclusion of this study is that the loss of heat sink problem is not serious, especially when the anticipatory scram signal from the intermediate heat exchanger is considered. Since a scram is called for when the primary sodium temperature rises from 650°F to 750°F at the exchanger discharge, the reactor will be essentially shut down when the surge enters the core.

Following a loss of heat sink, the system temperature will continue to rise unless a secondary heat sink is available. Both the shutdown gas cooling systems and the heat capacity of the fuel,

primary sodium and primary sodium piping will provide a sink for the heat generated after reactor shutdown. As discussed earlier, the shutdown gas cooling systems can remove reactor afterheat 30 min following a scram. If a power failure occurs such that the shutdown gas cooling system does not operate immediately, the heat capacity of the primary sodium system will prevent boiling in the core for about 6.7 hr after shutdown. This calculation neglected the heat losses to the neutron shields and header rooms and is therefore a conservative estimate of the inherent heat capacity. Similarly the system can absorb about 2.3 min of full reactor power before the outlet temperature of the reactor reaches 1620°F, the boiling point of sodium. These calculations show that while a rapid reactor scram is desirable following loss of heat sink, it is not essential for maintaining the safety of the plant.

7.5 SODIUM AND D₂O LEAKS

Separation of leaking sodium and D₂O is maintained by a system of barrier tubes and catch pans. Sodium and D₂O leaks occurring in pigtails, headers, fuel tubes, and in the calandria drain away from their sources to sumps in the following manner.

7.5.1 Leakage Paths

D₂O leaking to the inside of the calandria tubes will be deflected by the aluminum barrier tube. It will drain into the D₂O catch pan located on the floor of the calandria room through the outer annulus formed by the calandria tube and the barrier tube, and then into the D₂O sump. The point where the barrier tube passes through the catch pan is designed so that D₂O cannot drain into the lower header room. Any D₂O that is vaporized in the annulus will be condensed in the calandria tube gas cooling system and returned to the D₂O sump. An outer catch pan will drain D₂O leaking from the calandria shell, manifolds and piping. D₂O leaking from pipes, pumps, and other D₂O system components will be collected in catch pans and drained to the D₂O sump. D₂O cannot be returned to the system from this sump until its purity has been assured.

Sodium leaking from the upper pigtails, subheaders and outlet header will be collected in a catch pan located in the bottom of the upper header room. Squirts directed upwards or sideways will run on stainless steel sheathed walls into the catch pan. The fuel tubes passing through this catch pan are flashed and mechanically sealed so that sodium cannot drain through the upper thermal shield and fall on the calandria. The sodium is drained from the catch pan through a 6 in. drain line to the sump tank below the lower header room. Sodium leaking from a fuel coolant tube will be deflected by the aluminum barrier tube downward through the annulus formed by the barrier tube and the fuel coolant tube into the catch pan in the bottom of the lower header room. Sodium is drained from this catch pan to the sump through a 6 in. drain line located in the center of the catch pan. Sodium leaking from the lower pigtails, subheaders and headers will also be collected in the catch pan in the bottom of the lower header room and drained similarly. Leaks from pipes, pumps, heat exchangers, and other sodium-containing equipment will be channeled to the sodium pump. As discussed below, such leaks are very improbable.

7.5.2 Sodium Leak Sites and Leak Rates

Experience with major sodium installations shows that the most probable leak sites are at stressed welds^{12,13} and that the frequency of leaks can be reduced by a rigid inspection procedure.²⁰ In the SDR, the fuel-coolant tube is manufactured from a single piece of stainless steel, with welds used only above and below the reactor room (where the coolant tube is attached to the upper and lower pigtails). Inspection should assure that sound welds are obtained and failure of the coolant tube should only occur by direct corrosion of the tube wall, by melting, or by thermal stress fatigue. Corrosion in the primary sodium system is virtually negligible (~0.02 mg/cm²/month, equivalent to 0.01 mil/yr of uniform surface removal) because of the low level of oxide content maintained by the sodium cold traps. Melting of the coolant tube is unlikely, since the steel itself

is basically a high temperature material, melting at ~2700°F. In addition, the system instrumentation prevents temperature overrides to avert situations that could lead to coolant tube melting.

Thermal stress levels during steady state and transient operation are well below the design limits, and failures from thermal shock are extremely unlikely. Since no one portion of the primary system could be singled out as a possible leak site, a general review was made of the importance and probability of leaks from various locations. This review showed that the most significant leaks were those occurring either in the fuel-coolant tube below the core or in the lower pigtail-header complex. Since leaks in the upper header room piping will not uncover the fuel elements, leak rates for lower header room piping failures were evaluated.

Failures in the pigtails were analyzed, since these members are the most highly stressed, and a single pigtail failure could lead to the starving of flow in a particular lattice position. Leaks occurring at manifold and header welds are likely to be small,²¹ and their effect would be to slightly reduce flow to a number of lattice tubes, rather than produce a severe effect on one tube.

Because prediction of leak areas prior to the actual failure is impossible, an analysis of leak rate as a function of leak area in a lower pigtail was undertaken. Flow up through the channel was considered. For very large leaks, flow reversal from the outlet header down through the pigtail leak was also anticipated. The electrical analog method described in Appendix B was used for the calculation. The following three important facts are apparent:

1. Gross flow reversal as a result of leaks in the lower pigtail does not occur for leak areas of reasonable size. Flow reversal occurs only if there is an enormous hole in the lower pigtail. Experience with leaks at weldments has shown that most failures are in the 0.020 to 0.030 in. range;²¹ for leaks of this size, essentially no change occurs in the flow distribution of the failed pigtail lattice position.
2. Complete severing of the pigtail with flow reversal in the coolant channel delivers 350 gpm of sodium to the sodium spill-return system under full pump power. The spill-return system is sized to handle 500 gpm and can easily handle the leak from a completely severed pigtail. Under these conditions the reversed flow in the coolant channel is 70 gpm or 10% above normal flow in the channel.
3. Reactor scram followed by a cutback to 10% of full flow will reduce this maximum leak rate from 350 gpm to about 70 gpm.

The significance of these sodium leak rates is discussed in Section 7.10, Chemical Hazards Evaluation.

7.6 SPILL-RETURN SYSTEM

Safe and reliable operation of the spill-return system requires that:

1. the delay time involved in starting the spill-return system does not exhaust the reserve sodium in the expansion tank,
2. the sodium does not freeze on the catch pans or sump lines,
3. plugging or flow stoppage does not occur in the sump return lines.

An analysis of the delay time in channeling sodium from the catch pans to the sump tank is difficult because of the varying geometry of the open flow channel. A rough estimate using a severed pigtail leak in the lower header room gave a delay time of 1 min, indicating that a minimum sodium reserve volume of ~360 gal in the expansion tank is required to prevent uncovering of primary pump suction.

Should a leak occur in the upper header room, the maximum leakage rates will be appreciably smaller because of the decreased hydrostatic head of sodium there. In addition, the main pumps may be cut back to 10% of full flow as soon as the reactor is scrammed, reducing leakage from

the assumed upper piping leak. For this case in which the time delay is longer but the sodium leak rate smaller, the expansion tank can also provide enough reserve sodium volume to prevent uncovering the pump suction.

No freezing of sodium can occur in the header room or on the catch pans, since every portion of the sodium system is maintained above 300°F. The gas in both header rooms runs above 600°F during normal operation. During operation of the shutdown gas system, cooler gas is introduced in the lower header room. However, the room temperature will be maintained above 200°F to prevent any possibility of sodium freezeup.

Analysis of the permissible oxygen concentration in the header room gas has shown that no plugging by Na_2O can occur if the oxygen content of the gas is maintained at less than 1% by volume. In addition, the lines from the sump tank to the expansion tank are large, open channels, free from obstructions where the oxide could precipitate and cause a plug. Cold trap circulation will be needed to clean up the primary sodium system following a spill.

7.7 PLUGGING HAZARD

Loss of coolant flow through one or more fuel tubes can lead to serious overheating of the fuel elements. Three possible sources of plugging were considered:

1. an oxide plug following an undetected oxygen leak into the primary sodium system,
2. fuel element fracture leading to flow stoppage,
3. debris left in the system after fabrication.

The maximum rate at which an oxide plug can form can be calculated from the temperature, geometry, and flow rate of the primary sodium system, using certain simplifying assumptions. The most probable point for a plug to occur is in a lower pigtail, where it is estimated that a 6 in. plug averaging 80% Na_2O would shut off flow. Using the normal temperatures and flows, a plugging time of about 15 min results. The calculated time is quite long, showing that rapid loss of coolant flow in a fuel tube could not result.

The probability that such conditions could exist in an SDR (with a secondary liquid metal coolant circuit) is extremely small. It is difficult to postulate an undetected leak which lets oxygen in instead of letting sodium or cover gas out. Since mixing occurs in the piping leading to the lower header room, essentially the entire primary sodium flow would have to be saturated with oxygen, and plugging would occur to some extent in all pigtails, causing a noticeable increase in total flow resistance. Judging from experience with plugging meters and cold traps,¹² solid oxide precipitation begins as soon as supersaturation exists implying that some of the oxide will deposit in the piping upstream of the pigtails. Nucleation in the bulk of the fluid also occurs,¹³ so only a small fraction of the oxide that precipitates in a pigtail will stick to the pipe wall.

If the oxygen inleakage is detected and stopped, steps can be taken to avoid plugging. By stopping power generation and circulating sodium isothermally at a temperature high enough to dissolve any precipitated oxide, the oxide can be moved from plugs to the cold trap. To detect incipient plugging, thermocouples reading outlet sodium temperatures at the top of each fuel tube will be monitored and coupled to an automatic reactor scram circuit.

The hazard of a fuel element fracture is considerably less likely than oxide plugging. First, the element cladding is not under high stress, even during transients, and should be able to carry the load of each fuel pin. Second, the lower neutron shield plug grapnel forms an effective stop to hold up any debris should such a failure occur. Smaller pieces that could conceivably break off would be caught in the orifice plate assembly. The use of staggered holes in the orifice plates insures flow through the orifice assembly even if individual orifice holes are plugged with debris. Material accidentally left in the piping during construction will be screened out during cold runs.

Based on these design features, it is concluded that the hazard of unexpected loss of coolant flow in an SDR fuel tube either from oxide plugging or mechanical failures is negligible.

7.8 TEMPERATURE AND STRESS TRANSIENTS IN PRIMARY SYSTEM

Thermal transients and stresses arise in the primary sodium system from three sources: (1) full flow scram, (2) loss of the intermediate heat sink, (3) failure of one of the two primary pumps. In all cases, the transient is complete in less than one flow cycle through the primary loop and the initial transient is much more severe than subsequent transients as the hot or cold slug of sodium flows around the system.

The allowable stress is taken as twice the 0.2% offset yield strength of type 316 SS at the temperature at which the stress is imposed. For a given transient, the number of cycles that can be withstood will depend upon the temperature level reached and upon the magnitude of the total steady state and transient stress.

Experimental results of cyclic tests of austenitic stainless steel piping at a temperature of 1050°F are presented in Appendix D. The number of cycles to failure, employing stresses equivalent in magnitude to the allowable stresses established, is found to be of the order of several thousand cycles.¹¹ Since the experimental results are based on mechanically induced stresses, the number of cycles to failure will be conservative for transient stresses induced at the same temperature. The number of transient cycles occurring during reactor operation will be much smaller.

7.8.1 Scram with Full Flow

Analysis of the transient temperatures in the outlet pigtail-header system has shown that an average temperature drop of approximately 30°F/sec follows a scram with full sodium flow. The transient is similar to that shown in Fig. 7.2. The approximate thermal stresses at each location for this transient are given in Table 7.2. In no case is the allowable stress exceeded. Protection of the inlet nozzle of the intermediate heat exchanger is afforded by the expansion tank, which acts as a heat source and damps the initial transient, resulting in stresses well within the capabilities of established heat exchanger designs.

7.8.2 Loss of Heat Sink

The most severe shock to the lower sodium piping system follows the loss of intermediate heat sink. As a very conservative approximation of this condition, a step function in the primary outlet sodium temperature at the intermediate heat exchanger from 650°F to 950°F was assumed. The resultant high temperature front is attenuated as the shock approaches the lower pigtail. The time variation of the transient at important locations is presented in Fig. 7.4, with tabulated thermal and expansion stresses given in Table 7.3.

Even for this extremely pessimistic case, the thermal shock to the lower sodium piping is not excessive and the allowable stresses are not exceeded. Fig. 7.3 shows the front leaving the reactor and entering the upper header room piping array. The initial gradient is -27°F/sec which is less than that for the previous case with a scram at full flow. Consequently, the thermal stresses on the outlet system are less than for the full flow scram and no failure is anticipated.

7.8.3 Failure of One Main Sodium Pump

When a main sodium pump fails, the temperature transient shown in Fig. 7.2 is generated, which gives rise to thermal stresses in the upper sodium piping. During this transient, the sodium temperature rapidly rises to 1170°F and then decreases after the scram at a slower rate to 860°F. The time variation of the transient at important locations creates the transient stresses listed in Table 7.4. The combined stresses, as shown for this case are greater than those given in Table 7.2, but do not exceed the allowable stresses.

Table 7.2 — Total Stresses to Upper Header Room Piping Following Scram at Full Flow

Component	Wall Thickness, in.	Location	Max. Total Transient Thermal Stress, psi	Approx. Total Combined Stress, psi	Allowable Stress psi*
Fuel-coolant tube	0.250	At pigtail	23,000	25,000	45,000
Pigtail	0.125	At fuel-coolant tube	12,000	36,500	45,000
Subheader	0.226	At header	19,000	26,000	45,000
Header	0.322	At subheader	43,000	24,000	45,000

* Based on temperature of 850°F.

Table 7.3 — Total Stresses in Lower Header Room Piping Following Loss of Heat Sink

Component	Wall Thickness, in.	Location	Max. Total Transient Thermal Stress, psi	Approx. Total Combined Stress, psi	Allowable Stress psi*
Fuel-coolant tube	0.250	At bottom of lower neutron shield	16,900	18,600	43,000
Pigtail	0.125	At subheader	13,000	32,300	43,000
Subheader	0.226	At header	33,100	39,000	43,000
Header	0.322	At distributional header	40,100	42,000	43,000

* Based on temperature of 950°F.

Table 7.4 — Total Stresses to Upper Header Room Piping Following Primary Pump Failure

Component	Wall Thickness, in.	Location	Max. Total Transient Thermal Stress, psi	Approx. Total Combined Stress, psi	Allowable Stress, psi*
Fuel-Coolant Tube	0.250	At pigtail	32,000	34,000	40,500
Upper Pigtail	0.215	At fuel-coolant tube	18,500	33,500	40,500
Subheader	0.226	At pigtail	32,000	34,000	40,500
Header	0.322	At subheader	38,000	39,000	40,500

* Based on temperature of 1150°F.

7.9 NUCLEAR HAZARDS EVALUATION

7.9.1 Kinetic Behavior of the Reactor

The use of D_2O as a moderator results in slow reactor transients in response to reactivity changes, due to the large prompt neutron lifetime of 6.6×10^{-4} sec. A secondary cause is a 6.5% increase in the delayed neutron fraction to $\beta = 0.0080$ (at the beginning of life), arising from photo-neutron production in the D_2O .

As has been discussed, the reactivity coefficients for SDR are quite small. The absence of a large prompt negative temperature coefficient of reactivity results in there being no inherent power excursion limiting mechanism as a backup to draining the D_2O moderator or dropping control rods. However, the small reactivity coefficients also result in a great reduction in the severity of any postulated reactivity accident that could occur through the mechanism of a reactivity coefficient, e.g., cold moderator slug. Since the control rods move at a rate slow enough so as not to damage the reactor during any conditions and D_2O level changes are not permitted unless all control rods are fully inserted, the probability of a reactivity accident is very low.

In considering the relative importance of the various temperature coefficients (Table 5.3), temperature changes due to a given change in power must be kept in mind. The fuel temperature will undergo a large change, the sodium temperature a moderate change, and the D_2O temperature a negligible change following a variation in reactor power. For example, consider the reactivity effect arising during an increase from zero to full reactor power. Initially the fuel elements and sodium are at $250^\circ F$ and the D_2O temperature is $110^\circ F$. At full power the mean fuel temperature is almost $2000^\circ F$, the mean sodium temperature is $800^\circ F$, and the D_2O temperature is $137^\circ F$. The net reactivity changes are:

fuel	-0.0039 Δk
sodium	+0.0046 Δk
D_2O	-0.0010 Δk
control rods	-0.0010 Δk
net change	-0.0013 Δk

Thus the reactivity coefficients are not very significant.

7.9.2 Review of Nuclear Accidents

The reactor was carefully examined for possible reactivity accidents, including improper fuel location, control rod ejection, sudden addition of moderator, loss of coolant, and startup accident.

Improper Location of Fuel Element

During the loading process, assume that a fuel element is inadvertently located partially out of the core. When the reactor is brought to criticality, the stuck fuel element drops into its proper position, with a rapid positive reactivity change. The mechanical design of SDR makes this type of accident insignificant. If a fuel element is not in position, it will not be possible to weld the top cap to the fuel-coolant tube in place. The allowable clearance is $\frac{1}{4}$ in., which corresponds to a reactivity change of approximately $0.00005\Delta k$. This would result in a period of the order of several hours, hardly a noticeable perturbation. If the fuel element hanger rod fails, the element will fall about 3 in., and be stopped by the grapnel knob of the lower neutron plug. A negative reactivity effect will result.

Control Rod Ejection

Control rods in the SDR, unlike those in a pressure-vessel type of reactor, are completely isolated from the primary coolant system.

Control rod ejection due to gas pressure is impossible since the control rods are supported from above and are mechanically prevented from falling down through the reactor. The nitrogen coolant pressure drop is of the order of 4 psi and acts downward. Should the control rod coolant reverse direction by some means, it could not lift the weight of the rod mechanism or cause a mechanical failure of the tie-down devices. Vent provisions in the calandria room prevent a pressure buildup in any control thimble to above 6 psig, precluding rod ejection during any sudden pressure increases. Loss of coolant flow to the rods causes a reactor scram, preventing rod meltdown.

Sudden Addition of Moderator

Interlocks will always demand that all control rods are fully inserted unless the moderator level is properly flowing over the calandria weir. The only exception is during a fuel loading operation, where two rods are cocked for shutdown safety. If the overflow line became plugged, 3 in. of moderator, worth an additional $0.001\Delta k$, could be added. Assuming this amount of reactivity is added instantaneously, the reactor would achieve a stable period of 75 sec. This effect is minor and would be handled by automatic rundown of the reactor.

Another means of inadvertently adding reactivity through the moderator is the "cold slug" accident. If the D_2O inlet temperature to the calandria (normally $110^\circ F$) were reduced due to some system malfunction, a positive change in reactivity due to the negative D_2O temperature coefficient could result. The reactivity change is very slow since the large volume of D_2O results in a temperature time constant of about 4 min. Even neglecting this effect, the magnitude of the resultant change is small. Consider that the D_2O inlet temperature is suddenly reduced from $110^\circ F$ to $70^\circ F$. As the D_2O temperature coefficient is $-3.78 \times 10^{-5}/^\circ F$, the resultant step change in reactivity is $+0.0015\Delta k$ which corresponds to a stable period of 50 sec, again a minor effect.

The third moderator accident considered is that caused by filling the void barrier annuli with D_2O . Failure of a calandria tube, permitting D_2O to flow into a barrier space, results in a positive change in reactivity. Complete filling of both gas annuli in all 73 lattice positions gives an increase in reactivity of $0.026\Delta k$. Filling of all barrier spaces is virtually impossible since the D_2O would flow out of the bottom or would flash vaporize. A more likely situation would be the failure of one calandria tube, resulting in the flooding of one outer barrier space. This gives an insignificant reactivity increase; barrier flooding, therefore, is not a serious accident.

Loss of Coolant

Sodium boiling and sodium draining are the only two methods of losing coolant. As the sodium is a nuclear poison, it will have a positive void coefficient, and either of these accidents would introduce reactivity. Using the worth of the sodium in the core as 0.02 in Δk , (Section 5.1.2), the effect of sodium draining on reactivity was considered. A range insertion of reactivity at a rate equal to the maximum leak rate for a completely severed pigtail was assumed. This study showed that at least 60 sec would elapse before the sodium level would drop from the expansion tank to the top of the fuel elements, assuming a complete failure of the spill-return system to pump back any leakage sodium.²³ In this time, two scram signals would have been by-passed: (1) the drop in level in the expansion tank, and (2) the leak detectors in the lower header room catch pans. Even assuming that these three protective devices had failed, the resultant ramp insertion rate of $5.5 \times 10^{-4}\Delta k/\text{sec}$ is roughly equal to the rate arising from maximum-speed control rod withdrawal (Section 7.9.4), which is shown to be safe for both hot and cold systems. Hence, no serious nuclear energy release can occur. In actual practice, the spill-return system has been specifically designed to prevent sodium draining and subsequent uncovering of hot fuel elements.

Transient studies discussed in Section 7.8 have shown that gross sodium boiling in the core cannot occur during either normal operation or emergency operation. The only possible mechanism

to boil sodium is through a partial plugging of sodium flow in an inlet pigtail or orifice plate (see Section 7.7). The consequences of plugging are discussed in Section 7.10.

The nuclear consequences are relatively minor. Even if sodium boiling occurred throughout the lattice position, such that the sodium density were reduced to zero, the increase in Δk would be 0.00037, which is considerably less than the previous cases considered.

7.9.3 Startup Accident

For many reactors the startup accident represents one of the more severe nuclear incidents. For the SDR, however, the large heat capacity of the fuel elements minimizes the temperature rise resulting from large power bursts of short duration, which are characteristic of this type of incident. The startup accident for SDR was calculated under the following conditions:

1. The reactor is cold at the beginning of life with the sodium and fuel in equilibrium at 250°F and sodium flow at 10% of full flow.
2. The reactor power is the subcritical multiplication of the source, about 7×10^{11} below full power, with an initial k of 0.93.
3. All control rods are continuously withdrawn at a speed corresponding to a maximum reactivity insertion rate of $4.5 \times 10^{-4} \Delta k/\text{sec}$, which is about 50% greater than the maximum reactivity speed that can be obtained. All safety trips are assumed to fail to operate, except the flux limit trips which are set at 150% of full power.
4. A time delay of 0.1 sec was assumed for the safety amplifiers and clutch. (The actual time delay will be considerably less, about 0.05 sec.)

The results show a total energy release of about 50 MW-sec, which leads to an average rise in fuel temperature of about 130°F. Fig. 7.5 gives the power-time transient and the temperature responses of the fuel and the sodium. The resulting temperature profiles are less severe than the normal reactor operating conditions.

In addition to the cold, clean startup accident, an excursion was analyzed in which it was assumed that all control rods are inadvertently withdrawn at the maximum withdrawal rate, while the reactor is operating at full power. The reactor is shut down by the flux limit trips at 150% of full power, with a delay time of 0.1 sec. Fig. 7.6 shows the resulting temperature surges for the outlet sodium temperature and central fuel section temperature. The resulting power surge is also shown in Fig. 7.6. The excess energy released during the incident is about 38 MW-sec, which is about the same as that in the cold startup accident. The resultant temperatures are not excessive, and the average sodium outlet temperature peaks at 985°F. The mean temperature of the central portion of the fuel element (maximum mean fuel temperature) peaks at 2750°F.

7.10 CHEMICAL HAZARDS EVALUATION

In the SDR, the largest potential mechanism for energy release comes from the sodium-water reaction. However, no reasonable sequence of events can be postulated for mixing the two fluids. The design of the reactor provides for safe venting of the products if the reaction does happen to occur. No propagation of failures is anticipated if a sodium-water reaction occurs in a barrier space.

7.10.1 Assumptions

A thorough analysis was made of major accidents arising from the mechanical failures discussed previously. In all cases, reactor instrumentation adequately protects against overheating and fuel-coolant tube melting. The only accident that could result in fuel tube melting would arise from undetected plugging of one coolant tube. This could occur if the outlet thermocouple monitors on the plugged tube were inoperative. This accident has been chosen as a starting point for the sodium-water accident study. As discussed in Section 7.3.5, plugging is very unlikely unless it

occurs from debris in the sodium line as a result of unpredictable mechanical failures. Fig. 7.7 gives a schematic diagram of the sequence of events arising from a plugged fuel-coolant tube. Estimates of the time delays between consecutive steps are included, together with a list of alarm and scram signals. The probability of complete stagnation of the sodium immediately at the onset of the plug is very small. Therefore, a high temperature signal at the channel outlet is anticipated. For reduced flows greater than 30%, no failure is anticipated in the core. For less than 30% flow, sodium boiling will occur in the channel, leading quite rapidly to further choking of flow. A melt-down will occur in the center portion of each fuel rod, with the peripheral rods melting first. Heat losses by radiation and conduction will limit the amount of melting to the central portion of the rod cluster.

The subsequent steps in the accident are given in Fig. 7.7, showing an eventual failure of the fuel-coolant tube in the core. The hot sodium will be expelled from the fuel-coolant tube and will rapidly penetrate the barrier. Experimental evidence has shown that penetration of the calandria tube in a region that is backed by D_2O will not occur (Appendix D). For purposes of the maximum credible accident study, it was assumed, however, that the sodium penetrating the barrier would also penetrate the calandria tube in the core midplane region. D_2O and sodium were assumed to react during the elapsed time of the D_2O dump, starting when the calandria tube is breeched. The reaction is effectively quenched when the D_2O drains from the calandria and the dump valve is closed.

7.10.2 Results

The analysis evaluated the maximum amount of sodium and water that could react for various assumed sodium leak rates. Fig. 7.8 gives the results, showing the amount of sodium reacted during the D_2O dump (40 sec) as a function of various sodium leak rates, and the assumptions used for the calculation. The total energy release for various leak rates is also given. Both the sodium-water reaction and the hydrogen (D_2) oxygen reaction are included in the total release, since the pressure buildup will require venting into the air-filled containment building. This energy release will not be augmented by nuclear energy release, since the reactor will have been scrammed prior to the reaction and the calandria will be completely drained by the end of the incident.

Failure of the calandria tube will not cause subsequent failures in adjacent fuel tubes, since adequate expansion and venting provisions are available. The gaseous reaction products will be relieved through the helium cover gas line and the vent line connecting to the dump tank. Recent experiments at APDA¹⁶ support the conclusion that no damaging shock waves will arise from sodium-water reaction, as long as vent and expansion provisions are available. The APDA tests were performed on a water-filled, seven-tube heat exchanger bundle immersed in hot sodium. The central tube was opened suddenly, allowing a continuous flow of water to contact the sodium. No evidence of damage or distortion could be found on the surrounding tubes.

Since the failure is more likely to occur in only one fuel tube, the maximum leak rate for a failed fuel tube can be estimated. It is assumed that a plug occurs at the entrance to one fuel element, leading to the melting and rupture of the fuel tube. Assuming that the fuel tube and element are completely sheared at the core midplane, the sodium flowing from the break is approximately 120 gpm following a shutdown of one of the primary sodium pumps. This calculation was made on the basis of two conservative assumptions: (1) the coolant tube break is severe and (2) the plugging leading to the break resulted in reduction to 30% of normal flow, rather than complete plugging. (It is apparent, in this consideration, that a complete plug would result in a lower leakage rate through the break.) The sodium flowing from the break in the coolant tube must overcome the resistance offered by the penetrations in the barrier and calandria tubes; additional resistance is offered by the D_2O which is trying to flow in the opposite direction in the calandria tube. On the other hand, the barrier spaces offer alternate flow paths which divert the leaking sodium from entering the calandria. If we make the additional conservative assumption that all of the sodium

leaving the coolant tube during the D_2O drain enters the calandria and reacts with the D_2O , the total energy release is 3,000,000 Btu, corresponding to approximately 550 lb of sodium reacting. No aluminum-water reaction will occur since the aluminum calandria is cool. Appendix A summarizes the effects of various quantities of sodium-water reaction in representative containment buildings. For a typical 70-ft diameter containment building, with $\sim 500,000 \text{ ft}^3$ of free volume, the predicted equilibrium pressure buildup is ~ 4 psi. Specifications for the containment building now call for pressure testing at 30 psi.

Fig. 7.1 — One-line emergency power distribution diagram

Fig. 7.2 — Sodium and central fuel temperature rise following pump failure — 50% step-drop in flow; reactor scram at $T_{outlet} = 1050^{\circ}\text{F}$; $t_{delay} = 1$ sec for scram.

Fig. 7.3 — Sodium and central fuel temperatures upon loss of heat sink — reactor scrammed at time zero.

Fig. 7.4 — Temperature front following 300°F step increase in primary sodium temperature at intermediate heat exchanger. Curve 1 — main inlet header; curve 2 — lower pigtail-subheader joint; curve 3 — lower pigtail at reactor inlet.

Fig. 7.5 — Reactor power and temperature relationships for a startup accident employing a gravity scram at 150% power

Fig. 7.6 — Control rod withdrawal at an assumed rate of $4.5 \times 10^{-4} \Delta k$ per second at full power, with scram at 150% power. $t_{\text{delay}} = 0.1$ sec.

Fig. 7.7 — Sequential failures from plugged coolant tube

Fig. 7.8 — Sodium-water reaction data for plugged coolant tube incident. Assumptions: Na and D₂O react stoichiometrically; D₂O dump initiated when calandria is breached; reaction is quenched when all D₂O is drained from calandria; hydrogen generated is reacted with air in containment building.

APPENDIX A

CHEMICAL ENERGY RELEASES

Two potential chemical reactions^{6,17} are of prime concern from the safety standpoint for the SDR, namely, the sodium water reaction and the hydrogen-oxygen reaction.

SODIUM-WATER REACTION

Various reactions are possible between sodium and water,* depending on the relative amounts of reactant present initially. The major reactions and products are given in Table A1.

Table A1 — Major Reactions and Products

Reaction 1 occurs in excess water, giving end products of gaseous hydrogen and sodium hydroxide dissolved in the excess water. This is also the reaction that occurs with equimolar amounts of sodium and water.

Reaction 2 occurs in excess sodium where a further reaction is possible between the excess sodium and sodium hydroxide, giving sodium oxide and additional hydrogen. In excess sodium, the hydrogen liberated by the first two reactions can also combine with sodium to give the hydride, as shown in Reaction 3. The hydride decomposes at temperatures above 700°F.

As part of the early work on the SDR program, a critical review⁶ was made of the literature on sodium-water reactions. The major conclusions of this review were:

1. The reaction rate of sodium with water is essentially controlled by the contact surface available.
2. Attempts to inhibit the reactions by chemical means have failed.
3. A small expansion volume in either fluid system will eliminate shock-wave formation.

* From a thermodynamic standpoint, heavy water and light water are similar.

4. Pressure buildup can be relieved by providing an adequate expansion volume and conventional venting devices.
5. Temperature effects are relatively minor in sodium-water reactions.

Experience with sodium-water reactions indicates that the reaction is not explosive, and that severe shock waves do not occur. The maximum pressure peaks from a sodium-water reaction occur 2 to 5 millisecon after mixing of the fluids takes place; high explosives, however, generate peak pressures within microseconds of detonation. Thus, for the direct sodium-water reaction, where hydrogen is liberated, the usual design basis for safety calculations is the maximum equilibrium pressure developed from a reaction.

HYDROGEN-OXYGEN REACTION

When evaluating the hazard from a sodium-water reaction, the hydrogen-oxygen reaction must be considered if any air is present. The reaction

The hydrogen-oxygen explosion theoretically can generate detonation or shock waves that travel thousands of feet per second.²⁴ The conditions for detonation are strongly dependent on the experimental variables as well as the amount of inert gas diluent present in the mixture.¹⁴

Theoretical detonation pressures have been obtained as a function of oxidant to fuel ratio. For an initial pressure of 1 atm, the maximum theoretical detonation pressure is 265 psia at the stoichiometric ratio of oxidant to fuel. Addition of small amounts of an inert gas diluent has a relatively minor effect on lowering of the peak pressure. Experiments by Ordin have shown that detonation can be prevented by adding sufficiently large amounts of inert gas to place the mixture out of the flammable range. For nitrogen as a diluent, the flammability limits are shown in Fig. A1, taken from Reference 24.

When the oxygen content of the original air is reduced to less than 5%, no combustion can occur. This 5% limit has been experimentally observed for many fuel combustion processes, including sodium combustion.¹⁵ Thus, reduction of the oxygen content to less than 5% in rooms where sodium is present will prevent both sodium combustion and hydrogen-oxygen explosions.

One of the ground rules of the SDR design is that all sodium equipment rooms, including the header rooms and reactor room, be filled with nitrogen maintained at less than 1% oxygen. The nitrogen is purified by passing through NaK traps that remove essentially all of the oxygen present. The very low limit on oxygen content is not determined solely by explosion limits, but also by the requirement that the spill-return system function adequately without oxide contamination of any leakage sodium.

ULTIMATE PRESSURE FROM A SODIUM-WATER REACTION

A generalized analysis has been made of the final pressure and temperature of the gaseous mixture following a sodium-water reaction for various initial expansion volumes. This analysis will form a useful design basis for containment calculations to be performed in a more complete hazards study. The analysis assumes complete thermal equilibrium following an accident in which various amounts of sodium react with a given amount of water, with the water in excess. The hydrogen liberated is also assumed to react completely with oxygen in the surrounding atmosphere; this neglects any effects of the nitrogen blanket gas present in the lower sodium rooms. No heat is transferred to the building or structure, resulting in a very conservative approximation of the final state of the gases in the vessel.²⁵

The results of this analysis are shown in Fig. A2. The branched portions of the curves represent complete evaporation of the excess water, while the dashed portions represent a saturated equilibrium mixture for which additional water must be introduced in the building. Data, assumed initial conditions, and representative containment building sizes are given in Table A2. The cylindrical buildings are assumed to have standard ellipsoidal heads.

Table A2 — Initial Conditions and Containment Building Sizes

Initial Conditions			
Air Pressure: Atmospheric			
Air Temp.: 70°F			
Relative Humidity: 50%			
Amount of D ₂ O assumed in calculations: 15 tons			
D ₂ O Temp.: 130°F			
Mean Sodium Temp.: 800°F			
Gas Volume, ft ³	Spherical Building	Cylindrical Building	
	Diameter, ft	Diam, ft	Height, ft
2.5×10^5	84	67	100
5×10^5	106	84	126
12.5×10^5	145	114	171
40.0×10^5	213	108	252

SODIUM-ORGANIC COOLANT REACTION

Use of an organic coolant such as Dowtherm-A in the upper and lower neutron shields poses no significant safety problem. In the unlikely event of a leak of organic coolant from the shields and of sodium from the primary loop, mixing of the two fluids would occur in the lower catch pans. A small experimental study verified the general non-reactivity of Dowtherm-A with sodium.

Hydrocarbons are usually unreactive in the presence of sodium metal.¹³ The presence of reactive groups (such as conjugated unsaturation; oxygen in the form of alcohol, aldehyde, ketone, and ether linkages; amino nitrogen; and halogens) may cause hydrocarbons to become reactive even though these groups are present in impurity molecules associated with the hydrocarbons. These reactive groups in the presence of sodium act as catalysts for all types of possible reactions ranging from reduction of unsaturated hydrocarbons to polymerizations, degradation, and ring formation.

Specific tests using biphenyl have shown no reaction with sodium. Milich and King found that biphenyl and NaK contained in type 304 SS show no reaction at 800°F. Similarly, biphenyl when reacted at 800°F with 200°F water appears to be stable. Biphenyl alone shows no gas evolution at 400°C in over 100 hr of exposure.

STABILITY OF DOWTHERM-A IN THE PRESENCE OF MOLTEN SODIUM

Dowtherm-A consists of a mixture of biphenyl and biphenyl oxide. Experiments performed at NDA in which biphenyl was boiled (495°F) in the presence of sodium metal resulted in the formation of a black adherent covering on the surface of the sodium, the boiling biphenyl remain-

ing crystal clear. This black crust appeared to be a protective film and was sufficiently hard that its removal by filtration should not be difficult.

Mixtures of biphenyl and biphenyl oxide approximating the composition of Dowtherm-A turned dark brown when boiled ($\sim 500^{\circ}\text{F}$) in the presence of sodium for 1 hr in an argon atmosphere. No rapid reaction was observed, however, indicating that biphenyl oxide is not reactive toward sodium. The change in color of the Dowtherm-A is normal for this material when exposed to elevated temperatures.

It is concluded that no safety hazard will exist between Dowtherm-A and sodium in the SDR. In the event of leaks of the two fluids at operating temperatures, no significant chemical interaction is anticipated.

Fig. A1 — Limits of flammability of mixtures of hydrogen, air, and nitrogen²⁴

Fig. A2 — Ultimate pressure rise following sodium-water reactions in containment buildings of various free volumes. V free volume in building; — pressure, assuming no extra water; --- pressure, assuming additional water present as water spray.

APPENDIX B

CALCULATION OF LEAKAGE RATES RESULTING FROM FAILURE OF A LOWER PIGTAIL

INTRODUCTION

An analytical method has been developed to estimate the sodium leak rate from a leak in a lower pigtail. Two cases are considered: the case where a leak develops in the pigtail, and the case where complete severing of the pigtail occurs. The SDR primary sodium system consists of an external system and an internal system of about 60 parallel flow lattice positions in the reactor core. The effect of a leak in one position on the flow characteristics of the overall system will be considered.

ANALYSIS CONCEPT

The hydraulic problem considered the electrical analog of the primary sodium system, treating each of the flow channels as a resistance in the circuit. The head loss in any portion of the piping can be written as

$$\Delta H = \sum CV^2/2g = Q^2 \sum C/2gA^2$$

where ΔH = head loss, ft

C = head loss coefficient

V = flow velocity, ft/sec

g = gravitational acceleration, ft/sec².

If R is defined as an analogous resistance to flow (sec²/ft⁵), then $\Delta H = Q^2 \sum R_i$, where $R = C/2gA^2$.

The flow division for two or more parallel paths can be related by an equivalent resistance, R_e , as shown below.

$$\Delta H = R_1 Q_1^2 = R_2 Q_2^2 = R_e (Q_1 + Q_2)^2$$

$$R_e = R_2 \left(\frac{Q_2}{Q_1 + Q_2} \right)^2$$

$$\frac{1}{R_e} = \frac{1}{R_2} \left(\frac{Q_1 + Q_2}{Q_2} \right)^2 = \frac{1}{R_2} \left(1 + \frac{Q_1}{Q_2} \right)^2 .$$

Since

$$\frac{Q_1}{Q_2} = \frac{R_2}{R_1}$$

$$\frac{1}{\sqrt{R_e}} = \frac{1}{\sqrt{R_2}} \left(1 + \frac{\sqrt{R_2}}{\sqrt{R_1}} \right) = \frac{1}{\sqrt{R_2}} + \frac{1}{\sqrt{R_1}} .$$

If $R_1 = R_2$, then

$$\frac{1}{\sqrt{R_e}} = \frac{2}{\sqrt{R}}$$

and $R_e = R/2^2$; similarly if n equal resistances are in parallel, the equivalent resistance, R_e , is

$$R_e = \frac{R}{(n)^2} .$$

Case I – Small Leak in Lower Pigtail

Using this technique for lumping parallel resistances, the equivalent flow circuit for the SDR core is shown below.

where R_A, R_B, R_C, R_D, R_E = equivalent system resistances

Q_A = total flow from pump

Q_B = total flow to one of two main headers

Q_X = leakage flow.

Small resistances in various locations have been ignored or lumped in R_A, R_B, R_C, R_D , and R_E .

Three independent equations coupling the flows and resistances are obtained by traversing the three flow loops, viz., the two parallel loops involving the main pump, and the loop from the expansion tank out through the leaking pigtail. The three pertinent equations are:

$$\Delta H = R_A Q_A^2 + R_D (Q_A - Q_B)^2 \quad (1)$$

$$R_B Q_B^2 + R_C (Q_B - Q_X)^2 = R_D (Q_A - Q_B)^2 \quad (2)$$

$$\frac{\Delta p}{\rho} + \Delta Z - \frac{1}{2g} \frac{Q_X}{\left[D A_p \left(\frac{A_X}{A_p} \right) \right]^2} = R_E Q_X^2 - R_C (Q_B - Q_X)^2 + \frac{1}{2g} \left[A_p \frac{Q_X}{\left(\frac{A_X}{A_p} \right)} \right]^2 \quad (3)$$

where Δp = pressure differential between expansion tank and leak exit, psi

ρ = density of sodium, lb/ft³

ΔZ = elevation head between expansion tank and leak exit, ft

A_X = leak area, ft²

A_p = flow area in pigtail, ft²

D = discharge coefficient at vena contracta (assumed 0.6).

These equations were solved by a graphical method, and values of the unknown flows were obtained as functions of the parameter A_X/A_p for the actual pump discharge pressure of 50 psi. Fig. B1 gives the results of the calculation, showing that flow stagnation occurs when $A_X/A_p = 0.94$, corresponding to a 2.1 in.² hole in the pigtail. Leaks of this size are extremely unlikely and for more reasonable leak sizes, ($A_X/A_p < 0.1$) the leak rate is small and the resultant loss of flow to the channel is insignificant.

Case II – Completely Severed Pigtail

In a similar fashion, the analysis was carried out for the completely severed pigtail. Here, flow reversal does occur in the coolant channel, and the equivalent flow network is shown below.

Again, the three flow equations are written as

$$\Delta H_p = R_A Q_A^2 + R_D (Q_A - Q_B)^2 \quad (4)$$

$$\frac{1}{2g} \left(\frac{Q_B^2}{A_p^2} - \frac{1}{2g} \right) \left(\frac{Q_X - Q_B}{A_p} \right)^2 = R_D (Q_A - Q_B)^2 - R_B Q_B^2 + R_C (Q_X - Q_B)^2 \quad (5)$$

$$\frac{\Delta p}{\rho} + \Delta Z - \frac{1}{2g} \left(\frac{Q_X - Q_B}{A_p} \right)^2 = R_E Q_X^2 + R_C (Q_X - Q_B)^2. \quad (6)$$

Only one unique solution is obtained from this set, since the leak area of the severed pigtail is fixed. The sum of Q_C and Q_B which corresponds to the leak rate is 0.72 ft³/sec, or about 322 gpm. This is assumed to be the maximum credible leak rate for a severed pigtail. As a secondary result, the reversal flow (Q_C) through the failed lattice position is 0.19 ft³/sec, or about 95 gpm. This could lead to overheating in the position unless a scram has occurred. The net effect of a severed pigtail on flow distribution to the other lattice positions is essentially negligible and no overheating is anticipated in the other channels.

Fig. B1 — Leakage flow rate and flow rate in affected lattice position vs area ratio of leak to pigtail

APPENDIX C

CALCULATIONAL METHODS FOR REACTOR TRANSIENT DATA

CALCULATION OF FUEL-COOLANT TEMPERATURE COEFFICIENT

The fractional change in k caused by an increase in fuel and coolant temperature is given by:

$$\frac{1}{k} \frac{dk}{dT} \approx \frac{1}{f} \frac{df}{dT} + \frac{1}{p} \frac{dp}{dT}$$

where very small changes due to ϵ , η , and leakage are neglected.

Increasing coolant (sodium) temperature decreases its density, reducing parasitic absorption.

Since, by definition

$$f = \frac{\Sigma_a^u}{\Sigma_a^u + \Sigma_a^{\text{other}} + \Sigma_a^{\text{Na}}}$$

then,

$$\frac{1}{f} \frac{df}{dT} = -f_{\text{Na}} \frac{1}{\rho_{\text{Na}}} \left(\frac{d\rho}{dT} \right)_{\text{Na}}$$

At 800°F,

$$\frac{1}{\rho_{\text{Na}}} \left(\frac{d\rho}{dT} \right)_{\text{Na}} \approx -2.9 \times 10^{-4}/^{\circ}\text{C},$$

and f_{Na} for the reference reactor was calculated to be 0.03418, so that

$$\frac{1}{f} \frac{df}{dT} = +1 \times 10^{-5}/^{\circ}\text{C}.$$

An increase in fuel temperature effectively broadens the U^{238} resonances (Doppler effect) and thus reduces p . If resonance escape, p , is defined as:

$$p = e^{-1/W}$$

where

$$W = \frac{N_m V_m (\bar{\xi} \sigma_s)_m}{N_F V_F / \sigma_a^F \text{ eff } dE/E} F + X, \quad \text{or } W = \frac{C}{\phi_{\text{res}}} + X,$$

where

$$\phi_{\text{res}} = \int \sigma_a^F \text{ eff } dE/E \text{ and } C = \frac{N_m V_m (\bar{\xi} \sigma_s)_m}{N_F V_F} F,$$

then the derivative of p with respect to temperature, T , is

$$\frac{dp}{dT} = \frac{1}{W^2} p \frac{dW}{dT} \text{ and } \frac{dW}{dT} = -(W-X) \frac{1}{\phi_{\text{res}}} \frac{d\phi_{\text{res}}}{dT}.$$

Finally,

$$\frac{1}{p} \frac{dp}{dT} = (\ln p)^2 (1/\ln p + X) \frac{1}{\phi_{\text{res}}} \frac{d\phi_{\text{res}}}{dT}.$$

Experimental values from the literature show a reasonable value for

$$\frac{1}{\phi_{\text{res}}} \frac{d\phi_{\text{res}}}{dT}$$

to be $2 \times 10^{-4} (2R-1)/^\circ\text{C}$

where R = fraction of the total effective resonance integral that results from volume absorption (as opposed to surface absorption).

For the reference reactor

$$R = \frac{\text{volume contribution to effective resonance integral}}{\text{surface contribution} + \text{volume contribution}}$$

$$= 0.75$$

$$p = 0.96$$

$$X = 0.676$$

so that

$$\frac{1}{p} \frac{dp}{dT} = -0.4 \times 10^{-5}/^\circ\text{C}$$

Combining these two temperature effects we get:

$$\frac{1}{k} \frac{dk}{dT} = +1 \times 10^{-5}/^\circ\text{C} - 0.4 \times 10^{-5}/^\circ\text{C}$$

$$= +0.6 \times 10^{-5}/^\circ\text{C}$$

$$= +1.08 \times 10^{-5}/^\circ\text{F}.$$

Since the Doppler effect is practically instantaneous, to simply add these two effects is certainly an overestimate of this positive coefficient of reactivity.

CALCULATION OF D₂O TEMPERATURE COEFFICIENT

If one chooses to define reactivity (actually k_{eff}) as the ratio of the number of neutrons per fission observed experimentally ($\nu=2.46$ for U²³⁵) to the number required to make the reactor critical, then in principle the reactivity of any reactor under a given set of conditions can be calculated.

Two reactor criticality calculations were performed using D₂O temperatures of 68 °F and 135 °F respectively, with the appropriate two-group cross section changes at these temperatures. The change in effective neutron temperature was included. The sodium temperature was 800 °F in both cases.

The corresponding change in reactivity for these two reactors over this temperature range gives a value of $-3.78 \times 10^{-5}/^{\circ}\text{F}$ for the D₂O temperature coefficient.

CALCULATION OF PROMPT NEUTRON LIFETIME

The method³¹ used to calculate the neutron lifetime is that of introducing a small $1/v$ absorber uniformly throughout a critical reactor and then calculating the corresponding reactivity change.

A near critical reactor ($k_{\text{eff}}=1.0224$) was calculated. To this reactor an arbitrary, small amount of a $1/v$ absorber was added uniformly to all regions, both to the fast and thermal absorption cross sections, resulting in a $k_{\text{eff}} = 0.99917$.

The prompt neutron lifetime, ℓ , is related to δk and $\delta \Sigma_a$ in the following manner:

$$\ell = \frac{\sqrt{\pi}}{2} \frac{(\delta k)}{\Sigma_{a\text{th}} v_{\text{th}}},$$

where the $\Sigma_{a\text{th}}$ represents the added $1/v$ absorber. This procedure gave a value for $\ell = 6.6 \times 10^{-4}$ sec. An estimate of the average fast velocity is necessary in the above analysis and was found to be roughly $10v_{\text{th}}$. For this calculation, v_{th} was calculated to be 2400 m/sec, corresponding to a mean neutron energy of 0.03 ev.

REACTIVITY EFFECTS WITH TIME

Temperature Coefficient of Sodium

Long-term reactivity calculations based on NDA's Lottery Code (which solves the standard isotopic buildup equations in flux-time), shows that the absorption cross section of a fuel cluster decreases with time. This is due to the buildup of Pu²³⁹ and fission products and the U²³⁵ burnout.

The calculations gave:

$\Sigma_{a\text{cluster}}$ at start of life: 0.1434

$\Sigma_{a\text{cluster}}$ at end of life: 0.1037

Since the sodium utilization varies inversely as the $\Sigma_{a\text{cluster}}$ the sodium temperature coefficient increases to the value $+1.3 \times 10^{-5}/^{\circ}\text{C}$ at the end of life.

Zirconium and steel performed well, as was expected. While graphites in general performed poorly, AGOT-type (reactor grade) performed well enough to indicate that it might be developed into a suitable high-temperature barrier. Table D1 gives details of the more interesting tests.

MULTIPLE TUBE CONFIGURATION TESTS

The test section simulates a 24 in. length of SDR lattice position. A simulated fuel-coolant tube (~3 in. OD), filled with hot stagnant sodium, is encircled by the concentric barrier assembly to be tested. Outside the barrier is placed an annular aluminum water jacket, which simulates the calandria tube and its surrounding D_2O . The fuel-coolant tube and the water jacket each contain flush nozzles (approximately 0.093 in. bore), through which sodium and water, respectively, can be squirted radially into the barrier spaces. This test section is mounted in a containment vessel to which are piped the sodium and water required by the nozzles, together with inert cover gases and other service instrumentation. Tests can be operated for approximately 30 min at a sodium flow rate of 1 gpm, and temperatures of more than 1200°F at the sodium nozzle can be maintained. The two nozzles were directed radially toward each other for all tests reported here.

Table D2 presents results of tests of several barrier combinations, as well as results of experiments using no barriers. None of the barriers tested was penetrated by single jets of sodium or water, or by simultaneous jets of both fluids.

Some tests were run without any barrier present. Single jets of sodium (Test 1-5) or water were directed in the barrier space; neither the fuel-coolant nor water-backed calandria tubes were penetrated.

In Test 2-4, which was conducted at a temperature 50°F higher than the melting point of aluminum, the barrier was penetrated. Even at this temperature, however, the water-backed calandria tube maintained its mechanical integrity.

During the course of the experiments, some unplanned, direct mixing of sodium and water occurred within the test region. These mixings were not the result of tube configuration failures, but rather the result of test apparatus malfunctions. In one run, for example, sodium did not drain properly. Significantly, although a sodium-water reaction occurred, no damage to the barrier tube, fuel-coolant tube or calandria tube resulted. All components were used in subsequent tests.

The tests have demonstrated that the multiple-tube configuration will maintain isolation of sodium and water, even after breaching of any two of the three tube components. It has been further demonstrated that under conditions of actual mixing of sodium and water in the multiple-tube region (which in practice would require simultaneous failure of all three tubes) no mechanical damage to the system occurs.

TUBE AND HEADER TESTS

A program was conducted on various critical welded piping joints in the vicinity of the SDR core. The joints involved are tube-to-tube joints along the fuel-coolant tubes, and T-joints at both ends of the expansion piping joining the fuel-coolant tubes to headers.

The program was divided into three parts:

1. a weld development study to establish relative ease of fabrication of several proposed types of welds,
2. tensile tests of the tube-to-tube welds,
3. static bending tests of the T-joint welds, followed by fatigue tests of these welds.

The fatigue tests constituted the major effort. Both the static bending and fatigue tests were studied analytically, and the fatigue tests results were compared with existing information. The

test apparatus was mounted on a rigid steel frame. The test tee was clamped to the frame, heaters applied, and the pigtail deflected cyclically through predetermined deflections.

The tensile tests indicated that the weld types considered are approximately equivalent to each other in strength, and that all have higher tensile strength than the parent material. Test results were found to correlate reasonably well with existing information. It was found that a design lifetime of 10,000 cycles would permit an allowable stress of 29,100 psi. For a design stress of 16,500 psi, (which is used in the Chugach SDR) the corresponding safety factor is 1.8. At 16,500 psi a lifetime of 100,000 cycles before failure was predicted from the data.*

MOCKUP OPERATION

The previous component tests were applied to the designs of a reactor mockup. This integrated test of the performance of the entire sodium-water isolation system was made as a final measure of the reliability of the SDR design. The critical components of the sodium and water systems were assembled into a full-scale mockup and tested under simulated normal and aggravated operating conditions.

The mockup encompasses three lattice positions of the SDR. It includes the multiple-tube configuration, the sodium pigtail and header joints, and the heavy water calandria. Auxiliary apparatus is provided to supply (1) hot liquid sodium to the test section, (2) heated water to the test section, (3) cover gas for liquids and inert gas for barrier spaces and the upper and lower header boxes, (4) a leak detection system for both sodium and water in the barrier space.

By September 30, 1958 the mockup has successfully logged 2779 hr with sodium at 950°F and water at 150°F circulating in their respective systems. Unscheduled downtime was less than 0.5% (due to failure of a fuse). In addition, the test section was subjected to simulated emergency reactor operating conditions, including design pressures and temperatures higher than design, and rapid water dumps. During the entire operation, including both normal and abnormal operating conditions, there was not a single breach of any component of the sodium-water isolation system.

CONCLUSION

The basic conclusion drawn from the experimental program is that the materials selected and configurations developed are effective and reliable in maintaining isolation of the sodium and water systems in the SDR. Furthermore, as a result of complete isolation, the entire reactor is at low temperature except for the isolated fuel-coolant tubes, which are free to expand without restraint. This, combined with the low pressure of the sodium and heavy water, permits a reactor design within conventional engineering practice.

* All stresses are given in terms of the stress produced in a completely elastic pipe under the applied deflection. The design stress of 16,500 psi is one half of the actual allowable stress of 33,000 psi, disregarding fatigue. (The allowable stress was obtained in accordance with methods in the ASA Piping Code.) The reduction factor of 0.5 applied to account for fatigue is quite conservative; according to ASA standards such a reduction is recommended only when more than 250,000 cycles are anticipated.

Table D1 — Single-Failure Test Results

Specimen No.	Material	Thickness, in.	Jet Velocity, fps	Jet Temp., °F	Test Duration	Remarks (no failure unless noted)
11E	Al-1100	0.010	40	950	15 min	
13G	Al-1100	0.060	50	950	18 hr	Tiny pits at point of im- pact visible to naked eye
17A	Graphite (AGOT) perpendicular to grain)	0.250	50	950	15 min	
24A	Graphite (Nat. Carbon Code 82, perpendicu- lar to grain)	0.250	50	950	15 min	Intact, many cracks
Al foil	Al foil	0.005	50	950	See remarks	Punctured at point of impact after several minutes operation
31B	Al sprayed with 4 mil 18-8 SS	—	50	950	15 min	Warped, blistered
13K	Al-1100	0.060	40	1055; 1140	3 hr, 40 min; 20 min	Some pits in region of impact
13M	Al-1100	0.060	50	1150	3 hr, 30 min	Large hole eroded in re- gion of impact

Table D2 — Barrier Configuration Results

Test	Purpose	Sodium Squirt		H ₂ O Squirt		Test Duration, min	Results
		Velocity, fps	Temp., °F	Velocity, fps	Temp., °F		
1-4	Simulated water leak on SS fuel-coolant tube at operating temperature	none	—	50	150	17	No damage. Hot sodium in fuel-coolant tube chilled rapidly.
1-5	Simulated sodium leak on water-backed aluminum calandria tube	73	1000	none	—	10	No damage. Slight etching of metal surface at point of impingement
1-7	Simulated sodium and water leak on barrier	60	1050	50	155	12	No penetration or damage other than $\frac{3}{8}$ in. distortion from vertical axis
2-1	Repeat test 1-7 to confirm reliability of results	60	1050	50	155	19	Results same as test 1-7. No penetration or damage at point of impingement. Approx. $\frac{1}{4}$ in. distortion from vertical axis
2-3	Simulated sodium leak on aluminum barrier at max. transient reactor temperature	55	1150	none	—	13	No penetration or damage to barrier
2-4	Simulated sodium leak on aluminum barrier well above reactor max. temp.	60	1250	none	—	3	Barrier penetrated after ~20 sec. Water-backed calandria successfully withstood sodium jet for remainder of test
2-5	Repeat test 2-3 with 10-15% water vapor in barrier gas	~60	1135	none	—	18	No penetration or damage to barrier

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