CLINCH RIVER

BREEDER REACTOR PROJECT

50-537

PRELIMINARY SAFETY ANALYSIS REPORT

VOLUME 5

PROJECT MANAGEMENT CORPORATION

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PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 5

HEAT TRANSPORT AND CONNECTED SYSTEMS

PROJECT MANAGEMENT CORPORATION

CHAPTER 5.0 - HEAT TRANSPORT AND CONNECTED SYSTEMS

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5.0 HEAT TRANSPORT AND CONNECTED SYSTEMS

The Heat Transport and Connected Systems include those systems and boundaries which provide the necessary functions to safely remove and transport reactor heat to the steam generators under all plant operating conditions. These systems include: the Reactor Vessel, Closure Head and Guard Vessel, the Primary Heat Transport System (PHTS), the Intermediate Heat Transport System (IHTS), the Steam Generator System (SGS), the Steam Generator Auxiliary Heat Removal System (SGAHRS), the Direct Heat Removal Service (DHRS), and interconnected systems providing support functions to these systems.

The overall heat transport system provides three separate and independent paths for the transfer of heat from the reactor to the Turbine-Generator. Each of these three paths is also able to transport decay heat from the reactor to an ultimate heat sink following reactor shutdown. Each of these paths includes a component loop of the PHTS, a component loop of the IHTS and a component loop of the SGS. Heat is removed from the reactor by the forced circulation of sodium through the core and into the three piping loops of the PHTS. Each PHTS loop contains piping, a pump, a cold leg check valve (CLCV), and an Intermediate Heat Exchanger (IHX). Each PHTS loop is located in a nitrogen-inerted cell in the Reactor Containment Building. In each IHX heat is tranferred from one PHTS loop filled with activated sodium to the non-activated sodium circulating through one of the three piping loops of the IHTS. Each IHTS loop contains piping and a pump for circulating sodium between the IHX and the Superheaters and Evaporators in the SGS. The IHTS loops extend from the Reactor Containment Building Cells, containing the PHTS loops, to the Steam Generator Building containing the SGS evaporators and superheaters. Each SGS loop contains piping, two evaporators, one superheater, a recirculating pump and a steam drum, with connections to the BOP (Turbine-Generator, the Turbine Bypass, and the Main Feedwater System (MFS) and to the Steam Generator Auxiliary Heat Removal System (SGAHRS). In the evaporators, heat is transferred from the sodium in the IHTS loop to a steam/water mixture which circulates to and from the steam drum. In the superheaters heat is transferred from the sodium in the same IHTS loop to steam flowing from the steam drum to the turbine generator or turbine bypass. Condensate is returned from the main condenser to the steam drums via the MFS. The SGAHRS provides safety related alternate heat removal path from the SGS loops. These paths are used when the path through the Turbine Bypass becomes unavailable. SGAHRS includes safety related steam vent valves at each superheater outlet and each steam drum, one Protected Air Cooled Condenser (PACC) connected to each steam drum, and the Auxiliary Feedwater System (AFWS) for water inventory make-up to the steam-drums. The AFWS includes piping valves, a protected water storage tank (PWST) and 3 AFW pumps (2 motor driven and 1 turbine driven). Separate emergency power supplies are provided for each heat transport path to protect against the loss of power.

In addition to these three heat transport paths, a fourth, separate shutdown heat removal path is provided by the Direct Heat Removal Service (DHRS). The DHRS consists of two sequential loops. The first loop takes primary sodium from the reactor vessel through the overflow nozzle into the overflow vessel.

Dual (parallel) electro-magnetic pumps force the sodium from the overflow vessel through the Overflow Heat Exchanger (OHX) and back into the Reactor Vessel. In the OHX heat is transferred to the second loop which uses NaK as the process fluid. This loop contains dual (parallel) pumps and air blast heat exchangers.

Conditions for normal operations neglect the effects of heat gained by the system due to pump work and heat loss by the system through insulation losses, sodium purification systems and the heat loss from the Protected Air Cooled Condenser (PACC). When these effects are accounted for, it is shown that the heat transport requirements of each primary loop is slightly less than the 325 MW design basis. The same is true for each loop of the Intermediate Heat Transport System. The preliminary heat balance of the nuclear steam supply system (NSSS) is shown in Figure 5.1-1A. This balance indicates that when 975 MWt is transferred from the NSSS to the balance of plant (BOP), the required net reactor thermal power (total flowrate times the difference in enthalpies corresponding to the temperatures of the reactor outlet and inlet flows) would be 967.4 MWt. The thermal power transferred from the primary sodium to the intermediate sodium in the IHX would be 970.6 MWt. The difference between the design value of 975 MWt for the reactor and the 967.4 MW reactor power shown in Figure 5.1-1A represents a small (0.79%) margin in the rating of the reactor. If the NSSS were operated as shown in the figure, the PHTS hot-to-cold leg temperature difference would be 2.1oF less than the Thermal Hydraulic Design Value of 265oF. Final values for insulation losses and pump efficiencies (which will affect pump work) are expected to modify these values slightly but the present design basis for the reactor, IHX and steam generators will not be affected.

Following a normal scram, the turbine trips, the turbine bypass valves open, the PHTS and IHTS main pump motors trip, pony motors engage to drive the pumps, the PACCs louvers open and PACC fans start. The main feedwater pumps continue to provide feedwater to the steam drums. When the steam drum pressure falls to 1450 psig, the turbine bypass valves close and the total heat load is transferred to the PACCs which control steam drum pressure to 1450 psig and provide stable long term shutdown heat removal.

Off normal shutdown events could involve either the loss of off-site power or the loss of one or two of the heat transport paths. Upon loss of off-site power, emergency power supplies provide the required power to the PHTS and IHTS pony motors, to the SGAHRS vent valves, to the PACCs and to the AFWS. The turbine bypass and the main feedwater pumps are not available so the residual shutdown heat is removed through SGAHRS vent valves and the PACCs. The AFWS provides water inventory make-up to the steam drums for loss due to venting through the SGAHRS vent valves during initial pressure let down. Should there be a further loss of power from an emergency diesel generator concurrently with the loss of off-site power, only the heat transport path served by the affected diesel generator would be affected and shutdown heat removal would continue through the other two heat transport paths. In fact, a single loop of the HTS and the SGAHRS has sufficient capacity that reactor residual heat can be removed and the reactor cooled to a safe shutdown using that path alone. Additionally, capability for natural circulation is included in the HTS, and this provides a diverse means for heat transport from the reactor to ultimate heat sinks.

Principal features of the PHTS include a physical arrangement which ensures that, with the exception of the connection to the reactor vessel, each loop is physically separated from, and cannot physically interact with, any other loop. The major section of each loop, including the pump, CLCV, and IHX, is located in a separate inerted cell. Features of design, including guard vessels surrounding each major component (reactor vessel, PHTS pump, and IHX) and the elevated CLCV and piping arrangement between components, serve to ensure that, in the unlikely event of a breach of the primary coolant boundary, the resultant loss of coolant will not result in an unacceptable lowering of coolant level in the reactor; the reactor vessel coolant level will always remain sufficiently above the PHTS hot leg outlet nozzle to assure hydraulic integrity. Thus, even with a pipe break in one PHTS loop, the remaining two PHTS loops remain functionally unaffected.

Principal features of the IHTS include total separation in separate cells for each loop piping and pump. Likewise, the components (evaporators, superheater and steam drum) of each SGS loop are physically separated in different cells from the components of each of the other SGS loops.

Thus, the independence of the three heat transport system paths is attained. Each path is sized for sufficient capacity to accommodate reactor shutdown heat removal and maintain safe shutdown by itself, providing full redundancy of shutdown heat removal paths.

Principal features of the SGAHRS include a safety grade AFWS, with two motor driven pumps and one steam turbine driven pump, a seismically qualified, safety related water supply tank, and a closed water loop to the PACCs. The AFWS is designed to assure adequate water make-up flow for loss of inventory due to vent valve operation concurrent with a feedwater pipebreak. The Protected Water Storage Tank (PWST) is designed with a capacity to assure an adequate supply of make up water to accommodate concurrent losses from vent valve operation, a feedwater pipebreak, and expected system leakage.

Principal features of the DHRS include a diverse secondary side process fluid (NaK), separate and redundant primary and secondary side pumps, and separate air blast heat exchangers.

This chapter presents summary descriptions of these systems and boundaries in Sections 5.1.1 through 5.1.6; Section 5.1.7 outlines the features incorporated for system safety. The detailed descriptions of the systems and boundaries are given in Sections 5.2 through 5.6. Also included in this chapter is an overall heat transport system evaluation (Section 5.7).

The systems listed below interconnect with the Heat Transport and Steam Generation Systems and are discussed in the PSAR section noted. Failures in those systems which affect the Heat Transport and Steam Generation Systems are discussed in Sections 15.3 and 15.7.

Cover gas systems (Section 9.5)
Auxiliary Liquid Metal Systems (Section 9.3)
Electrical systems (Section 8.3)

Instrumentation and Control Systems (Chapter 7.0)
Steam, Feedwater and Power Conversion Systems (Chapter 10.0)
Piping and Equipment Electrical Heating and Control System (Section 9.4).

5.1 SUMMARY DESCRIPTIONS

5.1.1 Reactor Vessel, Closure Head and Guard Vessel (See Section 5.2)

The reactor vessel and closure head form the major portion of the primary containment system for the reactor core. As such, they see duty both as part of the Primary Heat Transport System and as the first line of defense for providing the degree of safety required to protect the general public. Thus,
58 these two structures were designed, and are being fabricated, tested, and installed to the most stringent codes and standards in order to obtain a very low probability of malfunction during service life. The guard vessel which surrounds part of the vessel and the inlet and outlet piping up to the second elbows assures even greater safety by providing for the retention of the sodium coolant during ponymotor flow and natural circulation in the event of a leak in this portion of the primary coolant boundary.

The reactor vessel is a top ring-supported cylindrical structure with a torispherical bottom head. It is 59 feet long with a diameter of 20 feet.
58 The sodium-containing portion is all stainless steel designed for up to $1100^{\circ}F$ in the outlet plenum region and $775^{\circ}F$ in the inlet plenum region. The top flange of the vessel and the vessel support ring are fabricated of SA 508 Class 2 low-alloy forgings. There is an Inconel 600 transition section between the low-alloy forgings at the top and the stainless steel in the remainder of the vessel.

The closure head consists of three massive eccentric rotating plugs contained within the top flange of the reactor vessel. These plugs will be fabricated from a low alloy steel, SA 508 Class 2, and are 22.0 inches thick. They are inter-connected by means of a series of plug risers with sealing accomplished by sodium dip seals and double inflatable seals. The nominal temperature of the closure head is 400°F. Over 4 feet of thermal and radiological shielding extends beneath each rotating plug. Argon serves as a cover gas for the system.

The guard vessel is a bottom-supported cylindrical vessel fabricated from 304 stainless steel. It conforms to the contours of inlet and outlet piping and the reactor vessel to such height as to assure outlet nozzle submergence in sodium in the event of a leak. The space between guard vessel and reactor vessel is adequate for in-service inspection. Exterior thermal insulation is provided to limit the heat load into the reactor cavity cell. A heating system for the reactor vessel is mounted on the guard vessel.

5.1.2 Primary Heat Transport System (See Section 5.3)

The Primary Heat Transport System (PHTS) consists of three parallel independent loops of piping and components required to transport heat from the reactor to the Intermediate Heat Exchangers (IHX). The PHTS coolant boundary is classified as Safety Class 1 commensurate with its importance to safety*. This classification requires the additional code classification of ASME Section III Class 1. The guard vessels are classified as Safety Class 2 and ASME Section III Class 2, but have been optionally upgraded designed and constructed to Class 1 requirements. The guard vessels are not code stamped. The PHTS loops transport the radioactive sodium coolant from the reactor vessel to the intermediate heat exchangers which thermally link the primary and intermediate loops. The three primary loops have common flow paths through the reactor vessel, but are otherwise independent in operation.

As shown in Figure 5.1-1, heated sodium flows from the reactor vessel outlet nozzle located above the reactor core to the suction of a free surface centrifugal pump. Sodium from the pump discharge is circulated through the shell side of the intermediate heat exchanger where heat is transferred to the intermediate sodium. From the intermediate heat exchanger, the primary sodium flows through the cold leg piping through a check valve to the reactor vessel inlet nozzle located near the lower end of the reactor vessel. The three loops are essentially identical and have been arranged to provide equal sodium transport times. Figure 5.1-1B shows the arrangement of the PHTS piping and its relationship to the reactor vessel, the IHXs and the PHTS pumps and other loop equipment. The system Piping and Instrumentation Diagrams (P&ID) are shown in Figures 5.1-2a and 2b.

*The PHTS is part, but not all of the Reactor Coolant Boundary as defined in Section 3.1. The IHX is a component of the PHTS. Other systems, part or all of which are included in the Reactor Coolant Boundary, are the Auxiliary Liquid Metal System (see Sections 5.6.2 and 9.3) and the cover gas system (see Section 9.5). However, the Cover Gas System and the primary sodium cold traps are classified less than Safety Class 1 consistent with the provisions of 10 CFR50.55a, Footnote 2.

Transfer of partial power by the heat transport system is achieved by varying system flow essentially in proportion to the power produced in the reactor.

The sodium flow rate is controlled by changing pump speed and is measured by a magnetic flowmeter located in the cold leg piping downstream of the IHX, between the check valve and the reactor vessel.

The principal thermal-hydraulic design parameters are given in
Table 5.1-1; the piping and instrumentation diagrams are shown in Figures 5.1-2a
53 and 2b. These figures show the system interfaces, instruments, controls, and
piping sizes.

As shown on the HTS Hydraulic Profile (Figure 5.1-3), the centerline
of the high point primary piping is at -19'-4" (elevation 796'-8") for the cold
40 leg and -18'-4" (elevation 797'-8") for the hot leg. The reactor vessel sodium
58 level is -21'3" feet (elevation 794 feet-9 inches). The high point piping is
located above the reactor vessel normal sodium level to permit sodium drainage
of a primary loop without lowering the reactor vessel sodium level.

The primary system hot leg piping from the reactor vessel outlet to the primary pump is 36 inches outside diameter x 0.500 inches wall thickness. The remaining hot leg piping from pump discharge to the IHX is 24 inches outside diameter x 0.500 inch wall thickness and the cold leg piping from the IHX to the reactor vessel inlet nozzle is 24 inches outside diameter x 0.500 inch wall thickness. Sizing of the primary system piping includes the following considerations:

- The sodium velocity in the primary loop piping at maximum flow is less than 30 ft/sec.

- An allocated pressure drop for the primary loop piping consistent
  with pressure drop allocations for the system components and the
  58 reactor and the maximum total developed head limitation of 458 feet
  for the single stage primary pump.

The primary system hot leg piping is type 316 stainless steel. The
40 cold leg piping is type 304 stainless steel. Welded pipe connections are used
throughout the system.

Curved pipe and elbows having a minimum bend radius of 1-1/2 pipe diameters are used for changes in direction of the piping system. The spacing of parallel runs of pipe and the spacing between pipe runs and walls, ceiling and components, provide adequate room for installation, inspection and maintenance, insulation, heaters and thermal expansion. Thermal expansion of the system is accommodated by loops and bends. Pipe support is provided by constant load hangers sized to support the insulated piping system when filled

with sodium. Mechanical snubbers are provided to limit pipe deflections during
58 seismic loading. Piping penetrations through equipment cell wealls are provided
with flexible seals. Piping penetrations through the reactor containment walls
are provided with rigid seals.

Table 5.1-2 lists the PHTS volumes and volume changes. The volume changes resulting from system temperature changes are accommodated by the Auxiliary Liquid Metal System overflow and makeup circuit.

Primary loop cover gas normal operating pressure is maintained at
47 6 ± 2 inches W.G. by the makeup and vent of argon cover gas (see Sec-
tion 9.5). The design of the pump is based on the concept for sodium level
control using the "standpipe-bubbler" system in which a continuous flow of
inert gas is supplied to the pump cover gas space. During pump shutdown or low
sodium flow conditions, the gas supply pressure will be sufficient to depress
the liquid sodium level to the standpipe nozzle elevation and the gas will bubble
up the standpipe to the reactor cover gas system. During higher sodium flow
rates where the normal draw-down almost uncovers the standpipe nozzle, the gas
will flow through the nearly empty standpipe to the cover gas system. A fixed
reference sodium level is maintained within the reactor vessel approximately
58 16 1/4 feet above the centerline of the outlet nozzles by means of sodium makeup
and overflow. 33

To preclude potential cover gas accumulation in the IHX, there is con-
tinuous venting of sodium from the top of the IHX to the primary pump tank via
the IHX vent return line. This sodium flow (~200 gpm at 100% primary sodium
flow) will carry any gas evolved in the IHX to the pump tank where the gas
will migrate to the cover gas space. The sodium flow is monitored by a
53 permanent magnet flowmeter. A trap in the line is provided to prevent cover gas
from being transferred to the IHX and breaking siphon should a primary boundary
3 leak cause the pump tank level to fall below the vent line nozzle. The IHX vent
return line is a 2 inch schedule 40 pipe of Type 316 stainless steel.

The primary heat transport system (PHTS) piping and components are located within the Seismic Category 1, tornado and missile hardened reactor containment building. The components and piping for each loop are located within three vaults (cells) in the reactor containment building: (1) an HTS cell which contains all of the major loop components, (2) an HTS pipeway, and (3) the reactor cavity which houses the reactor vessel and the associated primary loop piping. The cells are separated from each other by concrete shielding walls and are inerted with nitrogen which is circulated for cooling. Those parts of the PHTS equipment which come in contact with sodium are located in a nitrogen atmosphere below the level of the containment building operating floor. Each HTS cell has a separate atmosphere and the reactor cavity and the HTS pipeways have a common atmosphere. The pump drive systems (motors, speed controllers, and heating and seal assemblies) are located in an air environment above the operating floor. Separation of the equipment cells provides the capability of deinerting individual vaults for independent access for maintenance or inspections.

The intermediate heat exchangers and primary sodium pumps are rigidly supported to maintain fixed centerline locations and are enclosed by free standing, structurally independent guard vessels. Differential thermal expansions are accommodated by the inherent flexibility in the piping arrangement and by gaps existing between the major components and the guard vessel. The elevation of the lip (top) of the guard vessels is 7 feet above the minimum reactor vessel nozzle submergence level to limit loss of coolant and assure continued decay heat removal capability. All primary sodium piping, which is at an elevation below the top of the guard vessels, is contained within the guard vessel. All uncontained sections of primary sodium piping, inter-connecting the major components are at an elevation above the top of the guard vessels in a near horizontal plane.

There is no piping connection between the primary and intermediate sodium loops. The intermediate heat exchanger tube bundle isolates radioactive primary sodium from the intermediate sodium and provides a mechanical barrier to the transport of radioactive sodium out of containment. Intermediate sodium pressure within the IHX is maintained at a minimum of 10 psi above the shell side (primary) sodium so that any tube leaks will not result in leakage of radioactive sodium into the intermediate system. The intermediate sodium piping and components provide a second barrier to the release of radiation to the environment. This arrangement minimizes overall biological shielding requirements and permits design flexibility for locating the intermediate system components and the steam generators outside of containment in accessible, unshielded areas.

While the arrangement of the primary and intermediate loops is such that natural circulation will provide sufficient heat transport for safe decay heat removal in the event that all pump power is lost, the pumps are equipped with pony motors supplied with normal and emergency power to provide forced circulation decay heat removal. The pony motors are designed to provide ~7-1/2% of the system design flow. Figure 5.1-3 illustrates the elevational differences which provide the necessary flow potential for natural circulation decay heat removal capability.

5.1.3 Intermediate Heat Transport System

The Intermediate Heat Transport System (IHTS) consists of the piping and components required to transport reactor heat from the Intermediate Heat Exchanger (IHX) to the Steam Generator System. The IHTS and its components are classified as Safety Class 2. The corresponding ASME Section III Class 2 classification has been upgraded to Class I. The IHTS and its components will be designed, fabricated
35 and tested in accordance with ASME Section III Class I.

The system is comprised of three essentially identical, independent, cooling loops operating in parallel. As shown in Figure 5.1-1, each intermediate loop contains non-radioactive sodium 1
which is circulated by an intermediate sodium pump from the tube side of the IHX through the steam generator and back to the IHX. No direct sodium interconnections exist between the three loops.

The IHTS sodium piping from the IHX is routed inside a cell in the containment building until it exits through a containment penetration. The remainder of the loop is housed within a seismic Category I, tornado resistant, missile hardened, accessible cell within the Steam Generator Building. The piping arrangement for the portion of the IHTS in the containment building is shown in Figure 5.1-1B. The arrangement of the remainder of the IHTS piping is shown in Figure 5.1-1C. 1

53 As shown in Figure 5.1-2a, heated intermediate sodium leaving the IHX flows through 24-inch O.D. hot leg piping and enters the Steam Generator System (SGS) superheater module. At the superheater exit, the piping consists of two parallel runs, each 18-inch and extends to each of the two evaporator inlets. Cooled sodium from the evaporators flows through two parallel 18-inch runs; joined together at a tee and continues as a single run of 24-inch O.D. pipe to the IHTS pump suction. From pump discharge, the sodium flows through 24-inch O.D. piping to complete the circuit through the IHX. All piping with the exception of the superheater to evaporator piping, has a 0.500 inch wall thickness. The superheater to evaporator piping wall thickness is 0.562 inch. The principal thermal hydraulic design conditions are shown in Table 5.1-1.

Each cold leg of the intermediate loop contains a vertical, single stage, free surface centrifugal pump. The sodium flow rate is controlled by a variable speed pump drive. A pony motor is used to provide 7-1/2% of design flow for operation during start-up, shutdown, hot stand-by and following accident conditions.

A permanent magnet flowmeter is installed in the cold leg downstream from the pump in loops 1 and 3 (west and east loops respectively) and in the hot leg of loop 2 (middle loop). This flowmeter provides sodium flow rates 29 for process control and plant protection functions. A venturi flowmeter is provided to measure the inlet sodium flow to each evaporator module in one loop of the plant. This flow measurement is part of the diagnostic instrumentation for the steam generators.

An expansion tank is provided in each IHTS loop cold leg to accommodate the sodium volume change due to thermal expansion over the operating range. Each tank is located in the steam generator building near its corresponding sodium pump. The system cover gas operating pressure is maintained by control of the argon gas supply. The cover gas volumes in the expansion tank and pump tank are interconnected. A vent-sodium stream for the superheater and each evaporator flows from the high point bundle of the module. This stream provides positive flow through the upper tube bundle of the module to prevent gas build-up and provide a source of sodium for sodium-water leak detection.

While each loop has similar components, the loop piping differs in length and configuration because of the differences in distance between the IHX units and the piping penetrations into the steam generator cells. Piping configurations within the steam generator cells are identical for the three loops. Within each loop, equal flow to the two evaporators is maintained through the use of identical piping between the superheater discharge and the evaporator inlets. The piping from the evaporators to the IHTS mixing tee and from the mixing tee to the pump is also identical in all loops. 1

The sodium volume change per loop which must be accommodated in
53 the pump tank and expansion tank is shown in Table 5.1-3. The system
29 is initially filled to the pump pony motor operating level at 400°F. As
the system is heated to normal operating temperature, the sodium expands
and the pump level rises to the normal operating level.

29 Four-inch sodium dump lines are provided on the intermediate
system piping low points to drain the system in the event of a steam
generator leak. These lines are connected to the sodium dump tank in the
Steam Generator Protection System. Rupture discs supplied as part of the
Steam Generator System are provided for pressure relief protection. 2

The piping and instrumentation diagram for the IHTS is shown on
53 Figure 5.1-2a. This diagram shows the system interfaces, instruments, controls, piping sizes and materials. Instrumentation for the IHTS system includes that normally required for plant control as well as diagnostic instrumentation for evaluating equipment and system performance.

Temperatures are monitored at selected points throughout the sodium systems. Sodium pressure differentials are measured across the intermediate pump, the IHX (primary to intermediate), and in one loop only, across each steam generator module. Argon pressure is measured at the inlet to the expansion tank/pump cover gas. Level instrumentation is provided in the intermediate pump casing and in the expansion tanks. Intermediate sodium equipment areas are monitored for sodium oxide smoke by an electronic system or by a continuous chemical check on the atmosphere. Localized sodium leak detectors are installed under the components and piping of the system.

The IHTS is isolated from the PHTS in a passive manner in that the intermediate sodium is confined to the tube side of the IHX. The IHTS is isolated from water and steam in the steam generators in a passive manner in that the intermediate sodium is confined to the shell side of the steam generator modules. Manually operated isolation valves are provided to isolate the IHTS from the oxygen/hydrogen detectors and the sodium dump tank. The IHTS piping is equipped with double rupture disks in series to provide passive isolation from the steam generator protection system.

The relative elevations of the IHX and the steam generator modules are arranged to provide the required natural circulation of sodium in the IHTS for decay heat removal. The hydraulic profile showing these elevations is shown on Figure 5.1-3.

5.1.4 Steam Generation System (See Section 5.5)

The Steam Generation System accepts hot sodium from the Intermediate Heat Transport Loops, extracts heat and returns cooler sodium back to the Intermediate Loops. The Steam Generation System transfers a nominal 975 MW from the Intermediate Heat Transport System sodium and delivers superheated steam at 900°F and 1450 psig to the turbine generator. Required full-load operating parameters are given in Table 5.1-4.

The piping and instrumentation diagram for the Steam Generator System is shown on Figure 5.1-4. The Steam Generation System provides independent steam generation capability for each of the three reactor heat transport loops. Each independent Steam Generation System loop is comprised of the following:

Steam Generator Evaporator/Superheater Modules

Steam/Water Subsystem

Sodium-Water Reaction Pressure Relief Subsystem

Water Dump Subsystem

Sodium Dump Subsystem

Leak Detection Subsystem

In addition to providing for the removal of heat from the Intermediate System sodium by the superheater/evaporator components, the system provides the steam supply to the turbines, provides pressure relief protection for a sodium water reaction, water from the Steam/Water Subsystem, and provides for detection of water-to-sodium leakage in the steam generator modules.

5.1.5 Residual Heat Removal Systems (See Section 5.6)

The Residual Heat Removal Systems provide decay and sensible heat removal capability for all shutdown conditions. These systems consist of the
26 Steam Generator Auxiliary Heat Removal System and the Direct Heat Removal Service which are discussed below.

5.1.5.1 Steam Generator Auxiliary Heat Removal System (See Section 5.6.1)

The Steam Generator Auxiliary Heat Removal System (SGAHRS) is a safety related system whose main function is to provide redundant shutdown decay heat removal paths when the main NSSS heat sink or main feedwater supply is unavailable. It also functions to remove shutdown heat loads for refueling and other long term outages. It is part of the Residual Heat Removal Systems discussed in Section 5.6.

Figure 5.1-5 shows a piping and instrumentation diagram for the SGAHRS. The system performs its functions using two subsystems - short and long term heat removal subsystems. The short term subsystem removes heat received from the Heat Transport System by venting of steam from the steam
58 drum to the atmosphere through vent control valves. The expended water volume is replaced by the Auxiliary Feedwater Subsystem. This subsystem draws water from a protected water storage tank (PWST) and pumps it to the steam drum. Three pumps are provided to supply the total auxiliary feedwater flow rate required for all three loops. Two of these pumps are driven by electrical motors powered by normal or emergency plant A-C power; each of these pumps has the capacity of delivering 50% of the total auxiliary feedwater flow rate. The third pump is driven by a steam turbine which uses steam bled from the steam drum(s); this pump has the capacity to deliver 100% of the total auxiliary feedwater flow rate.

The long term heat removal subsystem, which also functions to assist the short term subsystem, is a protected air cooled condenser (PACC) located at a higher elevation than the steam drum. This PACC rejects heat to the atmosphere. Saturated steam is supplied to the condenser from the steam drum, and saturated water is returned by gravity flow (see Figure 5.1-6 for the hydraulic profile). The PACC utilizes a fan to force air across the condenser tubes for peak performance.

32 5.1.5.2 <u>Direct Heat Removal Service</u> (See Section 5.6.2)

The Reliability Program, discussed in Appendix C, will provide
41 verification that SGAHRS removes residual heat with a high level of reliability. Hence, it is judged that the steam and feed trains backed by SGAHRS for off-normal events will satisfy the requirement to safely remove residual heat. However, because of the developmental nature of certain equipment in the heat transport system, it is considered prudent to provide a supplementary means of removing long-term decay heat. The DHRS improves overall shutdown heat removal reliability by providing a fourth redundant
41 heat removal path and heat sink.

The DHRS safety function will be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that fuel element and reactor coolant boundaries are maintained. The DHRS will function to remove decay heat following a reactor shutdown in which all IHTS/SGS heat sinks are lost. For this event DHRS will be remote manually initiated within the first half - hour after shutdown. All three PHTS pony motors will provide core flow. For initiation after 24 hours, DHRS will have sufficient capacity to limit the maximum overflow sodium temperature below 900° F. DHRS will be called into service only after two main heat transport loops have been removed from decay heat service for any reason and will carry the total heat load only if three main loops have been removed from decay heat service for any reason.

26 A schematic diagram for the DHRS is shown on Figure 5.1-7. DHRS is provided by the Auxiliary Liquid Metal System overflow and makeup circuit acting in conjunction with the spent fuel cooling and cleaning system. Hot primary sodium overflows from the reactor vessel to the overflow vessel. The primary sodium makeup pumps pump the hot sodium through the overflow heat exchanger and back to the reactor vessel. Operation of the primary pump pony
16 motors provides sodium flow through the core. EVST NaK removes the primary sodium heat in the overflow heat exchanger. The heated EVST NaK is pumped to the EVST NaK air blast heat exchangers where heat is transferred to the atmosphere (See Section 9.3).

5.1.6 <u>Auxiliary Liquid Metal System</u> (See Section 9.3)

The Auxiliary Liquid Metal System is comprised of the piping and components which perform the following services for the Reactor Heat Transport System:

1. Maintain sodium purity levels in the sodium loops

2. Provide make-up capability to maintain reactor coolant level

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3. Provide overflow capability to ensure no excessive sodium level rise in the reactor vessel

4. Provide sodium storage capacity for the operational or shutdown conditions of the Reactor Heat Transport System

40 5. Provide sodium draining capability for the reactor vessel and individual primary and intermediate heat transport loops

6. Provide valving and piping arrangement of the drain and make-up lines to preclude accidental system draining of primary sodium

26 7. Provide decay heat removal through the Direct Heat Removal Service (see Section 5.1.5.2 above).

The Auxiliary Liquid Metal System is described in detail in Section 9.3.

5.1.7 Features for Heat Transport System Safety

The following is a list of safety considerations incorporated in the design and arrangement to assure continued decay heat removal under "off normal" conditions, and other related safety features.

a. The center line of the high point primary piping is at elevation 796 feet 8 inches for the cold leg and 797
0 41 feet 8 inches for the hot leg relative to a reactor vessel sodium level of 794 feet 9 inches. The high point piping is located above the reactor vessel sodium level to permit sodium drainage of a primary loop while maintaining the reactor vessel sodium level at an elevation which assures heat removal capability.

b. Guard Vessels are provided around the reactor vessel IHX primary pumps and all piping which is below the elevation of the tops of the guard vessels. The inventory of sodium in the reactor above the minimum safe level, the guard vessel elevations and net volume, and the maximum shut-off head of the pumps at pony motor speed are such that decay heat removal is assured following a major pipe leak. A trap in the IHX vent return line prevents pump tank cover gas from being transferred to the IHX should a primary boundary leak cause the pump tank sodium level to fall below the vent return line nozzle. This trap prevents a gas bubble
58 from forming at the top of the IHX that could block primary loop flow with the pump operating at pony motor speed. Details of these features are discussed in Section 5.3.2.1.1.

c. The relative elevation of the reactor core, IHX tube bundle and steam generator modules are arranged to assure natural circulation of sodium in the primary and intermediate loops in the event of loss of pumping power. Details are discussed in Section 5.3.3.2.

d. All piping is arranged to provide sufficient flexibility without the use of flexible expansion joints.

e. Design considerations to assure satisfactory capability for controlling and minimizing the effects of sodium leaks and fires.

   1. Primary sodium containing equipment and piping are installed inside cells constructed of fire retardant materials. The primary cells contain inert gas (nitrogen) during normal reactor operation.

   2. Experience has shown that a sodium leak in an austenitic stainless steel welded system can only originate as a small or weeping type leak, and that it propagates very slowly. The basic approach for the leak detection in the HTS is, therefore, early detection to minimize the potential for large sodium leaks. (See Section 5.3.3.6) Leak detectors will be provided for the HTS piping and components as described in Section 7.5.5.1.

   3. Smoke detectors are provided to signal the presence of a fire, including sodium fires in cells with an air atmosphere. Smoke detectors are provided in the inerted cells to signal the presence of fire when the cells are de-inerted.

   4. Powder or granular fire extinguishing agents are available for hand application to control small fires that may occur during normal operation in cells with an air atmosphere. When the inerted cells are de-inerted, similar equipment is available for placement in the cells.

   5. The HTS cells within containment are completely steel lined to protect the floor and lower portions of the cell walls in the event of a sodium leak.

5.1.8 Physical Arrangement

40| The General Arrangement drawings in Section 1.2 show the components and piping of the heat transport systems in relation to each other, plant structures and other CRBRP system components.

Amend. 40
July 1977

5.1.8.1 Intermediate Heat Transport System (IHTS)

The portions of the IHTS located outside the Reactor Containment Building are located within the Steam Generator Building which is of reinforced concrete, designed to withstand the extremes of environmental conditions, including tornadoes. Each of the three IHTS loops is independent of the others and located in separate cells which provide missile shielding and fire protection for adjacent loops.

There are no active components within the IHTS for which flooding or submergence has been identified as a problem. There are no safety related valves associated with the IHTS and the pump is located well above any potential flood level. Because of this, no special provisions for flood protection is required.

5.1.8.2 Steam Generator System (SGS)

All critical SGS components are located within the Steam Generator Building which is of reinforced concrete, designed to withstand the extremes of environmental conditions, including tornadoes. Each of the three SGS loops is essentially independent of the others and critical components in each loop are contained in separate cells which provide missile shielding. A further separation is provided by locating the steam drum, recirculation pump, majority of recirculation water piping, safety and power relief valves and the majority of the water side control and isolation valves in a cell which is separate from the cell containing the evaporator and superheater modules and sodium piping.

Valves in the Steam Generation System are elevated above the floor to provide ease of installation and maintenance. The recirculation pump inlet valve has a 3 foot clearance between the bottom of the valve and the floor. The maximum water depth in the Steam Generator Auxiliary Bay at this elevation (765') will be limited to 2 feet based on available water within the Steam Generator System and
45 provisions for isolation of feedwater flow (Section 7.6.5). All other SGS valves are located at higher elevations above floor levels. Therefore, submergence due to postulated water spills will not be a problem.

5.1.8.3 Steam Generator Auxiliary Heat Removal System (SGAHRS)

The arrangement of the components, valves and piping of the SGAHRS system in the Auxiliary Bay of the Steam Generator Building incorporates full safety considerations for environmental and operational accidents including flooding, missiles, jet impingement and fire. The redundancy of loops in the SGAHRS system is maintained by routing the loops through separate cells in the Steam Generator Building to protect adjacent loops from any accident in an adjoining loop.

Since the SGAHRS auxiliary feedwater supply is common to all three heat transport system loops, complete separation cannot be maintained such as in the case of the Intermediate Heat Transport and Steam Generator Systems. Because of this, special consideration is given to flooding, missiles, jet impingement and 9
fires within the SGAHRS cells.

Floods

At El. 733' the Auxiliary Feedwater (AFW) pumps and the AFW supply line valve stations are elevated above the maximum expected water level to preclude submergence during flooding. The sources for flooding at this level are: 1) the PWST and inlet lines to the AFW supply lines, 2) the alternate water supply, 3) the AFW supply lines, and 4) the drive turbine steam supply lines.

Flooding is confined to the cell containing the source of water.

The Protected Air Cooled Condenser (PACC) cells at El. 862' have floor drains for expected small leakages and any precipitation entering the cells.

Missiles

The SGAHRS piping, valves, and components are arranged and protected to prevent damage due to missiles. Missiles can be generated by failures in rotating components, loss of valve integrity, or by pipe ruptures and resulting pipe whip.

The primary barrier for missile protection consists of the Steam Generator Building reinforced concrete walls. The AFW supply lines, turbine-drive steam supply lines, and the PACC loops are routed through separate cells as much as possible to protect the SGAHRS redundancy. Protective missile sleeves will protect critical piping from damage by missiles. This includes piping common to more than one loop and piping from one loop passing through a cell containing an adjacent loop.

The turbine-driven AFW pump is located in a separate cell from the motor driven AFW pumps and the motor driven pumps are separated by reinforced concrete walls. The AFW supply line valve stations are separated from each other and from the AFW pumps by missile barriers to prevent damage by missiles generated in an adjoining loop.

There is no high energy piping in the PWST cell so the generation of missiles in this cell is not postulated.

Separation of the PACC cells as a natural consequence of PACC design protects the PACC loop redundancy from missile damage.

Jet Impingement

Jet impingement results from the rupture of either a steam line or a high pressure, high temperature water line. Protection requirements against jet impingement for critical components will be evaluated to determine if additional measures are required beyond those provide for the SGAHRS system.

19

Fire

The types of fire in the auxiliary bay cells are electrical and oil. The construction of the cell walls with non-combustible material and the confinement of the fire to the affected cell protects the redundancy of the multiple SGAHRS loops.

58 The PACC cell separation is employed to minimize fire damage and allow continued PACC operation, possibly at reduced loads.

The type of potential fire in the PWST cell is oil, the source of which is the 19
hydraulic fluid contained in the actuators for the AFW pump inlet valves. The separation of the valves and the limited quantity of combustibles involved is
58 such that the PWST availability is not jeopardized.

TABLE 5.1-1

HEAT TRANSPORT SYSTEM THERMAL HYDRAULIC DESIGN CONDITIONS

	Parameter	Thermal Hydraulic Design Value	
	Thermal Power (MWt)	975	
	<u>Primary System</u>		
	Hot leg temperature (°F)	995	
	Cold leg temperature (°F)	730	
45	Flow (per loop) million lb/hr	13.8	
	Pump Flow (gpm @ 995°F)	33,700	
45 36	Pump Head (Ft Na @ Design Flow and Temp)	450	33
	<u>Intermediate System</u>		
45 40	Hot leg temperature (°F)	936	
40	Cold leg temperature (°F)	651	
59	IHTS ΔT (°F)	285	
45 40	Flow (per loop) million lb/hr	12.8	
	Pump Flow gpm @ 651°F	29,500	
45	Pump Head (Ft Na @ Design Flow and Temp)	330	33

Amend. 59
Dec. 1980

TABLE 5.1-2

PHTS VOLUMES AND VOLUME CHANGES*

45 Component	Sodium Containment Volume $(ft)^3$ at Room Temperature	Sodium Containment Volume at Thermal/ Hydraulic Design Conditions
Primary System		
15 Primary System o R.V. to Pump	725, 717, 728	746, 738, 749
o Pump to IHX	235	242
o IHX to R.V.	426, 435, 448	434, 444, 457
43 IHX (Shell Side)	1348	1381
Pump - Tank, Suction, and Discharge Nozzle at Normal Operating Level	367	378 33
Check Valve	88	90
Total Volume (Per Loop)	3189, 3190, 3214	3271, 3273, 3297
43 Three Loop Total	9593	9841
Reactor Vessel	13629	13961
5 44 43 Total Primary Volume	23222	23802

---

56 Note: Net sodium overflow volume from a system fill temperature of 400°F to the thermal/hydraulic operating condition is 1439 FT^3 when corrected to an assumed 900°F in the overflow tank.

* Where three volumes are given, they refer to loops #1, #2 and #3 respectively.

TABLE 5.1-3

IHTS VOLUMES AND VOLUME CHANGES (PER LOOP)

	Component		Sodium Volume* (Ft^3)	ΔV** (Ft^3)	
	Piping, IHX to Superheater***				
		Loop #1	2068	40	
		#2	1393	27	
		#3	2309	45	
	Superheater (Ref. 1)		696	12	
	Piping, Superheater to Evaporator		384	7	
	Evaporators (Ref. 2)		1392	24	
	Piping, Evaporator to Pump		324	6	
	Pump	Loop #1	188	154	
		#2	188	131	
		#3	188	161	33
	Piping, Pump to IHX				
		Loop #1	1030	20	
		#2	563	11	
		#3	969	19	
7 43	IHX (tube side)		656	11	
	Expansion Tank				
		Loop #1	612	511	
		#2	612	449	
47		#3	612	546	
	Piping, Appendage		75	1	
	TOTAL				
		Loop #1	7425	786	
		#2	6283	679	
47 43		#3	7605	832	

*Volume based on component volumes at room temperature.

**ΔV = Sodium volume change from system fill temperature of 350°F to the maximum operating temperature of that component.

***Loop #1 = East Loop.

Amend. 47
Nov. 1978

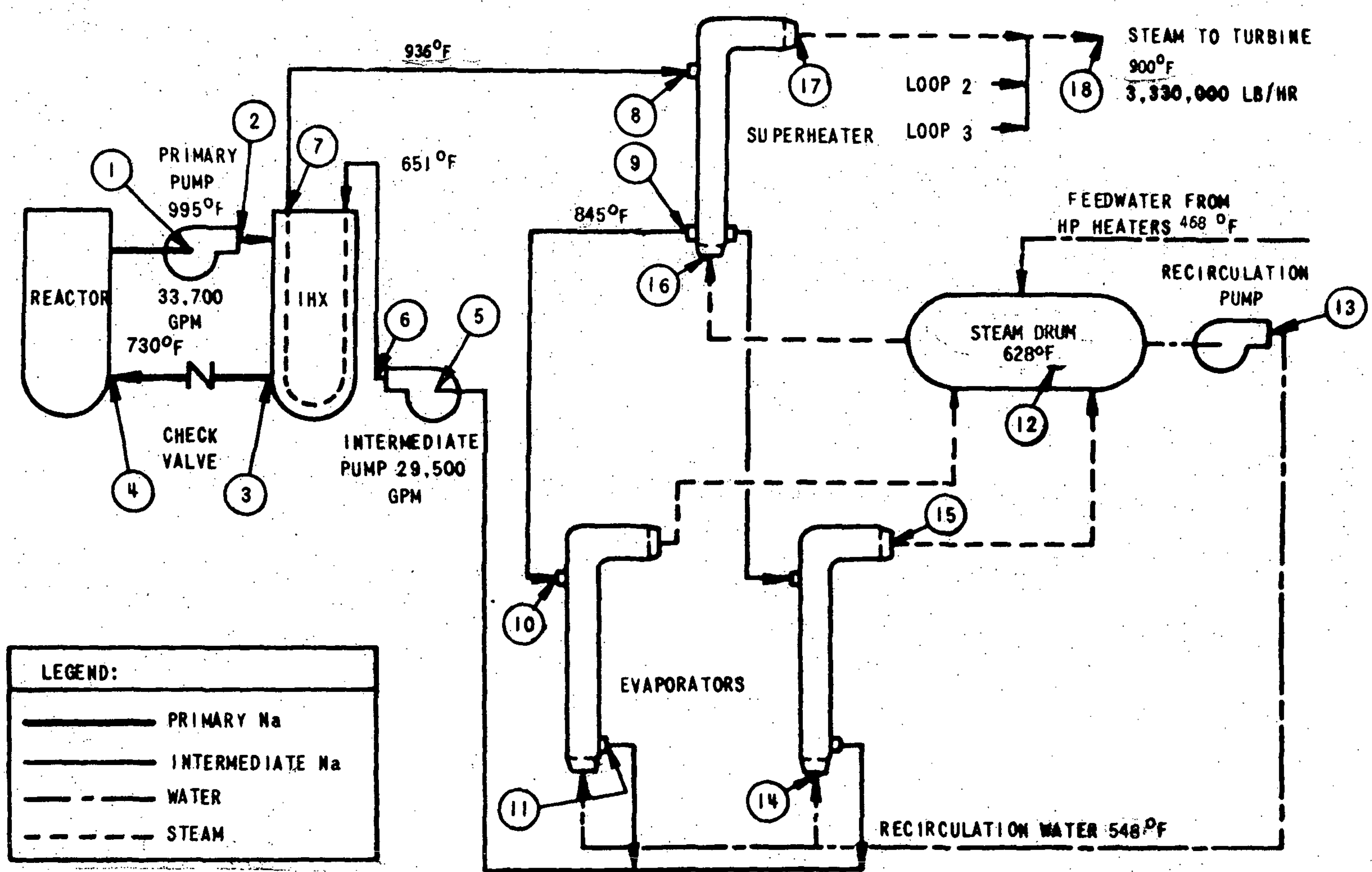
TABLE 5.1-4

STEAM GENERATION SYSTEM REQUIRED FULL-LOAD OPERATING PARAMETERS (REFERENCE VALUES AT RATED PLANT LOAD)

45

	Total Plant Duty	MWt	975
	No. of Steam Generator Loops		3
	Duty per Steam Generator Loop	MWt	325
	Sodium Flow per Loop	lb/hr	12.8×10^6
	Sodium Inlet Temperature	^oF	936^o
35	Sodium Outlet Temperature	^oF	651
	Sodium Inlet Pressure	psig (Nominal)	193
	Steam Pressure at Turbine	psig	1450
35, 45	Steam Temperature at Turbine	^oF	900
35	Feedwater Temperature	^oF	468
	Steam Flow (per Loop)	lb/hr	1.11×10^6

Amend. 45
July 1978



SEE TABLE ON FOLLOWING PAGE FOR
OPERATING PRESSURE AT NUMBERED LOCATIONS

Figure 5.1-1. General Configuration of the Heat Transport and Steam Generation Systems, One of Three Loops

HTS AND SGS NORMAL OPERATING PRESSURES

(See Figure 5.1-1)

	Location	Pressure (PSIG)
	1	5.3
	2	165.4
17	3	152.4
	4	136.3
	5	96
	6	247
	7	228
	8	193
	9	153
	10	134
43	11	135
	12	1874
	13	2019
	14	2009
	15	1881
35	16	1842
40	17	1535
	18	1450

Amend. 43
Jan. 1978

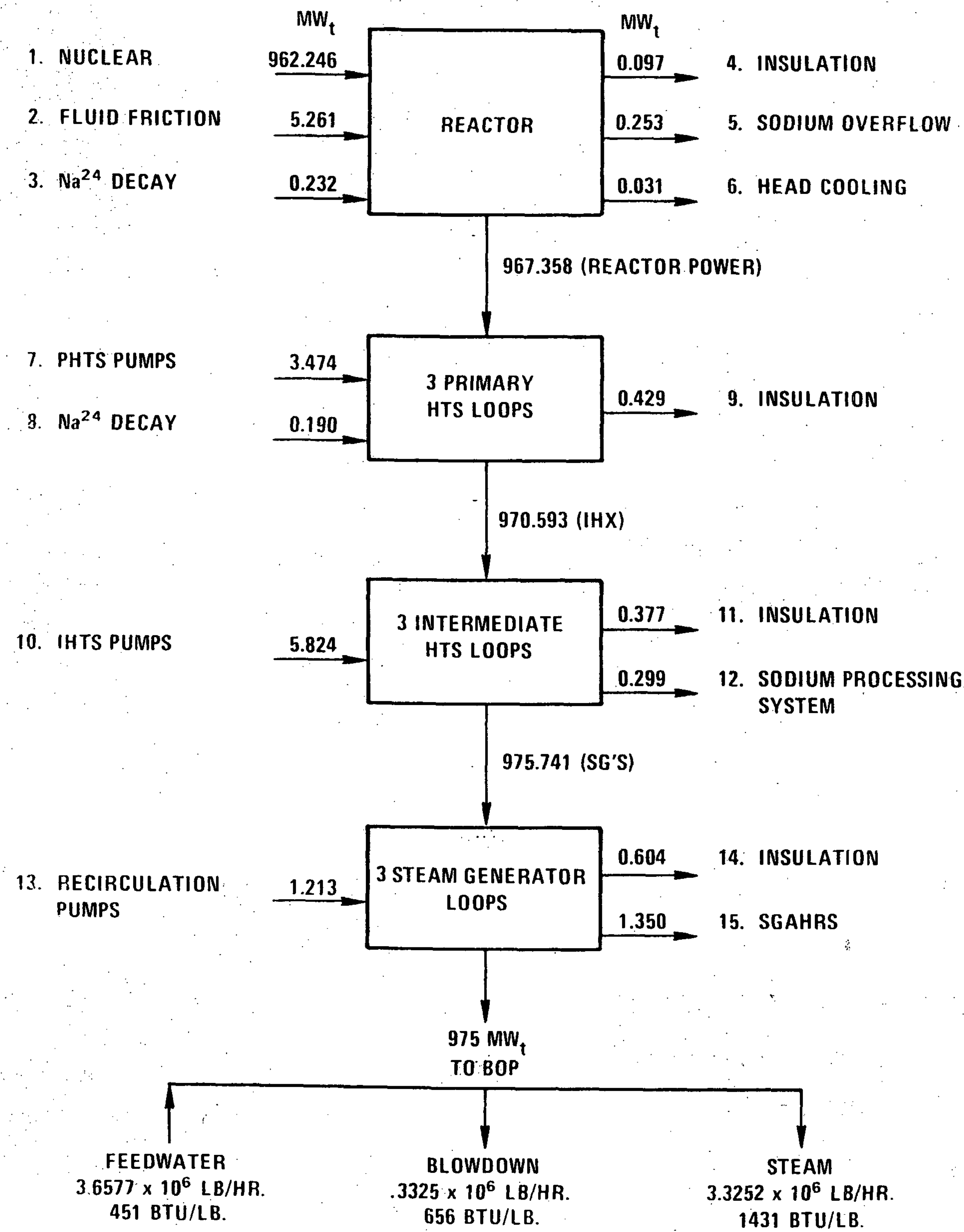


Figure 5.1-1A. Preliminary Nuclear Steam Supply System Heat Balance
975 MW$_t$ at NSSS/BOP Interface.

7683-8

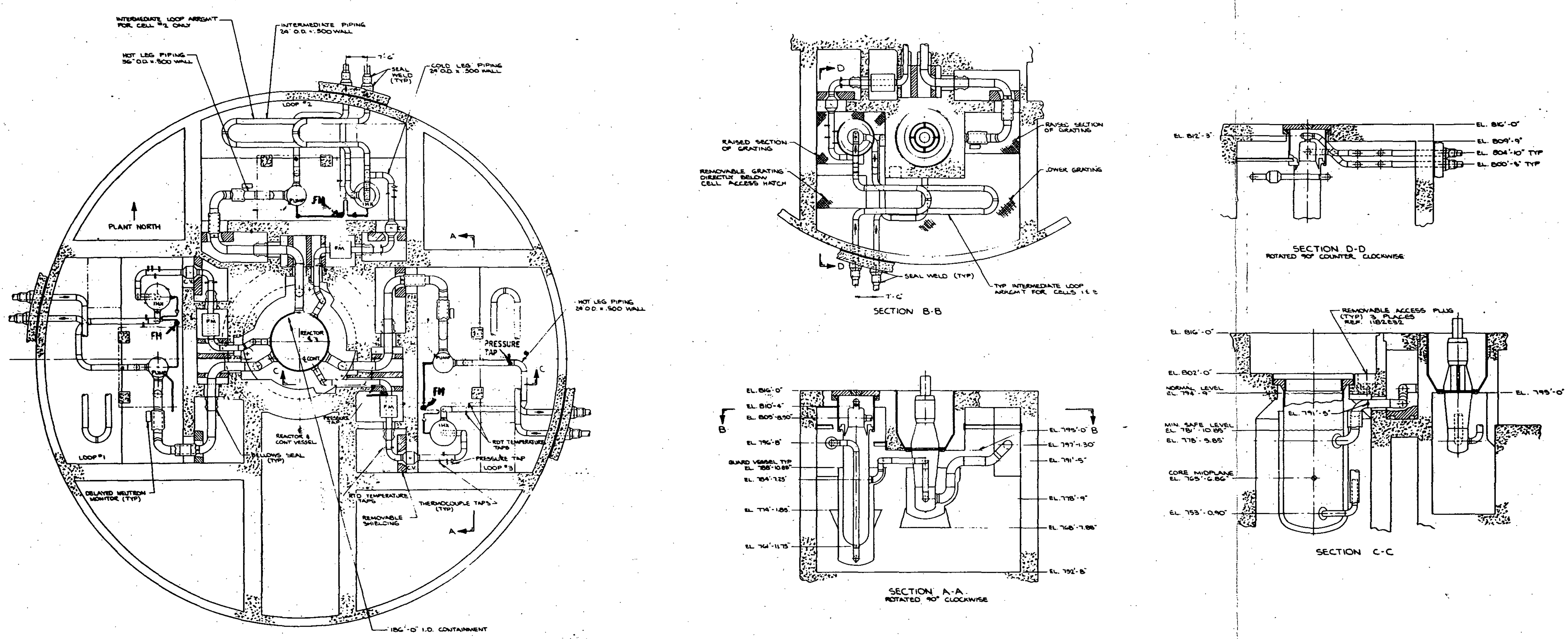
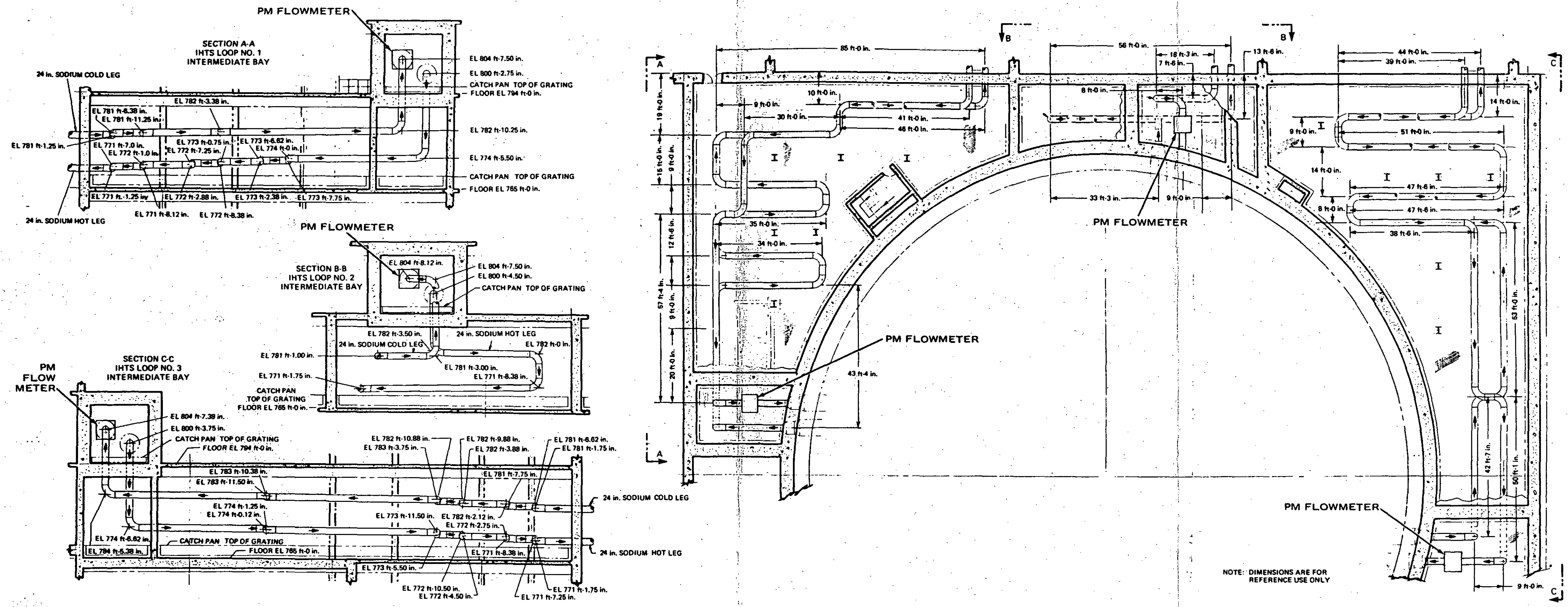


Figure 5.1-1B. Primary Heat Transport System General Arrangement

7683-30



81-850-01

FIGURE 5.1-1c. INTERMEDIATE HEAT TRANSPORT SYSTEM GENERAL ARRANGEMENT

(Sheet 1 of 3)

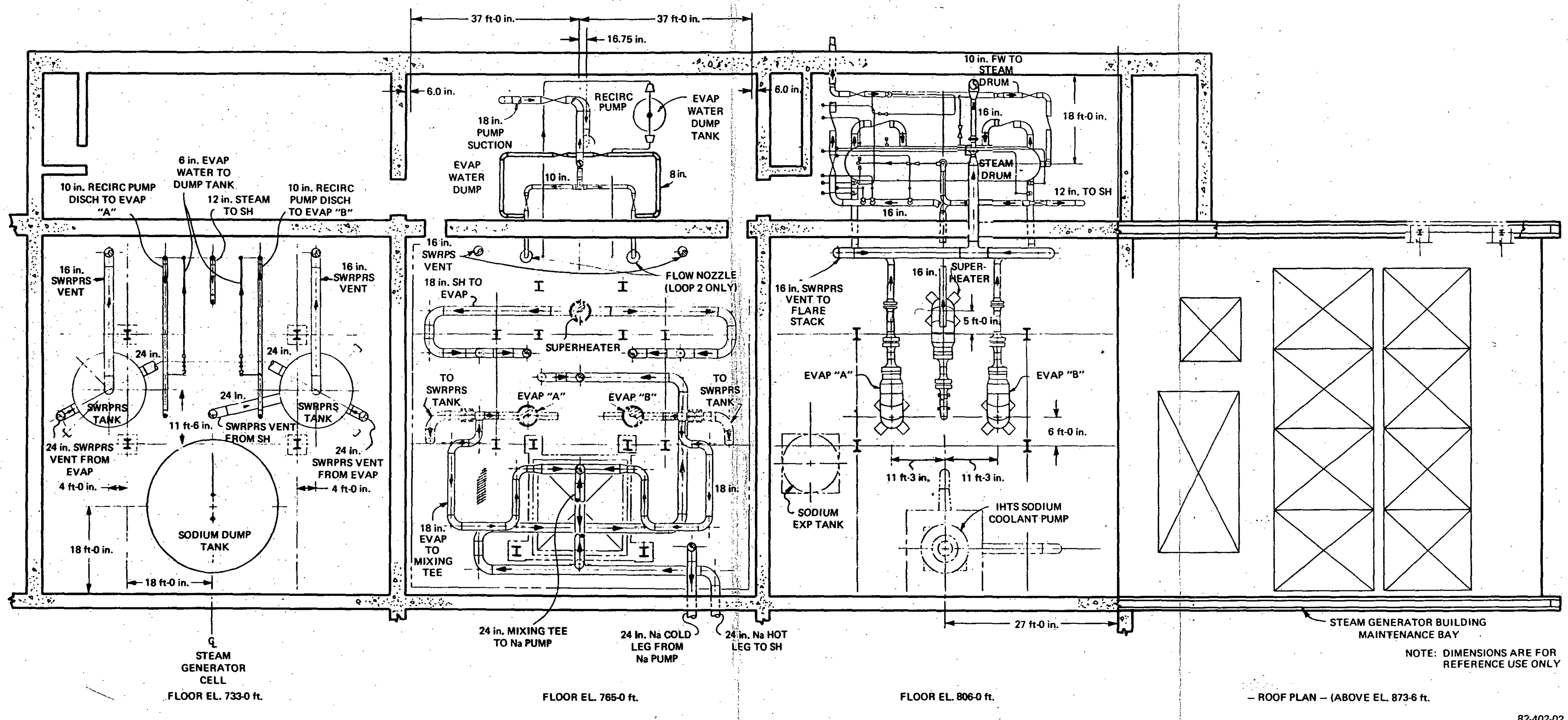
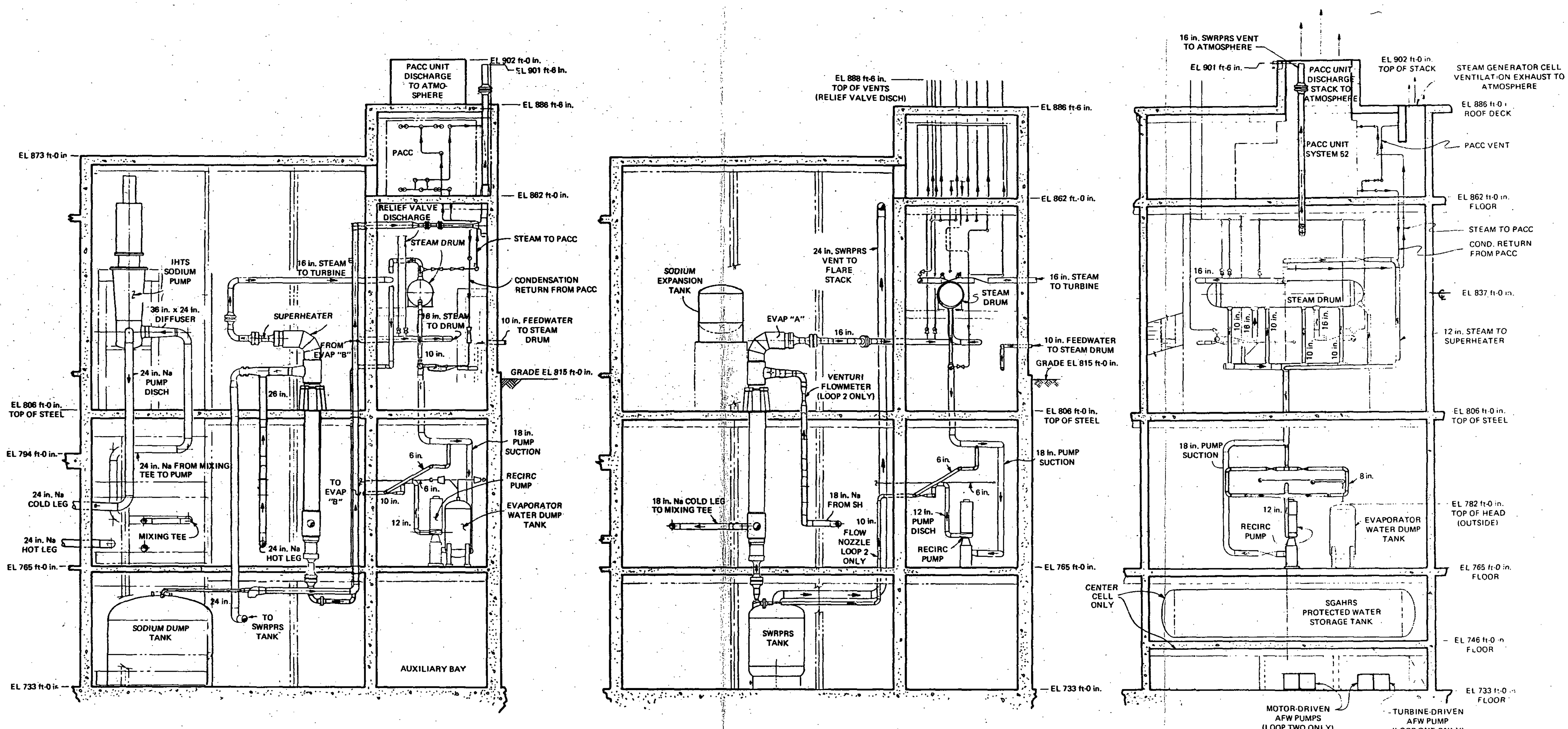


FIGURE 5.1-1c. INTERMEDIATE HEAT TRANSPORT SYSTEM GENERAL ARRANGEMENT

(Sheet 2 of 3)



TYPICAL CROSS SECTION FROM STEAM GENERATOR CELLS ONE, TWO AND THREE

TYPICAL CROSS SECTION FROM STEAM GENERATOR CELLS ONE, TWO AND THREE

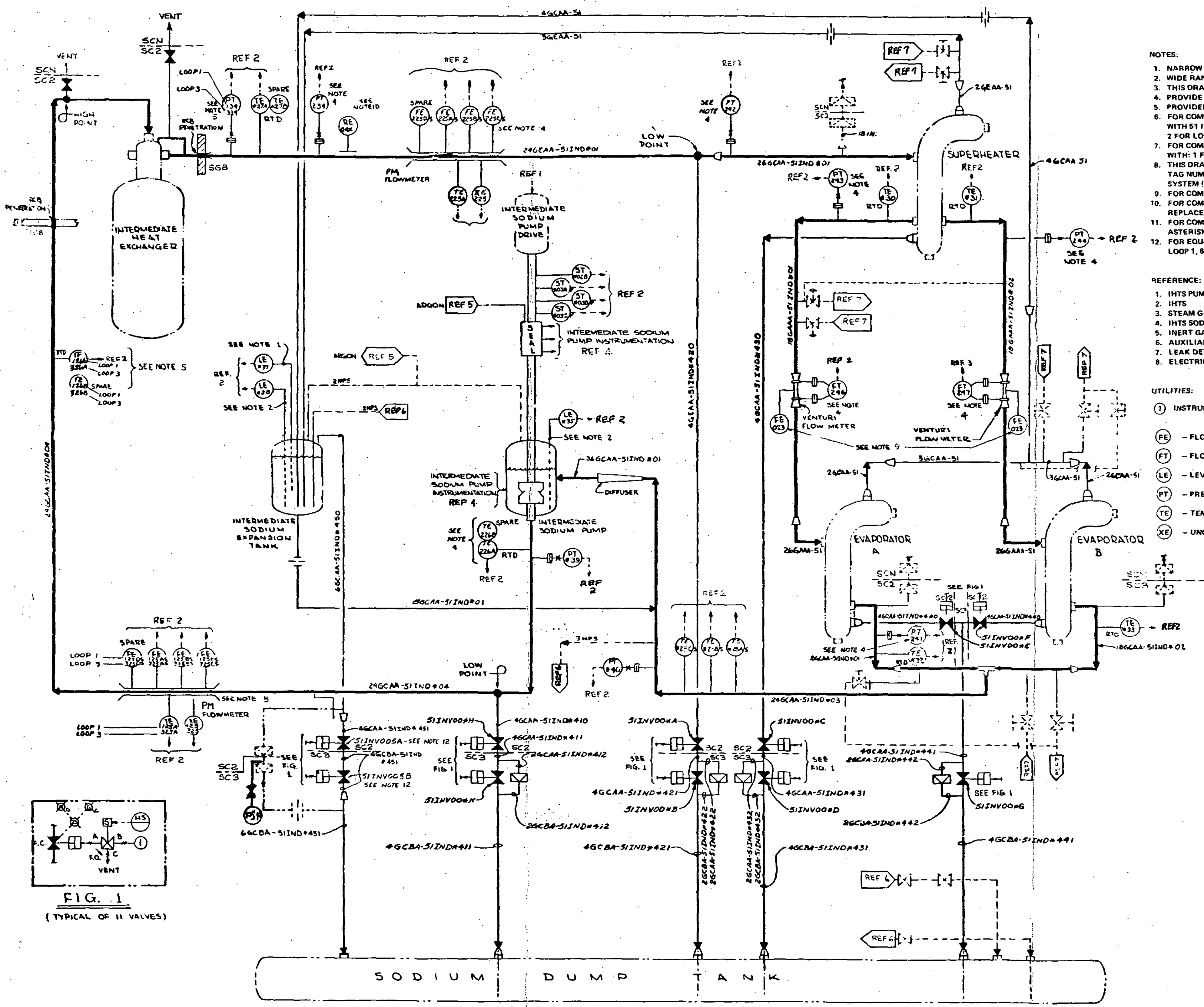
SECTION THROUGH AUXILIARY BAY

82-402-01

FIGURE 5.1-1c. INTERMEDIATE HEAT TRANSPORT SYSTEM GENERAL ARRANGEMENT

(Sheet 3 of 3)

53 Figure 5.1-2 DELETED



NOTES:

1. NARROW RANGE, HIGH ACCURACY.
2. WIDE RANGE.
3. THIS DRAWING IS TYPICAL FOR ALL THREE IHTS LOOPS.
4. PROVIDE ON LOOP NO. 2 ONLY.
5. PROVIDED ON LOOPS 1 & 3 ONLY.
6. FOR COMPLETE INSTRUMENT NUMBER DESIGNATION PRECEDE WITH 51 IN AND REPLACE THE ASTERISK WITH: 1 FOR LOOP 1, 2 FOR LOOP 2, 3 FOR LOOP 3.
7. FOR COMPLETE SPOOL NO. DESIGNATION REPLACE THE ASTERISK WITH: 1 FOR LOOP 1, 2 FOR LOOP 2, 3 FOR LOOP 3.
8. THIS DRAWING INCLUDES EQUIPMENT IDENTIFIED (BY "P" OR "S" TAG NUMBER SUFFIX) AS PART OF THE PLANT PROTECTION SYSTEM (PPS).
9. FOR COMPLETE EQUIPMENT NUMBER PRECEDE WITH 51INM.
10. FOR COMPLETE EQUIPMENT NUMBER PRECEDE WITH 96 PM AND REPLACE X WITH 3 FOR LOOP 1, 4 FOR LOOP 2, AND 5 FOR LOOP 3.
11. FOR COMPLETE VALVE NUMBER DESIGNATION REPLACE THE ASTERISK WITH: 1 FOR LOOP 1, 2 FOR LOOP 2 AND 3 FOR LOOP 3.
12. FOR EQUALIZATION LINE VALVES THE DESIGNATION IS 5 FOR LOOP 1, 6 FOR LOOP 2 AND 7 FOR LOOP 3.

REFERENCE:

1. IHTS PUMP DRIVE CONTROL SYSTEM
2. IHTS
3. STEAM GENERATOR SYS.
4. IHTS SODIUM PUMP
5. INERT GAS RECEIVING PROCESSING SYS.
6. AUXILIARY LIQUID METAL SYS.
7. LEAK DETECTION SYS.
8. ELECTRIC HEATING & CONTROL

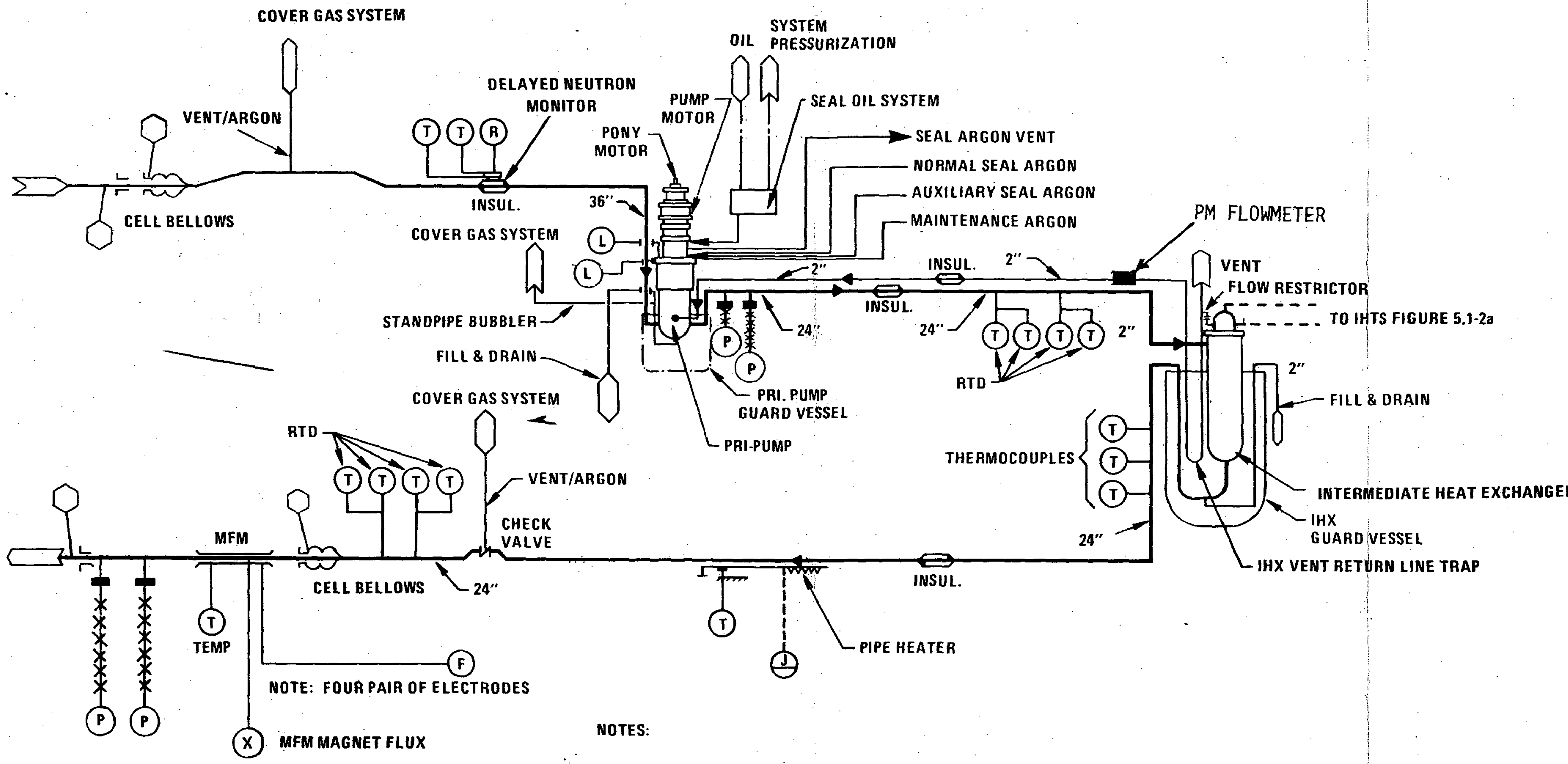
UTILITIES:

(1) INSTRUMENT AIR

(FE) – FLOW RATE ELEMENT
(FT) – FLOW RATE TRANSMITTER
(LE) – LEVEL ELEMENT
(PT) – PRESSURE TRANSMITTER
(TE) – TEMPERATURE ELEMENT
(XE) – UNCLASSIFIED ELEMENT

81-833-01

Figure 5.1-2a INTERMEDIATE HEAT TRANSPORT SYSTEM PIPING AND INSTRUMENTATION DIAGRAM



NOTES:

1. ALL PIPING AND COMPONENTS SHOWN ARE SEISMIC CATEGORY 1
2. ALL PIPING AND COMPONENTS SHOWN ARE ASME III, CLASS 1 EXCEPT FOR THE GUARD VESSELS WHICH WILL BE DESIGNED CLASS 1, BUT NOT CODE STAMPED

Figure 5.1-2b

Primary Heat Transport System

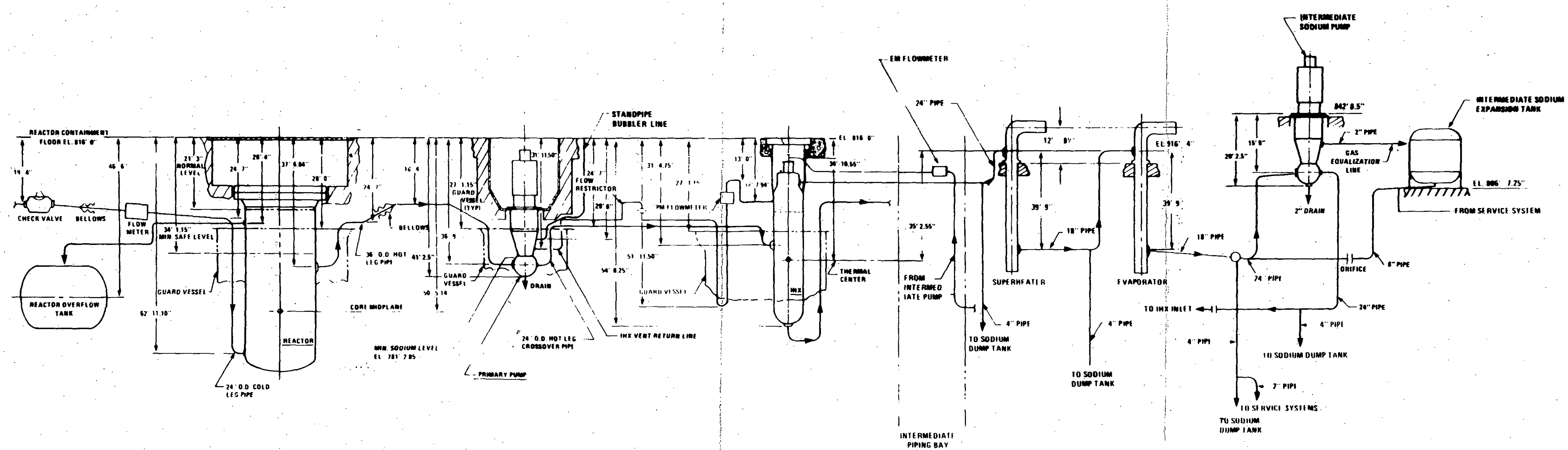


Figure 5.1-3. HTS Hydraulic Profile

PSAR 6250

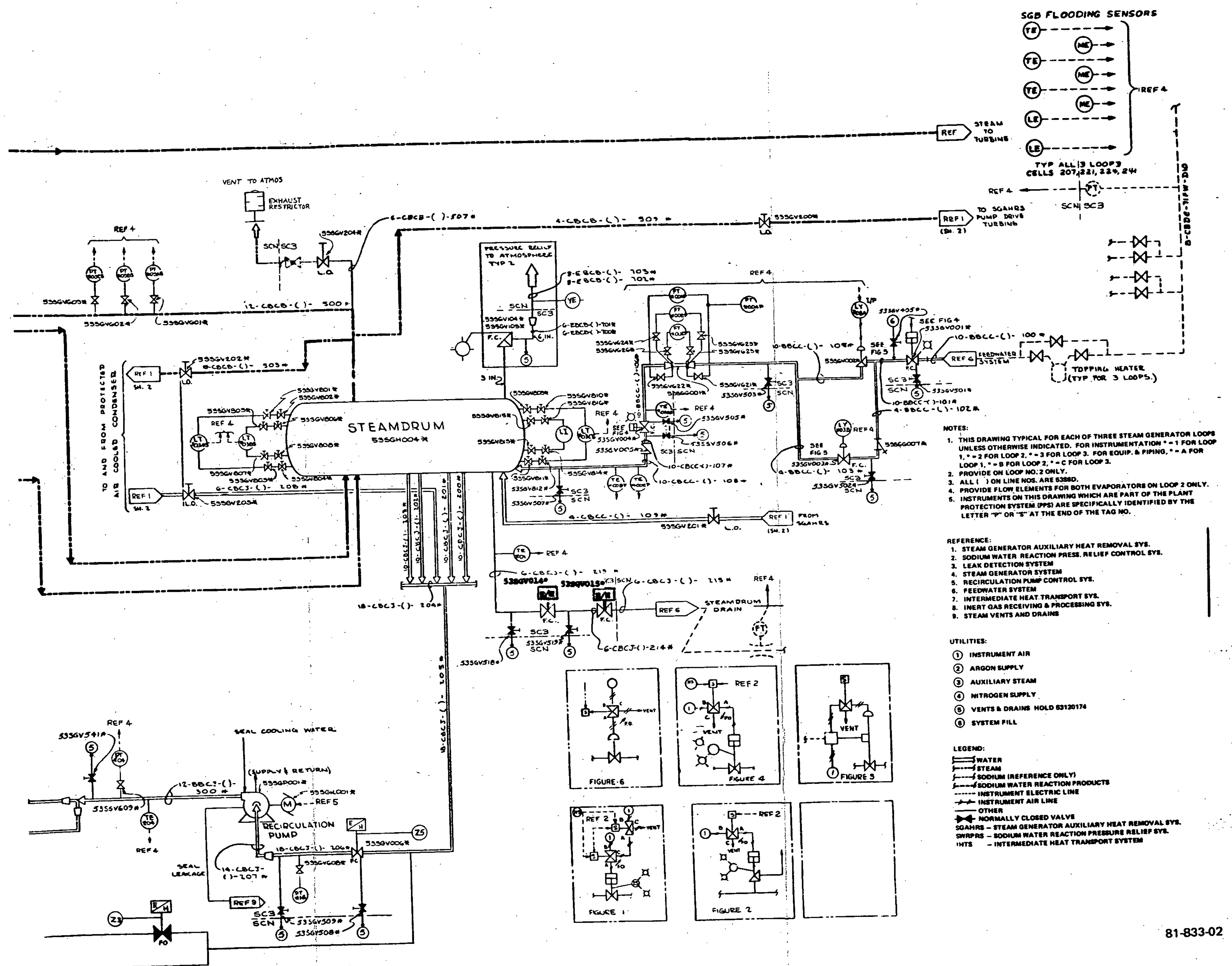
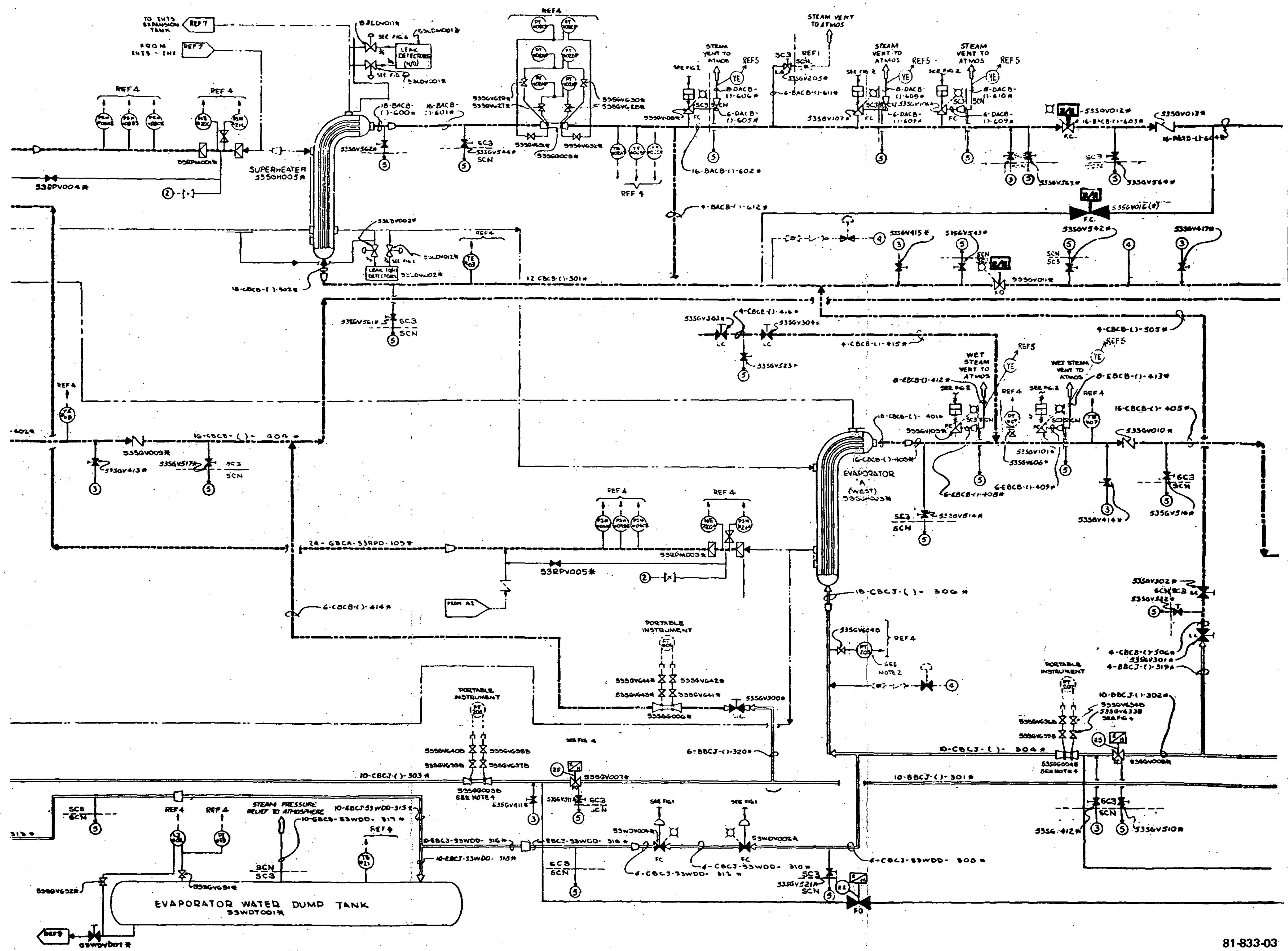


Figure 5.1-4 STEAM GENERATOR SCHEMATIC FLOW DIAGRAM (SHEET 1 OF 3)



81-833-03

Figure 5.1-4 STEAM GENERATOR SCHEMATIC FLOW DIAGRAM (SHEET 2 OF 3)

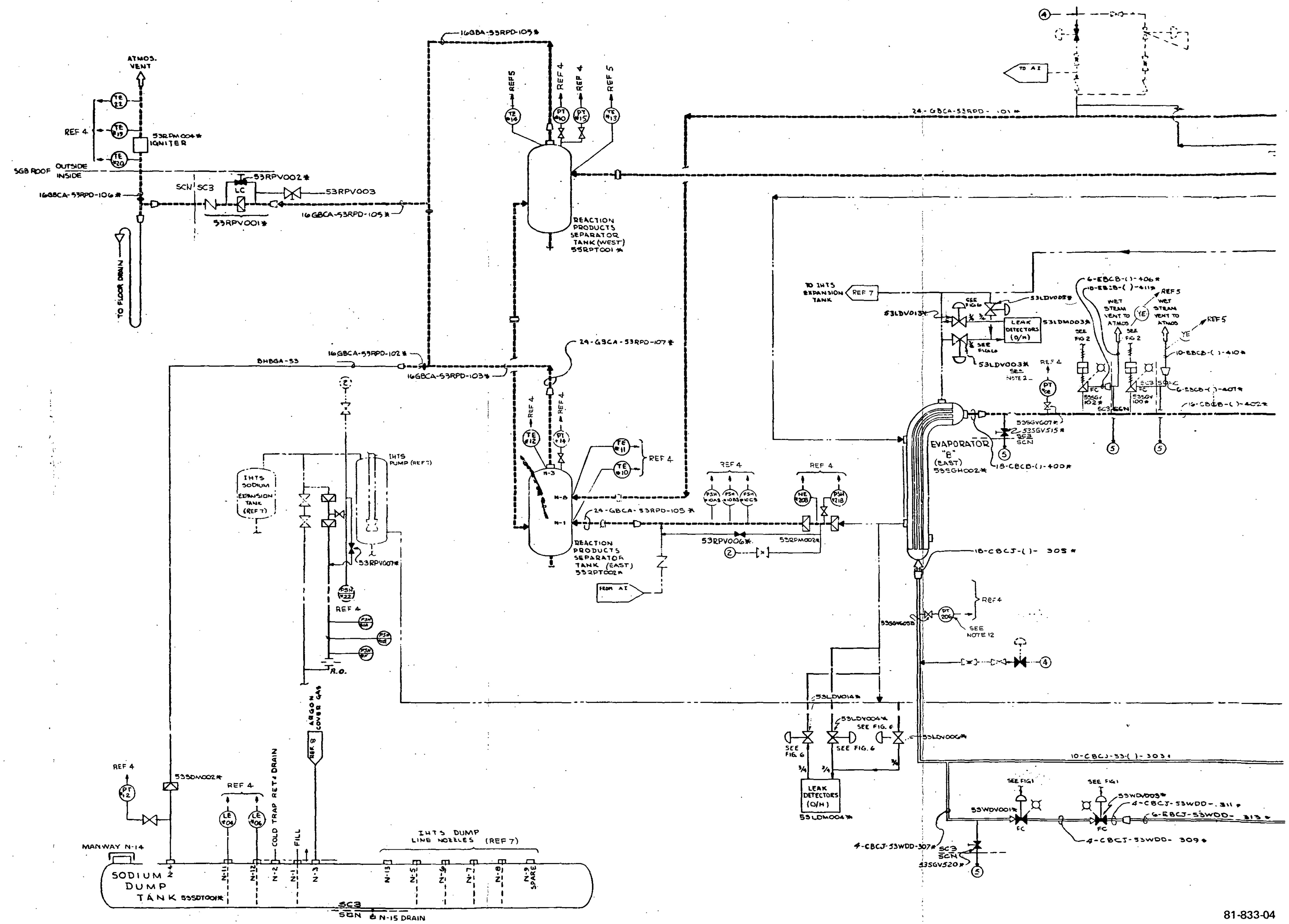


Figure 5.1-4 STEAM GENERATOR SCHEMATIC FLOW DIAGRAM (SHEET 3 OF 3)

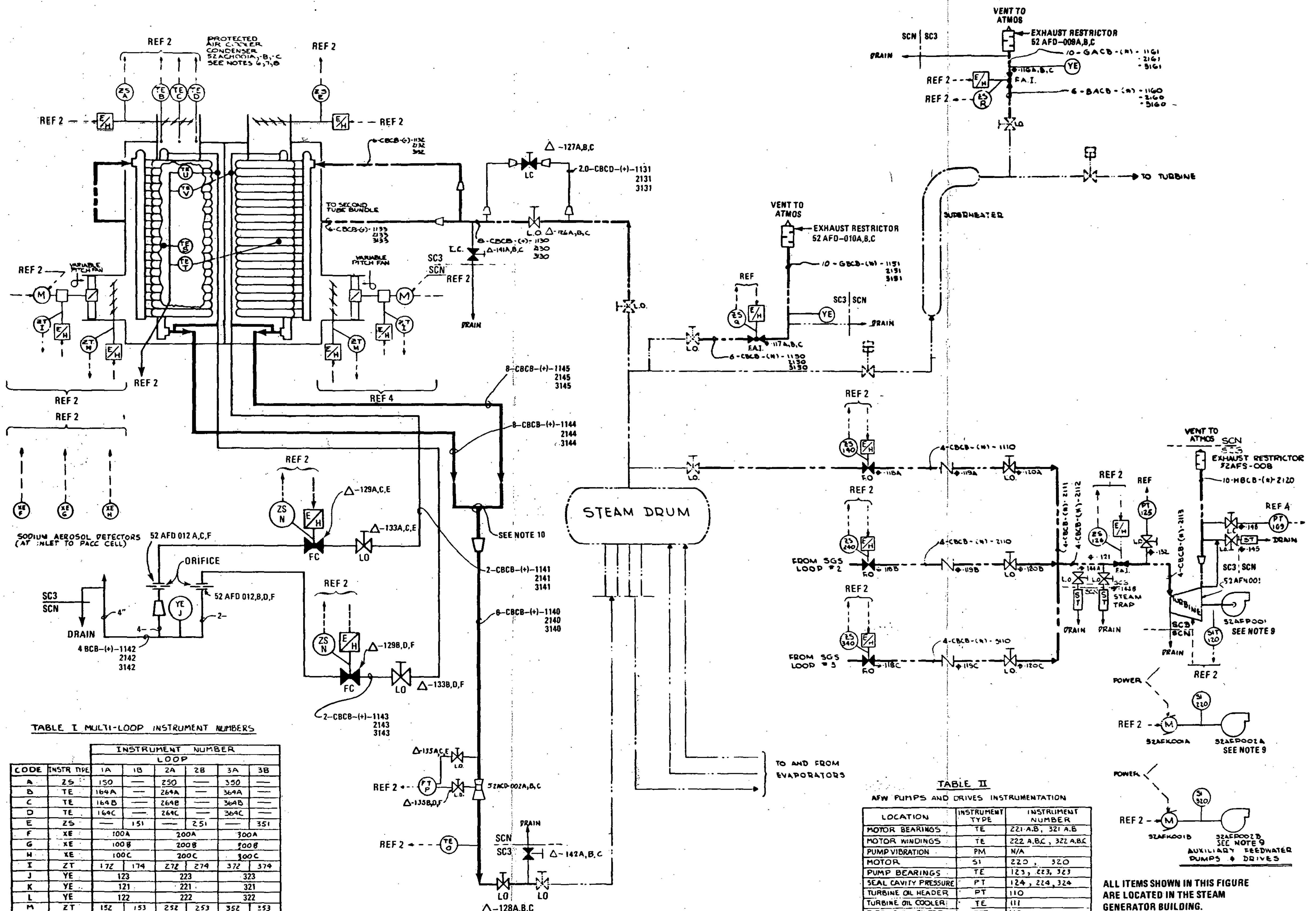


TABLE I MULTI-LOOP INSTRUMENT NUMBERS

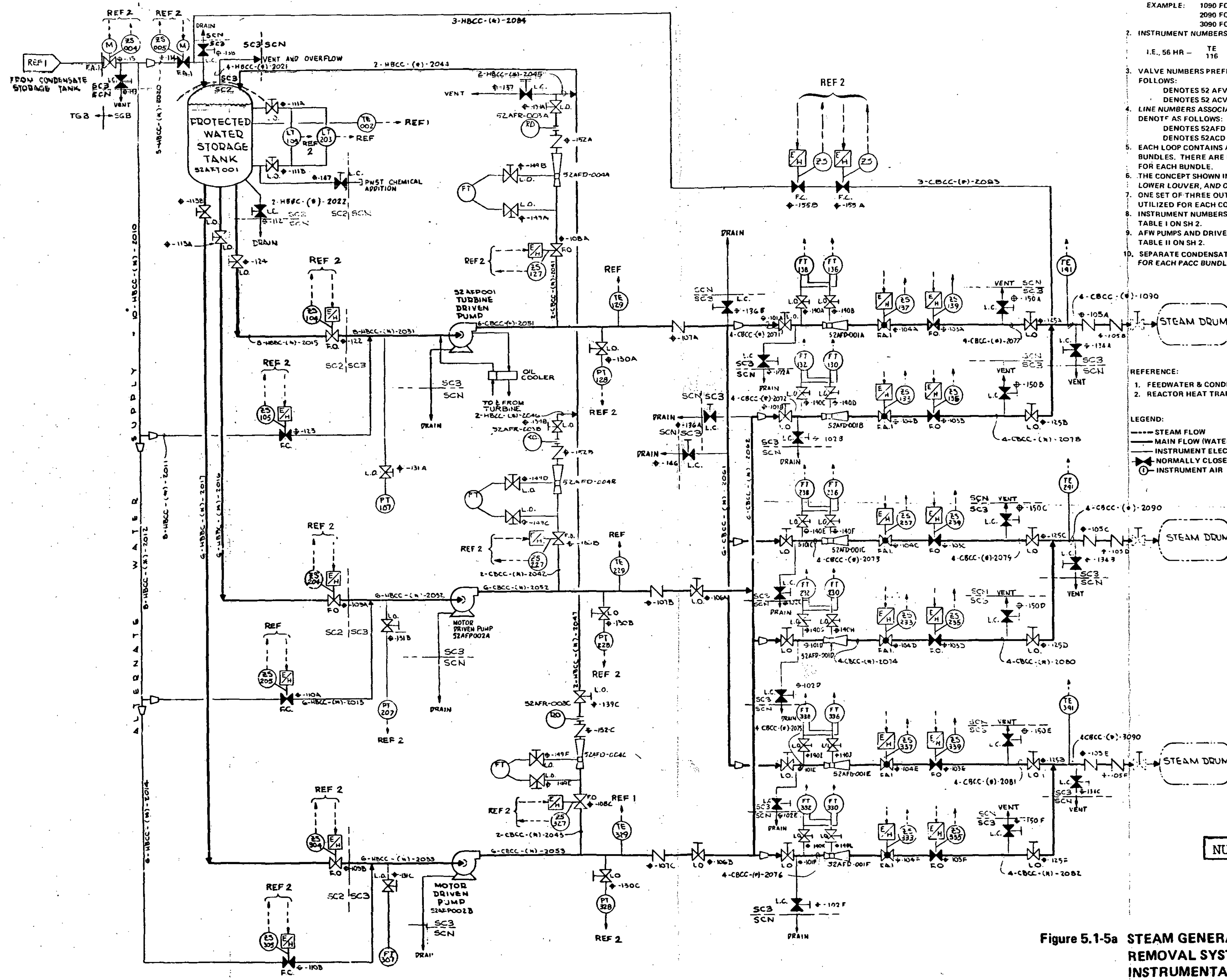
CODE	INSTR TYPE	INSTRUMENT NUMBER – LOOP 1A	1B	2A	2B	3A	3B
A	ZS	150	—	250	—	350	—
B	TE	164A	—	264A	—	364A	—
C	TE	164B	—	264B	—	364B	—
D	TE	164C	—	264C	—	364C	—
E	ZS	—	151	—	251	—	351
F	XE	100A		200A		300A	
G	XE	100B		200B		300B	
H	XE	100C		200C		300C	
I	ZT	172	174	272	274	372	374
J	YE	123		223		323	
K	YE	121		221		321	
L	YE	122		222		322	
M	ZT	152	153	252	253	352	353
N	ZS	147	155	247	255	347	355
O	TE	145	—	245	—	345	—
P	FT	144	—	244	—	344	—
Q	ZS	143	—	243	—	343	—
R	ZS	142	—	242	—	342	—
S	TE	172	—	272		372	—
T	TE	—	174	—	274	—	374
U	TE	152	—	252	—	352	—
V	TE	—	153	—	253	—	353

TABLE II

AFW PUMPS AND DRIVES INSTRUMENTATION

LOCATION	INSTRUMENT TYPE	INSTRUMENT NUMBER
MOTOR BEARINGS	TE	221 A.B, 321 A.B
MOTOR WINDINGS	TE	222 A.B.C, 322 A.B.C
PUMP VIBRATION	PM	N/A
MOTOR	SI	220, 320
PUMP BEARINGS	TE	123, 223, 323
SEAL CAVITY PRESSURE	PT	124, 224, 324
TURBINE OIL HEADER	PT	110
TURBINE OIL COOLER	TE	111
TURBINE OIL FILTER	DPT	112
TURBINE TRIP VALVE	ZS	113
TURBINE BEARINGS	YE	114 A.B
SUMP OIL	TI	108
TURBINE BEARINGS	TE	116, 117
TURBINE SPEED	SIT	120

Figure 5.1-5a STEAM GENERATOR AUXILLIARY HEAT REMOVAL SYSTEM PIPING AND INSTRUMENTATION DIAGRAM



NOTES:

1. FIRST DIGIT OF FOUR DIGIT PIPING SUFFIX NO. INDICATES LOOP NO.:
   EXAMPLE: 1090 FOR LOOP 1
   2090 FOR LOOP 2
   3090 FOR LOOP 3
2. INSTRUMENT NUMBERS ARE PREFIXED BY 56HR –
   I.E., 56 HR – TE 116
3. VALVE NUMBERS PREFIXED BY THESE SYMBOLS DENOTES AS FOLLOWS:
   DENOTES 52 AFV – I.E., 52AFC-101
   DENOTES 52 ACV – I.E., 52ACV-122
4. LINE NUMBERS ASSOCIATED WITH THE FOLLOWING SYMBOLS DENOTE AS FOLLOWS:
   DENOTES 52AFD – I.E., 4-CBCC-52AFD-1090
   DENOTES 52ACD – I.E., 6-CBCB-52ACD-1140
5. EACH LOOP CONTAINS A PACC WITH TWO HALF-SIZE TUBE BUNDLES. THERE ARE REDUNDANT UPPER AND LOWER LOUVERS FOR EACH BUNDLE.
6. THE CONCEPT SHOWN INCORPORATES ONE UPPER LOUVER, ONE LOWER LOUVER, AND ONE FAN PER EACH HALF-SIZE BUNDLE.
7. ONE SET OF THREE OUTLET TEMPERATURE SENSORS ARE UTILIZED FOR EACH COMPLETE PACC.
8. INSTRUMENT NUMBERS FOR MULTILOOP DEVICES ARE SHOWN IN TABLE I ON SH 2.
9. AFW PUMPS AND DRIVES INSTRUMENTATION ARE SHOWN IN TABLE II ON SH 2.
10. SEPARATE CONDENSATE RETURN LINES SHALL BE PROVIDED FOR EACH PACC BUNDLE DOWN TO ELEV. 834 OR LOWER.

REFERENCE:

1. FEEDWATER & CONDENSATE SYSTEM.
2. REACTOR HEAT TRANSPORT INSTRUMENTATION SYSTEM.

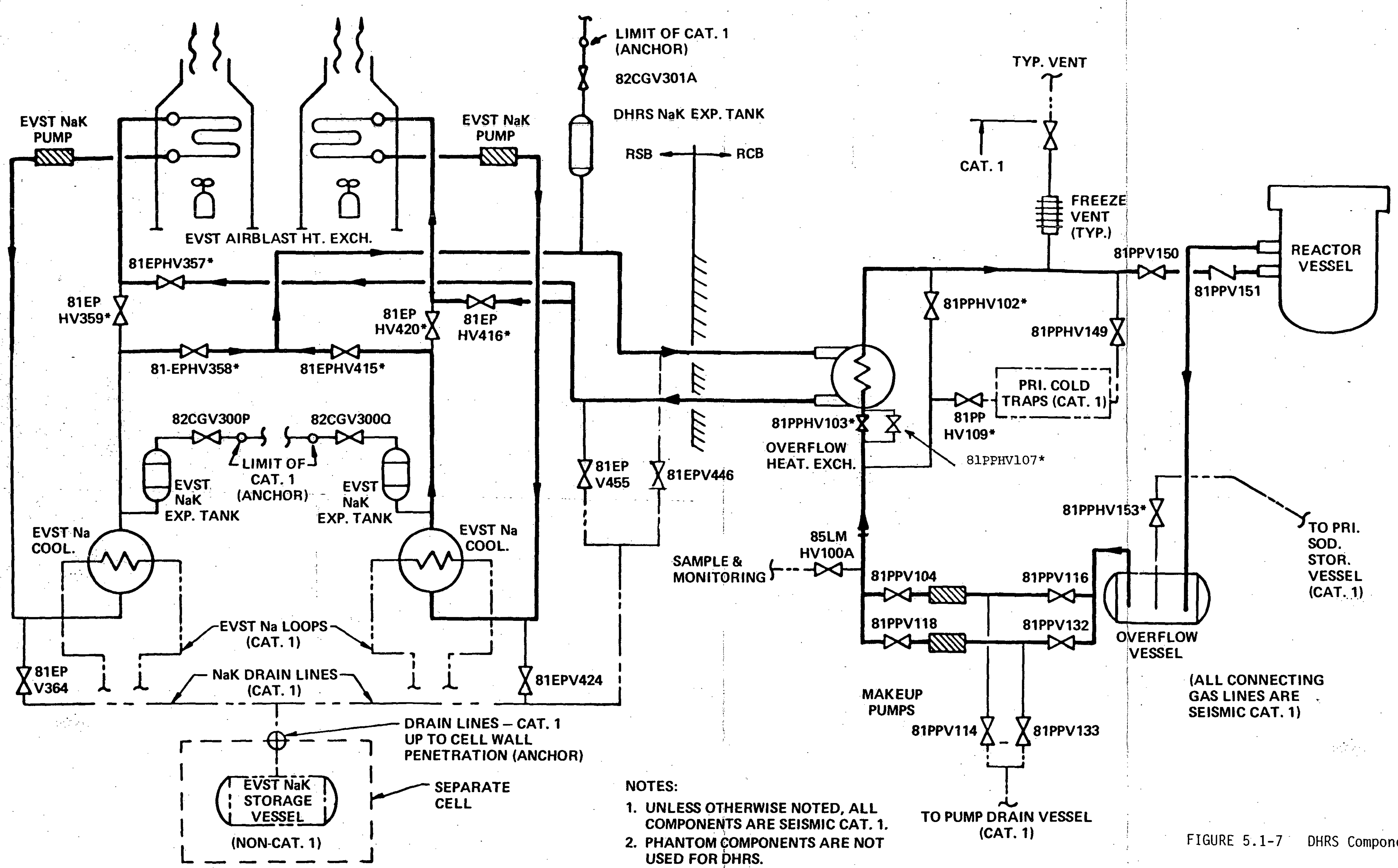
LEGEND:

STEAM FLOW
MAIN FLOW (WATER)
INSTRUMENT ELECTRIC LINE
NORMALLY CLOSED VALVE
INSTRUMENT AIR

NUCLEAR SAFETY RELATED

83-018-02

Figure 5.1-5a STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM PIPING AND INSTRUMENTATION DIAGRAM (Sheet 2 of 2)



NOTES:

1. UNLESS OTHERWISE NOTED, ALL COMPONENTS ARE SEISMIC CAT. 1.
2. PHANTOM COMPONENTS ARE NOT USED FOR DHRS.
3. VALVES SHOWN FOR LOCATION ONLY.
4. VALVE CLASSIFICATION – SEE TABLE 5.6-10.

*ACTIVE

FIGURE 5.1-7 DHRS Components

Amend. 75
Feb. 1983

5.2 REACTOR VESSEL, CLOSURE HEAD, AND GUARD VESSEL

5.2.1 Design Basis

5.2.1.1 General - Reactor Enclosure System

The major components of the reactor enclosure system are the reactor vessel, the closure head, and the guard vessel. The primary safety related function of these components is to provide containment, as appropriate, of coolant, cover gas, fuel, and associated thermal
17 and nuclear activities under all normal, upset, emergency and faulted conditions. These components shall be designed, fabricated and erected to quality standards that reflect the importance of this safety function. Where generally recognized codes or standards for design, materials, fabrication, and inspection are adequate, they shall be used. Where a component is not covered by nationally recognized codes or standards, specific and appropriate design requirements and acceptance criteria will be defined and provided in component specifications. 1

The reactor enclosure system provides radiation shielding as well as access for insertion and removal of surveillance material, for in-service inspection and for controlling, monitoring and servicing the core and its associated components and structures. The design transients for each of the components are described in Appendix B of this PSAR. In all cases the expected or hypothesized conditions shall not be more severe than the selected design criteria and transients.

41 With regard to Regulatory Guide 1.87 (June 1974, Rev. 0), "Construction Criteria for Class 1 Components in Elevated Temperature Reactors" (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596), portions relevant to component design and manufacture have been applied through the equipment specifications in the following manner:

Regulatory Position C.I.a:

41 All five Code Cases should be invoked, where applicable, for components in high-temperature gas-cooled reactors, gas-cooled fast breeder reactors, and liquid metal fast breeder reactors.

Implementation

The subject Code Cases are imposed by the equipment specifications except the Code Case revision distributed at time of contract placement, or later, will be used. For example, Code Case 1592-2 was received at the time of reactor vessel contract placement (April 18, 1975). Hence, either Code Case 1592-2 or subsequent revisions will be applicable to the reactor vessel, in lieu of Code Case 1592 as specified in Regulatory Guide 1.87.

Regulatory Position C.I.b:

41 These Code Cases may be used in conjunction with Subsection NB of Section III of the ASME Boiler and Pressure Vessel Code. Additional justification, relative to elevated temperature applicability, should 25

be provided in the Stress Report when other portions of Section III such as Appendixes E and F and Subsections NF and NG are used with these Code Cases.

Implementation

Application of the subject Code Cases to Section III Appendix E and Subsections NF and NG is not relevant to elevated temperature portions of the reactor vessel, closure head, or guard vessel (guard vessel support design to NB in lieu of NF). Supplementary rules for application of Code Case 1592 to Appendix F are imposed via RDT Standard F9-4T required by the equipment specifications.

Regulatory Position C.I.c:

Component designs should accommodate any required inservice inspection and surveillance programs to monitor and alert for material or component degradation such as creep rupture, creep deformation, creep-fatigue interaction, profusion of microcracks, and buckling. Representative environmental factors of concern which should be considered are the effects of the cooling fluid such as sodium, helium, air, and/or impurities; irradiation effects such as aging and ductility loss; and aging resulting from prolonged exposure to elevated temperature.

Implementation

The Reactor Enclosure System and components (reactor vessel, guard vessel and closure head) are being designed to accomodate inservice inspection. A surveillance program is being planned. Each equipment specification contains detailed requirements necessary for implementation of the system programs for inservice inspection and surveillance. Environmental effects on material properties are specifically considered in design through definition of effects in the equipment specifications.

Regulatory Position C.I.d:

When a Code Case refers to an Article in Subsection NB or that Article in turn references another Article in Subsection NB, it should be ascertained that all referenced Articles in Subsection NB are consistent with all applicable elevated-temperature Code Cases and the corresponding supplements in Part C of this guide. 25

Implementation

The CRBRP requires full compliance with the precise provisions of the Code in regard to the requirement of this Regulatory Position. Consistency and/or applicability is determined in strict accordance with the detailed requirements defined in the Code, including the
41 applicable Code Cases.

Regulatory Position C.2, Code Case 1592

41 a. The Stress Report should provide a description of analysis methods and clearly delineate areas which have been subjected to elastic analysis and areas which have been subjected in inelastic analysis. All computer programs used should be identified and sufficiently described with respect to those portions utilized to identify basic theory, assumptions, constitutive relations, extent of verifications, limitations, and justification of applicability including validation of computer models and modeling techniques.

41 b. The Stress Report should indicate for each part of the analysis whether the strength properties used in analysis are minimum, average, or maximum, except where they are specified by, or are an integral part of, the analytical method. Where an option exists, the use of minimum, average or maximum strength properties relative to critical failure modes, damage laws, and deformation limits should be justified on the basis of increased safety. If in this Code Case, adequate material property data for the materials to be used are not given, the data given are not used or are not appropriate, or extrapolation of given data is necessary for the specific use, the appropriate properties used in design should be documented and justified in the Stress Report. This material data based together with a description of the methods used to account for environmental effects throughout design life should be documented in the Stress Report.

41 c. The acceptability criteria and material properties given in Appendix T of this Code Case should be used to satisfy the strain, deformation, and fatigue limit requirements of 3250.

Implementation

These positions are implemented by application of RDT Standard F9-4T through the reactor enclosure components equipment specifications.

41 d. A full description of the buckling analyses pursuant to T-1500 should be documented in the Stress Report. This should also include the following:

(1) Indication of the margin for a design factor on load applied throughout service life in addition to the suggested design factor at end of life in Table T-1520-1.

(2) Justification that a process is purely strain controlled and not combined with load-controlled or significant elastic follow-up when the strain-controlled design factor in Table T-1520-1 is used. 2

(3) Description of the methods used to determine the minimum stress-strain curve suggested in T-1520 (c).

Implementation

The reactor enclosure component equipment specifications which invoke Code Case 1592 also invoke RDT Standard F9-4T, which requires planning of analysis to facilitate review and certification of the stress report. The intent of this position to provide full description of buckling analyses persuant to T-1500 will be fulfilled through the planning and subsequent execution of the structural evaluation program.

41 e. The intent of Appendix II (ASME B&PV Code) rules should be used for guidance when design is justified by experimental analysis. A description of procedures used for experimental analysis, the evaluation techniques, and acceptance criteria should be included in the Stress Report.

The CRBRP requires full compliance with the precise provisions of the Code in regard to the requirement of this Regulatory Position. Consistency and/or applicability is determined in strict accordance with the detailed requirements defined in the Code.

Regulatory Position C.3, Code Case 1593:

41 a. Implementation of this Code Case should comply with Regulatory Position C.1.d of this guide.

Implementation

The CRBRP application of this Code Case in the fabrication and installation of elevated temperature components requires the same consistency as is discussed in C.1.d implementation above.

41 Regulatory Position C.4, Code Case 1594:

a. Implementation of this Code Case should comply with Regulatory Position C.1.d of this guide.

Implementation

The CRBRP application of this Code Case in the examination of elevated temperature components requires the same consistency as is discussed in C.1.d implementation above.

Regulatory Position C.5, Code Case 1595:

41 a. The "non-hazardous liquid" in 6212(a) should be non-hazardous relative to possible reactions between residual test liquid and the normal coolant fluid and non-hazardous with respect to deleterious effects to the component (material) such as corrosion by either the test liquid or a fluid created by reaction of test liquid and coolant. 25

Implementation

a. Control of hydrostatic test fluid for the reactor vessel is limited by the specification to water of a certain purity. Requirements for drying the reactor vessel after hydrostatic test are given in the specification. Subsequent cleanliness requirements are also imposed by the reactor vessel specification to assure the component is delivered in a cleanliness condition suitable for liquid metal service. The closure head and guard vessel are not hydrostatically tested, hence not subject to this position.

41 b. Implementation of this Code Case should comply with Regulatory Position C.1.d. of this guide.

Implementation

The CRBRP application of this Code Case in the testing of elevated temperature components requires the same consistency as is discussed in
41 C.1.d implementation above.

Regulatory Position C.6, Code Case 1596:

41 a. The Overpressure Protection Report should indicate those components considered to be "non-critical" pursuant to 7110(a) (4) and 7110(b). The evaluation techniques and acceptance criteria used to justify designation as a "non-critical"component should also be included.

41 b. Implementation of this Code Case should comply with Regulatory Position C.1.d. of this guide.

41 c. The potential overpressure due to failure of a system component in 7121(d) should include consideration of pressure from an adjacent system by leaks or chemical reaction or both.

41 d. For those reactors using liquid sodium as the coolant, a description of the methods used to determine overpressure resulting from possible shock loads mentioned in 7122 should appear in the Overpressure Protection Report. This should include a definition of what constitutes rapid valve closure relative to pressure wave velocity and valve closing time. A description of how the pressure shock and momentum change effects have been accounted for with respect to pressure relief, piping design, and support systems should also be included.

Implementation

The CRBRP application of this Code Case, in a manner consistent
41 with these regulatory positions, is planned in the design of the overpressure protection system for the primary and intermediate heat transport system. An overpressure protection report will be written as required by the ASME Code giving a detail description of the overpressure protection system.

The reactor vessel, closure head, and guard vessel are designed
17 and manufactured in accordance with the ASME Boiler and Pressure Vessel
Code. Section III, Division 1, Nuclear Power Plant Components, with
addenda current at the time of contract placement. Applicable ASME
17 Code Cases for elevated temperature components and certain RDT Standards
and their effective dates were also imposed. See Table 5.2-1. Access for
58 surveillance materials and in-service inspection is provided.

17 Thermal and stress analyses of the reactor vessel, closure head,
and guard vessel account for all normal, upset, emergency, and faulted
conditions imposed on the plant. The analyses cover steady-state and
transient conditions for loads due to dead weights, temperature effects
seismic events, internal pressure, vibration, and fluid flow forces.

The reactor vessel and closure head have incorporated in their
design the ability to withstand hypothesized margin loading forces which
may result from core disruptive type accidents (see Section 4.2 of Reference
17 10a in Section 1.6).

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Amend. 58
Nov. 1980

The reactor vessel, guard vessel, and closure head together with their internal structures and supports are designed to withstand the OBE (Operating Basis Earthquake) and the SSE (Safe Shutdown Earthquake) as defined in Section 3.7.

5.2.1.2 Reactor Vessel and Support (See Figure 5.2-1)

The reactor vessel is a single-walled structure and constitutes the primary structural boundary which envelopes the nuclear reactor core, the core coolant, the core coolant cover gas, and the core ancillary components and structure. The vessel also provided positioning and support for related vessels internals and the core support structure. As the central component in the primary heat removal system, the reactor vessel distributes coolant flow to remove heat generated in the core, reflector, blanket, shield, and other
41 vessel internals including structural elements. It permits core replacement without structural modifications.

17 The vessel walls and outlet, makeup and overflow nozzle penetrations are cooled by primary sodium coolant bypass flow to keep
58 the steady-state metal temperatures below or equal to 900^{0} during normal two-loop and three-loop operation and to reduce the rate of vessel wall temperature change during operating transients. The maximum velocity of the coolant in the reactor vessel and adjacent piping, exclusive of the core and flow restricting passages, does not exceed
17 30 ft. per second.

Sodium is maintained at a minimum safe level (approximately 20 inches)
41 above the primary outlet nozzles (plant elevation approximately 782 ft.) to ensure continuity of the primary heat transport system (PHTS) loop syphon in undamaged loops for heat removal even in the event of a pipe leak. The excess coolant also ameliorates differences in channel discharge temperatures and mitigates the severity of coolant temperature transients. The sodium pool also serves to attenuate radiation from the core.

The primary inlet piping and the primary outlet piping are discussed in Section 5.3.

17 The reactor vessel design includes provision to accommodate in-service inspection. Capability to visually inspect the exterior surface of all critical pressure boundary welds is provided. Accessibility to critical areas between the guard vessel and reactor vessel is provided for in-service visual inspection throughout the life of the vessel.

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The reactor vessel is supported from its upper end. The vessel support system accommodates dead weight, seismic loads, and forces hypothesized under margin loading conditions from the assembled reactor vessel and closure
58 head to the reactor cavity wall through the support ledge.

5.2.1.3 Closure Head (See Figures 5.2-1, 5.2-2)

17 The closure head is attached to the vessel flange and completes the vessel pressure boundary. Access to the reactor vessel interior is provided through
5 various penetrations in the closure head. A primary function of the closure head is to provide positioning and support for the reactor control rods. instrumentation and refueling equipment. Refueling equipment positioning is provided by means of three eccentric rotating plugs which provide access to all routinely removable elements in the reactor core. Other functions performed by the closure head are as follows:

a. Attenuation of radiation emanating from within the vessel with a
42 limitation of 25 mr/hr at a distance of 3 ft. from the closest accessible surface.

b. Limiting leakage of radioisotopes such that in conjunction with head access area ventilation the dose rate due to leaked gas will not exceed 0.2 mRem/hr.

c. Thermal insulation of structural regions of the closure head and access area from the core coolant.

d. Prevention of excessive gas entrainment within the core coolant.

e. Support and attachment of all head mounted equipment.

f. Sealing all head penetrations during all normal, upset, emergency and faulted conditions.

g. Provision for maintenance of seals between rotating plugs.

h. Provisions for Structural Margins Beyond the Design Base (See Reference
60 10a, PSAR Section 1.6). 2

The closure head has an average temperature of 400°F under both normal operation and refueling conditions. This temperature assures that the liquid sodium will not freeze. It is low enough that the elastomer seals on the top
42 of the risers can be kept at approximately 125°F for adequate life though provision is made for seal replacement. The design life of these seals

is 5 years.

60 The seals for head mounted equipment are located in areas that are accessible for maintenance by special tools or for hands-on-maintenance. Those items requiring hands-on-service are kept at a maximum of 125° F for metalic and 140° F for insulated surfaces and most are elevated a distance above the 400° F head. All of the components requiring major repair are designed to be removed in modular assemblies. The following is a brief description of the major head mounted components.

PCRDM

Two types of seals are used to seal the primary pressure boundary of the CRDM and CRDM nozzle extension. The "conoseal" manufactured by Aeroquip Corp. will be removed and discarded during the maintenance procedure. These seals are located at the extreme top of the motor tube (elev. +137.87) and at the bottom of the motor tube in the mouth of the nozzle extension (elev. +89 and +87). At these elevations the temperature is approximately 150°F. The seals, in all cases, are inserted and removed with a conoseal installation and removal tool. The conoseal, which resembles the common "Belleville" spring washer, is actuated by the clamping force of the mechanism hold down bolts or packing screws.

The second seal is a canopy seal located at the interface of the nozzle extension and stub nozzle (Elev. +6). These seals will be broken only in the unlikely event that a shroud tube or nozzle extension requires replacement.

SCRDM

Any maintenance required on the SCRDM will be done after removal of the SCRDM from the reactor. Similar to the PCRDM, a "conoseal" joint (elevation 87") can be loosened and the SCRDM removed for repair. The seal between the SCRDM nozzle extension and stub nozzle (elevation 6") is identical to the Primary Control Rod System.

Riser Dip Seals

The sodium dip seals, operating at 400°F, will be cleaned by mechanical scraping tools through ports permanently located at the base of each riser assembly. A sodium dip seal supply system will also flush and fill the dip seal trough. All of this equipment will be either gas-actuated or manually actuated. Manual interfaces to the equipment will be located such that their 25
temperature is that of the head access area, 80°F.

Riser Elastomer Seals

The balance of the seals on the riser assembly operate at temperatures below 125°F.

Upper Internals Structure Jacking Mechanism

The UIS jacking mechanism utilizes metal buffered seals in the 400°F areas. These seals are part of the mechanical assemblies. The seals will be removed with components at the appropriate maintenance period. Elastomer seals are located in the cooler regions, have a service life of five years, and will be replaced using hands-on maintenance.

Liquid Level Monitor Ports Plugs

Four of these components, operating at 400°F, are located on the reactor vessel head and provide receptacles for holding the liquid level monitors. Three small port plugs are attached to the top surfaces of the closure head rotating plugs by partial penetration welds, two on the intermediate and one on the large rotating plug. One large port plug is bolted to the top surface of the large rotating plug and is sealed to the plug by double metal "O" rings. The seals remain attached to the port plug during installation and removal. Because the port plug remains stationary relative to the head assembly, the metal "O" rings beneath the plug flange are not expected to require maintenance.

5.2.1.4 Guard Vessel

The guard vessel provides for the retention of the primary sodium coolant in the event of a leak in the portion of the primary coolant boundary which it surrounds. The guard vessel geometry assures reactor vessel outlet nozzle submergence after such a leak which will maintain continuity in operating primary coolant loops to provide core cooling. The guard vessel also provides a uniform annulus for in-service inspection of the reactor vessel, with clearances that preclude contact with the reactor vessel and piping under accident conditions. Insulation for the reactor vessel and a heating system for the reactor vessel to be used prior to sodium fill and during prolonged shutdown are also mounted upon the guard vessel.

The Reactor Guard Vessel has pipes which surround the primary inlet and outlet piping to an elevation which matches the top flange of the guard vessel, 788' 0" (shown in Figure 5.2-1). The piping is welded to nozzles in the guard vessel shell. The piping elbows are formed from five mitered sections of piping. The mitered sections are welded together to form a 54" radius on the outlet pipes and a 36" radius on the inlet pipes. The piping is made from 1" plate which was rolled and welded with one longitudinal seam per section. the seams have been staggered on alternate sides of the mitered elbows and straight sections. The inside diameter of the outlet piping is 53" and the inlet piping is 41". The outlet piping has a 4'5.7" straight section above the elbow. The inlet piping has three 9'5" shell courses and one 1'6.5" shell course above the elbow. All sections are welded together with circumferential seams. The inlet and outlet piping has a plate flange at the top with a 60" O.D. on the outlet and 48" O.D. on the inlet. The flanges have been drilled and tapped with eighteen, 3/4" diameter holes equally spaced to provide for insulation support. The inlet piping has been provided with a seismic restraint which allows thermal growth by providing a support with a concentric gap around the piping but limits lateral pipe movement during a seismic event. The seismic restraint is attached to the guard vessel shell. All welds and NDT have been made in accordance with the ASME Code and Guard Vessel Equipment Specification.

The maximum and minimum widths of the radial gap between the guard vessel and the reactor vessel have been conservatively calculated, taking into account all relevent factors such as tolerances on the diameters of the two vessels, permissible out-of-roundness of the two vessels, possible deviations from straightness due to manufacture and subsequent operation, thermal expansion, initial deviations in the alignment of the two vessels, etc. The transporter for the television camera will be designed to accommodate itself to this maximum possible range of gaps as it moves in the space between the two vessels.

5.2.1.5 Reactor Vessel Preheat

The Reactor Vessel Preheat System will control the dry heat-up and cool down of the Guard Vessel, Reactor Vessel and Internals between ambient (70°F) and 400°F and if required will provide make-up heat for that lost to the Reactor Cavity during prolonged shutdowns.

The heat will be provided by tubular electrical heaters mounted between the Guard Vessel and insulation. These heaters will be arranged circumferentially around the Guard Vessel and will be grouped and controlled in zones of uniform heat output. Temperature sensing devices will monitor the Guard Vessel temperature in each of these zones and provide the necessary feedback for power level adjustments in the heaters.

The heaters will be mounted to the same framework which supports the Guard Vessel insulation. Attachment clips will offset the heaters from the Guard Vessel surface. Convective barriers, reflective sheaths and Guard Vessel insulation will be used to optimize heat input to the Guard Vessel and minimize losses to the Reactor Cavity.

Preliminary preheat, startup, shutdown analyses have been performed on the Reactor Vessel and Guard Vessel to determine the temperature differences which will result in opening and/or closure of the annular gap between the two vessels. By necessity the preheat analysis is very preliminary since no firm preheat procedure has yet been developed. Figures 5.2-4 through 5.2-6 show the temperature differences between the Reactor Vessel and Guard Vessel in the inlet and outlet plenum regions for the three transients in question. As shown the largest positive temperature difference between the Reactor Vessel and the Guard Vessel occurs in the outlet plenum region during startup (335°F) while the largest negative temperature difference occurs in the outlet plenum region during shutdown (-214°F). The nominal radial gap between the reactor vessel and guard vessel is 8 inches at assembly and at the end of preheat. This gap decreases to approximately 7.6 inches minimum during start-up and increases to approximately 8.3 inches maximum during shutdown. During preheat the gap also increases but to a lesser value than during shutdown due to the smaller maximum temperature difference.

Variations in the axial gap between the bottom of the reactor vessel and the inner surface of the guard vessel are noted between the states shown in the table. Thus, the largest axial gap is 11.0 inches at the dry cold condition and the smallest gap is 6.2 inches at the end of the heating phase of preheat.

5.2.1.6 Closure Head Heating

The Closure Head Heating Temperature Control System consists of a single master temperature set point device which is used as the set point reference by the individual heater zone controllers.

The individual heater zone controllers control the temperature of the closure head and reactor vessel support ring based on the temperature reference indicated and the individual zone temperatures as indicated by embedded temperature sensors located within the individual zones. Each zone has multiple temperature sensors. Failure of a single temperature sensor will be detected and alarmed.

The heaters are placed within individual zones to account for heat sinks, i.e., risers, nozzles and other head mounted equipment.

All elements of the head heating and control system are classified as Seismic Category II.

Sufficient redundancy has been incorporated into the design to assure that failures of individual components shall not cause degradation of system performance to the point where the thermal requirements are not met. A technical specification will be provided to address operations with failed heaters and temperature sensors.

5.2.2 Design Parameters

Overall schematic views of the reactor vessel, closure head assembly, inlet and outlet piping, and guard vessel are shown in Figures 5.2-1, 1A and 1B. The top view is given in Figure 5.2-2.

5.2.2.1 Reactor Vessel and Support

The reactor vessel and support will be constructed mainly of austenitic stainless and low alloy steels, and consists of six basic sections: the support ring, the vessel flange, the barrel, the core support forging and cone, the inlet plenum, and the vessel thermal liner. The support ring is a SA 508 Class 2 steel forging welded to the vessel flange. A box ring type of reactor vessel support interfaces with the vessel support ring and the reactor cavity support ledge. Holddown bolts pass through holes in the vessel support ring, the reactor vessel support and the support ledge clamping the three together. The vessel support is a ring structure with a box type cross section. The vertical sides of the box are Inconel 600 to limit the heat flow from the reactor vessel. The top and bottom plates of the box cross-section are SA 543 Class 2. The bolts are SA 193 Type B7 with 3.50-8UN threads. The ring supports the reactor vessel and internals and closure head. The vessel flange is a second SA 508 Class 2 steel ring forging welded to an Inconel 600 transition section. The latter is, in turn, welded to the barrel. Radiation shielding in the form of a boron carbide collar surrounding the vessel near the flange is provided in the annulus between the reactor vessel and the vessel support ledge. The barrel comprises the upper cylindrical portion of the vessel and has an inside diameter of 243 in. with a minimum wall thickness of 2.38 in. The lower end of the barrel is joined to the core support forging and cone, which provide support for the core support structure. The overall height of the reactor vessel and support is nominally 704 in. (58 ft. 8 in.). The inlet plenum is designed for 200 psig at 775°F and -15 psig at 600°F, the stainless steel portion of the outlet plenum is designed for 15 psig plus head of sodium at 900°F and -15 psig at 600°F.

Coolant enters the reactor vessel through three 24-inch nozzles located 120° apart in the inlet plenum below the core support structure. Core effluent and bypass flow are mixed in the outlet plenum region above the core, and the

coolant is discharged from the vessel through three 36-inch nozzles. The
6 inlet and outlet nozzles are designed for a total sodium flow rate of
17 41.46×10^6 lbs/hour (42.3×10^6 lbs/hour stretched condition).

The vessel thermal liner protects the barrel from the high-temperature sodium in the outlet plenum and from excessive thermal
47 transients. The annulus between the vessel and liner is 2.5 inches. By-pass coolant from the inlet plenum is diverted to this annulus to maintain
17 the vessel wall temperature at $900^{\circ}F$ or below during normal operation. Two
47 holes with special pressure reducers are provided in the core support cone to vent gas from underneath the cone to the outlet plenum.

The bypass flow which cools the vessel wall behind the thermal liner enters the outlet plenum through bypass flow holes in the thermal liner located a few inches beneath the sodium pool free surface level. Sufficient mixing occurs away from metal surfaces so that excessive fatigue damage to the Type 316 stainless steel will not occur.
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Four auxiliary nozzles are provided in the upper part of the
17 barrel. Two of the nozzles are for the auxiliary sodium system: one inlet nozzle for sodium makeup and one outlet nozzle for sodium overflow. The other two nozzles are for the cover gas system: one gas inlet nozzle and one gas outlet nozzle. The sodium makeup nozzle is 4 in.
17 diameter while the overflow is 8 in. in diameter. The nozzles provide a loop whereby the sodium level can be controlled, impurities can be removed from the sodium and samples of the coolant taken. Both cover gas nozzles are 3 in. in diameter. They provide for circulation of the cover gas to and from the Radioactive Argon Processing System and for controlling the pressure of the primary coolant system.

The reactor vessel support box ring, the seismic keys, and the anchor bolts form the reactor vessel support system. In addition, a design
17 requirement of the vessel support system is to extend plant capability by providing Structural Margin Beyond the Design Base (see Section 4.2 of
58 Reference 10a of Section 1.6).

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36 5.2.2.2 Closure Head

The closure head consists of three rotating plugs which are constructed of SA 508 Class 2 steel. Each plug contains a major penetration eccentric to its outside diameter. These rotating plugs are interconnected
17 by means of a series of plug risers. Sealing between the plugs is accomplished by sodium dip seals and double inflatable seals of elastomer material. At
36 its top, the large rotating plug has an outer diameter of 257 in., and an inner diameter of 176 in. The large rotating plug provides access to the vessel interior for the ex-vessel transfer machine and the core coolant liquid level monitors. The intermediate rotating plug (175 in. O.D. and 68 in. I.D.) provides access to the vessel interior for the control rod
41 drivelines, upper intervals support columns, and the liquid level monitors. The small rotating plug (67 in. O.D.) provides access to the vessel interior for the In-Vessel Transfer Machine. The thickness of each rotating plug is
41 22 in. Rotation of the plugs will be accomplished by a gearing and bearing system attached to the plug risers. The nozzles for each penetration
41 are constructed of an austenitic stainless steel.

58 Each rotating plug is provided with a mechanical lock and electrical interlocks which prevent plug rotation during reactor operation and refueling when plug rotation is not desired.

The mechanical locks include the following:

a. Each plug includes a separate positive lock to assure that the plug cannot be moved, and will not drift from its normal operating position during reactor operation. This lock will be installed to prevent relative rotation
58 between each bull gear and the bearing outer riser whenever the control rod drivelines are connected. The locks shall be manually installed at the end of each refueling cycle, and will be removed only during the refueling period when plug rotation is necessary.

b. The plug drives are designed to be self locking to react to any seismic torque occurring during refueling, which could rotate the plugs and thus damage a fuel or blanket assembly during removal from the core.

The electrical interlocks include the following:

a. During reactor operation, the plug drive and control system keyswitch is in the OFF position, the control system is deenergized, and there is no power to the
24 plug drive motors.

b. Electrical interlocks are provided to prevent the plugs from being inadvertently rotated by their drive system unless the upper internals are raised and locked, and the IVTM and EVTM are in a safe condition.

c. An electrical interlock is also provided to prevent vertical operation of the IVTM or operation of the EVTM over the HAA during operation of
24 the plug drive system.

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Each rotating plug has attendant thermal and radiological shielding extended to a depth of 74.65 in. beneath the top of each plug forging. The shielding is composed of a series of plates fabricated from carbon steel, and stainless
17 steel. The cover gas between each set of plates attenuates thermal conduction and thereby acts to decrease the heat flux imparted to the rotating plug. A heating and cooling system is provided to maintain the closure head at 400°F (nominal) as well as providing heating and cooling for other small head
17 mounted subassemblies.

45 A gas entrainment suppressor plate assembly is positioned beneath the head
17 thermal and radiological shielding at a depth of 122.65 in. beneath the top of each rotating plug. It protects the head shielding from being contacted by the core coolant and minimizes the amount of cover gas entrained in the core
17 coolant. The assembly is designed to accommodate all normal, upset, emergency, and faulted conditions.

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42 In plan view, the subassembly consists of 33 plates at the same elevation with horizontal gaps between them. (Fig. 5.2-3) These plates have penetrations in line with the head penetrations to allow the passage of the head mounted components into the outlet plenum. Each plate is supported by means of a central support column affixed to the lower shield plate. These central columns, when possible, consist of tubes which surround closure head penetrations. Support columns which do not surround penetrating equipment will be capped to minimize the amount of cover gas entrained in the sodium pool. The support columns will be inserted through oversized penetrations in the lower shield plate, accurately positioned and then attached to the top surface of the lower plate by means of bolting. The support columns will be attached to the suppressor plate by means of welding. This attachment weld is located above the region of the suppressor plate where high thermal gradients occur by using a plate with an extruded weld neck. The top end of the support column, which protrudes through the lower shield plate is composed of 2 1/4 Cr-1Mo. material to minimize the differential expansion with the carbon steel
57 42 shield plate. The lower, in sodium, portion is austenitic stainless steel. The use of a single support provides adequate support while lessening the thermal stresses by permitting the plates to flex freely under the expected 2
thermal gradient.

The riser has been designed to maintain a maximum temperature of 125°F in the region of the elastomer seals. Thermal analysis has been completed for this design which shows that this temperature (125°F) is maintained by the head access area cooling system.

5.2.2.3 Guard Vessel

The guard vessel is a bottom-supported, right circular cylindrical vessel surrounding the reactor vessel. It was fabricated from SA240 Type 304 stainless steel. The purpose of the guard vessel is to assure outlet nozzle submergence in the event of a leak in the reactor vessel nozzles, piping, or piping connections. To fulfill this requirement the guard vessel extends to approximately 6 ft. above the minimum safe sodium level providing for sodium shrinkage and pumping head differential. Also, the guard vessel permits inservice inspection of the reactor vessel by providing a nominal clearance of 8 in. between the two. The guard vessel is insulated on the outer surface to limit the heat load into the reactor cavity cell and to reduce heat loads to the Reactor Cavity Heating and Ventilating System. A trace heating system is mounted on the outside surface of the vessel for pre-heating and heating during prolonged plant shutdown.

Flux monitors for low, intermediate and full power operation are provided in the annulus between the guard vessel and reactor cavity cell walls. This annulus will be filled with nitrogen gas ($\leq 2\%$ oxygen by volume). A discussion of the reactor cavity cell is found in Section 3.8 and 3.A.1.

Continuity detectors and aerosol sampling lines are mounted inside the guard vessel to detect potential leaks in the reactor vessel or inlet or outlet piping. See Section 7.5.5.1.

5.2.3 Special Processes for Fabrication and Inspection

5.2.3.1 Nondestructive Examination

Nondestructive examination of materials and welds will be performed in accordance with the ASME Boiler and Pressure Vessel Code and RDT Standards. The techniques employed, as appropriate for the respective product forms, materials, and weld configurations comprising the reactor vessel, closure head, and guard vessel, are liquid penetrant, magnetic particle, ultrasonics, and radiography. Surface finish and cleanliness will also meet all requirements of the ASME Code and the other contract documents. Periodic swab tests of stainless steel surfaces during fabrication in the shop will be performed to assure that potentially harmful substances such as chlorides do not contact the components in concentrations greater than specified in applicable codes and standards.

5.2.3.2 Controlled Welding to Maintain Alignments

Specified alignments must be maintained between the core support structure and the upper end of the reactor vessel. Where welds such as girth seams in the vessel and the weld attaching the core support structure to the vessel influence these alignments, special welding procedures and processes utilizing proven technology will be used to control the relative alignments of the parts being joined by welding.

Prior to any welding, the core support structure will be aligned by equalizing the gap between the core barrel and thermal liner support ring at four points located 90° apart. Also, the weld preps on the core support structure and core support cone will be aligned vertically and radially using the respective weld lands as the reference surface. Four wedges, which have been contour machined to match half the weld geometry, are placed in the top of the joint. Their purpose is to prevent movement during initial welding.

Welding will be accomplished by using four welders positioned 90° apart. Movement of the core support structure during welding will be monitored by measuring the distances between the core barrel and the thermal liner support ring at four equally spaced locations. If 1/16 inch or more distortion occurs, welding will stop on one side and continue on the opposite side until re-alignment occurs. Continuous monitoring will be performed until 1/2 inch of weld has been deposited. At that point, the wedges will be removed and periodic monitoring will be performed during the remainder of the welding.

After welding is complete, the weld prep for the top portion of the vessel at the top of the thermal liner support ring will be machined concentric to the centerline of the core support structure. The top portion of the vessel is fabricated so that the bottom weld prep is concentric to the vessel flange. The top portion of the vessel is then assembled to the lower assembly using a ship-lap joint (sometimes known as a spigot fit). With this joint, no special welding techniques are required to maintain alignment, it is purely manual metal-arc welding. By having precise alignment within the two sub-assemblies, that is, centerline to weld prep and by using a precision fit-up of these sub-assemblies, the core support structure is located to vessel flange within the required tolerance.

The weld circumference was divided into four quadrants, each of which was divided further into 12-inch increments. The first weld pass was made using four welders working simultaneously, one welder per quadrant. The position of the core support structure then was measured. If a significant movement was found to occur, it was corrected by welding 12-inch increments which were selected by the welding engineer. The subsequent passes were welded and corrections made as necessary. This was repeated until movement of the core support structure ceased.

The selective placement of weld passes to control distortion during welding does not result in localization of overlaps or start-stops. The weld overlaps or start stops are no different from those encountered in normal arc welding. Sensitization is controlled, as it is in other shop fabrication and field welds, by limiting the interpass temperature to 350°F maximum per RDT Standard E15-2-NB, which is imposed by appropriate equipment specifications.

5.2.3.3 Dimensional Checks

All dimensions of the reactor vessel, closure head, and guard vessel will be measured and checked against the dimensions and tolerances specified on the manufacturing drawings. Any deviations will be documented by Supplier

Nonconformance Reports. Approval of dimensions not in accordance with the drawings will be granted only after determining that safety and operability of the plant will not be affected adversely. Deviations which do not meet the requirements of the ASME Code will not be permitted.

5.2.3.4 ASME Code Pressure Tests

Pressure tests will be performed on the completed reactor vessel and on the completed closure head as required by the ASME Code.

The high-pressure inlet plenum portion of the reactor vessel has been pressure tested to a pressure of 250 psig. This pressure test took place after the installation of the core support structure. The pressure test also provided structural verification of the core support structure, although not required by the ASME Code. Following the pressure test of the inlet plenum, the entire vessel was pneumatically tested; during this test, the upper end of the vessel was sealed by a test head.

The closure head was pneumatically tested to a pressure of 18 psig. A suitable test fixture was used to retain the head and apply the test pressure to it.

5.2.4 Features for Improved Reliability

5.2.4.1 Reactor Vessel Thermal Liner and Nozzle Liners

In order to protect the pressure boundary of the vessel in the outlet plenum region from high temperatures and severe temperature gradients during steady-state and transient conditions, the reactor vessel is provided with a thermal liner that extends downward from above the sodium pool level to an elevation below the core support horizontal baffle. Nozzle liners for this purpose also are provided for the three outlet nozzles and for the makeup nozzle.

5.2.4.2 Internal Elbows in Reactor Vessel Inlet Plenum

In order to promote mixing of the three inlet streams in the reactor vessel inlet plenum and minimize thermal gradients in the pressure boundary of the inlet plenum, each inlet nozzle is provided with an internal pipe elbow that deflects the flow downward and away from the wall. In this manner, the mixing of the entering sodium occurs in the interior of the plenum, providing coolant uniform temperature to core components.

5.2.4.3 <u>Deleted</u>

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5.2.4.4 Plug Seals

The annuli between the plugs are designed with offsets to minimize radiation streaming. Dip seals provided in the closure head plugs block the transport of radioisotopes into the upper annuli and maintain dose rates due to leakage below acceptable limits. A dip seal is an annular trough filled with liquid sodium. This forms a liquid sodium trap around the periphery of the annulus, sealing the upper annulus above the dip seal from cover gas space. This arrangement is shown on Figure 5.2-7.

The dip seals are located in the closure head, 400°F region, providing a high integrity seal at the base of the annular section formed by the closure head and riser assembly. The upper end is closed by a combination of two inflatable elastomer seals and a set of elastomer static o-ring seals. The elastomer seals are separated by a buffer space to give assurance that oxygen and moisture leakage from the Head Access Area is limited to amounts that give acceptable sodium frost deposit rates and to insure proper dip seal pressure capability. 20

Figure 5.2-8 gives enlarged views of the elastomer seal areas to show the sealing arrangement. The upper view shows typical seals for the top end of the riser. The inflatable dynamics seals are mounted into a removable seal retainer ring, which is sealed to the riser top 20
by elastomer o-ring seals. The space enclosed by the two inflatable seals is a static buffer.

The CRBRP inflatable seals are the same in cross-section and made of the same material (nylon-reinforced nitrile rubber) as the FFTF-IVHM inflatable seals. Extensive testing was conducted to qualify the IVHM inflatable seal at an operating temperature of 150°F (Reference 6)
41 Additional performance testing has been conducted (Reference 7) on an FFTF-IVTM type inflatable seal to assure these seals will
58 function reliably under the CRBRP speed and pressure operating conditions.

Specific objectives of the CRBRP program included buffer cavity leakage and seal drag measurements under combinations of seal and buffer cavity pressure, measurement of gas diffusion through the seal, breakaway seal drag as a function of dwell times, effect of lubrication and long duration (life) cycling on the wear characteristics of the seal, and the effect of horizontal and vertical runout of the seal runner on seal performance. The verification of elastomer life at 125°F is documented in
58 Reference 8. The life of the elastomers for upward of 5 years in this temperature and environment has been demonstrated by tests. 25

The seal retainer ring is sealed to the inner riser by two solid elastomer o-ring seals. These have an inerted, static buffer space between them which acts similar to the one described above. The seal follower is sealed to the outer riser in the same way, but with the space between seals 20
49 purged with a small argon flow instead of static buffering.

The primary source of experimental data for CRBRP elastomer seal design is the Cover Gas Seal Development Program performed by Atomics International (AI).
58 Testing of elastomer seals and materials included static, rotary, and reciprocating seal leakage tests, compression set tests, gas permeability measurements, and seal lubricant evaluations including elastomer compatibility, thermal stability and friction testing. Elastomers tested include silicone, ethylene-propylene, urethane, nitrile (Buna N) and butyl rubber supplied by three different seal vendors. Test temperatures ranged from 100 to 300°F depending upon the elastomer being tested. In addition to the numerous quarterly reports which have been published, two summary reports
41 58 have been issued (References 8 and 9). The information contained in these reports amply demonstrate the ability of several elastometers to meet CRBRP design requirements at 125°F. These tests demonstrate that 25
elastomer life of upward of 5 years in this temperature and environment is achievable.

A test program on the performance of sodium dip seals was carried out by Atomics International. The results of these tests were reported in Reference 10. The testing verified the acceptability of leak rates, dynamic stability, annulus frosting, effect of solids buildup in the dip
58 seal, wetting and cleaning.

The closure head has several penetrations that permit access to the reactor interior. Penetrations will be sealed in one of two ways. One is to provide two seals with a pressurized buffer space between them. In the event of a failure of the inner seal, that between the reactor interior and the buffer space, buffer gas will leak into the vessel. In the event of a failure in the outer seal, buffer gas will leak into the head access area.

In each case, the leak will be detected and repaired before a leak can occur between the reactor interior and the head access area. The other method is to use a hermetic seal.

Figure 5.2-8 shows the method of sealing the riser bases to
the vessel head. Soft coated metallic o-rings with an inerted, purged
space between them are used in the same way as the elastomers at the 20
49 riser top. A continuous metal "C" ring or canopy seal is welded at both
ends completely around the periphery to provide a hermetic seal at the base
5 of the large outer riser. The base of the small and intermediate outer
risers are welded directly to the closure head and therefore have no leak
57 path at the juncture.

General approach to seal selection is:

1. For openings which are at high temperature and/or require long life, metallic seals, hermetic or double buffered, are used.

2. Seals which operate at low temperature and can be replaced relatively frequently are elastomer sealed.

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A margin seal is provided to meet the leakage requirements associated with Structural Margin Beyond the Design Base (SMBDB) (see Section 4.2 of Reference 10a of Section 1.6). The riser drawing, Figure 5.2-7, shows the margin seals at the upper end of the riser assembly. The seals are in contact with the bearing and risers thus sealing the upper end of the annulus against radioactivity that is not accompanied by significant pressure increase. SMBDB sodium slug forces and residual pressure are expected to cause the inner riser to rise relative to the seal until the margin shear ring in the head stops the upward motion. Sufficient seal surface overlap will be provided to accommodate this motion. The pressure increases will seat the seal even more tightly, thus providing increased sealing capability. The margin seal is expected to meet the gas and sodium leakage requirements.

The internal riser space between the margin seals and the inflatable seals is purged with a small flow of clean argon gas. The purging of this seal space, along with the buffer space between the two elastomer o-rings in the bottom of the seal follower and between the two metallic o-rings in the bottom of the inner riser, is provided by a seal service system which also provides continuous pressure for the inflatable seals and argon gas for the static buffer space between the pairs of inflatable seals and between the pairs of elastomer o-ring seals in the bottom of the seal retainer rings.

5.2.4.5 Surveillance and In-Service Inspection

5.2.4.5.1 Surveillance

Representative surveillance materials will be obtained from the various product forms, including weldments, from which the reactor vessel and guard vessel are fabricated. The requirements of Appendix H to 10CFR50 will not necessarily be followed since they were generated for ferritic material and the CRBRP reactor vessel will be austenitic. Appropriate surveillance samples will be placed inside the reactor vessel and/or guard vessel, thus providing means for monitoring and evaluating potential material degradations. In-service inspection and monitoring shall include as objectives the requirements listed in Table 5.2-2.

To provide additional assurance in the area of fracture toughness, the project will either; 1) provide confirmatory data supporting that the loss of fracture toughness for values around one dpa are neglible, or 2) include fracture toughness surveillance in CRBRP.

The philosophy of the CRBRP Materials Surveillance Program is to demonstrate continuing safe operation of permanent (30 year) reactor components by monitoring radiation induced property changes in the component materials. Monitoring of change is accomplished by testing materials specimens irradiated in the reactor, at intervals during the life of the plant and by comparison of the test data with data from unirradiated materials. 19

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The method of obtaining data representative of irradiated permanent component materials consist of 1) selection of coupons of component materials used at locations where it is predicted that detectable change will occur, 2) fabrication of test specimens from the coupons, 3) irradiation of the specimens in the reactor in environments which will provide advanced data and 4) withdrawal of the specimens at planned intervals during the plant life and testing of the specimens at component anticipated service temperature. Details of the coupon/speciment selection, irradiation and testing requirements are as follows: 19

Coupon Selection Requirements

1. The materials of the permanent Reactor System components, which are designed for the full life of the plant, shall be considered for representation by surveillance coupons.

2. Materials surveillance coupons, to monitor radiation effects in the materials of permanent reactor system components, shall be required if the predicted fluence is greater than 1×10^{21} n/cm^2, E > 0.0, in the component material.

3. Subject to requirements 1 and 2, base metal and weld metal coupons shall be required.

4. For each location defined by the application of requirements 1 through 3, sufficient material shall be obtained, during fabrication, to produce coupons from which 15 test specimens shall be fabricated.

5. The test specimens shall be sub-size tension specimens as indicated in ASTM E-8, having a gage diameter of 1/4 inch and a gage length of 1 inch and an overall length of 2 5/8 inch.

Test Specimen Irradiation Requirements

1. Surveillance test specimens shall be irradiated in the Removable Radial Shields and/or the Fuel Transfer and Storage Assembly as required to obtain environmental conditions as noted below.

2. Three test specimens of each component material, defined by the coupon selection requirements, shall be placed in a capsule set. (A capsule set shall be one or more individual capsules as required to obtain environmental conditions as noted below). Four capsule sets shall be assembled. 19

3. The four capsule sets shall be in place in the reactor at startup. One set shall be withdrawn at each of 1/4, 1/2 and 3/4 of plant life (to the nearest normal refueling interval). The fourth set shall remain in the reactor as a contingency set.

4. Positioning of the capsule sets and the distribution of test specimens within the capsules shall be such that the minimum anticipated fluence on the test specimens shall be as follows:

   o Test Specimen to be withdrawn at 1/4 of plant life shall have anticipated total fluence at least equal to the anticipated total fluence on the component material at 1/2 plant life.

   o Test Specimens to be withdrawn at 1/2 of plant life shall have anticipated total fluence at least equal to the anticipated total fluence on the component material at 3/4 plant life.

   o Test Specimens to be withdrawn at 3/4 of plant life shall have anticipated total fluence at least equal to the anticipated total fluence on the component material at full plant life.

   o Irradiation of the contingency test specimens shall essentially duplicate irradiation of the test specimens scheduled for withdrawal at 3/4 of plant life.

5. The test specimens shall be positioned so the anticipated total flux shall not exceed three times the anticipated total flux on the component material.

6. The test specimens shall be positioned to best simulate other component material service conditions after fluence criteria are met.

<u>Test Specimen Testing Requirements</u>

1. Three test specimens of each component material, defined by the coupon selection requirements, shall be tested in the unirradiated condition to provide reference data.

2. Irradiated specimens shall be tested after removal from the reactor according to the schedule defined by the irradiation requirements.

3. Specimens shall be tested at a strain rate of 3×10^{-5} in/in/ sec and at the anticipated service temperature of the component material.

4. Testing procedures shall include the use of extensometers and other devices to produce a record of load and elongation data.

5.2.4.5.2 <u>In-Service Inspection</u>

In-service inspection (ISI) equipment is provided to perform a visual examination of the outer surface of the welds on the reactor vessel and nozzles, and the inner surface of the welds of the reactor guard vessel. These examinations are to be performed during those periods when reactor coolant temperature is approximately 400°F. The ISI equipment for the reactor vessel/guard vessel annulus consists of a TV camera, transporter, and cabling to provide for cooling and appropriate electrical interfaces.

The overall sensitivity of the TV camera will be such that accumulations of liquids, liquid streams, liquid drops and smoke are discernible. The TV examination will also be capable of determining the presence of loose parts and debris.

The reactor vessel, guard vessel, guard vessel extension, and support ledge insulation form an assembly designed to provide transporter access to all reactor vessel welds, excepting portions of three reactor vessel longitudinal welds masked by the reactor cavity radiological shield and the reactor guard vessel longitudinal welds covered by the leak detector tubes mounted to the guard vessel. The transporter will be similar to the transporter design employed on FFTF.

5.2.5 Quality Assurance Surveillance

Quality assurance surveillance for the reactor vessel and reactor vessel guard vessel has been performed by quality assurance personnel who were present at the fabricator's facility during all important phases. Quality assurance personnel have monitored all important phases of fabrication for the closure head. The interfaces between the various QA organizations are given in Chapter 17.0.

5.2.6 Materials and Inspections

The materials used in fabricating the reactor vessel, closure head and guard vessel are summarized in Table 5.2-3. In general these materials (for the reactor vessel and closure head) conform to ASME Boiler and Pressure Vessel Code, Section III, and the supplemental requirements of RDT Standard E15-2NB-T. The materials for the guard vessel conform to ASME Boiler and Pressure Vessel Code, Section III, requirements for Class I Vessel.

Requirements for delta ferrite content are given in the ASME Code, Section III, ASME Code Case 1592, RDT Standards M1-1T and M1-2T, and the reactor vessel specification. In all cases, the determination of delta ferrite content will be made from chemical analyses of welding materials as applied to the Shaeffler Diagram in the ASME Code. There is no requirement that delta ferrite determinations be made from production welds.

The service environment and temperature will not result in material degradation effects in the combination of dissimilar metals and weldments utilized between the Type 304 stainless steel vessel and the SA 508 class 2 ferrite vessel flange.

The transition region of the vessel is located in a position which has a total fluence of less than 10^{15} n/cm^2 (E>0.0 MeV). This low fluence level is considered to be below the threshold level for mechanical property degradation of the materials involved (Ref. 1).

The service temperature for the SA 508 to Inconel 600 weld is approximately 465°F and that for the weldment between the Inconel 600 and Type 304 stainless steel is about 650°F. At these operating temperatures, the three base metals involved together with the Inconel 82 weld filler metal are metallurgically stable. (Ref. 2 and 3).

Both the internal and external environment are considered benign with regard to degradation of the various materials in the transition region. The internal environment will be argon gas and sodium vapor and essentially no sodium mass transfer or interstitial transfer effects occur at temperatures below 700°F, especially when the sodium is present as a vapor or thin condensed layer (Ref. 4 and 5). The external environment is reactor cavity gas, nitrogen plus approximately 2% oxygen, and the interaction of the materials involved with this reactor cavity gas are negligible at the low service temperatures.

Selection of the materials for this transition joint was made based on the above considerations coupled with the requirement to minimize thermal expansion differences which could cause high stresses to be built up during thermal cycling. In addition, the use of the nickel base alloy filler metal, Inconel 82, minimizes the depletion of carbon from the fusion zone of the SA 508 during welding and subsequent high temperature stress relief.

Inspection of materials for the primary pressure boundary (Reactor
Vessel and Closure) will be in accordance with the RDT material standards for
the particular materials. Inspection during fabrication will be in accor-
41 27 dance with Section III of the ASME Code and RDT E15-2 NB-T, Class I Nuclear
Components. Inspection of materials and inspection during fabrication of
the Guard Vessel will be in accordance with ASME Code, Class I and RDT E15-
41 27 2NB-T. The overall inspection and test plans for the three structures will
17 be prepared by the fabricator and approved by the purchaser prior to fabri-
cation.

5.2.7 Packing, Packaging & Storage

Applicable requirements to assure adequate quality during shipping
58 43 and storage are in the respective equipment specifications rather than
in RDT Standards.

58 The specifications require that packaging and packing be adequate
to protect items while the suppliers' facilities, during transportation to
31 the delivery point and during storage at the site.

58 The specifications do, where appropriate, provide requirements for
sealing the openings in the components, purging the components and/or their
31 containers, selecting and using desiccants, selecting and using materials
contacting the components which are suitably free of chlorides, flourides,
lead, copper, zinc, cadmium, sulfur, mercury, etc.

57 During storage, the equipment is being maintained in a dry gas
31 environment, where appropriate, to protect it from contamination. The purge
gas, container integrity, etc., is being monitored to assure compliance
58 with previously prepared procedures.

57 Protective measures to be taken during construction will be
provided by the construction contractor. 25

REFERENCES FOR SECTION 5.2

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2. "Inconel Alloy 600", Huntington Alloy Products Bulletin T-7, 1969.

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4. S. Schrock, S. Shiels, C. Bagnall, "Carbon - Nitrogen Transportation in Sodium Systems", Summaries of Technical Papers, CONF-76-05-3-SUM.

5. "Heat Treating, Cleaning and Forming", Metals Handbook, Vol. 2, ASME 1964, pp. 149-166.

6. Atomics International Rerport No. TR-707-810-004, "Test Report (Development) IVHM Reactor Refueling Plug Dynamic Seal Test", May 8, 1974. 22

7. Atomics International Report No. TR-707-810-009, "Test Report (Development) CRBRP Rotating Plug Inflatable Seal", July 9, 1975.

8. Atomics International Report No. AI-AEC-13145, "Design Guide for Reactor Cover Gas Elastomeric Seals", March 7, 1975.

9. Atomics International Report No. AI-AEC-13146, "Penetration, Leakage, and Compression Set Testing of Elastomeric Seals for LMFBR Use", April 2, 1975. 25

58 10. Atomics International Report No. AI-DOE-13226, "Performance Tests of Sodium Dip Seals", T. T. Shimazaki, February 28, 1978.

TABLE 5.2-1

SUMMARY OF CODE, CODE CASES AND RDT STANDARDS APPLICABLE TO DESIGN AND MANUFACTURE OF REACTOR VESSEL, CLOSURE HEAD AND GUARD VESSEL

Component/Criteria	Reactor Vessel	Closure Head* Pressure Boundary	Closure Head* Internals (as appropriate)	Guard Vessel
Section III ASME Code, 1974 Edition	Addenda thru Winter '74	Addenda thru Winter '74	Addenda thru Winter '74	Addenda thru Summer '75
	Class 1	Class 1	Class 1	Class 2**
ASME Code Cases	1521-1,1592-2,1593-0,1594-1,1595-1, 1596-1 (1682,1690 Optional)	1682,1690	1521-1 1592-4,1593-1	1592-4,1593-1,1594-1 If elected by supplier 1521-1 & 1682
RDT Standards Mandatory	E8-18T, 2/75 E15-2NB-T, 11/74 Amend thru 1/75	E15-2NB-T, 11/74 Amend thru 6/75	E15-NB-T, 11/74 Amend thru 6/75	E15-NB-T, 11/74 Amend thru 6/76
	F2-2, 8/73 Amend thru 7/75	F2-2, 8/73 Amend thru 7/75	F2-2, 8/73 Amend thru 7/75	F2-2, 8/73 Amend thru 7/75
	F3-6T, 12/74	F3-6T, 12/74***	F9-4, 9/74	F3-6T, 10/75 With Amend 1/75
	F6-5T, 8/74 Amend thru 2/75	F6-5T, 8/74 Amend thru 2/75		F6-5T, 8/74 Amend thru 11/75
	F7-3T, 11/74	F7-3T, 6/75		F7-3T, 6/75
	F9-4T, 9/74	M1-1T, 3/75 M1-2, 3/75 Amend thru 7/75		F9-4, 9/74

*For those reactor vessel and closure head components internal to the pressure boundary special purpose high cycle fatigue curves and creep fatigue damage rules have been developed as discussed in Section 4.2.2.3.2.3.

TABLE 5.2-1 (Continued)

Component/Criteria	Reactor Vessel	Closure Head: Pressure Boundary	Closure Head: Internals (as appropriate)	Guard Vessel
RDT Standards	M1-1T, 3/75			
	M1-2T, 4/75	M1-4T, 3/75		
	Amend. 6/75			
	M1-4T, 3/75	M1-6T, 4/75		
		Amend 1-7/75		
	M1-6T, 4/75	M1-10T, 3/75		
	Amend. 6/75			
	M1-10T, 3/75	M1-11T, 3/75		
		Amend 1-7/75		
	M1-11T, 3/75	M1-17T, 3/75		
	Amend. 6/75			
	M1-17T, 3/75	M2-2T, 12/74		
	M2-2T, 12/74	M2-7T, 3/75***		
	M2-5T, 1/75	M3-10T, 7/75		
	Amend 1-2/75			
	M2-7T, 2/75	M7-4T, 3/75		
	M2-18T, 4/76			
	M2-21T, 12/77			
	M3-6T, 3/75			
	M3-7T, 4/75			
	M5-1T, 11/74			
	M5-2T, 5/73			
	M5-3T, 12/74			
	M5-4T, 1/75			
	M6-3T, 2/75			
	M6-4T, 2/75			
	M7-3T, 11/74			
	M7-4T, 4/76			
Non-Mandatory	F9-5T, 9/74		F9-5T, 9/74	F9-5T, 9/74

**Functionally designated Class 2, and constructed to rules for Class 1, but not hydrostatically tested or code stamped.

***Except for the three rotating plugs, for which the applicable issues are: F3-6T, 3/69 for LRP & SRP; F3-6T, 5/74 for IRP. M2-7T, 2/69 for LRP & SRP; M2-7T, 2/74 for IRP.

TABLE 5.2-2

OBJECTIVES FOR IN-SERVICE INSPECTION

Components	Surface[a]	Type of Inspection		On-line Monitoring	
		ISI Transporter & TV Camera	Penetrant, Ultrasonic Magnetic Particle, Visual	Temperature	Leak Detection
Reactor Vessel	o	X		X	
Guard Vessel	i	X			
Closure Head	o		X	X	
Closure Head	i			X	
Vessel Nozzles[b]	i, o	X		X	
Between Reactor Vessel and Guard Vessel					X
Head Access Area					X

[a]i - inside, o - outside

[b]nozzle to vessel and nozzle to extension joint; reactor vessel-outside, guard vessel nozzle to vessel-inside

TABLE 5.2-3

MATERIALS FROM WHICH THE REACTOR VESSEL, CLOSURE HEAD AND GUARD VESSEL ARE FABRICATED

	Reactor Vessel	Product Form	Material	Comment
	Support Ring	Ring Forging	SA 508 Class 2	
	Vessel Flange	Ring Forging	SA 508 Class 2	
1	Transition Shell	Plate	SB 168	Inconel 600
17	Shell Cources	Plate	SA 240, Type 304	Austenitic stainless steel
	Core Support Ring	Forging	SA 182, Type F304	Austenitic stainless steel formed into arcs and welded
	Core Support Cone	Plate	SA 240, Type 304	
	Inlet Plenum	Plate	SA 240, Type 304	
	Thermal Liner	Plate	SA 240, Type 316	Austenitic stainless steel
1	Thermal Liner Support Ring	Forging	SA 182, Type F304	Austenitic stainless, formed into segments and welded
	Nozzles	Forging	SA 182, Type F304	
	Closure Head			
	Rotating Plugs	Forging	SA-508, Class 2	
	Penetration Nozzles	Forging	SA-182, Type F304	Austenitic stainless steel
	Shield Plates	Plate	SA-516, Grade 60	
	Shielding Support Skirts	Plate	SA-516, Grade 60	Plate formed into arcs and welded
	Reflector Plates	Plate	SA-240, Type 304	Austenitic stainless steel
	Reflector Plate Supports	Forging or Pipe	SA-182, Type F304 or SA-312, Type 304	Austenitic stainless steel
	Suppressor Plates	Plate	SA-240, Type 316H	Austenitic stainless steel
	Suppressor Plate Support Column	Forging	SA-182, Type F316H and SA-336, Gr F22	Austenitic stainless steel Lower portion welded to 2¼ C. - 1 Mo. upper portion
	Spacer Bars	Bar	SA-387, Class 2, Grade 22	
	Margin Ring	Bar	SA-540, Class 1, Grade B-24	
	Margin Ring Keeper	Bar	SA-533, Class 1	
57	Suppressor Plate Column Caps	Plate or Forging	SA-387, Grade 22, Class 2 or SA-336, Grade F22	

TABLE 5.2-3 (Cont'd.)

	Guard Vessel	Product Form	Material	Comment
1				
	Vessel top flange	Bar, Plate, Forging	SA 479, SA 240, SA 182, Type 304	
	Vessel	Plate	SA 240 Type 304	
	Vessel to support skirt ring	Bar, Forging	SA 479, SA 182 Type 304	
	Support Skirt	Plate	SA 240 Type 304	
	Support Flange	Plate	SA 240 Type 304	
	Nozzles	Plate, Forging	SA 240, SA 182 Type 304	
	Guard Pipe Flanges	Bar, Plate	SA 479, SA 240 Type 304	
	Guard Pipe	Welded Pipe, Plate	SA 409, SA 240 Type 304	
17	Guard Pipe Elbows	Welded Fitting, Plate	SA 403, SA 240 Type WP 304	
	Cleanout Nozzle	Forging	SA 182 Type F 304	
	Cleanout Nozzle Cap	Forging, Plate	SA 182, SA 240 Type F 304	
41				

56 TABLE 5.2-4 HAS BEEN DELETED

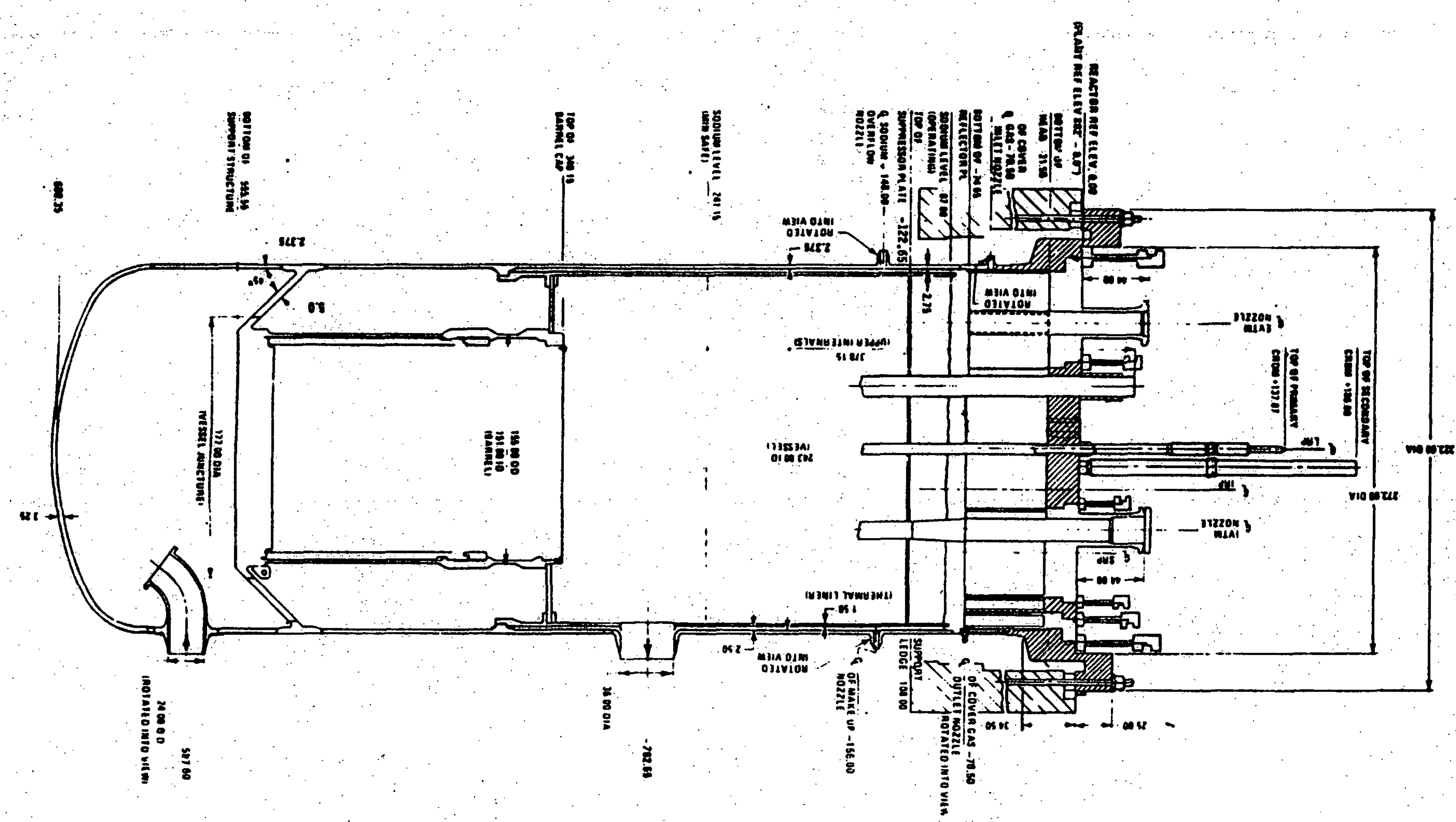


FIGURE 5.2-1 EXPANDED SCHEMATIC OF REACTOR VESSEL AND CLOSURE HEAD

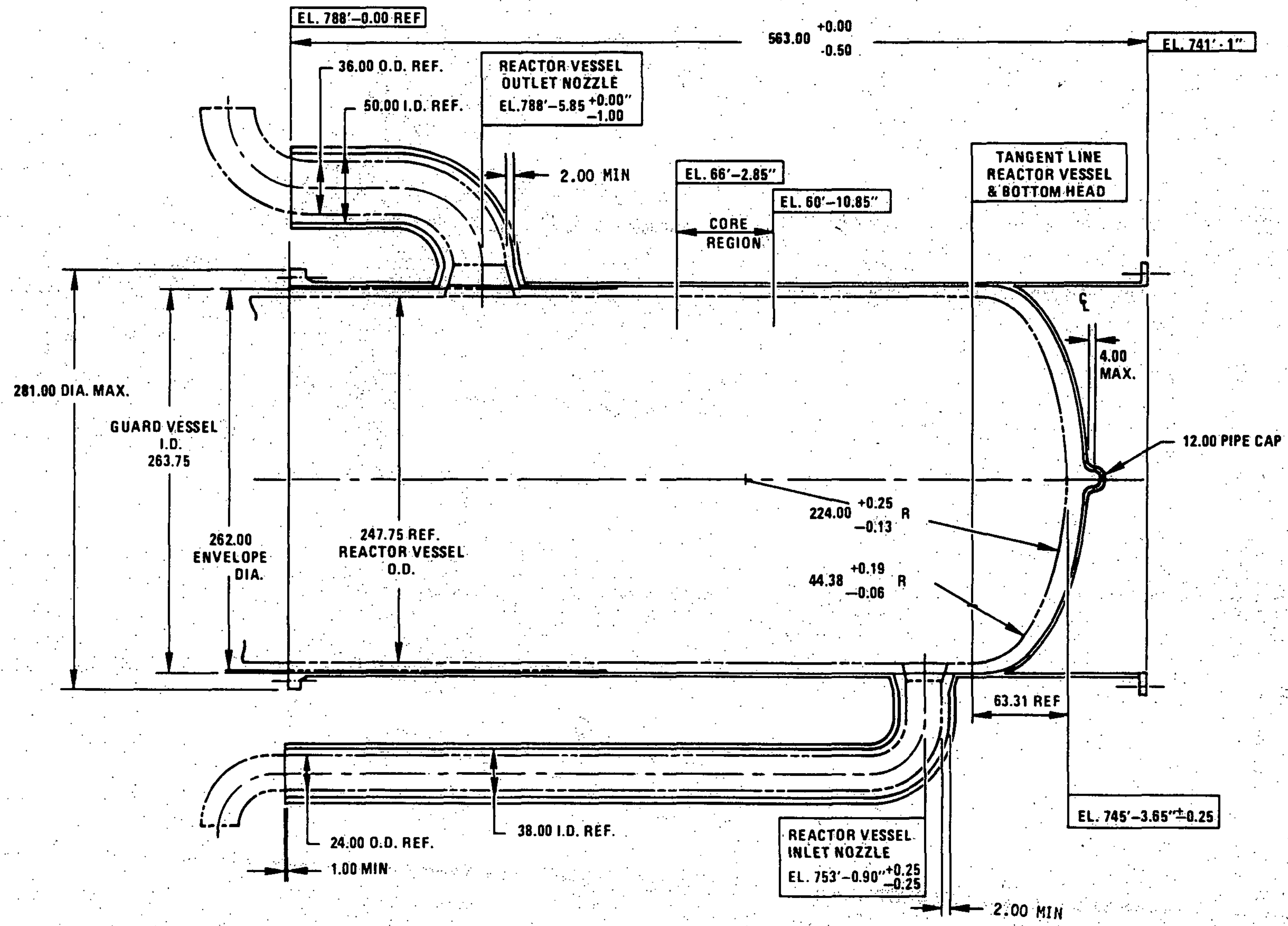


Figure 5.2-1A Reactor Vessel Guard Vessel

0285-1

Amend. 46
August 1978

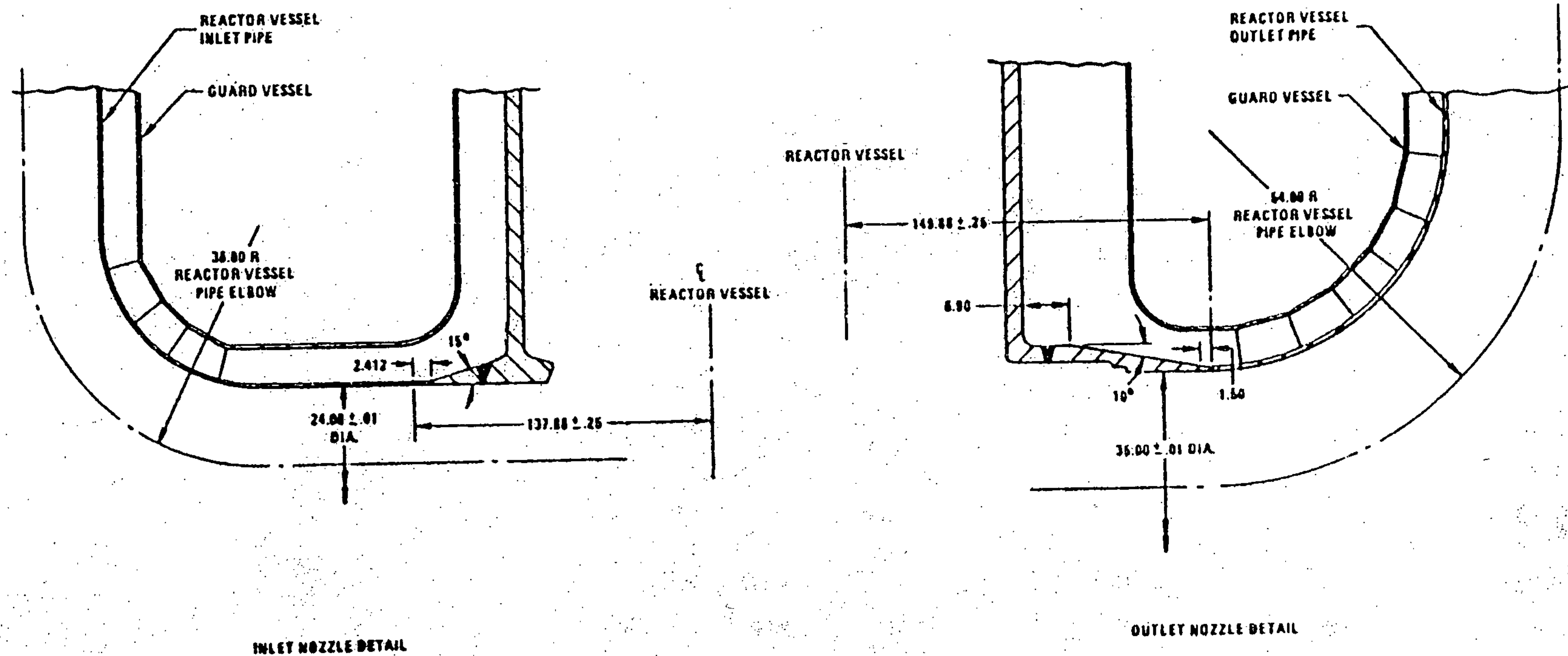


FIGURE 5.2-1B REACTOR VESSEL INLET AND OUTLET NOZZLE AND PIPE DETAIL

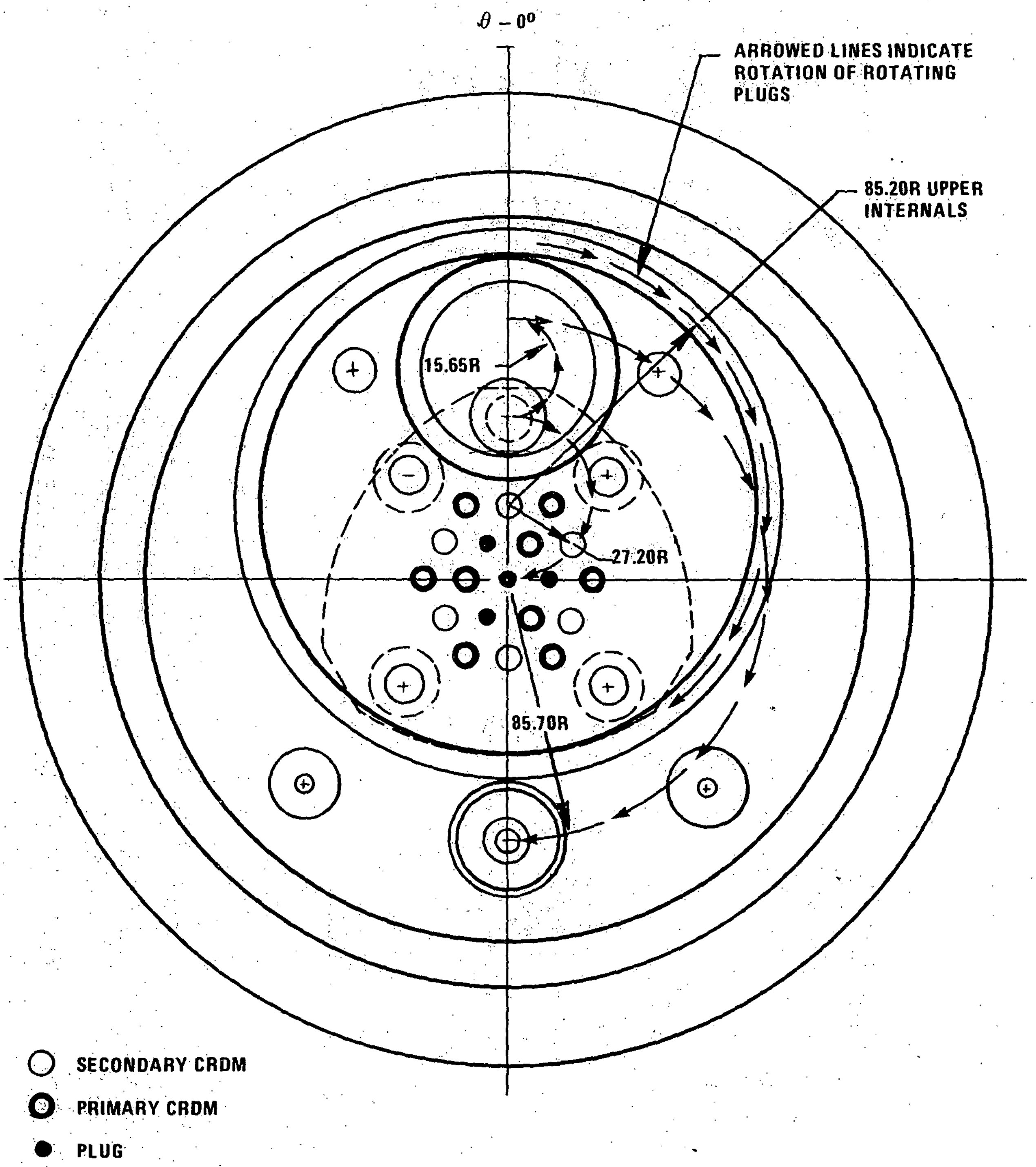
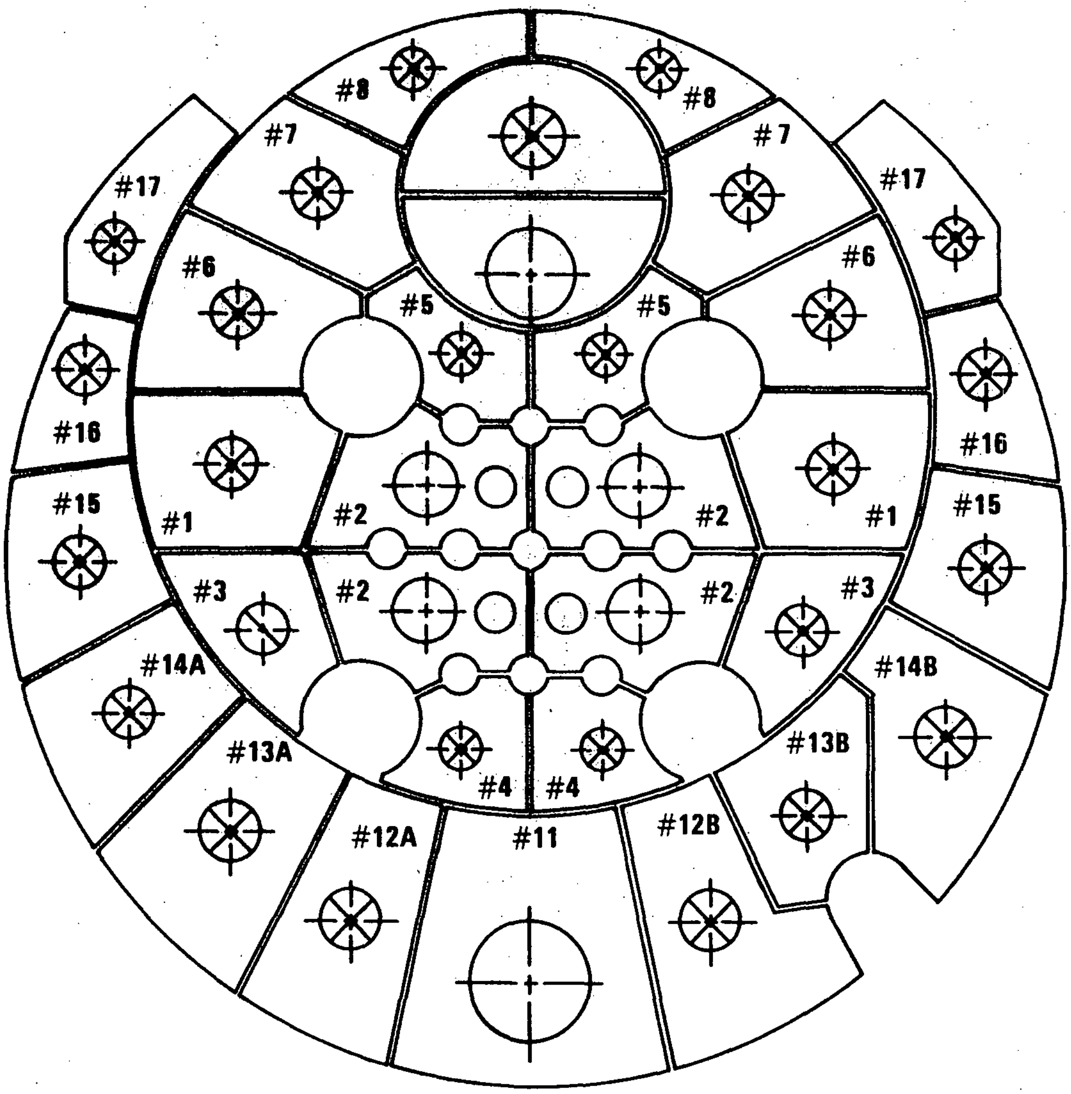


Figure 5.2-2. Closure Head Top Schematic View

1614-5

Amend. 51
Sept. 1979



KEY:

SUPPORT COLUMN ONLY – PENETRATING EQUIPMENT PASSES THROUGH

SUPPORT COLUMN WITH CAP

HOLE FOR PENETRATION EQUIPMENT

Figure 5.2-3 Suppressor Plate Plan View

Amend. 57
Nov. 1980

3975-1

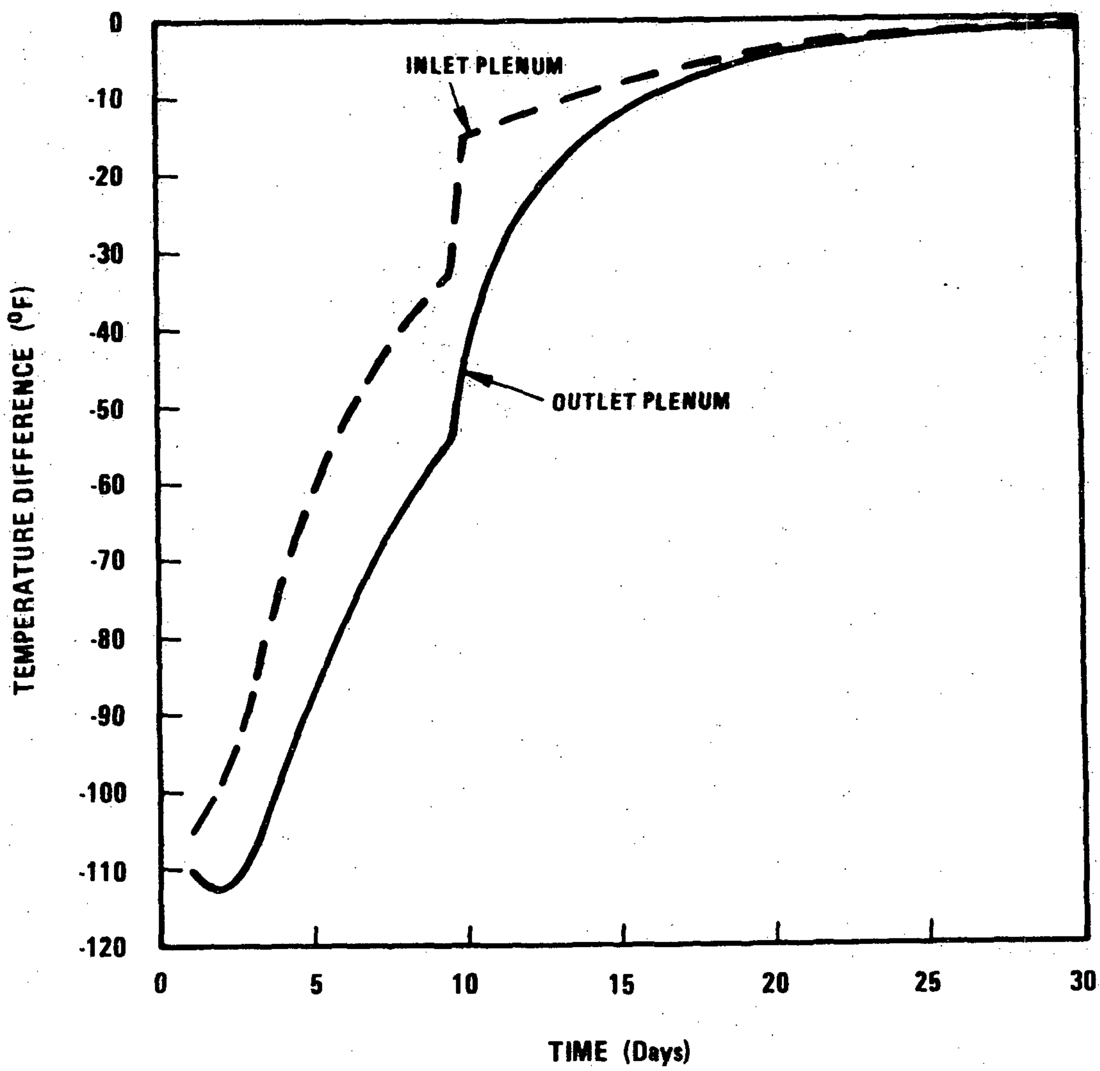


Figure 5.2-4 Temperature Difference Between The Reactor Vessel And Guard Vessel (Trv-Tgv) During Preheat

7683-104

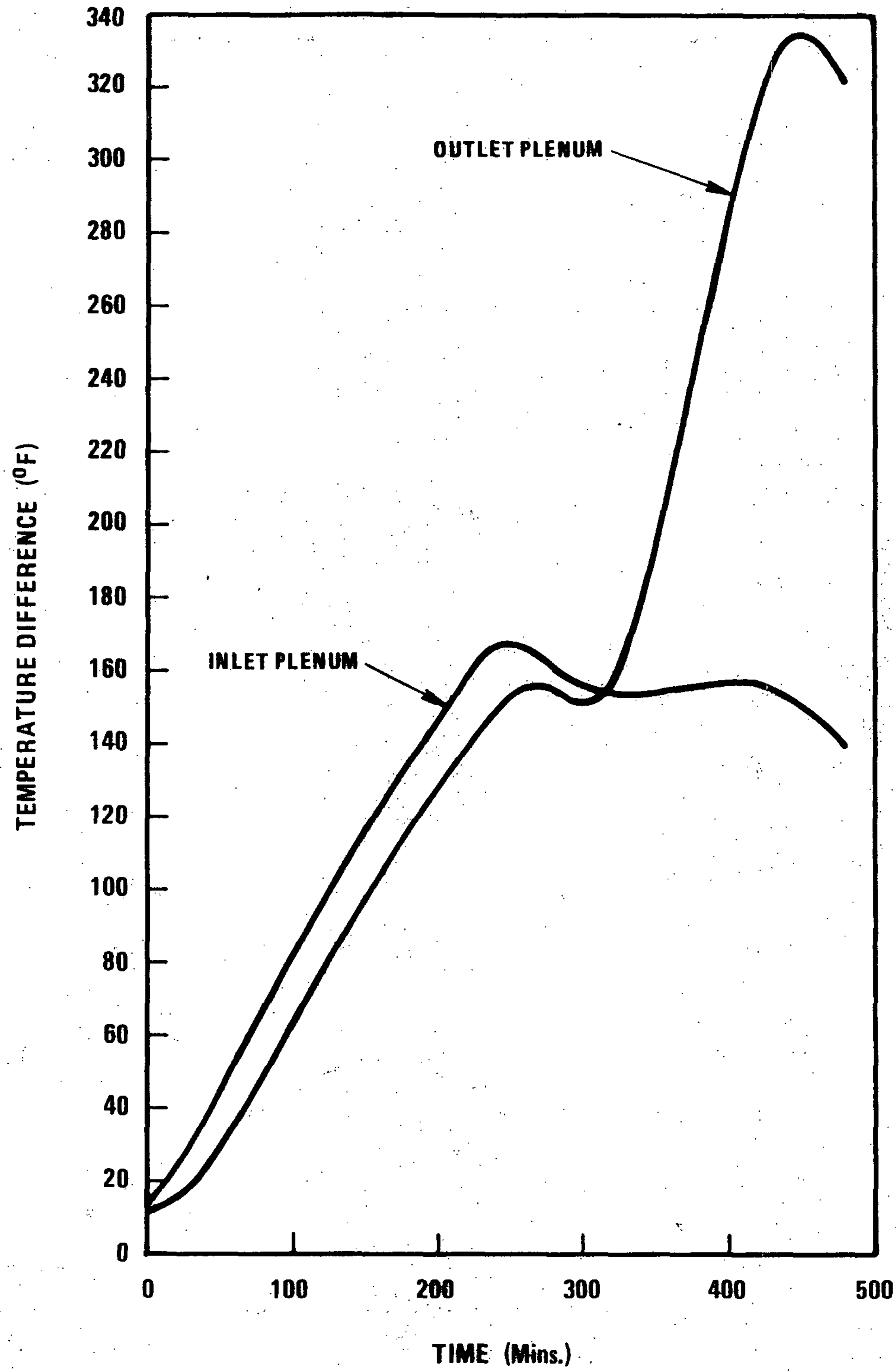


Figure 5.2-5 Temperature Difference Between The Reactor Vessel And Guard Vessel (Trv-Tgv) During Startup

7683-105

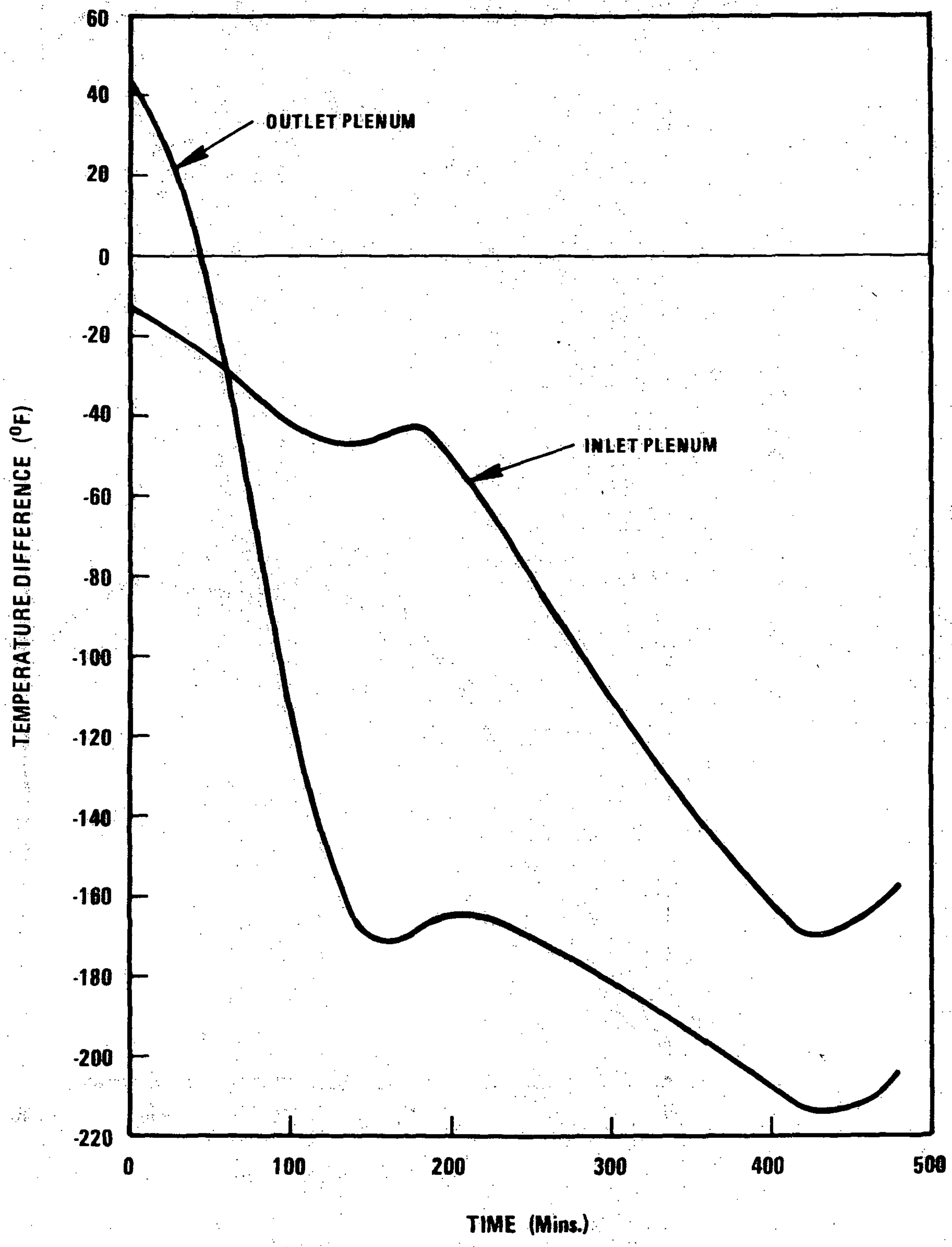


Figure 5.2-6 Temperature Difference Between The Reactor Vessel And Guard Vessel (Trv-Tgv) During Shutdown

7683-106

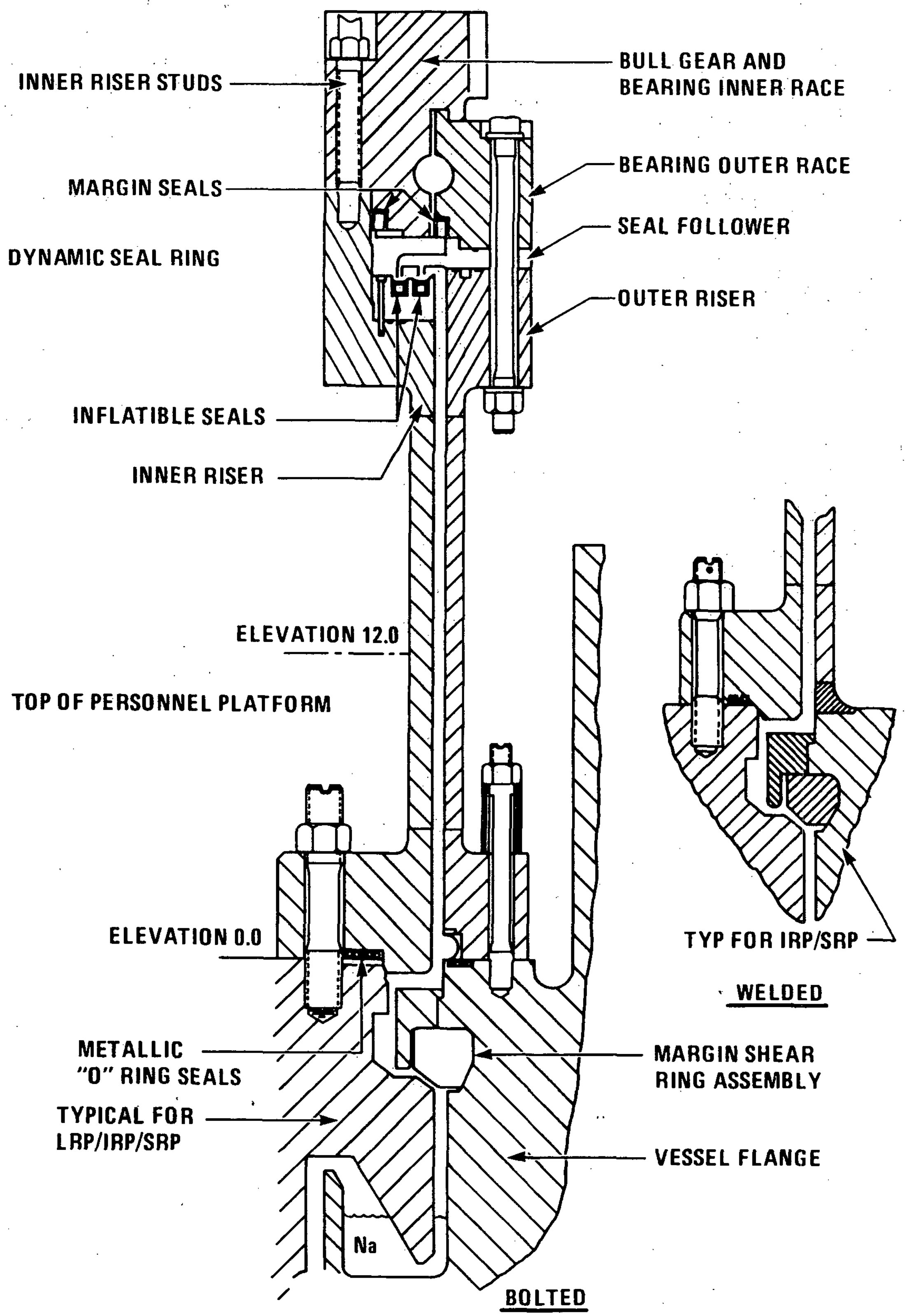


Figure 5.2-7. Typical Riser Assembly

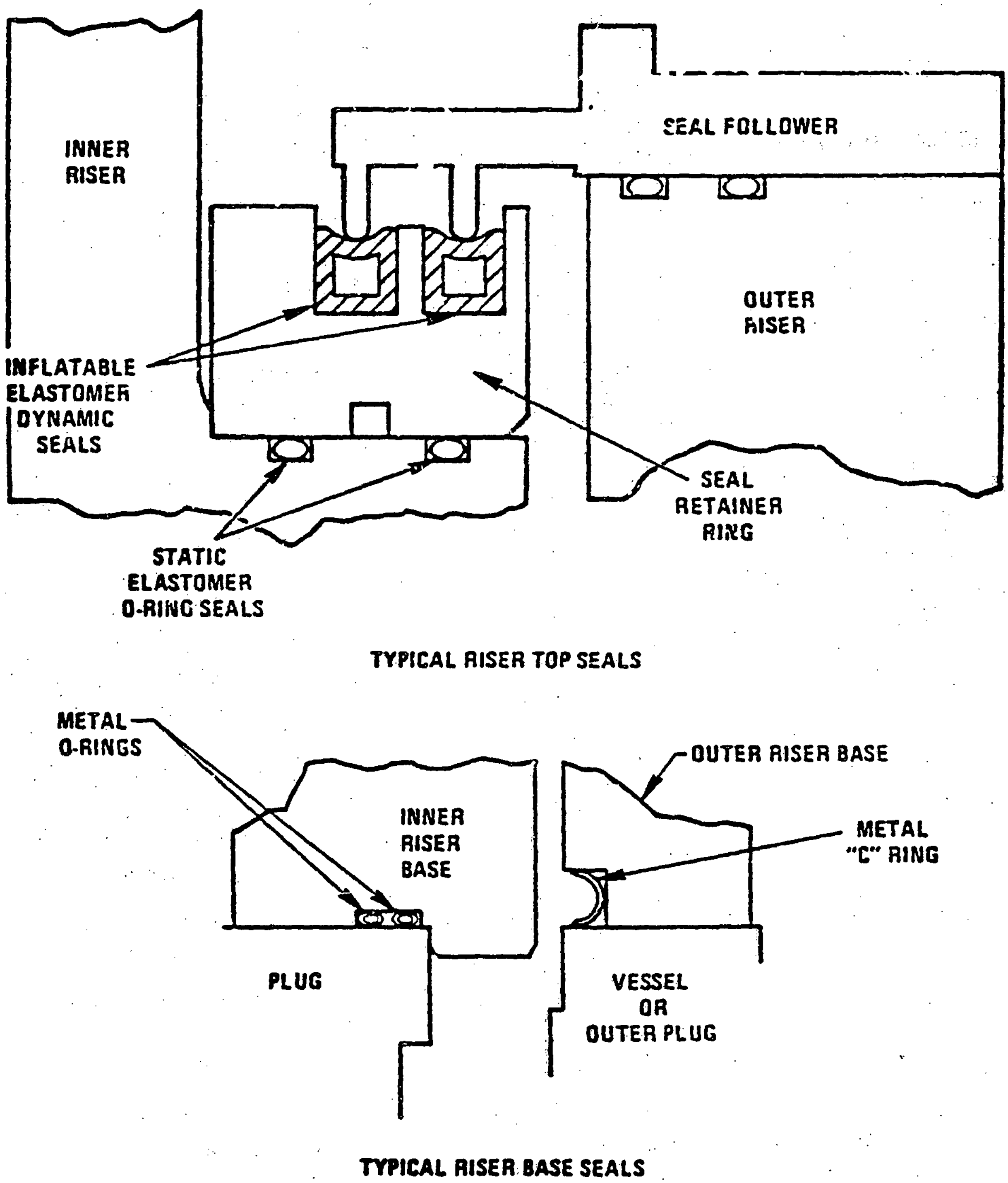


Figure 5.2-8 Riser Sealing Details

5.3 PRIMARY HEAT TRANSPORT SYSTEM (PHTS)

5.3.1 Design Bases

5.3.1.1 Performance Requirements 17

The Primary Heat Transport System extracts reactor generated heat and delivers this heat to the intermediate coolant system under all normal and off-normal operating conditions. Specific performance requirements include: 17

Heat Transport and Flow Performance

a. Transport of reactor generated heat (975 MWt) through the primary to intermediate coolant system while maintaining an adequate flow rate for controlling reactor temperature conditions within limits which preclude damage to the reactor vessel, fuel and reactor internals.

b. Regulation of heat transport system flow in response to plant process control over the full operating power range of 40 to 100 percent reactor thermal power.

c. Transfer of decay heat to the intermediate coolant system by pony motor operation or natural circulation under all normal and off-normal conditions including failure of a heat transport system component or loop. Specifically, there will be capability to remove decay heat by natural circulation with two or three loops following operation on two loops at the appropriate power. 17

d. Containment of primary sodium coolant and radioactive fission products within the primary coolant system by providing a boundary for primary coolant confinement and a separation of the primary and intermediate coolant systems all within the confines of the containment building.

e. Transport of reactor generated heat to the intermediate coolant system with two-loop operation at the appropriate power input. 17

f. Provide a sodium coolant system which can be easily filled, vented and drained.

g. Support of operation in a hot stand-by condition - nominally 7 1/2 to 10% of full flow at a normal temperature of 600°F.

Structural Performance

a. Design, fabrication, erection and testing of the PHTS components which comprise the sodium boundary shall be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1,
57 Nuclear Power Plant Components, 1974 Edition. The addenda to

Section III that are applicable for the various components are given in Section 5.3.1.2. The applicable ASME Code Cases (1592, 1593, 1594, 1595, 1596) for elevated temperature components shall also be employed. In addition, the RDT Standards E15-2NB-T and F9-4T with revisions as given in Section 5.3.1.2 shall apply, along with the RDT Standards F2-2, F3-6T and F6-5T of appropriate revisions coincident with the component contract
7 | 57 date of concern.

b. The natural frequencies of all components will, where possible, avoid resonance with all expected pump driving frequencies. Where not possible, the component design shall insure that structural damage will not occur as a result of resonance.

c. Structural design shall provide for dry HTS piping and component heat up at a rate of 3°F/hr.

d. Structural design shall provide for a system fill under conditions of full vacuum with system components at an average temperature of 400°F and hot spot temperatures of 600°F.

Transients

a. The primary heat transport system shall be designed to accommodate the thermal transients resulting from the normal, upset, emergency and faulted conditions described in Appendix B of this document.

b. The system shall be designed such that a normal or upset event does not adversely affect the useful life of any HTS component.

c. Following an emergency condition, resumption of operation must be possible following repair and re-inspection of the components, except that the primary coolant pumps (damaged or undamaged) must maintain capability to provide pony motor flow following all emergency conditions except in the affected loop for pump mechanical failure.

d. During and following a faulted condition, the heat transport system 33
must remain sufficiently intact to be capable of performing its decay heat removal function, including maintenance of primary coolant pump pony motor flow.

e. The primary heat transport system will accommodate, without loss of decay heat removal capability, the pressures imposed on the intermediate system, by a major sodium water reaction and the sodium hammer resulting from a primary loop check valve closure caused by the most severe flow degradation, such as a primary pump seizure.

60 33

<u>Seismic Loads</u>

The PHTS components and supports (ASME Class 1) under the jurisdiction of the ASME Code, Section III, shall be designed to accommodate the load combinations (including seismic loads) prescribed herein without producing total combined stresses, total strains (for all loading cycles) and cumulative creep/fatigue damage (for all loading cycles) in excess of those allowed by the Code. No component of stress or strain of an individual loading condition shall be included which would render the combination non-conservative. Transient
27 loadings shall be included as required by the Code. For elevated
57 temperatures, Code Case 1592 supplemented by RDT Standard F9-4T will apply.

The ASME Code Class 1 PHTS seismic Category I components shall be designed to withstand the effects of the SSE and of the OBE and remain functional. The 1
OBE will be considered as an upset condition and the SSE as a faulted
5 condition. For the low-temperature portion of the design analysis, the rules in the ASME-III code proper will be followed. Accordingly, the OBE shall be included in the analysis to the Code "Design Condition" requirements and limits. For the elevated-temperature portion of the design analysis, the requirements set forth in Code Case 1592 will be followed. Therefore, the OBE shall be included in the analysis to the Code "Upset Condition" requirements and limits. The reason for this difference is that the rules of the Code proper do not require an assessment of Primary Membrane stress plus Bending except in the "Design Condition" case.

Complete details for loading combinations of normal and transient condition loadings with seismic loads are provided in Section 3.9.1.5.

Thermal and Hydraulic design basis parameters are given in Table 5.3-1. Component structural design pressures and temperatures are given in Table 5.3-2. Pressures and temperatures used for structural evaluation at steady state operating conditions are given in Table 5.3-3. 35

5.3.1.2 Applicable Code Criteria and Cases

The PHTS pressure containing components shall be designed, fabricated, erected, constructed, tested and inspected (in compliance with 10CFR50, Section 50.55a) to the standards listed below:

COMPONENT	APPLICABLE STANDARD AND CLASS
PHTS Pump	ASME III, Class 1
PHTS IHX	ASME III, Class 1
PHTS Check Valve	ASME III, Class 1
PHTS Piping (including flowmeter, thermowells and pressure tap penetrations, etc.)	ASME III, Class 1

All primary heat transport system components shall be analyzed as Class 1 nuclear components in accordance with the rules indicated by the documents in the following:

COMPONENT	ASME Code[1] EDITION & ADDENDA	RDT-E15-2NB-T[2] EDITION & AMENDMENTS	RDT F9-4T[3] EDITION	CODE CASE 1592[4] REVISION
PHTS IHX	1974 plus Summer 1974 Addenda	November 1974 Amendment 1	September 1974	1592-1
Primary Pump Guard Vessel	1974 plus Summer 1974 Addenda	November 1974 Amendments 1, 2, 3	September 1974	1592-4
IHX Guard Vessel	1974 plus Summer 1974 Addenda	November 1974 Amendments 1, 2, 3	September 1974	1592-4
PHTS Piping	1974 plus Addenda to Summer 1975	November 1974 Amendments 1, 2, 3	January 1976	1592-7
PHTS Check Valve	1974 plus Addenda to Summer 1975	November 1974 Amendments 1, 2	September 1974	1592-2

NOTES: 1) ASME Boiler and Pressure Vessel Code, Section III
2) RDT E15-2NB-T (Supplement to Section III)
3) RDT F9-4T (Supplement to Code Case 1592)
4) ASME Code Case 1592, "Class 1 Nuclear Components in Elevated Temperature Service"

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57 The "Nuclear Systems Materials Handbook", (Ref. 1) shall be used to obtain material properties data not available from the above sources. As required in RDT F9-4T, the use of additional or alternative material properties shall require the approval of the purchaser. Code Case 1521, "Use of H-Grades of SA-240, SA-479, SA-336, and SA-358, Section III," may be used for H-Grades of Type 304 and 316 austenitic stainless steels. RDT F9-5T, Sept. 1974 (Section 6) provides alternative procedures for satisfying the strain limits of Appendix T of Code Case 1592 which are acceptable
27 to the purchaser. Section 6 of RDT F9-5T, Sept. 1974 also provides time/temper-
36 atures below which the primary plus secondary and peak stress limits of Section III may be used in place of the limits of Appendix T of Code Case 1592. The scope of the analysis of Code Case 1592 shall be used even if the limits from Section III are used. For example, the primary plus secondary stress intensity range due to emergency as well as normal plus upset conditions is limited.

In addition, Code Cases 1593, for fabrication and installation of elevated temperature components, 1594 for their examination, 1595 for their testing, and 1596 for their overpressure protection shall apply for the primary heat transport system components.

5.3.1.3 Surveillance Requirements

Changes in fracture toughness of the PHTS piping and components may be caused by carburization, plastic creep straining and the thermal environment though not radiation effects.

The need for surveillance of stainless steel piping and components for changes in fracture toughness will be determined by ongoing programs.

If a requirement is identified by ongoing programs, a surveillance program will be designed in accordance with the philosophy of 10CFR50, Appendix H.

5.3.1.4 Materials Considerations

5.3.1.4.1 Basis for High Temperature Design/Analysis

Rules governing the construction of Class 1 components which are to experience temperatures above those now provided in Section III shall be constructed in accordance with the following considerations:

a. The rules for materials in NB-2000 shall apply except as modified by Code Case 1592, and

b. The rules for design in Code Case 1592 shall replace rules for NB-3000.

27 c. RDT Standards F9-4T, Jan. 1976 and F9-5T, Sept. 1974 shall be used to supplement the rules of ASME Code Case 1592 for the construction of Class 1 nuclear components.

Code Case 1592 contains rules for materials and design of Class 1 components, parts, and appurtenances which are expected to function with metal temperatures which sometimes exceed those covered by the rules and stress limits of subsection NB of Section III, ASME B&PV Code. In extending the Class 1 component rules to higher temperatures, Code Case 1592 retains part and scope-of-coverage descriptions identical to those in subsection NB; that is, the rules of this case apply to structures which, upon failure, can lead to a gross violation of the pressure boundary of a Class 1 component.

The rules of the Code Case are applicable to Class 1 nuclear components independent of the type of contained fluid.

The stress limits and design rules of subsection NB are applicable only to service conditions where creep and relaxation effects are negligible. Consequently, the rules of subsection NB guard only against the time-independent failure modes--ductile rupture, gross distortion (buckling and incremental collapse) and fatigue. The material behavior models for a subsection NB analysis are restricted to those exhibiting only elastic or elastic-plastic behavior. However, the rules in the Code Case allow similar material behavior models to be used at elevated temperatures provided times are short, stress levels are low, and temperatures are not too high.

At temperature and loading conditions where creep effects are significant, the design analysis must also guard against time-dependent failure modes. The rules and allowable limits of the Code Case reflect both time-independent and time-dependent material properties and structural behavior by considering the following modes of failure:

- ductile rupture from short-term loadings
- creep rupture from long-term loadings
- creep fatigue failure
- gross distortion due to incremental collapse and ratcheting

Brief guidelines are also provided in the Code Case for the following modes of failure:

- loss of function due to excessive deformation
- buckling due to short-term loadings
- creep buckling due to long-term loadings

Design procedures and materials data not contained in the Code Case may be required to insure the integrity or the continued functioning of the structural part during the specified service life. For example, the rules do not provide methods to evaluate deterioration which may occur in service as a result of corrosion, mass transfer phenomena, radiation effects, or other materials instabilities. Nor do the rules ensure continued functional performance of deformation-sensitive structures such as valves and pumps. The heat transport system component design shall account for the degradation in material properties for the effects of the sodium environment as discussed in Section 5.3.3.10.

The requirements of the Code Case do not apply to items not associated with the pressure retaining function of a component. Items normally not covered by these rules include solid shafts, stems, trim, bearings, bushings, wear-plates, seals, packing gaskets, and valve seats.

5.3.1.4.2 <u>Materials of Construction</u>

All materials for the primary system heat transport system piping, pumps, valves, heat exchangers and guard vessels are specified to minimize corrosion and erosion and ensure compatibility with the environment.

In order to avoid the possibility of accelerated erosion, the internal coolant velocity is limited to less than 30 feet per second.

The materials used for the primary heat transport system components shall conform to the ASME Code and the supplemental requirements of RDT E15-2NB-T, Oct. 1975. All parts of the primary heat transport system that form part of the reactor coolant pressure-retaining boundaries shall be fabricated from either Type 304 or Type 316 austenitic stainless steel. Material for
42 41 service above 800°F will contain 0.055 percent carbon minimum.

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. All smaller piping which comprise part of the primary heat transport system such as the loop drains and connecting lines to other systems are also austenitic stainless steel.

The check valve in the primary heat transport system which is in contact with the sodium coolant is constructed primarily of stainless steel. Other materials in contact with the coolant are special materials, such as those for hard surfacing.

All parts of the primary heat transport coolant pump in contact with the reactor sodium coolant are austenitic stainless steel except for bearings and special parts.

All parts of the IHX that form part of the pressure retaining boundaries or that are exposed to liquid sodium containing environments shall be fabricated from either Type 304 or Type 316 austenitic stainless steel.

The guard vessels for the IHX and primary pump components shall
44 be fabricated from Type 304 Austenitic Stainless Steel.

The material specifications for the construction of the primary heat transport system components are given in Tables 5.3-4 thru 5.3-8.

The supplier may substitute other ASME-approved material than that specified in Table 5.3-4 thru 8 after review and approval by purchaser prior to use or procurement. If the supplier elects to use other ASME-approved materials, the supplier's request for approval shall contain topical reports demonstrating the adequacy of the selected alternate material.

Selection of alternate materials shall be based on the mechanical properties, metallurgical stability, sodium compatibility, and response to radiation under the applicable design and environmental conditions. When recommending the use of alternate materials, the supplier shall document the justification which shall include, as a minimum, a summary of available test or experience data and a discussion of the adequacy of the recommended materials relative to 304 or 316 stainless steel or other purchaser-specified alternate.

5.3.1.4.3 Additional Requirements

The following requirements are modifications or additions to the requirements of the materials specifications identified in Tables 5.3-4 thru 5.3-8.

Hydrostatic Test of PHTS Piping

The PHTS piping will be hydrostatically tested in accordance with the ASME Code, Section III, Article NB6000 and Code Case 1595 (supplemented by RDT E15-2NBT, October 1975), as described below. The lengths of straight pipe will be tested by the manufacturer at his facility to satisfy the Code material specifications. The thermowell body sub-assemblies will be tested under external pressure, by the manufacturer. The IHX Vent-line flow restrictor will be tested by its manufacturer. All other items are considered by the Code to be materials and will not be tested separately. Rather, each completed PHTS piping sub-assembly (spool) will be hydrostatically tested by the spool fabricator at his facility prior to their being shipped to the plant site. Thus, all piping items such as fittings and branch connections will be tested, as well as the spool welds, prior to installation. Specific procedures have not yet been written, but they will be in compliance with the Code requirements noted above as they
57 apply to the design conditions at each location in the system.

Strength Tests of HTS Components

Hydrostatic or pneumatic strength tests shall be conducted in
accordance with the ASME Code Section III Code Case 1595-1 and implement 16
27 any requirements of RDT Std E15-2NB-T, Oct. 1975.

Water shall be used as the test medium for hydrostatic tests unless it cannot be shown that residual water can be completely removed. If complete water removal cannot be accomplished, a substitute liquid may be used. For example, the cold leg check valve hydrostatic test will be performed using liquid Freon as the test medium. In any case where a suitable liquid cannot be used, a pneumatic strength test will be employed in accordance with the Code. Guard vessels are not pressure tested.

In addition to hydrostatic and/or pneumatic strength tests, helium leak tests of certain liquid metal containing parts or assemblies will be required. Particularly, tube to tube sheet welds, tube bundles, single pass welds and thin sections (e.g., bellows, rupture discs and seal welds) will be helium leak tested in accordance with the ASME Code, Section V, Article 10. Helium leak tests will be performed after a pneumatic test so that any porosity or minute defect will be exaggerated thereby increasing the sensitivity of the helium leak test. If a component to be hydrostatically tested will also be helium leak tested, the helium leak test will be performed first to preclude water molecules from plugging minute leak paths which the helium leak test otherwise would detect.

Gas Pressure Test - Leak Test

Prior to sodium fill of the reactor and the PHTS, a system pneumatic test will be performed in accordance with 6320 of ASME Code Case 1595-1. This Code Case requires that the test pressure shall not exceed the lowest of the maximum test pressures allowed for any of the components in the system which in the case of the CRBRP would be 18 psig (1.2 times the 15 psig design pressure of the reactor vessel upper plenum). The principal purpose of the test is to leak check field welds. Individual components and piping spool pieces will have been pressure tested prior to installation in accordance with individual specification requirements as noted above. This test will be conducted to supplement visual penetrant and radiographic examination of field welds by checking for gas leaks.

A trace gas (helium) may be added to the test medium or a bubble test may be performed. These tests may be supplemented by a gross leak rate test. The various options that may be used in preparing a specific test procedure have not been evaluated, but will be included in the FSAR. 16

Chemical Analysis

The materials specified in Tables 5.3-4 thru 5.3-8 for the primary heat transport system components shall conform to the chemical compositions specified by the applicable RDT Standard and the additional requirements noted.

The check analysis shall be made on samples taken from broken tension test specimens or material samples taken from tension test specimen location. The analysis results shall meet the chemical requirements of the material specification. Failure to meet the chemical composition requirements shall be cause for rejection of the entire lot.

5.3.1.4.4 Welding

Welding Filler Materials

The weld filler materials to be used for the Primary Heat Transport System components and primary piping fabrication shall conform to the requirements of the RDT Standards for weld filler materials identified in Table 5.3-9.

The supplier may substitute other RDT Standard filler material types for those specified in Table 5.3-9 after review and approval by the purchaser prior to procurement. If the supplier elects to use other approved filler materials, the supplier's request for approval shall provide technical justification for the proposed substitution. Alternate materials shall conform to applicable RDT Standard requirements.

Delta Ferrite Control

The delta ferrite requirements and determination for filler classifications E308L, E308, ER308L, and ER308 shall conform to the requirements of Section III of the 1974 ASME Boiler and Pressure Vessel Code. The delta ferrite range of 5 to 9 percent shall be used since these limits are consistent with the Code and RDT Standards.

Welds on Associated Equipment

Weld filler materials for associated equipment welds shall conform with ASME Code, Section II, Part C, Welding Rods, Electrodes and Filler Materials.

Welding Qualifications

Welding qualifications shall be in accordance with the Code as supplemented by RDT F6-5, Nov. 1975 and RDT E15-2NB-T, Oct. 1975. RDT E15-2NB-T, paragraph ANB-4331, specifies provisions under which the supplier's previously qualified procedures, welder, welding operators, and welding machines need not be requalified in accordance with RDT F6-5, Nov. 1975. RDT E15-2NB-T, paragraph NNB-4325 fufills the requirements of Regulatory Guide 1.71 by requiring that preparation of weld qualification mockups must include simulation of any environmental or accessibility restrictions that might be encountered in making the actual weld.

Production Welding

Production welding shall be in accordance with the Code as supplemented by RDT E15-2NB-T, Oct. 1975, with modifications set forth in the equipment specifications for each major heat transport system component.

Weld Repair

Purchaser approval is required to make weld repairs to correct defects that recur:

a. After the first cycle of weld repair of defects in the fusion zone and adjacent base metal where welds are not heat treated following repair.

b. After the second cycle or repair in welds that are heat treated after each repair cycle.

Purchaser approval is required for repair of crater cracks restricted to the crater of any weld pass if the third repair cycle results are not acceptable.

5.3.1.5 Leak Detection Requirement

The PHTS Leak Detection Subsystem, part of the Sodium/Gas Leak Detection System described in Section 7.5.5.1, will provide indication and location information to the operator in the event of a sodium leak from the primary sodium coolant boundary, in a timely manner in order that action may be taken before a critical size crack in the primary boundary develops. (A critical size crack is a crack that would bulge open due to operating stresses. See Section 5.3.3.6.)

The detection system sensitivity requirements are discussed in Section 7.5.5.1.

5.3.1.6 Instrumentation Requirements

The primary system is provided with an instrumentation system which monitors the process variables within the PHTS and which provides signals for safety action and operational information. The measured variables and instrumentation provided are discussed in Section 7.5.2.

5.3.2 Design Description

5.3.2.1 Design Methods and Procedures

5.3.2.1.1 Identification of Active and Passive Components which Inhibit Leaks

In the primary heat transport system, the only active component which is considered a part of the PHTS is the primary pump (see Table 5.3-10 for a list of pumps and valves). In the event of pipe leaks, the primary pumps are automatically reduced to pony motor flow following manual reactor shutdown or
59 40 reactor scram due to low reactor vessel sodium level.

In the unlikely event of a primary pipe or component boundary failure, the PHTS has been designed to limit the loss of reactor coolant and assure that for any boundary failure, continued reactor cooling is provided. The PHTS design features which limit loss of coolant and assure reactor cooling are the
56 combined use of elevated piping, use of a guard vessel around major equipment and a five foot pony motor shutoff head. The PHTS guard vessels have been designed such that the tops of the guard vessels are at an elevation which is approximately 9 feet above the tops of the reactor vessel discharge nozzles.
56 This level is based on the combination of the pony motor shut-off head of 5 feet and the minimum safe reactor vessel level which is two feet above the top
40 of the reactor discharge nozzle, plus an additional two feet to accommodate sodium shrinkage and hydraulic uncertainties.

The volume of the guard vessel and the volume of sodium above the minimum safe
level (MSL) in the reactor vessel and pump tank have been sized to assure that
the guard vessel volume will be less than or equal to the volume loss from the
reactor and pump tanks for any leak condition assuming contraction and cooldown
after the leak assuming no reactor vessel makeup. At minimum operating sodium
level the volume of sodium above MSL in the reactor vessel and pump tanks is
4430 ft^3. The net volume of each PHTS guard vessel is 2700 ft^3. The net
59 volume of the reactor guard vessel is 3100 ft^3.

Continued reactor cooling is provided in the unlikely event of a pipe failure by the PHTS elevated piping arrangement. All PHTS piping is routed at an elevation above the tops of the PHTS guard vessels thereby limiting the loss of coolant in the unlikely event of a pipe failure.

The combination of guard vessel elevation, guard vessel volume, reactor vessel
59 and pump tank sodium inventory above the minimum safe level, pony motor
shutdown head and elevated piping assures a limited loss of reactor coolant and
continued reactor cooling capability.

Within the PHTS, there are two general types of failures of the pressure
59 containing boundary. They are (1) failures which occur within a guard vessel
and (2) failures in elevated piping outside of the PHTS guard vessels.

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Amend. 59
Dec. 1980

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59 Leak of PHTS Outside Guard Vessels

For a leak in the elevated pipework between the reactor vessel and the pump suction, leaks will stop when the level of sodium in the reactor vessel drops about one foot below the level of the leak, because the syphon effect is broken. At most, in postulated cases where the check valve fails to close, reactor sodium level will drop no further than the frictional head across the reactor with pumps operating at pony motor speed. It should be noted that independent of the check valve the level in the reactor vessel will not fall below the minimum safe level. The unaffected loops continue to operate because, with the low reactor outlet nozzles, the syphon effect in these loops is unimpaired. The pony motor is required only to overcome the frictional losses in the unaffected loops, so they continue to provide decay heat cooling.

For a leak in the elevated piping between pump discharge and reactor vessel inlet, sodium will flow out until the level of sodium in the reactor vessel falls to a level where the head developed by the pump at pony motor speed is insufficient to raise fluid to the height of the break. At this point, the
59 sodium level in the reactor will be approximately one foot above the minimum safe level. If the pony motor in a failed loop is stopped, the loss of sodium will be reduced to that generated by reverse flow from the active pumps. The rate of sodium loss from the defective loops will be restricted by check valve action. However, plant safety is independent of operation of the check valves. For small leaks, check valve action will produce little mitigation of the leak.

The potential for cell atmosphere ingress into the coolant system in the event of a leak exists at any point in the system that operates under a negative pressure relative to atmospheric. Therefore, the potential for cell atmosphere ingress into the coolant system depends on whether the loop is shut down, the pump is on pony motor operation, or the pump is on main motor operation. Each 25
of these three loop conditions produce a unique pressure profile around the loop, and therefore, must be addressed separately.

Loop Shutdown - Pumps Idle

When the loop is shut down, the potential for gas getting into the system exists in three different places: the elevated section of the 36" hot leg piping, the upper region of the IHX (above the upper tubesheets), and in the elevated section of the cold leg piping including the check valve. These three areas and the connecting vent lines are the only points in the system that are above the normal sodium level in the reactor. The high point vent lines
59 contain freeze seals backed up by locked closed valves in the Argon cover gas system.

Loop Operation at Pony Motor Flow

Operation at pony motor flow in the loop eliminates the negative pressure in the elevated section of the primary cold leg piping (including the check valve) and therefore, eliminates the potential for cell atmosphere ingress. Negative pressure does exist in the IHX and the elevated section of the 36" hot leg piping, and therefore, the potential for gas ingress does exist. The potential for cell atmosphere ingress into the primary system in the IHX exists at two separate places; the other annulus to which the IHX to pump tank vent line is attached, and at the top of the IHX near the bellows seal. Gas ingress near the bellows seal would be self limiting because any gas in-leakage in this area would have to travel the entire length of the tube bundle through the annulus between the intermediate (tube side) downcomer and the inner shroud of the tube bundle (See Figure 5.3-15) before it could enter the main sodium stream. Gas entering the main stream from this point would require a pressure of approximately 10 psi higher than the normal cell pressure. Therefore, the only part of the IHX that has a potential for gas inleakage is the outer annulus of
59 the IHX to which the IHX vent return line is attached. However, this region of the IHX is not in the main stream and in-leakage would be self limiting. Any gas accumulation would be forced to the gas space of the pump upon start-up of the pumps on main motors.

Loop Operation with Pumps Driven by Main Motors

Operation of the pumps on main motors eliminates negative pressure in all of the systems downstream of the pump and leaves only the elevated section of the 36" primary hot leg piping at a negative pressure. At full flow in the loop, the system pressure is negative from the second elbow from the reactor vessel to the second elbow before the pump (approximately 73 ft. in each loop). This section of piping has only one penetration (high point vent connection). It contains no thermowells, connections to pressure transducers or other penetrations to the coolant boundary. The high point vent contains a freeze seal backed up by redundant "locked closed" valves. 25

Discussions of protective system action and delay times can be found in Section 7.2.

5.3.2.1.2 Design of Active Pumps and Valves

The applicable ASME Boiler and Pressure Vessel Code, Addenda, Code Cases, and
59 RDT Standards were used as the design basis for the Primary

Heat Transport System (PHTS) sodium pumps. Detailed analyses are being made to
59 40 verify structural performance within the design basis limitations.

The primary pump normally operates in a temperature range between 400°F to 1015°F and a pressure range between 0 to 200 psi. Since the operating temperatures exceed 800°F (for austenitic steels), materials time-dependent behavior must be considered and the high temperature design criteria must be invoked for structural analysis. The analytical criteria to be employed are the Class 1 requirements of the 1974 Edition of the ASME Boiler and Pressure Vessel Code; Section III, Nuclear Power Plant Components with Addenda through winter of 1974, as supplemented by RDT Standard E15-2NB-T with Amendments and addenda through June, 1975. The elevated temperature Code Cases (i.e. 1592
59 37 through 1596) and RDT Standard F9-4T also apply.

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The pump stress analysis is being performed using verified and documented computer programs as well as standard hand calculations. Where possible, the design of the pumps is such as to keep the membrane stresses in the elastic range; where this is not possible, inelastic analysis techniques are being used. The pump unit and all its parts are being designed such that no damage or malfunction will result from internally or externally generated operational vibrations, including shaft rotating frequencies, impeller vane wake passing frequencies, flow induced pressure oscillations, or vibrations and shock loads anticipated during shipping and installation. Vibration of the pump components and the pump's response to seismic excitation are being determined by analysis. Amplitude and frequency limits imposed are being limited to accumulated fatigue damage and consider proper function of the pump parts. RDT Pump Standards require that the first rotor bending natural frequency must be at least 25 percent higher than the maximum pump shaft speed. This requirement is being augmented with amplitude restrictions; the pump generated vibrations measured at the discharge and suction nozzles shall not exceed .010 in. peak amplitude within the continuous operating ranges of the pump.

Pump Operability

The PHTS pump manufacturer is required to assure operability under accident conditions and during seismic events in accordance with Reference 12, PSAR Section 1.6.

Prototype pump testing includes acceptance testing in water at the pump suppliers' facility and performance testing in sodium at the Sodium Pump Test Facility. The prototype pump was tested in water at 130% of full flow design conditions to verify hydraulic and mechanical performance. Sodium performance testing is being done at full rated flow, at expected operating head and temperature. Testing in sodium includes mapping of head and flow or both maximum and minimum plant loop impedences. Testing of the pump's performance when subjected to fluid borne temperature transients includes the plant predicted upset and emergency transients up to capability of the facility. Operability of the pump during and after the emergency and faulted plant conditions is being verified by analysis, since comprehensive accident and seismic qualification testing is not possible due to test facility limitations.

A dynamic analytical model which includes foundation mass and stiffness, piping mass and stiffness, drive motors, pump tank and all internal pump parts including sodium masses has been constructed for these analyses.

This model is used to calculate displacements and loads during normal operation and during the specified seismic events. For the prototype pump the model was modified to change the foundation from that the CRBRP to the water test pump mounting stand. Using the modified model predictions of pump dynamic performance were made and correlated with measurements taken during water test thereby verifying the adequacy of the model.

Each plant pump will be assembled and water tested by the pump supplier before final cleaning and shipment to the site. This test will confirm that each pump assembly is properly balanced and that it will operate within acceptable vibration limits. Similarly each shaft seal cartridge assembly is operated prior to shipment to insure its proper operation.

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A listing of the analytical methods anticipated to be used in the evaluation of stresses is presented in Section 5.3.3.1.5.

5.3.2.1.3 Surveillance and Inservice Inspection

Introduction

The Inservice Inspection Program consists of three types of inspection which are listed in decreasing frequency occurrence.

Visual condition inspection to be conducted during periods of plant shutdown such as for refueling.

Weld inspections to be conducted during periods of prolonged plant shutdown such as for maintenance.

Metallurgical and component internals inspections to be conducted when components are removed for maintenance.

Inservice inspection will be planned and conducted according to the 6
59 appropriate requirements of Appendix G. 33

Visual Condition Inspections

Visual inspections will consist of remote visual viewing in the PHTS cells and pipeways (between the PHTS cell and reactor cavity), and visual examination of the intermediate (excontainment) system. Remote viewing capabilities will include the following:

> Ability to view the primary coolant boundary (in-so-far as practical) in the PHTS cell and pipeway including capability to conduct visual examinations of the annulus between the pump or IHX and the respective pump or IHX guard vessels.

Ability to examine pipe hangers for proper adjustment, in addition to examination of the pump, IHX, piping and guard vessel support structures for structural integrity.

Viewing within the PHTS cells, guard vessels and pipeways will not require deinerting the cell or pipeway. And, inspection capability shall not require decay of Na_{24} prior to inspection, however, inspections within the PHTS guard vessels can require cool down to 400°F prior to inspection.

Such inspections shall consist of visual examination of pipe insulation for evidence of sodium leakage, insulation integrity, set point positions of hangers (baseline records should be kept for indication of gross changes) in addition to structural examination for evidence of structural degradation (such as concrete foundation spalling, structural weld failures, structural buckling, etc.).

Weld Inspection

Weld inspections shall be conducted during periods of prolonged plant shutdown during which the PHTS is accessible for insulation removal. Those areas subject to inspection shall be established on a priority basis which will depend on ease of access, duration of plant outage and loop condition (drained or full of sodium, radiation levels, probability of failure and the number of inspection personnel available). Such inspections should include "non-destructive testing" (NDT) of any bi-metallic welds, component nozzle welds, highly stressed sections of piping, etc.

Metallurgical and Component Internals Inspection

Metallurgical and component internals inspection will be conducted only when components or sections of pipe are removed from the system for maintenance.

The details of the examination conducted will depend on what component is removed, however, the inspection will include examination of any surfaces subject to wear such as the pump bearings, examination of the effect of material sensitization, stress corrosion cracking, interstitial transfer of carbon, nitrogen, or hydrogen or mass transfer from hot leg to cold leg of carbon or chromium, etc. Such testing may require removal of samples for laboratory study. Removal samples will be identified. 27

Surveillance

Surveillance capability will be provided to monitor PHTS pumps and piping during plant operation. Piping displacement will be checked by remote visual examination. Instrumentation to measure piping hanger loads and pump parameters will be determined as design progresses.

In addition, sodium leakage from the HTS piping and components will be monitored by the leak detection system. For additional information on leak detection see Section 7.5.5.

5.3.2.1.4 Protection Against Accelerated Corrosion and Material Degradation

Stringent precautions will be taken to minimize corrosive attack and stress-corrosion cracking in susceptible materials. Since ferritic and high-nickel alloys are relatively immune to these types of failure in environments normally experienced during fabrication and service, most attention will be focused on unstabilized austenitic stainless steel components. These steels, under certain conditions, have been shown to be susceptible to failure in halide and caustic-contaminated environments.

Sensitization usually encourages intergranular attack. In fact, the heavier the sensitization, the faster the rate of attack. Because of this the following general precautions will be taken to minimize intergranular attack: 1

a. Minimize the exposure of components to moisture by the use of protective coatings or enclosures.

b. Conduct any necessary stress-relieving heat treatments at temperatures above the sensitization range ($T \gtrsim 1600°F$) and rapidly cool to ambient temperatures upon completion.

c. Avoid the use of any potentially corrosive environment during all stages of component handling, fabrication, transportation, installation and service.

With respect to stress corrosion, sensitization will encourage intergranular cracking in halide environments. For caustic contaminated solutions, the role of sensitization is unclear. In some instances it seems to enhance attack whereas in others it retards it. As an overall precaution, therefore, it is desirable to strictly limit sensitization in austenitic stainless steel. Additional measures to mitigate stress corrosion will include the following:

a. Welding will be with low fluoride fluxes to minimize contamination from fluoride ions.

b. Components will be stress relieved to minimize stress assisted crack propagation. Heat treatments will be above 1600°F followed by rapid cooling to avoid sensitization.

c. Currently, there are no established procedures which will definitely eliminate stress-corrosion cracking in stainless steel components during the periods of fabrication and service. However, the principal contributory factors in stress-corrosion failure are known, and procedures for monitoring and minimizing the effects of potentially dangerous environments have been developed and will be employed in the fabrication of CRBRP components. These cover all stages of component handling, fabrication, transportation, storage, installation and service.

As explained above, the sensitization of unstabilized austenitic stainless steels will be minimized in order to restrict possible intergranular attack and stress-corrosion cracking if the material is inadvertently exposed to a corrosive environment. This will require careful control of all heat treatments to ensure that if it is necessary to traverse the temperature range in which sensitization can occur (800-1600°F) the time of exposure in this range is kept to a minimum.

For large complex components, however, rapid traversal of this range during the cool from the annealing temperature will be avoided in order to prevent the reintroduction of residual stresses.

Each component will be evaluated independently to determine whether heat treatment is necessary. If it is required the heat treatment will be optimized to give the maximum degree of stress relief together with the minimum degree of sensitization. 1

The inerted cell atmosphere water vapor concentration will be
44 held at approximately 1000 vppm during plant operation.

Sodium leak tests have shown corrosion rates to be 0.12 mil/hr at 1050°F, 1000 vppm H_2O and 1.2 v/o O_2 or less, and 0.014 mil/hr at 800°F, 1300 vppm H_2O and 1.2 v/o O_2. (References 35 and 36).

This "accelerated corrosion" from the presence of water vapor and sodium is acceptable in that propagation of a leak from corrosion at this rate will not significantly affect the plant capability to safety shutdown and maintain safe shutdown conditions. 23

5.3.2.1.5 Material Inspection Program

Materials test reports and letters of compliance will be obtained from the vendor. Chemical check analyses will be conducted on all materials, and archive samples will be maintained.

Completed components will be non-destructively tested (ultrasonic, eddy current, liquid penetrant, and/or X-ray) to ensure integrity. Particular attention will be given to critical areas such as welds. For austenitic stainless steel components undergoing heat treatment the absence of sensitization will be verified by the Strauss test (ASTM A-262, Practice E) on prototypic material which has been given the same heat treatment.

5.3.2.2 Material Properties

The coolant boundary sections of the primary heat transport system are made from unstabilized austenitic stainless steels, viz. Type 304 and Type 316 stainless steel. Depending on the application, minimum lower limits ranging from 0.04% to 0.055% will be placed on the carbon level. In all applications governed by the rules of ASME Code Case 1592, a minimum carbon content of at least 0.04% is specified in the material purchase specification. In certain applications (e.g., primary hot leg piping) where interstitial loss in sodium becomes appreciable, a minimum carbon content of 0.055% is specified in the material purchase specification to minimize the impact of interstitial transfer on design. Specific procedures for accounting for environmental effects on material properties (including interstitial transfer) are provided in an appendix to the Equipment Specifications. In order to minimize adverse effects due to too high a carbon level, an upper limit of 0.08% will be maintained. Various properties of the austenitic stainless steels are described below.

5.3.2.2.1 Short Term Tensile Properties

With respect to short term tensile properties it may be shown that both carbon and nitrogen levels have a strong influence and the following equations may be used to estimate the yield strength (σy), the ultimate tensile strength (σu), and the total elongation (ε) from the carbon and nitrogen contents C and N.

For solution treated Type 304 stainless steel

$$\sigma y = 89.952 + 181.167\ (C+N) - 0.148\ T\ (C+N)$$

$$-5.727\ T^{1/2} + 0.105\ T \qquad (1)$$

standard deviation = 2.236

[Valid for 297 ≤ T ≤ 922°K, 75-1200°F]

$$\sigma_u = -1.806 - 227.391\ (C+N) + 1218.392\ T^{1/2}$$

$$+ 1.127\ T\ (C+N) - 8.446 \times 10^{-4}\ T^2\ (C+N) \qquad (2)$$

standard deviation = 2.645

[Valid for 366 ≤ T ≤ 866°K, 200-1100°F]

$$\varepsilon = 29.595 - 89.898\ (C+N) + 468.414\ T^{-1/2} \qquad (3)$$

standard deviation = 2.955

[Valid for 477 ≤ T ≤ 922°K, 400-1200°F]

The above three equations are valid for (C+N) ≤0.13 weight percent and apply to plate, bar, pipe, and forged material. σ_y and σ_u are in ksi, ε is expressed in percent, and T is in degrees Kelvin.

In the case of solution annealed Type 316 stainless steel

$$\sigma_y = 88.474 + 211.640\ (C+N) - 0.206\ T\ (C+N)$$

$$- 5.762\ T^{1/2} + 0.111\ T \qquad (4)$$

standard deviation = 2.806

[Valid for 297 ≤ T ≤ 977°K, 75-1300°F]

17 $$\sigma_u = 8.563 - 461.665\ (C+N) + 1064.154\ T^{-1/2} \qquad (5)$$

$$+ 2.722\ T\ (C+N) - 2.330 \times 10^{-3}\ T^2\ (C+N)$$

standard deviation = 6.458

[Valid for 297 ≤ T ≤ 977°K, 75-1300°F]

17 $$\varepsilon = 28.000 - 99.139\ (C+N) + 653.926\ T^{-1/2} \qquad (6)$$

standard deviation = 6.703

[Valid for 297 ≤ T ≤ 977°K, 75-1300°F]

The above three equations are valid for (C+N) $\lesssim$ 0.10 weight percent, but at room temperature, 1100°F, and 1300°F the range of validity may be increased to 0.167, 0.141, and 0.141, respectively. The equations may be used for plate, bar, pipe, rod sheet, strip, and forged material.

Based on the above equations a general comparison may be made between Type 304 and Type 316 stainless steel. From Figures 5.3-1 and 5.3-2 it may be seen that for alloys containing equal quantities of (C+N) the ultimate tensile strength and ductility of Type 316 stainless steel is far superior. However, the yield strength of Type 316 is only slightly higher.

5.3.2.2.2 Stress Rupture Properties

The rupture strength of austenitic stainless steel is also strongly dependent on C+N content as shown in Figures 5.3-3 and 5.3-4. To a good approximation the rupture strength is a linear function of C+N.

Although the figures show that Type 316 SS has a higher rupture strength than Type 304 SS for a given temperature, the effect of a change in the C+N on the rupture strength is basically the same for each alloy. As shown in Figure 5.3-5 a given change in C+N changes the rupture strength by the same amount for each alloy at a given temperature. There is a temperature effect however with a change in C+N influencing the rupture strength more at the lower temperatures. The percentage change in rupture strength for Type 304 and Type 316 SS for a given change in C+N is shown in Figures 5.3-6 and 5.3-7.

A comparison of allowable design stresses for the unstabilized austenitic stainless steels, as specified in the ASME Boiler and Pressure Vessel Code, Section III, and ASME Code Case 1592 is given in Table 5.3-11. Note that the use of L grade materials is restricted to a temperature of less than, or equal to, 800°F.

5.3.2.2.3 Fatigue Properties

A comparison between the fatigue behavior of Type 304 and Type 316 stainless steel has been made for temperatures between 806 and 1202°F. The Type 304 contained 0.053 and 0.052 weight percent carbon and nitrogen, respectively, and the Type 316, 0.06 and 0.048 weight percent carbon and nitrogen, respectively. For equivalent heat treatments and test temperatures Type 316 has a shorter fatigue life than Type 304 at the same total strain range. If the comparison is based on the stress amplitude, however, Type 316 is superior. This is because for a given stress the corresponding strain range for Type 316 is somewhat smaller than that for Type 304 owing to the higher yield strength of Type 316 stainless steel. Hence, a selection of Type 304 or Type 316 for fatigue resistance will depend on whether the fatigue condition will be strain or stress controlled.

The effect of carbon content on the fatigue behavior of 18 CR-12 Ni iron based alloys has been studied. It was shown that at elevated temperatures in the range of 1112 to 1472°F, for a wide range of stress amplitudes, decreasing the carbon level from 0.05 to 0.004 weight percent causes a much shorter fatigue life and lower endurance limit. This was attributed to the beneficial effect of carbides in effectively blocking grain boundary sliding and migration, thereby inhibiting the nucleation of grain boundary cracks. At room temperature, however, it has been shown that for Type 316 stainless steel fatigued at strain amplitudes in the range ± 1 to ± 4 percent, the presence of 2 to 3 volume percent of chromium carbide can decrease the fatigue life to about on one-third of that in solution-treated material. This was attributed to the brittle fracture of the grain-boundary carbides. A comprehensive research program is underway on investigating the effect of interstitial transfer on the fatigue of LMFBR materials. This includes work at Westinghouse and Argonne National Laboratory on interstitial loss effects in Types 304 and 316 stainless steel, and a General Electric Company effort on the effects of carburization in both austenitic and ferritic alloys. The Westinghouse and Argonne work will be mainly at elevated test temperatures whereas the latter program will be focused on lower temperatures (<1000°F) where carburization of austenitic stainless steel prevails. 1

No reduction of fatigue properties due to carbide precipitation is expected in either Type 304 or Type 316 austenitic stainless steel at the operating temperatures of the CRBR. Various investigators have shown that at temperatures above 750°F there is either no effect or a beneficial effect due to the precipitation of carbides on the fatigue properties of either Type 304 austenitic stainless steel. Driver (Ref. 5.3-42) has shown that a 0.05C austenitic stainless steel, heat treated 24 hours at 1292°F to produce $M_{23}C_6$ carbides, had a higher fatigue strength than a 0.004C austenitic stainless steel. Driver's results for tests at 1292°F are shown in Figure 5.3-39. Kanazawa and Yoshida (Ref. 5.3-43) have also observed increased fatigue strength of both Type 304 and Type 316 which they attribute to aging at temperatures of 750°F and 1110°F during fatigue testing. Cheng et al (Ref. 5.3-44) have shown that aging both Types 304 and 316 stainless steel at 1050°F (for 1000 hours) had a beneficial effect on the 1050°F fatigue life. For material aged at 1200°F (for 160 hours) there was no significant difference in fatigue life over that of annealed material. Brinkman et al (Ref. 5.3-45) have shown improved 1100°F fatigue properties after aging at 750°F or 1110°F.

In terms of high temperature fatigue crack propagation rate, the effect of carbides in Type 304 and Type 316 appears to be negligible or slightly beneficial. James and Knecht (Ref. 5.3-46) found no effect of aging for 1150 hours at 891°F on the crack paopagation rate in Type 304 or Type 316. Mahoney and Patton (Ref. 5.3-47) have found that crack propagation rates at 1200°F were not dependent on either carbon content or carbide morphology after aging for 18 hours at 1300°F. James (Ref. 5.3-6) has shown a decrease in the crack growth rate of both Type 304 and Type 316 stainless steel. Material aged at 1000°F or 1200°F had crack growth rates lower than those of the annealed material as shown in Figures 40 and 41. The improvement in the elevated temperature fatigue properties of aged Type 304 and 316 stainless steel has been attributed to carbides effectively blocking grain boundary sliding and migration and to a blunting effect when a crack encounters a carbide particle. (Ref.5.3-42).

Although the above results are based on short time aging data, relative to CRBRP operating life, the results are considered representative of those that will be observed in CRBRP. Weiss and Stickler (Ref. 5.3-48) have shown the precipitation of carbides in Type 316 to follow the time-temperature curve given in Figure 5.3-42. Precipitation of the 25 carbides in Type 304 would follow a similar curve). It is apparent from this curve that the carbide precipitates will be present for long aging times. This has been confirmed by data of Mochal et al (Ref. 5.3-49). They observed that carbides considered to precipitate in 316 SS throughout the duration of a 5 year test program which included approximately 30,000 hours at 1225°F. Although these data indicate that carbide precipitation will occur over a very long period in CRBRP, the effect on the fatigue properties is expected to be more immediate. James (Ref. 5.3-6) has shown that increasing the aging time from 1500 hours to 6000 hours did not significantly change the aging effect.

As noted above it has been shown that there may be a reduction in the room temperature fatigue properties of aging austenitic stainless steels. Barnby and Pearce (Ref. 5.3-50) observed a 1/3 reduction in the fatigue life of Type 316 stainless steel which they attributed to the presence of 2-3 volume percent carbides.* However, Mahoney and Patton (Ref. 5.3-47) did not observe any significant difference in fatigue crack propagation rates in Type temperature crack propagation rates in Type 304 stainless steel after aging for 3000 hours at 1200°F.

*Test temperature not stated but assumed to be room temperature.

5.3.2.2.4 Effect of Neutron Irradiation on Mechanical Properties

Neutron irradiation has the following effect on the mechanical properties of austenitic stainless steels:

a. Yield and tensile strengths are rapidly increased (see Figures 5.3-9 and 5.3-10).

b. Ductility is decreased (see Figure 5.3-11).

c. Creep rupture strength and creep ductility are greatly reduced (see Figure 5.3-12).

d. Fatigue life decreases (see Figure 5.3-13).

The figures indicate that for fluences less than 10^{21} n/cm^2, the effects are not significant. Fluence seen by the primary head transport system piping and components are several decades below this level. For example the maximum fluences external to the reactor vessel (at the core midplane) are 3.4×10^{18} n/cm^2 and 2×10^{20} n/cm^2 for fast (E > 0.1 MeV) and total fluences, respectively. Therefore the effect of neutron irradiation on mechanical properties is not a consideration for PHTS materials.

5.3.2.2.5 Mass Transfer Properties 17

In the primary heat transport system, which is constructed mainly from Types 304 and 316 stainless steels, there will be a gradual loss of metallic elements from the hotter regions of the system during service. This will cause a measurable corrosion loss which will depend on metal temperature, sodium velocity, and oxygen content in the sodium. The following correlations have been derived for Types 304 and 316 stainless corrosion rates.

For sodium velocities from 10 to 40 ft/sec., and greater

$$R/\phi = 6.68525 \times 10^7 \text{ Exp } (-18120/T_K) \qquad (1)$$ 17

with R = mils/year, T_K = °K, and ϕ is the oxygen level in ppm as measured by the vanadium wire technique.

For velocities less than 10 ft/sec.

$$R/\phi = (1.1704 \times 10^7 + 3.496 \times 10^6 V) \text{ Exp } (-18120/T_K) \qquad (2)$$ 17

in which V is in ft/sec.

Corrosion allowances for PHTS components will be based on the above equations. For all but thin sections, such as IHX tubing where the allowance is 0.001 inches on both the inside and outside of the 0.045 inch (minimum) wall tubes, the corrosion allowance over the 30 year lifetime of the plant is a minor consideration even at the highest service temperature.

5.3.2.2.6 Interstitial Transfer Properties

The effect of sodium exposure upon the mechanical properties of stainless steel is expected to involve the change of the concentration and distribution of the interstitial alloying elements, carbon and nitrogen. Recent observations suggest that both long-term properties, such as creep rupture strength, and short-term tensile properties are thus altered (Ref. 1a thru 1j). Independent observations (Ref. 1k, 1m, 1n) in polythermal sodium systems show that carbon and nitrogen are transferred from stainless steel in high temperature regions, such as upper core components, to stainless steel in low temperature regions, such as the IHX and cold leg return lines. Several theoretical treatments of carbon transfer predict this movement, although the predicted values vary (Ref. 1p, 1q, 1r).

In order to predict variations in mechanical properties due to sodium effects, computational techniques based on two sets of information are required. These are the effects of the operating environment (i.e., temperature, time, sodium chemistry, and location) on the interstitial concentration in a particular component, and the relationship between the interstitial concentration and the required mechanical properties. The latter relationship has already been described in Section 5.3.2.2.1; a description of the former effects follows.

Work at Westinghouse and elsewhere (Ref. 1s, 1t) has concentrated on determining the carbon partitioning behavior between isothermal elements of sodium and steel as a function of sodium temperature and carbon activity or carbon potential. An arbitrary technique for measuring the carbon activity of sodium has been developed, the measured parameter being the equilibrium carbon level in a Type 304 stainless steel foil exposed at 1300°F to the system sodium. This value, in ppm, was termed the "carbon potential" C_s of the sodium. C_s can be translated into a carbon activity measurement.

Equilibrium carbon values (C_e) were obtained for various structural materials exposed to sodium of known carbon potential and temperature. The relationships given below between C_s, C_e, and temperature were derived from the experimental data for Types 304 and 316 stainless steels. (Ref. 1s)

For Type 304 Stainless Steel (1a)

$$\log_{10} \frac{C_e}{C_s} = 3.85973 - 0.002973T(°F)$$ 17

For Type 316 Stainless Steel (1b)

$$\log_{10} \frac{C_e}{C_s} = 3.03454 - 0.002313T(°F)$$

These correlations were obtained under low sodium velocity, essentially isothermal conditions, with little or no metallic mass transfer.

Effective diffusion coefficients, D_{eff}, for carbon were measured for Types 304 and 316 stainless steels as a function of temperature (Ref. 1u). The effective diffusion coefficient is considered as describing the net effect of all rate dependent processes involved in the transport of carbon into or out of an element of stainless steel. Rate dependent processes include solid state diffusion, precipitation and dissolution of carbides, and surface reactions. It was assumed that the net process could be described by solutions to the diffusion equation with constant boundary conditions.

D_{eff} was determined by exposing 0.003-inch foils in a sodium test loop under a decarburizing environment for various times and plotting C (average carbon content), the bulk carbon concentration, as a function of time. A computer program was used to fit a solution of Fick's second law to the data in a least squares sense. The program determined both the values of D_{eff} and C_e.

Measurements were made in the temperature range 1050°-1400°F. The data are described by the equations, where D is given in cm^2/sec.

$$D_{eff} = 1.75 \exp(-43{,}234/RT) \text{ for Type 304SS} \quad (2a)$$

$$D_{eff} = 40.91 \exp(-49{,}851/RT) \text{ for Type 316SS} \quad (2b)$$

where R is the gas constant and T the absolute temperature in degrees Kelvin.

The equilibrium and diffusion data were applied to reactor situations (Ref. 1v, 1w) by making certain critical assumptions:

- The operating loop rapidly reaches a steady state situation, i.e., the carbon potential, C_s, does not change as a function of time. (Note: the value of C_s = 30 used in Ref. 1w is now considered conservative.)
- The carbon concentration of the sodium remains constant, regardless of position and temperature in the system.
- An element of steel surface is in equilibrium with the sodium in contact with it. 17

- The equations derived from the experimental data can be linearly extrapolated to temperatures outside the range of experimentation.

- The carbon transfer process through the steel could be described by the classical diffusion equations.

The equations can be used to calculate the carbon gradient through a stainless steel component at any time and to calculate the bulk carbon concentration in the component as a function of time.

The bulk carbon concentration for finite systems exposed to the sodium on both sides is given by the equation:

$$\bar{C}(t) = C_e + \frac{8}{\pi^2} (C_o - C_e) \sum_{i=1}^{\infty} \frac{1}{(2i-1)^2} \exp\left\{-\left[\frac{(2i-1)\pi}{na}\right]^2 Dt\right\} \quad (3)$$

The absolute carbon value at a depth x after time t, C(x,t), is given by:

$$C(x,t) = C_e + \frac{4}{\pi} (C_o - C_e) \sum_{i=1}^{\infty} \frac{1}{(2i-1)} \sin\left[\frac{(2i-1)\pi x}{na}\right] \exp\left\{-\left[\frac{(2i-1)\pi}{na}\right]^2 Dt\right\} \quad (4)$$

where: $\bar{C}(t)$ = average carbon content at time, t (ppm)

C_e = surface carbon concentration (ppm)

C_o = initial carbon concentration (ppm)

D = diffusion coefficient for carbon (cm^2/sec)

t = time (sec)

C(x,t) = carbon concentration (ppm) at any position, x, and time, t

x = distance from wall-sodium interface (cm)

a = component wall thickness (cm)

n = 1 if exposed from two sides (cm)

n = 4 if exposed from one side (cm)

C_e and D are calculated from Equations (1) and (2). The computer program is listed in Reference 1v.

In the case of nitrogen, there are no data available to allow the formulation of relationships similar to those described above for carbon. Thus assumptions are made that nitrogen behaves similarly to carbon and the nitrogen equilibrium concentration is equal to C_e for any given temperature. 17

Values of (C + N) can thus be estimated and these values substituted into the appropriate equations, 1 through 6, in 5.3.2.2.1 in order to determine mechanical property changes arising from interstitial transfer during sodium exposure. 17

5.3.2.3 Component Descriptions

The primary sodium pumps are free surface, single stage, vertically mounted, drawdown type centrifugal pumps driven by a variable speed 5000 hp squirrel cage induction motor. An auxiliary 75 hp pony motor, on each pump unit, provides low flow capability (<10%) for decay heat removal and other low power standby conditions. Variable pump speed is achieved by having the main drive motor supplied with variable frequency power from a fluid coupled MG set. Each
59 pump is designed to deliver 33,700 GPM of $995^{\circ}F$ sodium at a 458 foot head. 33

The design envelope for the primary pump is shown in Figure 5.3-14. The pump tank incorporates a 36 inch side suction nozzle. Pump discharge is through a 24 inch nozzle.

Sodium flow enters the pump tank through the horizontal 36" diameter nozzle on the equator of a spherical tank. From the nozzle the flow splits and enters the inlet guide structure into the upward and downward facing impeller inlets. The flow leaving the impeller is joined and the combined flow passes through a
59 triple volute pump casing. The flow is then channeled through the 24" 33
horizontal exit nozzle on the pump tank equator at a location 90° clockwise from the suction nozzle as viewed from above. A 20 inch diameter balancing pressure port is located on the high pressure side of the pump, 180 degrees away from the discharge opening. This feature is included to reduce side loads exerted by discharging fluid, thereby reducing any creep enhanced rachetting
49 57 and to prevent excessive loss of bearing clearance. Separation of the suction and discharge pressure in the pump tank is achieved by controlled clearances between the impeller, pump casings and ducts within the pump tank sphere.
59

The impeller and impeller shaft assembly is supported in the pump casing by dual sodium lubricated hydrostatic bearings above and below the impeller. The bearings are supplied with sodium at near pump discharge pressure. The sodium
49 flows through the hydrostatic bearing pads and returns through the bearing 33
clearances to the low pressure side of the pump.

Pump design is based upon sodium level control using a standpipe bubbler system
59 in which a continuous flow of argon gas is supplied to the pump cover gas space above the sodium free surface. During pump shutdown or low sodium flow conditions, the gas supply pressure is sufficient to depress the liquid sodium level to the standpipe nozzle elevation and gas will bubble up the standpipe to the reactor cover gas 33

system. During higher sodium flow rates, normal drawdown will almost uncover the standpipe nozzle, purge gas will further lower the sodium level, and gas
41 will flow (bubble) into the nearly empty standpipe to the cover gas system.

49 The standpipe is essentially one leg of a manometer which automatically balances the gas pressure in the pump tank. The standpipe-bubbler will automatically perform the function of a pump cover gas vent and relief valve to the cover gas system, and will not rely on the signal of a liquid level sensor for level control. Level sensors are available for alarm.
49

The standpipe bubbler nozzle location is shown on Figure 5.3-14.

41 Since the standpipe bubbler is physically just a pipe connecting the pump tank to the gas equalization line between the reactor and the overflow tank, there are no malfunctions which could result in overpressuring the cover gas region of the pump such that the sodium level in the pump would drop to a point which
59 would permit this gas to enter the hydraulics of the pump.

The vent line from the top of the IHX to the pump tank enters the pump tank at an elevation about 30 inches below the standpipe bubbler nozzle. A trip in the IHX vent return line prevents pump cover gas from flowing to the IHX should the
54 pump sodium level drop to the IHX vent return nozzle elevation. During full
52 power operation, 200 gpm of sodium flows from the IHX to the pump to vent any gas tending to accumulate at the top of the IHX through the vent line to the cover gas system of the pump. During rapid flow changes such as a scram, the flow from the IHX to the pump through this line decreases with the speed of the pump. Since the vent line connection to the pump is located above the hydraulics of the pump, any gas entering through this line will bubble up to the free surface rather than enter the main sodium pump. In the case of bubbler system, the standpipe is 6 inches in diameter connected to a 2-inch gas equalization line which is more than adequate to vent all of the gases which could inadvertently enter the primary pump tank. Therefore, there is no way that this gas could be bled to the core. Even if gas entering the pump should get into the pump hydraulics and into the main sodium piping, it would accumulate at the top of the IHX and be vented from the heat exchanger back to the pump gas system via the IHX vent line. Therefore, any gas introduced into the PHTS from the pump is precluded from entering the core of the reactor.

The gas blanket in the pump tank will serve as a damper on level swings - for 25
example during pump trip the sodium level will tend to rise from the 100% flow, drawn-down level to the normal system level. During this event the sodium fluid will rise to a level somewhat above the gas-bubbler nozzle working against the entrapped and compressing gas in the pump upper tank ullage. As the gas supply continues to enter the pump tank, the liquid level will stop its rise, then fall until the bubbling action through the standpipe is re-initiated. 33

Both PHTS and IHTS pumps require a shaft seal to effect a zero leak seal from cover gas to atmosphere. This shaft seal, shown schematically in Figure 5.3-14A, is an oil lubricated, double rubbing face seal. The seal has a shaft driven internal oil circulator and an integral air to oil heat exchanger, with oil supply to make up for oil leakage past the rubbing faces. Oil leakage from the seal assembly into the sodium coolant is prevented by two barriers. The first barrier is an oil dam approximately 1.2 inches above the lower face seal. The normal leakage from the lower face seal is diverted by this oil dam into the oil leakage drain passage into the lower seal leakage collection reservoir. A second barrier is the collar above the drop down seal located just above the purge labyrinth. This collar extends beyond and over the labyrinth, thereby shunting any oil to a drain plenum. For oil to penetrate into the sodium, three things must happen:

- o Failure of the oil dam
- o Failure of the collar to divert oil
- o Overflow of the plenum drainage over the drop down seal lip

A positive pressure is maintained in the shaft seal oil at all times by means of an oil supply tank which will be pressurized above the loop operating pressure. The oil feed line to the seal will be oriented to preclude seal drainage in the event of a line break. The seal is capable of many hours of operation on the self contained fluid.

The oil system supporting the shaft seal contains three tanks, each of which will have a level probe, thereby permitting monitoring of total oil inventory, its location, and permitting calculation of seal leak rate. The lower seal leakage collection tank is sized to hold the entire system's oil inventory of approximately 41 gallons.

Oil vapors which may potentially be drawn from the lower seal leakage collection tank into the tank ullage during draw down (pump speed up) are retarded from such passage by means of a split flow purge gas feed of recycled argon into the purge labyrinth. This gas feed splits and flows up and down the shaft from the feedpoint. This gas input is flow controlled at the inlet, and flow controlled at the discharge from the lower seal leakage collection tank. If feed pressure into the tank is detected to be low (by the gas feed system) the discharge of gas from the tank will be closed. In event of gas line rupture at the oil tank discharge, the orificing by the line will retard loss of cover gas pressure.

Radioactive vapors from the tank ullage are prevented from escape to the atmosphere by the two barriers consisting of the gas downflow at the purge labyrinth and the oil lubricated double shaft seal. Radioactive purging is continuous by means of the bubbling in the standpipe, which is connected to RAPS.

The kinetic energy of the total rotating mass coupled to the pump impeller following a <u>primary pump trip</u> is as follows:

Pump (impeller, bearing, and shaft)	660,000 ft-lbs
Motor	5,540,000 ft-lbs
Total estimated kinetic energy (1116 rpm) 33	6,195,000 ft-lbs 16

Cracking or internal flaws in the rotating assembly could result in a fracture-type failure; however, the rotational inertia forces are sufficiently low that a structural failure has little potential for serious consequences. It is a design requirement and it has been demonstrated analytically that failure of any part of the pump rotating assembly will not affect the integrity of the pressure boundary. It has been demonstrated analytically that failure of any
59 part of the rotating assembly or the lower region of the inner structure (including the static part of the pump hydraulic system and the shaft lower bearing support) will not reduce the flow path area below a required minimum for natural circulation of the sodium under shutdown conditions.

The pump equipment specification requires the maximum energy of missiles resulting from a fracture of the pump rotating assembly at maximum overspeed to be dissipated before a coolant boundary is breached. An examination of the design of the pump static hydraulic machinery shows that there is a minimum of one and as many as four layers of over one inch thick stainless steel between the impeller and the outer shell of the coolant boundary. It was shown that this structure will easily absorb the energy of a ruptured impeller. The 14
analysis determined that the maximum kinetic energy available in a ruptured piece from the rotating assembly (at maximum pump overspeed) was less than the energy required to rupture the weakest section of the coolant boundary.

The pump has been designed such that the first bending critical speed of the shaft is at least 125 percent of the design speed. General allowable pump vibrations were specified based on the Hydraulic Institute Standards, 12th Edition. For design conservatism, the vibrational amplitudes of the primary pumps are to be less than those specified by the Hydraulic Institute. Water performance tests of the pump and drive motor assembly substantiated the
59 analytical models used for the dynamics analysis of the pump.

59 The PHTS pump design minimizes the potential for amplification of impeller-volute reaction pressures by acoustic resonance within various portions of the pump and pumping system. Such amplification results from some portion of a pressure pulse arriving at a specific location in phase with other pressure pulses so that the effects are additive. Such an increasing pulse is finally limited by energy dissipation due to flexure of the containment walls, fluid friction, inefficiency of the reflection, etc. Based on fluid borne noise data collected by the pump manufacturer, the fluid borne peak-to-peak pressures have been demonstrated in water tests to be less than 0.5 psi for the PHTS pumps.

59

Section 5.3.1.1 stated that the natural frequencies of components will be different from pump driving frequencies to avoid resonance. The frequencies of vibrations generated by the pumps, both primary and intermediate are given in Table 5.3-23.

The lowest natural frequency mode of the IHX tubes occurs at about 63 hz. The calculation of the IHX tube frequency is for a single full length active tube, in 995°F sodium, constrained by the tube supports provided between the tube sheets. Water tests of a full scale, 30° segment IHX model are planned to verify the analysis.

The estimated fuel assembly and fuel rod bundle first mode natural frequency is in the range of 50 hertz. PSAR Table 4.2-19 identifies a planned fuel assembly flow and vibration test. Objectives of this test include determining the vibration dynamic characteristics of the assembly. 25

5.3.2.3.2 <u>Intermediate Heat Exchanger</u>

The CRBRP IHX is a straight tube, flexible downcomer design. It is comprised of an enclosure shell, tube bundle with replaceable 11
bellows and cylindrical hanging support. The main subassemblies and other design features are shown in Figure 5.3-15 and described below.

<u>Hanging Support</u>

The main support for the IHX shell and bundle assembly is the hanging support which consists of a cylindrical shell with an upper flange. The hanger flange has bolting holes used to anchor the support to the operating floor. Four inspection parts are located the the hanger flange.

The cylinderical portion of the hanging support contains openings for the intermediate piping. The hanger is fabricated from type 304 and type 316 stainless steel. The lower edge of the hanging support is welded
27 to the shell and bundle through a "Z" shaped junction forging.

27

Shell Assembly

The shell, which is the main IHX enclosure, is fabricated from Types 304 and 316 stainless steel. The shell is welded directly to
27 the lower edge of the cylindrical hanging support through the "Z" junction.

The shell cylinder is fabricated from three cylindrical sections. The use of Type 316 stainless steel reduces the thermal and creep ratcheting problems, thus
27 providing a shell design without the need for a thermal liner in the high temperature region. The bottom portion of the shell consists of a shell cylinder, lateral support ring, hemi-head and primary outlet nozzle, all of Type 304
27 stainless steel. The lateral support ring has sixteen spacer guides attached to it which serve as guides and restraints for the tube bundle lower tubesheet.

To preclude potential cover gas accumulation in the IHX, there is continuous venting sodium from the top of the IHX shell to the primary pump tank below the minimum safe sodium level. This sodium flow will carry any gas evolved in the IHX to the pump tank where the gas will migrate to the cover gas space.

Tube Bundle Assembly

The tube bundle assembly consists of two major sub-assemblies: (1) the bundle, consisting of tubesheets, tubes, support plates, tierods and spacers, outer shroud, hemi-head, downcomer, strongback and by-pass seal, and (2) the channel assembly consisting of replaceable bellows, upper head, intermediate outlet nozzle, intermediate vent, inner and outer channel
27 cylinders, upper downcomer pipes, and "Z" junction forging.

Tube Bundle Subassembly

The tube bundle contains 2850 7/8 inch O.D. x 0.045 inch minimum wall tubes spaced on a 1-5/16 inch triangular pitch. The tubes are joined to both the upper and lower tubesheets. The tube-to-tubesheet joints are made by front face fillet welding the tube ends to specially prepared stubs in the tube sheet face. An automatic T.I.G. welding procedure is used to join tubes to tubesheets. The tube ends will be expanded into the tubesheet holes. The upper and lower tubesheets are designed with a minimum inner and outer rim
27 so as to permit better thermal response with the perforated portion.

The strongback pipe, 34-5/8 inches diameter, is welded to the upper tubesheet and extends down to approximately 6 inches above the lower tubesheet. The O.D. of the strongback is machined to provide little distance
7 between it and the inner support plates reducing by-pass flow away from the tubes. The lower portion of the strongback contains three slots which engage matching keys attached to the downcomer. The slot and key combination provides torsional and lateral stiffness for the lower tubesheet-head complex
7 during shipping and operation. A mechanism (gas trap) is incorporated into the lower downcomer region to prevent gas from being entrapped in the annulus between the strongback and downcomer.

The outer shroud is provided as the outer bound of the flow in the heat transfer zone of the bundle. The uppermost portion of the outer shroud is the distribution cylinder which forms the entrance zone to the heat transfer tubes. This distribution cylinder is designed to insure uniform circumferential flow into the bundle. This perforated entrance cylinder is welded to the upper tubesheet. Eight additional rings and eight cylindrical sections welded together make up the remainder of the outer shroud complex. The machined rings serve as stiffeners. One ring is utilized as a primary bypass seal attachment point. The ring at the primary inlet nozzle is a support point for saddle and strap supports during shipment. The outer shroud terminates 1 to 2 inches below the lowermost support plate to allow for exit flow from the heat transfer zone
57 to the primary outlet nozzle.

The tubes in the tube bundle are supported by nineteen support plates.
7 The uppermost support plate extends to the I.D. of the shell. This plate forms the upper boundary of the inlet plenum. The plate limits direct impingement of sodium upon the face of the upper tubesheet. The second support plate is a complete plate, i.e., it extends from the O.D. of the strongback to the I.D. of the outer shroud. This plate contains flow holes preferentially drilled to further ensure uniform bundle flow. The lowermost plate is also complete and
7 aids in uniform distribution through the tube bundle exit. The sixteen intermediate support-baffles are overlapping "doughnut" baffles which contain uniform flow holes resulting in a combination of cross and axial flow.

The plates are supported by eighteen tierods and spacers, 6 at the I.D., 6 intermediate and 6 at the O.D. The tierods are threaded into tapped holes in the shell side of the upper tubesheet. This is a standard method of support for heat exchangers. The lower tubesheet and hemispherical head is a floating assembly. The head is welded to the lower tubesheet and serves as a pressure boundary between the intermediate and primary systems. A baffle flow ring is provided to uniformly distribute the intermediate flow

27 into the heat transfer tubes. The lower head complex is guided by the sixteen spacers attached to the shell lateral support ring. The guides provide lateral restraint against shipping and seismic forces.

Multiple drain holes are located in areas where gas could be trapped during filling or sodium could be trapped during drainage. Vent lines
27 (2 inch) are provided on the intermediate side for use during fill operations.

The recommended reference design is a bellows type primary by-pass seal. The bellows type seal represents a simple, basic design utilizing the minimum number of pieces. The design is sub-assembled to the bundle as a complete unit. The assembly is clamped to a machined ring which is an integral part of the bundle outer shroud. The clamp is made up of split rings that are applied to the bundle in 180° overlapping segments and pin welded together (i.e., the segments are pinned and the pins are welded in place). The bellows flange is then bolted to this ring and held in place with Class A of RDT Standard E8-18T locking devices. A split ring, is bolted to the bellows lower flange to form a containment chamber in the event of bellows failure. The O.D. of the bellows is protected by a 1/4 inch thick cylinder. This sub-assembly can be inspected prior to insertion of the bundle into the shell. The mating of the lower seal face and the seal ring in the shell, upon insertion of the bundle into the shell, is not anticipated since reasonable misalignments can be accepted by the bellows. The calculated primary by-pass flow thru the seal is less than 2.8 percent of the primary loop design flow. There may be local depressions or swells in the machined face of the mating flanges, but leakage through these areas will result in acceptable by-pass flow. The bellows will be precompressed to ensure a load on the sealing flange faces during operating conditions. Once the thermal movements between shell and bundle have been determined, the precompression of the bellows will be finalized.

27

Channel Subassembly

A "vacuum bottle" design forms the inner intermediate channel cylinder reducing the radial thermal gradient across the walls and eliminating the problem of the juncture discontinuity that would result if the inner channel cylinder were welded directly to the head and tubesheet. To make the welded connection between the channel, shell, hanging support and integral bundle it is necessary to use a "Z" section forgoing. The "Z" section branches out to form the channel outer wall and the bundle support. In addition the "Z" section allows for minimization of the quantity of entrapped gas above the
27 primary vent nozzle.

7

The uppermost portion of the channel contains the removable IHX bellows assembly. The bellows permits the differential axial thermal growth between the downcomer and straight tubes and also serves as a portion of the pressure boundary between the intermediate and primary systems. 11

The bellows is designed as a replaceable unit. The bellows housing contains two 6 inch diameter handholes to permit required NDE during assembly and replacement.

The bellows will be constructed as a Class I Component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, and the Supplementary Requirements for RDT E15-2NB-T. The analytical methods for the bellows will be in accordance with PSAR Section 5.3.3.1.5, Analytical Methods for Pumps, Valves and Heat Exchangers.

The IHX Bellows are presently undergoing testing at Foster Wheeler Energy Corporation. The test bellows will undergo a life cycle test with pressure/deflection cycles equal to four times expected life. Two test bellows have already successfully completed this test with no failure or cracks detectable by Liquid Penetrant Examination. Strain gauge measurements indicate good correlation between analytical and experimental results. 19

A 2 inch connection in the bellows housing serves as a start-up and/or intermittant vent to prevent excessive gas accumulation.

5.3.2.3.3 <u>Valves</u>

The only valves located in the primary heat transport system are the cold leg check valves. The function of these valves is to limit flow reversal in a loop which has a pump failure. The valve is physically located in the 24 inch diameter cold leg piping on the horizontal section between the intermediate heat exchanger and the reactor inlet.

The check valve design is scaled-up from the FFTF design with improvements for maintainability. The concept employs a tilting disc seating against a free-standing seat. The moving member and the seat are structurally remote from the main body to minimize distortion of the seat due to mechanical or thermal loading of the body. Figure 5.3-16 shows the direction of flow during normal operation, seat and disc, and the principle of operation. A dash pot is provided to control the closing rate of the disc upon flow reversal to limit pressure surges in the loop. To provide natur- 1

al circulation flow, the disc hangs at a slightly open angle and the dash
pot and pivot friction forces are spring balanced to insure the disc re-
turns to this position. This concept was proven in high cyclic tests on the
FFTF program and extensive design studies have proven feasibility of scale-
up to larger sizes. The dash pot performance will be verified by cyclic
41 tests in water and freon.

An access opening of welded or flanged seal-welded construction
will be provided at the top of the valve for removal of the disc assembly
and the dash pot. A vent line connection in the top of the access closure
permits removal of entrapped gas during loop-filling operation. 1

The current valve concept under consideration is fabricated of Type 304 austenitic stainless steel to be compatible with the 304 cold leg piping. Stellite handfacing is employed at the disc and seat mating surfaces to minimize wear. Similarly stellite bushings are used to reduce wear on rubbing surfaces in the dashpot and disc hingle pins.

Check Valves

The design limits for PHTS Check Valves are as follows:

1) Normal and Upset Operating Conditions
Figure NB-3222-1 ASME Code Section III

2) Emergency Conditions
Figure NB-3224-1 ASME Code Section III

3) Faulted Conditions
Table F-1322.2-1 ASME Code Section III, Appendix F

4) Normal Plus Upset, Emergency and Faulted at Elevated Temperature
Figure 3220-1 ASME Code Case 1592

The standard valve design rules in Section NB-3512 apply to valves
of conventional shape having generally cylindrical or spherical bodies with
a single neck of a diameter commensurate with that of the main body portion,
such as having a neck inside diameter less than twice the main run inside
diameter in the neck region. The check valve is a design previously de-
veloped for high temperature service in the FFTF Plant, and is of such
shape that the standard rules do not apply, therefore, the alternate design
rules of Section NB-3200 are used. These rules for design require very
thorough mechanical and thermal analysis of the entire structure and permit
use of higher stress levels in evaluation. 19

5.3.2.3.4 Piping and Support

Each PHTS piping loop is provided with the necessary elbows, supports, and expansion loops to provide adequate flexibility for thermal expansion as well as the necessary constant load hangers, rigid restraints and dynamic snubbers to react deadweight and seismic loads. Each loop has similar components, and the layout of each loop is identical except for the piping within the reactor guard vessel. This is necessary to maintain axisymmetric flow within the reactor. The PHTS loop piping consists of the main hot leg, the crossover hot leg, and the cold leg. All piping is of 0.500 inch wall thickness.

Piping

The main hot leg piping is 36-inch, stainless steel Type 316 and extends from the reactor vessel through the reactor cavity penetration, then on to the primary pump. At the pump exit the crossover piping consists of 24-inch Type 316 stainless steel which extends to the IHX primary inlet.

The cold leg piping commences at the IHX outlet and consists of a 24-inch Type 304 stainless steel. The cold leg continues through the reactor cavity penetration and completes the loop at the reactor vessel inlet nozzle. Located in this leg, in the horizontal runs, are the check valve and flow metering device.

All large PHTS piping will be supplied as spool pieces to the site.

Section 5.1.2 states that the PHTS piping is designed to limit sodium velocities to 30 feet per second.

This limit was selected based on optimization of pipe size, required insulation, pre-heat, sodium inventory and pipe corrosion. This optimization is based on consideration that the higher the allowable sodium velocity the smaller is the diameter of the PHTS piping, which results in shorter pipe runs due to increased flexibility, short elbows and a more compact PHTS layout. This results in cost savings in insulation, decreased size of the containment structure, pre-heaters, piping, HTS cell heat load, and sodium storage. These savings justify the corresponding increase in pump head.

The above was also considered in conjunction with the effect of corrosion. It was found that the corrosion rate remains nearly constant over a range of sodium velocities from 10 to 40 fps, thereby verifying the acceptability of the higher sodium velocity allowance of 30 fps. The 30 fps was chosen to ensure minimal erosion/corrosion in the PHTS piping. The PHTS piping system has been sized such that 30 fps sodium velocity will not be reached or exceeded during normal operation.

The erosion-corrosion rate as used as a function of temperature, velocity and oxide concentration used for the PHTS, is specified in Section 5.3.2.2.5.

Section 5.1.2 describes the primary system hot and cold leg welded piping. The selection of welded pipe was based on a detailed evaluation of stress criteria, available analytical results and pipe fabrication variables reported in Reference 5.3-51. The criteria for structural evaluation are those shown in Code Case 1592.

The design transient description for normal, upset, emergency and faulted plant conditions used in the design of the primary system piping runs are given in Sections 5.7.3 and 5.7.4. The process for the selection of transient umbrella events and the duty cycle is explained in Appendix B.

The in-service inspection requirements for the PHTS piping are defined in Appendix G.

Pipe Supports and Restraints

All piping loops for the Primary Sodium Piping System will be supported at specific locations along their lengths and will be restrained, at specific locations, against seismic disturbances.

The location and spacing of the pipe supports will conform to the following guidelines:

a. The maximum span between supports is based on limiting the piping dead weight stress to approximately 1500 psi. When calculating these stresses, the weight of sodium in the pipe, the pipe, insulation, trace heaters and other local appurtenances will be considered.

b. Pipe supports are located at points of concentrated loads, if possible, such as at the primary cold leg check valve.

c. If the thermal expansion or growth of the pipe is minimal at a support location, rigid restraints are used, otherwise constant load type hangers are used.

The supports for the piping and associated components for the PHTS consist of assemblies made up of commercially available constant load pipe hangers and rigid restraints, commercially available mechanical snubbers, and specially designed pipe clamps. These supports will be designed to meet the requirements of the ASME Code, Section III, Subsection NF for Class 1 supports.

The load carrying capacity requirements of the constant load hangers and rigid restraints were determined by deadweight analysis and the travel requirements of the hanger were determined from thermal expansion analysis. The minimum travel to be accommodated by the all metal constant load hangers used in the PHTS will be the maximum thermal expansion of the pipe at the hanger location plus 20 percent of that travel. The minimum travel must not be less than 1.0 inch. The constant load hangers will be provided with travel stops to limit the vertical travel of the pipe run when the pipe is empty.

The seismic snubbers are of the mechanical type (non-hydraulic). The snubber allows the free movement of the pipe during normal expansion and contraction of the piping system, but will restrain pipe movement during seismic motions.

The pipe clamps which are located on the horizontal runs of piping will be of the "non-integral" type. Typically, the horizontal pipe clamps will consist of a segmented outer steel clamp ring, and a segmented steel sheathed load bearing insulation inner ring. the outer segments are held together by a system of bolts and belleville spring washers. The clamping loads exerted by the belleville spring washers are designed to prevent lift off of the clamp from the pipe under design seismic conditions, but will not cause any undue stress on the pipe wall or cause damage to the sheathed load bearing insulation. Reference Figure 5.3-36.

The pipe clamps which are located on the vertial runs of piping engage the horizontal surfaces of the transition pieces which are integral parts of the vertical piping to satisfactorily transmit the pipe loads to the pipe hangers without slippage. The vertical pipe clamp consists of three major components; the outer support ring, the steel-sheathed load-bearing insulation inner bands, and the pipe transistion section
56 which includes an axial support ledge. The transition piece is designed to be welded to the vertical run of the pipe line. It has the same inner diameter of the pipe, and the wall thickness is the same as the pipe line at the point of attachment to the pipe line, but gradually slopes to a larger wall thickness, thus forming a support ledge which contacts the load bearing insulation and transmits the pipe load to the clamp ring. The outer support ring is made up of two Type 304 stainless steel semi-circular rings which are bolted together to form a very stiff circular ring. The inside surface of the ring is machined to a "channel" shape to receive and capture the canned load-bearing insulation. Attachment lugs for snubbers and hangers are welded to the outer surface of the
56 15 ring (see Figure 5.3-38). 1

The constant load hangers being used for pipe support in the reactor cavity area are of "all metal" construction, and are designed to meet the requirements of the ASME Code subsection NF, components supports. Although there is no directly applicable experience with constant load pipe hangers in areas such as the CRBRP reactor cavity, it is expected that no material property degradation will take place due to the radiation and temperature environment because of the type of materials being used.

The radiation fluence in the vicinity of the hangers nearest the reactor core has been estimated to be a 5×10^{18} nvt when integrated over all energy neutrons essentially no neutrons have energy >1 MeV). For this type of neutron energy spectrum, a fluence of 10^{19} nvt corresponds to the onset of shift in nil-ductility temperature (Ref. 29). It is concluded that no significant material degradation will accrue from this phenomenon.

Access to the pipe hangers in the reactor cavity for maintenance and inspection is provided by removable access plugs located in the HAA ledge. There are no inaccessible pipe hangers. 16

The design, fabrication and testing of supports for piping and associated components are given special attention to assure operability and structural integrity will be maintained during the CRBRP lifetime. The Class 1, 2 and 3 supports all have common basic features, controls and requirements like those defined in Section 3.9.2.6.

Reactor Cavity Attachment

The primary heat transport system penetrates the reactor cavity at six locations; the 36 inch hot leg between the reactor vessel and the primary pump of each loop and the 24 inch cold leg between the reactor vessel and IHX of each loop. At the penetration the piping is fitted with a "flued head" piping transition piece which becomes an integral part of the piping run. The flued head in turn is welded to a bellows seal which is attached to the cavity wall forming a flexible seal. The bellows, although not code stamped, will be designed to the criteria of ASME Section III, Subsection NF.

5.3.2.3.5 Guard Vessels

The IHX's and primary sodium pumps are located in structurally independent, free standing guard vessels. These are fabricated from Type 304 stainless steel. Location and general arrangement of these vessels is shown in Figure 5.1-3.

The IHX and primary pump guard vessels enclose the component and its primary system inlet and outlet piping up to a level which will preserve the reactor system minimum safe sodium level in the unlikely event of component leakage. The top elevation of each guard vessel is sufficient to prevent spillage of sodium over the lip when the pump is running at pony motor speed and the sodium is at the minimum safe level in the reactor. The guard vessels have segmented covers. These covers do not establish a pressure boundary. The covers contain penetrations for a drain and purge pipe, a gas sample tube and leak detector guide tubes. Additionally, inspection ports are provided to facilitate periscope inspection of the component during refueling, maintenance or other shutdown periods. The maximum pressure which can be experienced by the guard vessels is equal to the sodium head of the filled vessel.

5.3.2.3.6 Insulation

The thermal insulation used for the PHTS is a refractory fiber (alumina silica) in blanket form. The blanket insulation is sheathed in reflective steel. Where in-service inspection requirements are identified the insulation is pre-fabricated in encapsulated reflective steel modules to facilitate quick removal without disassembly of large sections of insulation. All materials used on the PHTS insulation system are compatible with the PHTS materials.

Piping

The PHTS insulation system incorporates a one inch annulus between the pipe and the insulation. The annulus houses trace heaters and stand-off heater retention units. To insure the one inch annulus, there is an inner reflective sheath. The appropriate thickness of thermal insulation is covered by an outer protective sheath with banding straps.

The present trace heating design locates the heaters at the bottom of the piping. The heater power is selected so as to produce a specific heatup rate for the piping and to limit the diametral temperature gradient across the pipe. The stresses that result from the diametral temperature gradient on the piping will be combined with other stresses in the piping during the filling condition to ensure that code stress limits are not exceeded. Preliminary evaluations of this condition have shown that the stresses resulting from the temperature 25
gradients are well within code allowables.

Components

For the PHTS components (IHX, pump) the insulation and trace heaters (IHX only) will be supported by a fabricated stainless steel framework supported from the cell structure. Annuli between the vessel surfaces and the insides of the insulation system structures will prevent metal-to-metal contact. For the PHTS component guard vessels (IHX-GV and pump GV) the fabricated insulation system frameworks are supported from appurtenances provided on the vessels and secured to the vessels by banding straps. A network of pins and banding straps will support layers of alumina-silica blanket insulation which can be readily removed so the trace heaters can be serviced if necessary. An outer protective
59 sheath with banding straps will keep the insulation in place.

Thermal Analysis

59 The heat loss per linear foot of piping versus insulation thickness is
calculated using the following equation:

$$q = \frac{2\pi r_0\ (\Delta T)}{\frac{r_0}{r_1 h_1} + \frac{r_0 \ln \frac{r_2}{r_1}}{k_1} + \ldots\ldots + \frac{r_0 \ln \frac{rx+1}{rx}}{k_x} + \frac{1}{h_r + h_c}} \quad \left(\frac{BTU}{hr\text{-}ft} \right)$$

The design requirement for in-containment insulation exterior surface
59 temperature is 140°F maximum with a cell atmosphere temperature of 90°F. Since
the temperature drop from the pipe to the outer surface is proportional to the
59 thermal resistances of the materials, the surface temperatures were calculated
using the following equation:

$$T_{surface} = T_{max} - (\Delta T) \frac{R_{insul}}{R_{total}}$$

The insulation thicknesses are, generally 14" on components, 1" on hot leg
pipes and 8" on cold leg pipes. In some limited envelope applications, reduced 1
59 thicknesses of special high temperature insulation are used.

5.3.2.4 Overpressure Protection

The reactor coolant system contains no pressure relieving devices for the sodium coolant. Pressures in the reactor coolant system are normally a function of the reactor cover gas pressure and the pressure developed by the main circulation pumps. Since the system contains no isolation valves, isolation of individual loops or components in those loops is not possible.

The reactor cover gas pressure is protected from pressures in excess of 15 psig by relief valves connected to the cover gas space of the primary overflow tank. The overflow tank gas space is connected to the reactor vessel cover gas space through a pressure equalization line which contains no isolation valves. (See Section 9.5) System pressures with pumps running at speeds coincided with the synchronous speed of the drive motors and maximum (15 psig) cover gas pressure are less than the design pressure of the system (200 psig downstream of the pump, 30 psig upstream of the pump).

Code Case 1596, "Protection Against Overpressure of Elevated Temperature Components," Section III, Class 1, is complied with by the following method.

Specifications for components and piping in the reactor coolant boundary include off-normal dynamic and sustained overpressure loads resulting from:

- Check valve slams resulting from a primary pump seizure
- Dynamic loads associated with a sodium/water reaction (IHX)

The plant protection system trips the pumps on primary to intermediate flow mismatches. Therefore, any decrease in primary flow due to some flow blockage in one primary loop causing the pressure to increase as the pump approaches its shutoff head would be limited. The plant protection system also trips the reactor and pumps on a flux to flow mismatch, thereby providing protection against overpressure due to core blockage during power operation.

5.3.2.5 Leak Detection System

5.3.2.5.1 Leak Detection Methods

Leaks from the liquid metal circuits of the reactor coolant system can
be detected by measurement of changes in liquid metal inventory, detection of 28
radioactivity and a separate leak detection system.
40 28
The leak detection system will detect leaks, if they should occur,
in piping, and inside of major component guard vessels as well as below large
tanks such as the reactor overflow tank. Details of the methods for detection
40 of liquid metal to gas leaks are discussed in Section 7.5.5.1. 28

5.3.2.5.2 Indication in Control Room

Detection of a leak will activate an annunciator in the control room.
As discussed in Section 7.5.5.1, leak location will be identifiable from either
40 the Plant Data Handling and Display System or by reference to the leak monitoring
panels.

Since these loops do not contain isolation valves, isolation of the leaking loop is not possible; however, following shutdown, the affected loop may be partially or completely drained to limit the leak without jeopardizing the function of the other two loops.

5.3.2.5.3 PHTS Coolant Volume Monitoring

The total sodium inventory in the Primary Heat Transport System is monitored by the Data Handling and Display System, using sodium level detectors in the Reactor Vessel (see Section 7.5.3), in the Primary Sodium Pump Tank (see Section 7.5.2), and in the Auxiliary Liquid Metal Overflow Tank (see Section 9.3.5).

5.3.2.5.4 Critical Leaks

Reference 2 of Section 1.6 provides the results of the investigation of potential cracks in the PHTS and the capability to detect leakage from
45 such cracks.

5.3.2.5.5 Sensitivity and Operability Tests

The sodium-to-gas leak detection system, as described in Sec-
56 tion 7.5.5.1, is continuously self-monitoring for channel malfunction. Periodic maintenance procedures will provide for additional checking of operating characteristics. During installation and check-out, the correct electrical functioning of each contact and cable detector will be tested.

5.3.2.5.6 Confinement of Leaked Coolant

All cells and pipe chases within the Reactor Containment Building that house coolant (sodium) equipment and/or piping are operated with an inert atmosphere (nitrogen gas) containing a low oxygen concentration. Information concerning the design of these cells and pipeways is contained in Paragraph 3.A.1.1. The operation of these cells in an inerted atmosphere ensures that a minimum coolant fire will result from any spill or leak.

Inerting gas for individual cells or groups of cells is treated separately to prevent the spread of aerosol vapors beyond the confines of the area where the leakage or spill occurred. Separate cells are provided for redundant and/or safeguard equipment or systems to preclude the loss of that equipment or system in the event of a coolant leak or spill.

Provisions have been made for reducing any splash effect around coolant handling equipment or instrumentation by approximately locating such equipment within the cell to minimize such consequences.

The following design bases have been used to minimize the effects of leaks (including splash effects) from other components:

a) All piping and major components are suspended from the ceiling or side walls above the floor.

b) All piping and major components have an outer layer of metal sheathing over the insulation.

c) All instrumentation in general penetrates the top of the pipe and
56 is protected by metal guard boxes. 25

Provisions have been made for confining coolant leaking from major components of the primary system by containing these components within the guard vessels. (See Section 5.3.2.1.1.)

5.3.2.5.7 Primary/Intermediate Coolant Leakage

Primary to Intermediate coolant leakage is very unlikely due to the higher operating pressure of the intermediate system. The diffusion rate of radioactive sodium during normal design conditions through the tube walls is also zero. But, following IHX tube failures, IHX tube leaks and other faulted conditions in the IHX, back diffusion can be initiated. The rate of back diffusion determined from previous work and analysis is insignificant at design conditions. Detection of intermediate to primary coolant leakage is discussed in Section 7.5.5.2.

5.3.2.6 Reactor Coolant Volume System

The reactor coolant volume control is accomplished in that part of the Primary Sodium Storage and Processing System which consists of the primary sodium overflow vessel, primary sodium makeup pumps, primary sodium cold traps, and associated piping and valve. These components operate continuously during reactor operation to provide sodium-level control within the reactor vessel, to accommodate volumetric changes in primary sodium, and to limit oxygen and hydrogen impurities in primary sodium. The overflow heat exchanger, located in a bypass on the reactor makeup return line, is not normally operated. This component is only operated in the event that the overflow circuit
26 is used for removal of reactor decay heat. The Primary Sodium Storage and Processing System is shown on Figure 9.3-2 (included in Section 9.3) and is described in more detail in Section 9.3.

The overflow-makeup components are the only components within the Auxiliary Liquid Metal System which, during normal reactor operation, circulate reactor coolant. The overflow heat exchanger, used only for decay heat removal, is normally bypassed. All these components are located in inerted cells within the Reactor Containment Building (RCB). Fill and drain piping connect these components with sodium storage vessels, one inside and others outside of the RCB. All connecting lines to the normally operating components are isolated from the storage components by manually operated, locked-closed valves. Lines penetrating containment are isolated by additional, locked-closed valves, located outside the containment wall. Lines penetrating containment are anchored by steel plate members welded to the pipe and to pipe sleeves imbedded in the containment wall, whch are, in turn, welded to the inner containment liner.

All overflow-makeup components are Seismic Category I components, and all component supports are designed and analyzed to ensure component integrity and operability during and after the safe shutdown earthquake. The largest component, the overflow vessel, is hung from overhead structures, and laterally braced to provide both earthquake protection and to permit unrestricted thermal expansion during all operating conditions.

The method of controlling impurities and permissible levels of contamination in the reactor coolant is by use of the cold traps, which are part of the Primary Sodium Storage and Processing System, described in more detail in Section 9.3. Operation of the cold trap will maintain oxygen and hydrogen levels in primary sodium at or below 2 ppm and 0.2 ppm, respectively. Operation of the primary cold trap, in conjunction with operation of cold traps in the Intermediate Sodium Purification System, will maintain a tritium con-
46 | centration in the primary sodium at, or below 3 μ Ci/gm Na.

5.3.3 Design Evaluation

5.3.3.1 Analytical Methods and Data

The thermal and hydraulic characteristics of the reactor coolant system have been determined using a steady-state thermal/hydraulic CRBRP plant model. The analysis recognizes that uncertainties exist in such thermal hydraulic design characteristics such as heat transfer coefficients, pump characteristics, component and piping pressure losses. The model sizes the plant equipment for the worst expected conditions (nominal thermal hydraulic design characteristic values degraded two standard deviations) for a set of thermal hydraulic design points (turbine throttle conditions, feedwater temperature and pressure, sodium hot and cold leg temperatures and flows and steam generator recirculation ratio), the heat exchanger internal geometries, and other thermal and hydraulic design parameters. Actual expected operating conditions (sodium flows and temperatures, sodium pump heads, and steam/water conditions within the Steam Generator System) and deviations of these conditions are computed by a Monte Carlo technique employing a randomly selected set of heat transfer and other thermal/hydraulic characteristic values within their uncertainty ranges.

The Thermal/Hydraulic Design Conditions given in Table 5.3-1 had their origin in a system analysis which used a Monte Carlo technique for establishing these parametric values. The idea was that if a set of principal plant parameters were selected, and individual components such as the IHX, the steam generator modules, the main sodium pumps and the recirculating pumps were designed with their own margins to account for uncertainties in heat transfer coefficients, pressure drops and head-flow capabilities, the probability of actually operating the plant at the selected set of principal plant parameters would be small. For the final set of Thermal/Hydraulic Design Conditions, for example, which include a 995°F reactor outlet temperature to provide 900°F steam at the turbine throttle valve, the actual expected reactor inlet and outlet temperatures (based on nominal values for the various parameters which can affect the reactor outlet temperature required to produce 900°F at the turbine throttle) would be significantly lower than 730°F and 995°F, respectively. Furthermore, it is possible to randomly select values for the various parameters (such as heat transfer coefficients) within their uncertainty range and to use these values to compute what the various hot and cold leg temperatures would be for a set of cases. The results for the set of performance calculations could then be used to provide the variance from the nominal for each of the principal plant parameters. The utility of this analysis is 1) to provide expected reactor inlet and outlet temperatures for the purpose of predicting fuel burnup, and 2) to provide an estimate of what the "stretch power capability" may be for the NSSS. | 25

The method of approach was to use a trial set of thermal/hydraulic plant parameters. A "design case" was then computed based on pessimistic values for the various parameters affecting component sizes to determine heat transfer areas, etc. The plant was then analyzed to determine the expected conditions (based on nominal system flow resistances, nominal heat transfer coefficients), and the maximum value (nominal value plus two standard deviations, as determined by the statistical calculations of the plant performance). These values were then compared to the maximum allowable cladding mid-wall temperature to determine if criteria were satisfied. The Monte Carlo analysis involved 33 performance calculations for a given plant "design", each set of calculations having randomly selected values for 36 different thermal hydraulic design characteristics. The value for a particular characteristic was selected by a random number generator which effectively selected a value within the uncertainty range for that characteristic.

It should be recognized that this analysis was a priority analysis. It was used only as a tool for providing a rational basis for selecting a set of flows, temperatures, and pressure drops to be used as a basis for design. Once the values given in Table 5.3-1 were established, the Monte Carlo technique was no longer part of the design process, per se.

The thermal/hydraulic requirements for a particular permanent plant component as specified in equipment specifications are based on the Thermal/Hydraulic Design Conditions. For example, each IHX must be capable of transferring 325 MWt at the end of its 30 year life when the primary flow in each loop is 13.82×10^6 pounds per hour at an inlet temperature of 995°F, and the intermediate flow is 12.78×10^6 pounds per hour at an inlet temperature of 651°F. The component designer provides excess heat transfer area, internal primary flow bypass, maximum tube wall thickness and flow maldistribution. The primary pump design head flow requirement is based on the same loop flow (13.82×10^6 pounds per hour of 995°F sodium) at the maximum loop pressure drop of 160.7 psi. In other words, the Thermal/Hydraulic Design Conditions provide values for designing (sizing) components just as a balance-of-plant heat balance is used to size condensers, feedwater heaters and pumps. There are no confidence levels used for design parameters, per se. Individual component design margins are established on the basis of experience and applicable experimental data.

In addition to these values used to size components, a set of conditions used for the fatigue evaluation of components has been chosen. These values are given for the HTS components in Table 5.3-3. These "design" parameters reflect a power of 115% of rated power, with primary and intermediate hot leg temperatures at their structural design temperatures of 1015°F and 965°F respectively, and hot-to-cold-leg temperature differences of 300°F for the primary system and 340°F for the intermediate system. These temperature differences, while larger than those which will actually occur during steady state operation, were selected to provide conservatism in the transients provided for structural evaluation. For those events which result in down transients in cold leg components, different conditions are selected for conservatism. 25

Although equipment sizing calculations, plant transient analysis and safety evaluations are not based on expected operating conditions, predictions of fuel life are computed based on nominal parameters plus 2 standard deviations of the reactor inlet and outlet temperatures based on the above mentioned Monte Carlo technique.

31 It is to be emphasized that accident analysis, structural evaluation of permanent plant components and component sizing are based on a fixed set of parameters conservatively chosen and are not based on the expected temperatures and flows resulting from a statistical analysis. 25

The thermodynamic and physical properties of the fluids and materials are based on References 1 and 2.

5.3.3.1.1 Structural Evaluation Plan (SEP)

A structural evaluation plan for each segment of the Heat Transport System is not presently available. However, to facilitate the orderly review and verification of the stress report to be provided to the owner by the manufacturer, structural evaluation plans shall be prepared for each major Class 1 component of the primary heat transport system. The SEP for each component shall provide a description of the methods of analysis which the manufacturer contemplates using in various phases of the structural analysis. The plan shall indicate the degree to which the manufacturer anticipates using elastic, simplified inelastic and rigorous inelastic methods of analysis in design iterations and for demonstrating compliance with the requirements of RDT Standard F9-4T. The manufacturer shall identify any computer programs to be used and shall describe, or provide the basic theory of the program, and identify the assumptions involved in their use. The manufacturer shall also

57 5.3.3.1.2 <u>Stress Analysis Verification</u>

The SEP for each major Class 1 component of the primary heat transport system (i.e., primary piping, primary pump, IHX, guard vessel*, and primary check valve) will specify that a checklist be provided for identifying the anticipated analytical requirements for each component under normal, upset, emergency and faulted plant conditions. A sample of the structural design and analysis checklist to be used for each component is given in Figure 5.3-17.

Structural evaluation details of how the manufacturer of each primary system Class 1 component intends to demonstrate compliance with structural requirements shall be as described in the following categories.

<u>Duty Cycle</u>

To confirm the structural integrity of the PHTS equipment and piping, the sequence of the application of loads or cycles must be selected to provide the most conservative loading history of the applicable events. This is especially important if inelastic analysis methods are being used. Per the PHTS Equipment Specifications, the designer or manufacturer must determine the most conservative sequence of the application of the start-up to shutdown cycles using simplified analysis techniques subsequent to the detailed thermal transient and stress analysis. The equipment designer
57 in the ASME Code stress report must substantiate and document the load history used in the analysis of the equipment for the most severe cases. The application of material properties to be used in structural analysis of the PHTS equipment and piping is in accordance with ASME Code Case 1592 and RDT Standard F9-4T, and this application will give conservative results; sensitivity studies are not required. For example, the standard requires average yield stress properties to be used when calculating residual stress rupture damage. 19

<u>Failure Modes</u>

The manufacturer shall identify the locations and failure modes which are expected to be dominant and the load conditions (pressure, thermal, seismic, etc.) associated therewith. Any failure modes not identified in the
57 ASME Code Section III, Code Case 1592 and/or RDT Standard F9-4T to be guarded against for specified loads shall be identified.

*Guard vessels, although classified as ASME III, Class 2, will be designed and analyzed as ASME Class 1 components but will not be code stamped.

Analysis of any component which has an operating temperature above 800°F must account for the time dependent materials properties. Components in the primary heat transport system which operate above 800°F are the hot leg piping, pump, and IHX. 20

Screening Analysis

Where appropriate the manufacturer shall outline a screening analysis to facilitate the early stages of design, and for the purpose of ordering long lead items such as forgings. A screening analysis is also useful in the performance of early design iterations and identification of locations in the structure requiring detailed inelastic analysis. It is often possible to establish conservative upper bounds on strains using linear elastic and simplified inelastic methods of analysis.

A typical screening analysis may consist of an initial linear elastic analysis with one or more design iterations, and comparison with environmentally adjusted design limits to scope the effect of inelastic material behavior. If design limits cannot be met by elastic calculations, simplified inelastic calculations may be made for comparison with design limits. Simplified inelastic analysis may take the form of approximate methods such as one dimensional approximation of two and three dimensional geometries, or simplified material models. A sample flow chart for a screening analysis is shown in Figure 5.3-18.

Detailed Inelastic Analysis

The extent of inelastic analysis required will generally not be known until completion of screening analysis. However, the manufacturers of the individual primary system components shall identify, in the respective SEP's, regions where initial detailed inelastic analysis may be required and the extent to which the analysis may realistically be anticipated.

The manufacturer shall modify the SEP to include the extent of inelastic analysis required to satisfy design criteria. Detailed inelastic analysis shall generally be performed in accordance with the procedures in
7 RDT Standard F9-5T, Sept. 1974, Section 4. Alternate procedures proposed for use shall be included in the SEP. The plan and schedule for furnishing substantiation and justification of the alternate techniques and methods shall be given.

Support Test Program for Component Design Analysis

The manufacturer of each Class 1 primary system component shall identify structural tests required in support of design analysis in two categories: those which appear necessary from the outset, and those which may be required depending on the outcome of the screening analysis. Although test programs falling in the latter category may be highly speculative, it is important that some reasonable estimate be included in the SEP in order to scope the magnitude of the test program. Upon completion of the screening analysis, the total test program shall be identified and submitted in the modified SEP.

In addition to structural tests, any basic material property tests required to obtain data in support of design analysis shall be identified. Such data will normally be provided as references in the design specification or design code for the component, but certain analysis procedures may require data not provided, or data may be required for specific heats of material to evaluate test results. Where additional material property data are required, the manufacturer shall identify the types to be performed, number of samples desired, and specific data required.

Environmental Effects

Environmental effects shall be accounted for in all structural and functional evaluations.

The environmental factors which may modify material behavior include:

a. The environmental fluid (e.g., sodium, inert gas, air, and vacuum).

b. The exposure to elevated temperature for long time durations.

c. Irradiation.

The possible effects of these factors which may modify material behavior include:

a. Aging

b. Irradiation induced loss of ductility

c. Loss of alloying elements (including N, C, B, M_n) on both local and an overall basis

d. Loss of surface material due to erosion, corrosion, or oxidation

e. Alteration of the microstructure, especially at the surface of the component.

Guidance for the treatment of these effects is included in Sec-
27 tion 5.3.3.10 of this document and in RDT Standard F9-5T, Sept. 1974, Section 7. The manufacturer shall indicate how he intends to account for these environmental effects (e.g., whether in accordance with procedures provided or through alternate procedures).

5.3.3.1.3 <u>Compliance with Code Requirements</u>

All systems and components of the primary heat transport system under the jurisdiction of the ASME Code, Section III for Nuclear Power Plant Components shall be designed to accommodate the load combination prescribed therein without producing total combined stresses in excess of those allowed by the Code. Additional loading combinations may be set forth in the project specifications. No component of an individual loading condition shall be included which would render the combination non-conservative. When a particular loading condition does not apply to a system or component, that loading condition shall be deleted from the load combination. Transient loadings shall be included as required by the Code. For elevated temperatures Code Case 1592 supplemented by RDT Standard F9-4T, Jan. 1976, will apply.
27

As specified by Regulatory Guide 1.48, the primary system ASME Class 1 or Seismic Category I components shall be designed to withstand the concurrent loadings associated with the upset plant condition and the vibra-
46 tory motion of 50 percent of the Safe Shutdown Earthquake (SSE). The design limits for this case are specified in NB-3223 and NB-3654 of the ASME Code for vessels and piping. Also per Regulatory Guide 1.48, the design limits specified in NB-3225 and NB-3656 of the ASME Code for vessels and piping, respectively, shall not be exceeded when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condtion.

For components at elevated temperature, the design stress limits as specified by paragraph 3223, "Normal and Upset Conditions" and by paragraph 3225, "Faulted Conditions" of Code Case 1592 shall be used for the OBE load condition and the SSE load condition, respectively. The strain and creep fatigue damage resulting from the OBE load conditions shall be included in the limits specified by paragraph 5.0 of the Code Case.

For Code Class 2 components in the primary heat transport system, the design limits shall be as specified in Section 3.9.2, in accord with the intent of Regulatory Guide 1.48.

5.3.3.1.4 <u>Category I Seismic Design</u>

The main components and piping of the primary heat transport system shall all be designed as Category I Seismic structures. A list of components and supporting structures for the primary heat transport system that are to be designed to comply with Category I seismic requirements is shown in Table 5.3-12. In order to get the proper coupling of the dynamic interaction between components and piping, the components will be modeled and included in the seismic analysis of the piping loops.

Diagrams of the seismic models used for the components and piping loops will be shown in the FSAR. Resulting stress levels for the OBE and SSE, combined with all other loadings, will be shown in the FSAR at all points in the piping loops, including the points at which there are large changes in flexibility in the system.

26 5.3.3.1.5 <u>Analytical Methods for Piping Pumps, Valves, and Heat Exchangers</u>

The Intermediate Heat Exchanger, Primary Sodium Pump and Check Valve shall be constructed as Class I components in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant
27 Components and the supplementary requirements of RDT E15-2NB-T, Oct. 1975.

The construction of parts and components for design temperatures exceeding 800°F shall be in accordance with Code Cases 1592-1596 and the supplemental requirements of RDT F9-4T. 33

Those parts of the components outside the limits of Code jurisdiction shall be designed by methods equivalent to those required by the Code. The supplier shall submit such design rules, methods or standards to the purchaser for approval prior to use.

The designer is required to perform thermal stress analyses using the CRBRP plant transients applicable to the component. Initially it is required that the designer perform elastic and simplified inelastic analyses to check the basic structural criteria and to assess the capability of the equipment in meeting the transient requirements and to provide these analyses and assessments to the purchaser early in the design process. If the limits cannot be satisfied with the more conservative elastic methods, inelastic methods may have to be considered, pending review and approval by 33 the purchaser.

The designer is required to perform seismic analysis using the methods discussed in Section 3.7. The components of the primary heat transport system shall be analyzed by a detailed dynamic analysis using either time history methods or the response spectrum method. Other methods of dynamic analysis which provide an acceptable solution may be used but the justifications and procedures shall be submitted to the purchaser for review and approval. A simplified analysis such as that based on an equivalent static load method may be used if it can be demonstrated that the simplified method provides adequate conservatism.

A tentative list of the types of computer programs that may be used in the evaluation of stresses in the PHTS pumps, valves and heat exchangers is given below.

SAP IV STARDYNE 33

CREEP-PLAST

MARC-CDC

44

ANSYS

44 TFEAFTS

TAP-A

A general description of the application of each is given in Appendix A. The computer programs utilized will be verified to assure their accuracy and applicability.

The following paragraphs discuss plans to be used in analyzing areas of the pump and heat exchanger where the general plan outlined above is not expected to be adequate to demonstrate component integrity.

Intermediate Heat Exchanger

The IHX vendor will perform hydraulic flow model testing to verify the performance characteristics of the IHX. The objectives of the IHX hydraulic model test are to:

a. Establish the uniformity of the flow distribution in the IHX to assure predictable heat transfer performance and flow stability.

b. Determine the overall pressure loss characteristics.

c. Demonstrate the absence of damaging tube vibrations.

The tests under (a) and (b) shall cover the flow conditions between 100% and 7-1/2% of the hydraulic design flow rate. The vibration tests shall cover a range from zero flow to at least 120% of the hydraulic design flow rate. The upper limit of flow for the tube vibration test shall be recommended by the supplier for the purchaser's approval.

The IHX and all its parts shall be designed so that they will not be damaged or caused to malfunction either by internally generated vibrations, such as flow-induced vibrations, or by mechanical vibrations, such as pump vibrations, check valve slam vibration, or vibrations or shocks caused by shipping.

Baffles and tube support plates, tie rods, impingement plates, etc., shall be provided so that natural frequencies of all unsupported tube spans are at least 50 percent higher than hydrodynamically generated frequencies in the flow range from zero flow up to 100 percent of the hydraulic design flow rate. Provisions shall be made to prevent damaging vibrations in areas where localized fluid velocities are high.

Engineering vibration analyses of the tube bundle structural design covering peak velocities over the range of flows from zero to 100 percent of deisgn flow rate shall be included in the Design Report. The complete analytical method shall be described in detail giving all references and assumptions in an orderly way to facilitate verification. The analyses shall show that the maximum amplitude of tube vibration will not exceed 25 percent of the nominal distance between the outer surfaces of adjacent tubes.

The vibration analysis shall cover vibrations and shock during shipment. Dynamic loadings for use as a reference basis shall be recommended by the supplier for purchaser approval. Complete substantiation of recommended loadings shall be submitted by the supplier.

The dominant failure mode of those portions of the CRBRP-IHX, which operate in the creep regime during normal 100% power operation, is creep-fatigue with creep damage the major contributor to the cumulative damage sum. The creep damage is a result of residual stresses which are set up by the upset and emergency condition thermal transients. In the case of the primary inlet nozzle, intermediate outlet nozzle and the upper portion of the primary shell, the calculated residual stresses are increased by the OBE loads which are superimposed on the peak thermal stresses of upset transient 13U. The cyclic portion of the OBE load does not contribute significant damage, since fatigue is only a very small portion of the accumulated damage. In other portions of the CRBRP-IHX, such as the upper tubesheet, upper portion of tubes, and intermediate channel complex, which are also creep fatigue limited, the OBE loads are inconsequential. The remainder of the CRBRP-IHX is fatigue limited, but as in the case of the previously mentioned portions, fatigue damage is small.

Two areas of the CRBRP-IHX, in which the seismic loads in themselves must be considered, are the hanger and the upper portion of the primary shell. Both these areas experience larger overturning moments due to the horizontal portion of the seismic motion. The OBE is controlling and the limit is time
26 independent load controlled buckling.

For those portions of the CRBRP-IHX, which operate below the creep regime during normal 100% power operation, the complete design process will be carried out using elastic analysis methods and design criteria. For the remainder of the CRBRP-IHX elastic and simplified inelastic, analysis is being used in the design phase and fabrication release phase. The simplified inelastic analysis involves the use of infinite, thick walled cylinder ; finite thin shell cylinder; and coarse mesh two dimensional MARC analysis. The seismic loads used in these analyses were developed using response spectrum methods as contained in the ANSYS program. The complete CRBRP-IHX was modeled using continuum and shell-type axisymmetric finite elements. The nonaxisymmetric motion which results from the horizontal seismic motion was evaluated using the first harmonic of the Fourier series expansion of the displacement function for axisymmetric structures subject to nonaxisymmetric loads. While the seismic loads are nonaxisymmetric, the simplified inelastic analyses are axisymmetric. This necessitated taking the worst case portion of the seismic loads and making it axisymmetric.

The final evaluation of the critical areas of the CRBRP-IHX will be carried out using either detailed or coarse mesh two dimensional inelastic MARC analyses if found necessary. The areas involved are:

i. Upper Tubesheet
ii. Inner Channel Junction
iii. Hanger/Shell Junction
iv. Primary Inlet Nozzle
v. Intermediate Outlet Nozzle
vi. Upper Portion of Primary Shell

Component degradation will be accounted for using the methods contained
26 in Code Case 1592-1 of the ASME Boiler & Pressure Vessel Code.

In evaluating the structural adequacy of the IHX with respect to the design basis Sodium Water Reaction, the dynamic nature of the intermediate sodium pressure history is being accounted for by using dynamic load factors. The factor will be applied to the maximum intermediate pressure which in turn is used to determine the pressure-induced primary stresses. These primary stresses are limited by the emergency condition allowables of Code Case 1592. Paragraph 3224, as modified by RDT F9-4T, Jan. 1976. The fatigue damage associated with the cyclic nature of the pressure history will be accounted for per Paragraph T-1400 of Code Case 1592.

This event consists of an instantaneous rupture of an evaporator or superheater tube followed by the rupture of 6 adjacent tubes, which result in rupture disk actuation, automatic isolation and blowdown of all evaporator modules and the superheater in the affected loop, and manual activation of the sodium rapid dump system. In addition, a trip of the reactor, turbine, and sodium pumps occurs. The intermediate sodium system
experiences a pressure transient resulting from the reaction. This event is 16

classified as faulted for the affected steam generator module. Differential pressure between the primary and intermediate sides during this event is conservatively evaluated by assuming that the primary side pressure is that resulting from pony motor speed (approximately 6 psig). For the rest of the loop, the occurrence is classified as an emergency event.

For the unaffected loops, the event is similar to the reactor trip from full power. Decay heat removal is maintained throughout the two remaining loops. The transient responses of temperature, flow and pressure on both the primary and intermediate side of the IHX in the affected loop are presented in Figures 5.3-18A through 5.3-18G. Particular attention is directed to Figures 5.3-18F and G which show intermediate side short term and long term pressure effects.

In evaluating the structural adequacy of the IHX, with respect to the check valve slam, the dynamic nature of the primary sodium pressure history is being accounted for by using dynamic load factors. The factor will be applied to the maximum primary pressure, which in turn, is used to determine the pressure-induced primary stresses. These primary stresses are limited by the emergency condition allowables of Code Case 1592, Paragraph 3224, as modified by RDT F9-4T. The fatigue damage associated with the cyclic nature of the pressure history is accounted for per Paragraph T-1400 of Code Case 1592. The description of the pressure pulses for the sodium-water reaction and check valve closure is included in the equipment specification. The curves define the amplitudes, duration and number of cycles.

Rapid check valve closure can only occur as a result of primary pump mechanical failure. The event involves a postulated instantaneous stoppage of the impeller of one primary pump, while the system is operating at 100% power. The failure may be a seizure or breakage of the shaft or impeller. Primary system sodium flow in the affected loop decreases rapidly to zero as the pumps in the unaffected loops seat the check valve (thereby causing a rapid check valve closure or slam). A reactor trip will be initiated by the primary intermediate flow ratio subsystem. Sodium flow in the intermediate circuit of the affected loop decays as in a reactor trip from full power, modified by changes in natural circulation head. The event is characterized by a down transient in the hot leg of the intermediate circuit of the affected loop. The transient responses of temperature, flow and pressure on both the primary and intermediate side of the IHX in the affected loop are presented in Figures 5.3-18H through 5.3-18M. Particular attention is directed to Figure 5.3-18J which shows primary pressure effects.

Both the sodium water reaction and check valve closure events are classified as emergency events for the IHX. As such, the IHX designer is required to determine which of the six emergency events is most severe to the IHX. The selected event is then applied with a periodicity of two consecutive occurrences during the first three years of operation, and thereafter five times over the remaining 27 years (or once every six year period). If vendor analysis indicate either as the most severe event, the occurrence of the two consecutive events will be moved to the most stringent time in the life for the event to occur. Preliminary analysis indicates that damage from either of these events will be insignificant.

Pump

Inelastic analyses of the pumps was required to demonstrate conformance with the ASME Code. Paragraph 4 of RDT Standard F9-5T, Sept. 1974 gives a description of acceptable methods for time-independent elastic-plastic analysis and time-dependent creep analysis. Some of the computer programs listed above have inelastic capabilities, and will be used where applicable.

For the purposes of loads and analysis the pump R-Spec divides the pump into four areas. These are: Subcomponent 1 which consists of the pump tank, Subcomponent 2 which is the upper inner structure including the pressure bulkhead, Subcomponent 3 which is the rotating machinery and Subcomponent 4 which is the static hydraulics.

Subcomponent 1 is designed to the ASME Boiler and Pressure Vessel Code Section III, Subsection NB Class 1 and Code Case 1592 where applicable. The cone and cylinder are designed mainly by dynamic stiffness requirements. These include seismic loads and the necessity of keeping the natural frequency of the structure above the operating speed of the impeller. SAP IV and the "NASTRAN" computer codes were used for this analysis. The analysis has been qualified by comparing the results of one analysis against the other. The sphere sealing ring and cone-sphere support ring are designed by sealing ring leakage which requires elastic response during normal and upset conditions. A failure will reduce pump efficiency below plant criteria. These areas have been analyzed by 3D global analysis using NASTRAN. The nozzles are designed by pressure, pipe nozzle loads, and thermal transients. The failure modes associated are creep and creep fatigue. 2D elastic analysis is required. The design is being made with sufficient space for thermal baffles and liners to keep it elastic as much as is possible. But it may be necessary to qualify it using simple inelastic analysis. Hydraulic leakage test data has been obtained which determined the relation of sealing gap to leakage rate.

Subcomponent 2 conforms to the same Code requirements as Subcomponent 1. The upper closure plate and radiation shield are designed by the design pressure and temperature requirements. Elastic failure is the predominant mode. The heat shield has steady state thermal gradients which are determined by a 2D axisymmetric model and stresses are calculated with a 2D stress model. The motor stand has been designed by the stiffness requirements of the motor and seismic loads. The principle failure mode is excess vibration leading to fatigue failures.

Subcomponent 3 can be removed and inspected after an emergency or faulted event and repaired before the plant is placed in service again. Therefore, this section was designed and analyzed to the ASME Boiler and Pressure-Vessel Code, Section III, Subsection NB for Class 1 Components and Code Case 1592 where applicable. However for emergency events Code Case 1592 is used and the design rules for load controlled stresses (Section 3227) applies. Strain deformation and fatigue analysis need only be performed up to the emergency event and the limits will apply only to the pumps ability to operate at pony motor speed after the event. This area has been designed by critical frequency requirements, inertial loads, torque and thermal transients. It was analyzed with a 2D axisymmetric model. The loads caused by bearing
59 misalignment were accounted for. A general 1/2 scale model hydraulic

performance test was run using water as the pumped fluid. This test provided information on the pump NPSH and internal leakage flows. Inelastic analysis of the upper journal impeller weld region of the rotating assembly using ANSYS, was required to show adequate ratchetting strain margins for various upset events.

Subcomponent 4 consists of the lower removable region of the pump inner structure. It was analyzed to the same Code rules as Subcomponent 3. The principle loads are thermal transients, hydraulic pressure, containment of a failed impeller, reaction loads against the hydraulic machinery due to deformation of the sphere during the thermal transients and bearing loads during asymmetrical heating. Principle failure modes associated are elastic failure, creep and creep fatigue. The hydraulic casting has been analyzed by a 3D global model using NASTRAN. The bearings are fed directly from the pump discharge so they are exposed to thermal transients. They have been analyzed with a 2D axisymmetric analysis to develop loads and stresses. An axisymmetric 2D model was used to calculate the stresses in the static shroud around the impeller. Inelastic analysis was required in the bearing support region using MARC and ANSYS.

Piping

The incontainment sodium piping shall be designed and analyzed to the Class 1 requirements of the ASME Code, Section III and Code Case 1592. The piping will be designed to assure that piping stresses, strains and deformations are within the applicable Code criteria and system functional limits. The analyses to satisfy these limits shall reflect both time-independent and time-dependent material properties and structural behavior (elastic and inelastic) by considering all of the relevant modes of failure listed below:

1. Ductile rupture from short-term loadings
2. Creep rupture from long-term loadings
3. Creep-fatigue failure
4. Gross distortion due to incremental collapse and ratchetting
5. Loss of function due to excessive deformation
6. Buckling due to short-term loadings
7. Creep buckling due to long-term loadings

To perform the structural evaluation of the primary piping, the loadings on the piping loop that result from the usual load effects including internal pressure, deadweight, support movements, thermal expansion, seismic, and thermal temperature gradients must be obtained at particular locations in the piping system (usually at piping components such as elbows, tees, reducers, girth welds, etc.).

Formulae given in the ASME Code, Section III are used to determine stresses throughout the piping resulting from internal and external pressure.

General purpose finite element codes are used to perform piping system flexibility analyses which determine the forces and moments acting on the piping system due to various loading conditions. Even though there are no specific guidelines for modeling runs of pipe using pipe or beam elements, most codes check the assembled model for disparities in the assembled stiffness matrix such as large stiffness differences between elements, small stiffnesses or lack of symmetry. Typically such codes can handle a large range of stiffness values. One commonly used code checks the ratio of the maximum to minimum stiffnesses and prints a warning message if this ratio exceeds 1×10^8.

Stresses that result from these loads will be considered in evaluating the failure modes of the piping and piping material.

The types of analysis required to verify the design of the piping will include elastic, simplified inelastic and detailed inelastic. Simplified inelastic and detailed inelastic methods that are to be used will conform to the requirements of RDT Standard F9-4T and the guidelines of RDT Standard F9-5T.

Degradation of material properties over the lifetime of the piping in accordance with the requirements identified in the equipment specification will be accounted for.

26 No structural verification testing is planned for the main sodium piping.

5.3.3.1.6 Analytical Methods for Evaluation of Pump Speed and Bearing Integrity

The primary pump shaft is supported at the impeller end by a sodium bearing. Hard facing material is applied to the bearing and journal surfaces to provide
59 wear-resistant surfaces for startup and shutdown transients during journal
lift-off and touchdown. 33

The approach to reduce or eliminate large thermal stresses in the sodium
bearings is to provide two-sided exposure to the thermal transients over most
of the bearing/journal supports and to use baffling to isolate the bearings
from the large enclosed sodium pools. The bearing supports were analyzed for
59 temperature differences between the hydraulic assembly casting and the bearing
supports during thermal transients, including thermal shear loads on bolts. 33

Because the bearing hard face surfaces are to be protected from the rapid
thermal transients, a two-step approach to analysis was taken. The first step
was to determine the transient temperature distribution in the structure.
These temperature distributions were then used in the second step to establish
59 the thermally induced loads.

59 Pump structural lateral and vertical natural frequency and critical speed
analyses were performed using SAP IV and STARDYNE computer programs. The shaft
torsional natural frequency and critical speed analyses was performed using SAP
59 IV. These analyses included calculation of the natural frequency by eigen-
value extraction from the dynamical matrix based on the standard finite-element 33
stiffness formulation. In addition, time history and response spectrum dynamic
response calculations were performed.

59

5.3.3.1.7 Operation of Active Valves Under Transient Loadings

There are no Active valves in the PHTS.

44

19

5.3.3.1.8 Analytical Method for Component Supports (Vessels, Piping, Pumps and Valves)

In accordance with the ASME Code, component supports will have the same code classification as the components they support. Design of each component support will comply with the ASME Section III, paragraph NF, design rules corresponding to the component support classification. In order to provide assurance that the component support stresses comply with limits specified in paragraph 5.3.3.1.2, analysis of each component support will be performed. The analytical techniques and applicable computer codes discussed in paragraph 5.3.3.1.5 and the loading combinations prescribed in Section 3.7 will apply to detailed analysis of support components. The stress limits on the supports for the normal, upset, emergency and faulted conditions shall be as those listed in Fig. NF-3221-1 of section NF of ASME Code Section III. If the component support design temperatures exceed those for which allowable stress values are given by Section III, the rules covered in Code 17
Case 1592 and RDT F9-41 will be satisfied. By design, the check valve free standing disc-seat assembly is isolated from the body and supports so that valve function would not be impaired even by gross deflections of the piping and valve support system. 13

5.3.3.2 Natural Circulation

Although the HTS normally transfers reactor decay heat to the steam generator system with forced sodium circulation in at least one of the three main heat transport loops using pony motor drives on the main sodium pumps, the PHTS, IHTS and SGS have been designed to assure adequate
24 decay heat removal by natural circulation.

5.3.3.2.1 Preliminary Assessment 25

A preliminary assessment of the CRBRP natural circulation decay heat removal capability has been made for several cases including: heat removal through three loops following three loop rated power operation, two loops following three loop rated power operation, and two and one
27 loops following 2/3 rated power operation on two loops. (See Reference 37). The principal case is the first of these cases (3 to 3). It has been analyzed on the basis of the following event description.

From initial conditions of three loop full power operation,
44 complete loss of all electrical power to the main motors and pony motors is assumed, initiating a PPS trip in 0.5 seconds. Immediately upon loss
44 of electrical power, the sodium pumps coast down and stop (in approximately 55 seconds) when the normalized speed has decreased to approximately 1% and cannot overcome internal friction. Flow in all sodium loops is then maintained by the natural circulation thermal driving heads. Auxiliary feedwater flow at 70°F is available from the auxiliary feedwater portion of the SGAHRS in approximately 30 seconds. The turbine driven auxiliary feed pump takes suction from the protected water storage tank to maintain drum levels during the transient. Final decay heat rejection is through SGAHRS via the superheater outlet and steam drum vent valves and the exhaust from the auxiliary feed pump turbine.

The reactor temperatures during the period of concern (the initial two minutes of the transient) are relatively insensitive to the particular IHTS and SGS recirculation flow transients. The emphasis of the assessment was placed, therefore, on factors which directly affect the primary flow transient, the temperatures resulting from this flow transient and the thermal
51 hydraulic behavior in the fuel and blanket regions. To assure that the natural circulation transients predicted by the analysis were conservative, "worst case" conditions were used for those factors which would affect the
51 maximum cladding temperature in the blanket and fuel assemblies. These temperatures are maximized by a combination of minimum flow (following the trip), maximum reactor inlet temperatures, maximum decay power and conservative calculational assumptions.

Selected parameters for the three to three loop case are shown in
51 Figures 5.3-18P and Q. Figure 5.3-18P shows HTS cold leg temperatures as noted to illustrate the transport delays characteristic of the plant's operation on natural circulation. The fluctuations in cold leg temperatures and the movement of the cold sodium through the piping system results in the flow fluctuations seen in Figure 5.3-18Q which in turn yield the fluctuations seen in the temperature transients between 600 and 850 seconds after scram. Results of preliminary transient analyslis for the heterogeneous core scheme show that the peak core temperatures for the 3 to 3 loop natural circulation
51 24 event will not exceed allowable design values.

Analysis of the other cases indicated that the temperature transients in the fuel and radial blankets would not be significantly different for the first 600 seconds. Beyond this time, the magnitude of the temperatures for any of the cases are not high enough to be of concern. 32

Based on analysis of the set of events mentioned above, it has been concluded that:

- The arrangement of components within the plant (i.e., the relative elevations of the IHX with respect to the core, the steam generator modules with respect to the IHX and the steam drum with respect to the evaporator modules) will provide thermal driving heads necessary to promote adequate flows in the natural circulation mode.

- The specified pump coastdown characteristics satisfy the minimum flow decay requirements immediately following a
24 plant trip.

The analyses supporting these conclusions were limited in scope to five specific cases analyzed for a maximum of about 17 minutes. The trends shown at 17 minutes (steady flow, decreasing cladding temperature, etc.) taken with the various areas of conservatism indicate that the
27 conclusions above, for the long term, are justified.

5.3.3.2.2 Verification of Preliminary Assessment

The verification of natural circulation decay heat removal capability of CRBRP will be based on analysis using data from component tests.

The system resistance at very low flows (~3% of full flow) will be verified flow tests on key components. That is, the specified requirements for ΔP as a function of flow will be verified by test on an individual component basis. For example, the specified requirement that the pressure drop of the check valve at a flow of 670 gpm of 730°F sodium (2% of design flow) shall be $\leq$0.03 psi, and the requirement that the locked pump rotor pressure drop in feet of sodium, shall be $\leq$ 250 x the normalized flow squared will both be verified by test prior to installation. The reactor pressure drops will be determined by out-of-reactor tests on individual components. Based on these results, the actual system resistance curve (head vs. flow) can be constructed for flows in the natural circulation range (2.5 to 3.5% of full flow). Tests will also be performed on simulated fuel and radial blanket assemblies to ascertain the intra-assembly buoyancy
31 effects. 25

5.3.3.3 Pump Characteristics

The pump characteristic curves of head, efficiency, brake horsepower, and net positive suction head (NPSH) versus volume flow at constant
33 speed are plotted in Figure 5.3-19. The information is a first approximation

and is based on pump vendor test data from similar pump designs. For every point on the head-capacity curve when the pump speed is changed, the capacity, head, and brake horsepower will vary according to the standard pump affinity laws.

The rated hydraulic torque at the primary pump design speed of 1116 RPM is 15,096 ft-lbs. This can be expressed as a function of flow and speed by the following equation:

$$\bar{T}_{Hyd}/\bar{N}^2 = 0.584 + 0.033\ (\bar{Q}/\bar{N}) + 0.553\ (\bar{Q}/\bar{N})^2$$
$$-0.152\ (\bar{Q}/\bar{N})^3 - 0.019\ (\bar{Q}/\bar{N})^4$$

where,

$\bar{T}_{Hyd} = T_{Hyd\ Actual}/T_{Hyd\ Rated}$; $T_{Hyd\ Rated} = 15{,}096$ ft-lbs

$\bar{N} = N_{Actual}/N_{Design}$; $N_{Design} = 1116$ RPM

$\bar{Q} = Q_{Actual}/Q_{Design}$; $Q_{Design} = 33{,}700$ gpm

The rated frictional primary pump torque at the primary pump design speed is 5524 ft-lbs.

The primary pump inertia is 3086 lb-ft^2 including a sodium inertia of 311 lb-ft^2. 34

The flow head for zero speed (locked rotor) is plotted in Figure 5.3-20. The percentage of peak head is plotted versus the percentage of peak reverse flow in the second quadrant. The percertage peak head is plotted versus the percentage of peak forward flow in the fourth quadrant.

The following expressions for pump locked rotor impedance are based on Figure 5.3-20.

Forward flow $H = 250\bar{\omega}^2$

43 Reverse flow $H = 320\,\bar{\omega}^2$

where,

H = pressure drop through pump, feet

$\bar{\omega}$ = normalized flow, i.e., the ratio of actual system flow to design flow.

The first expression represents the internal pump impedance to flow in the positive direction when the impeller is locked. This requirement is established to assure that the pressure loss through the pump will not provide an unacceptable impedance to flow when the system is removing reactor decay heat under natural circulation.

The pump drive system will incorporate separately powered pony motors which have the advantage of being able to set speeds to obtain desired flows during hot functional testing. Basically, the requirement is to provide approximately 7.5 percent flow (2528 gpm) at a head of about 4.0 feet of sodium. The system impedance curves at <10 percent flow for three-loop operation are shown in Figure 5.3-21. The head versus flow for one-loop operating and reverse flow in the other two loops is also shown. The curve specifies that the required pony motor flow operating range is along the pump characteristic curve from zero flow to the intersection with the minimum one-loop system impedance curve. Operating requirements specify stable pony motor operation at shutoff conditions.

Pump coastdown to pony motor speed and an automatic changeover from main motor to pony motor takes place following reactor trip. Flow coastdown characteristics following a primary pump trip are shown in Figure 5.3-22. The curve shows mass flow vs. time following pump trip to pony motor. The pony motor is energized at all times during operation and therefore is effective the instant the pump coasts down to pony motor speed. This will occur at approximately 30 seconds after pump trip.

The sodium temperature will reduce with time resulting in an increase in sodium density and mass flow pumped by the pony motor. This increase tends to be compensated by the loss in thermal driving head in the system. The resultant increase in flow is small particularly during the time span shown on Figure 5.3-22. For simplification in analysis a constant flow rate for pony motor operation is shown on the graph, since this assumption has little effect on the overall system performance.

There will be no vortexing or gas entrainment at flows in the operating range of the PHTS pumps. Entrainment of argon gas in the inlet sodium is minimized by a vortex suppressor plate in the reactor vessel and a low exit velocity

(11 ft/sec) at design flow from the reactor vessel. Gas entrainment within the pump is minimized by designing and testing, to ensure that all pump parts which need to be submerged during operation at pony motor speed or restart to pony motor speed are located below the minimum sodium level.

The pump standpipe bubbler also provides a vent for gas de-entrained from the PHTS. The maximum gas flow that the standpipe can accommodate without flow instability corresponds to an entrained gas volume fraction of 3×10^{-4}. This volume fraction has been shown to have no impact on pump performance or heat transfer in the IHX. Outlet plenum tests which give expected gas entrainment levels are discussed in Section 4.4.4.1.

The maximum oxygen content in the primary system sodium is specified to be ≤ 2 ppm at 800°F or above and ≤ 5 ppm below 800°F. This level of sodium impurity will not affect the pump operating characteristics.

The biological shielding for the PHTS sodium pumps as well as the pump assemblies is designed to withstand the loadings associated with the Safe Shutdown Earthquake (SSE) and the transient overpressures for extremely unlikely plant conditions. The analyses required to demonstrate this treated the PHTS pump as a Class I component in accordance with the rules of the ASME Code Section III and modifying RDT Standards.

The biological shielding for the PHTS sodium pumps is provided by (1) an annular shield tank which surrounds the pump shaft, (2) the pump shaft itself which is designed to provide an integral part of the shield requirements, (3) the pump support structure which is part of the operating floor, and (4) special precautions to preclude streaming along the instrumentation penetrations. The annular shield assembly is integral with the top closure flange of the pump pressure boundary containment vessel which is designed in accordance with the ASME Code for Class 1 nuclear components. The pump shaft supporting assembly and the annular shield structural assembly are supported on the pump tank flange which is mounted on a pump support ledge designed into the operating floor pump motor well. The design of this joint provides for the dual function of resisting static and dynamic loads and provides a seal at the boundary between the pump atmosphere and the RCB atmosphere that consists of a double metallic "o" ring seal with argon purge gas in the annulus between rings.

For the PHTS pump SSE seismic analysis, a 2% damping value was used. The SSE loadings were considered to occur in conjunction with a plant trip. Following the SSE, the Intermediate Heat Transport System, Steam Generator System, and Steam Generator Auxiliary Heat Removal System must provide for removal of stored and decay heat. The primary pump is designed to maintain pony motor flow without loss of structural integrity after the SSE. Computer programs, such as SAP IV and ANSYS, will be utilized to perform seismic analyses on the primary pumps. Descriptions of these computer programs can be found in Appendix A.

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5.3.3.4 Valve Characteristics

The essential characteristics and features for the check valve design to satisfy the system operating requirements are included in Table 5.3-13.

The CRBRP Cold Leg Check Valve (CLCV) is hydraulically similar to the FFTF 16" check valve. As noted in Reference 62, extensive tests have been performed on the FFTF valve to confirm its hydraulic performance. In operation at FFTF, no problems due to gas accumulation in the CLCV have been identified. Secondly, tests performed on the FFTF hot leg isolation valve verified the self-venting characteristics (that any accumulated gas will be swept out) of the main valve body of both valves. Since the valve bodies of both CRBRP and FFTF CLCV's and the FFTF hot leg isolation valve are geometrically equivalent, the results of these tests are applicable to the CRBRP CLCV. Air deliberately introduced into the valve was quickly entrained and removed by the turbulent fluid at pipe flow velocities greater than 7 fps, which correspond to a 2 fps (average) velocity in the valve. The experimeters noted that, "This is the approximate limiting value for air removal from pipelines as established by previous research" (Reference 63). Under full flow conditions the flow velocity in the CLCV inlet pipe is about 25 fps and about 6 fps (average) in the valve body. At 40 percent flow the velocity in the valve body is still above the 2 fps level, and at this reduced flow rate there is little likelihood for any gas entrainment at the principal gas entrainment site (the reactor vessel outlet plenum). Thus, no gas accumulation is expected in the main body of the CLCV.

Although no mechanism has been identified which would allow the de-entrainment of gas in the valve, an evaluation has been performed with the following conservative assumptions. The dome on the top of the check valve is initially full of gas at the conditions consistent with the thermal hydraulic design condition, the pumps are tripped, and the pressure in the valve has been minimized by assuming that the sodium level in the reactor is at the minimum safe level. The evaluation showed no break of siphon. Therefore, gas accumulation in the check valve would not be a problem for CRBRP.

The design transients for normal, upset, emergency and faulted plant conditions are described in detail in Appendix B of this PSAR. The valve design takes into account the effects of corrosion of the pressure boundary.

Preliminary analyses indicate that the check valve pressure drop at 3% flow will be less than 21% of the total system pressure drop for the three loop natural circulation case. 25

The design for over pressurization resulting from check valve slam, is included as part of the plant duty cycle.

5.3.3.5 Intermediate Heat Exchanger Characteristics

The IHX shall be thermally and hydraulically designed to permit safe, stable and predictable operation throughout the operating range. For these conditions, the IHX shall be designed for uniform flow distribution of the primary and secondary sides and to prevent thermal stratification of liquid metal, internal recirculation, and reverse flow. Direct impingement of inlet liquid metal on the heat transfer tubes shall be avoided. Areas of low flow, or packets or erosion entrapment, shall be avoided to the maximum extent possible.

The IHX thermal and hydraulic sizing shall be sized to dissipate 325 MWt for the full load process temperatures and flows given below:

Parameter	Primary	Intermediate
Inlet temperature (°F)	995	651
Flow rate (full glow) (lb/hr)	13.82×10^6	12.78×10^6
Pressure drop (nozzle to nozzle) (psig)	$13 \pm 20\%$	$14 \pm 20\%$
Inlet pressure (psig)	170	195
LMTD (°F)	68.5	

The IHX nominal overall heat transfer coefficient is 1373.8 Btu/hr-ft^2-°F for the temperature and flow conditions given above. This yields a required heat transfer area of 11810 ft^2 (or a length of 18.09 feet for each of the 2850 tubes which have an outside diameter of 0.8750 inches and a nominal wall thickness of 0.0484 inches). The actual effective tube length is 24.21 feet. This increase in length (33%) accounts for the uncertainties in tube and shell side heat transfer coefficients and tube wall thickness as well as allowances for flow bypass, fouling and tube plugging. The actual tube length (between the tube sheets) is 25.84 feet versus the effective length of 24,21 feet. The difference includes a 5-1/4 inches stagnant region below the upper tube sheet and 14-1/4 inch inactive length through the support plates. This sizing will assure that under the most pessimistic conditions, the unit will transfer 325 MWt with primary temperature of 995 and 730 °F, and intermediate temperatures of 936 and 651°F at the flow rates given above. The implication of this oversizing is that the primary temperatures will not have to be as high as 995/730°F when the plant is operating at rated power. The nozzle to nozzle pressure drops at rated 1

flow are 13.7 psi nominal for the shell (primary) side and 8.5 psi nominal for
59 the tube (intermediate) side.

During full power operation, the thermal center of the unit is 15 feet above the core midplane. (The geometric center of the tube bundle will be approximately 16'-3" above the core midplane). The thermal center is that point along the axis of the tube bundle at which half of the heat transfer has taken place; that is, the enthalpy at the thermal center is the average of the inlet and exit enthalpies. During a transient, the thermal center shifts. When the pumps are tripped, the primary and intermediate flows tend to collapse together. However, the intermediate flow decreases more rapidly (proportionately) because there is less momentum stored in the intermediate system. This "mismatch" of flows causes a slight lowering of the thermal center in the IHX. A preliminary analysis indicated that the thermal center would drop 1 foot in the first 10 seconds and an additional 3 feet in the next 90 seconds. As natural circulation continues, the thermal center gradually moves towards the top of the tube bundle because under steady state natural circulation conditions, there is more flow in the intermediate loop than in the primary loop. This condition arises primarily as a result of the smaller system flow impedance of the intermediate system and its larger elevation difference by comparison to the primary system. These effects are accounted
15 for in DEMO, the system transient analysis code (see Appendix A).

Internal convection within the unit is not expected to be significant. The shell side of the unit is baffled to create crossflow in addition to axial flow, and this feature is expected to minimize the tendency to develop maldistribution of flow even at low flows. Tube to tube flow variations on the intermediate side would be expected to be self-correcting due to buoyancy effects.

The design transients for normal, upset, emergency and faulted plant conditions are described in detail in Appendix B of this PSAR. Also, factored into the design is the effect of corrosive oxide on heat transfer.

The level of sodium oxide and other corrosive impurities, in the primary system, will be maintained and controlled by continuous cold trapping at 60 gpm. as discussed in Section 9.3. The present oxide levels during normal system operation is maintained below 2 ppm.

As mentioned above, the heat transfer area includes an allowance for fouling. This allowance is consistent with a 9% degradation of the overall heat transfer coefficient due to mass transfer deposits which is based on experimental data reported in Reference 2a.

The IHX is designed to use tubes with 0.045 inch -0 wall thickness (i.e., 0.045 inch min.). Allowing for 0.001 inch corrosion on either side of the tube wall, and 0.005 inch for scratches on the surface, the minimum available wall thickness for analysis is 0.038 inch.

Analysis per ASME code for 200 psi design pressure at 775°F requires a minimum wall of 0.030 inch. The available wall thickness of 0.038 inch would permit an external pressure of 280 psi (per code) with an inherent safety factor of 3. Actual calculated collapse pressure is therefore 840 psi for the min. inches available wall thickness of 0.038 inch. Margin over the specified design pressure is therefore 4.2.

The maximum design leak rate to the intermediate heat exchanger from the reactor coolant system to the intermediate coolant is zero by definition of the design. This design value is justified by provision of all welded joints and elimination of seals between the intermediate and primary systems. If a leak did occur the over pressurization of the intermediate system to the primary would prevent sodium leakage contamination of the intermediate system.

5.3.3.6 Coolant Boundary Integrity

41 The following is a summary of the piping integrity assurance program which is detailed in Reference 2 of Section 1.6.

The highest quality of engineering, fabrication, installation and inspection will go into the primary piping. The primary piping is a Class 1 item and will require a detailed stress analysis as required by the ASME Code. The analyses of the primary piping will assure that the design 41 will be able to meet all anticipated service requirements.

The system will be designed to assure that piping stresses, strains and deformation are within the applicable code criteria and system functional limits. The analyses to satisfy these limits shall reflect both time-independent and time-dependent materials properties and structural behavior (elastic and inelastic) by considering all of the relevant modes of failure.

(1) Ductile rupture from short-term loadings;

(2) Creep rupture from long-term loadings;

(3) Creep-fatigue failure;

(4) Gross distortion due to incremental collapse and ratcheting;

(5) Loss of function due to excessive deformation;

(6) Buckling due to short-term loadings;

(7) Creep buckling due to long-term loadings.

The RDT inspection limits placed on the pipe material and welds of the primary piping are much more demanding that those of the ASME Code. The stringent inspections, controls and checks will assure that the probability of an undetected defect in the piping, larger than the allowable, is extremely small. With this as background, the potential for a piping rupture is assessed for the 24-inch primary cold leg of the primary system as a representative and pertinent example. The approach used in this initial assessment and the results therefore will be summarized. As the design of the primary piping system progresses and more detailed stress analysis results become available, along with data from the fracture mechanisms test programs, the assessment for potential crack growth will include the entire primary piping system; i.e., the primary 36-inch hot leg, the primary 24-inch hot leg as well as the 24-inch cold leg. Using the basic approach outlined in this section, potential for a failure in the primary piping system will be investigated at elbows, straight-pipe sections, girth welds, structural discontinuities, piping penetrations and branch connections. At these locations the potential for longitudinal as well as circumferential cracking will be investigated.

The integrity analysis which follows takes into account the piping stresses due to all design loadings. These stresses are then used with a postulated defect much larger than the rejection limit imposed by the applied RDT Standard. Furthermore, the postulated defect is assumed to be at the location of highest stress in the piping system. Since the fatigue crack growth properties of the material are well-known, the techniques of fracture mechanics can be employed to predict the growth rates of these postulated defects under the service loadings. The significance of the predicted crack elongation is assessed by comparison with the critical crack size which has been experimentally verified. The critical crack size in a low pressure (i.e., 200 psi) LMFBR system is that length at which the crack will bulge open under operating stresses.

In addition, the potential for piping degradation due to caustic corrosion caused by a postulated leak has been investigated experimentally. Experiments have been performed to investigate sodium leakage from cracks in test pipes in PHTS cell environments.

The results of the piping integrity assessment in this section are summarized below.

It is shown by a scoping calculation that a hypothetical pipe crack many times deeper than the maximum QA allowable flaw would not grow to the critical size, would not penetrate the pipe wall, nor would it even grow significantly. A flaw which is postulated to be one-quarter of the way through the wall would grow an insignificant amount during the lifetime of the plant even if it were located at the point of highest stress. Thus, to reach the critical crack size, any defect would have to be subjected to cyclic stresses well above those predicted.

Even if, by some inconceivable means, a through-the-wall crack could be obtained, it would leak and the sodium would be detected before the crack could grow. Further, a small leak would not cause a crack to grow significantly by caustic corrosion because the reactor cavity moisture
41 level is low. The corrosion attack from preliminary tests at -3°F dewpoint rate is low ($\sim$1 mil per month for a leak of 50 gram/hr) and the leak would be detected by one or more of several leak detection methods (See Section 7.5.5.1). Thus, a leaking crack would be detected and no viable mechanism exists (neither fatigue crack growth or corrosion) to significantly enlarge a postulated through-the-wall crack.

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41 Pages 5.3-48 through 5.3-65

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5.3.3.7 Inadvertent Operation of Valves

The only valves which, if inadvertently operated, could perform action on or compound the consequences of transients are the vent and drain valves adjacent to the primary heat transport system. The primary HTS does not have any isolation valves or other non-passive valves for performing action during transients or other design events.

46 The inadvertent opening of the reactor make-up pump drain valves
(would require inadvertent opening of at least one manually operated valve as
46 well as one remotely operated valve), with a make-up line check valve malfunction, could siphon the reactor sodium level down to an elevation less than six feet below the normal operating condition. This sodium level is over
41 seven feet above the minimum safe level and consequently will have no adverse effect upon the system performing action during transients or other operating conditions.

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The IHX and pump drain valves must be manually operated in series, with a vacuum drawn in the drain lines or pressurization of the loop in order to cause a siphon, before draining of the primary loop can be initiated. The occurrence of such an event is highly improbable. However, in the event such an inadvertent operation of the valves does occur, the reactor vessel sodium level will be lowered at a rate equal to the drain rate minus the reactor make-up rate. (The design flow rate for the 3 inch drain line is 200 gpm, the reactor make-up rate is 150 gpm). Since the reactor level is continu-
36 ously being monitored by redundant level instruments, the reactor would be scrammed when the reactor level fell to the trip point.

The inadvertent operation of the high point vent valves in the primary system consist of sequential opening in series of manually operated valves before draining can be initiated. The opening of the high point vent valves will initiate sodium flow into the gas system. However, with a sodium freeze vent, no sodium will be able to penetrate the valve, unless heater units are inadvertently activated to melt the sodium. Thus no effective way for breaking syphon can be postulated by inadvertent vent valve operations.

Design measures will be taken to prevent inadvertent opening of such valves. These measures include:

1. Double valves manually operated in series.

2. One valve locked closed with the key under administrative control.

5.3.3.8 Performance of Pressure Relief Devices

For discussion of HTS overpressure protection, see Section 5.3.2.4. See Section 9.5 for a discussion of the cover gas system.

5.3.3.9 Operational Characteristics

A complete list of events to be considered in the structural design and fatigue analysis of the system and its components is included in Appendix B of this PSAR.

5.3.3.10 Material Considerations

5.3.3.10.1 Structural Materials for Elevated Temperature Service

Data on the high temperature mechanical behavior of heat transfer system materials are being generated by ongoing experimental programs and improved analytical techniques and methods of extrapolating properties to system design times are concurrently being developed. Because of this there is likely to be a continuous optimization of mechanical design and analysis procedures. However, the following general baseline procedures, given below, will be adopted for extrapolating mechanical property data.

5.3.3.10.1.1 Extrapolation of Creep-Rupture Data

The maximum design temperature for the primary heat transport system is 1015°F with the corresponding expected operating temperature approximately 50°F lower. Within the range of operating temperatures in the system, creep or material time dependent behavior must be considered. Stress-rupture times for temperatures in the creep regime will be estimated from ASME Code Case 1592 for which rupture times of up to 2 x 10^5 hours are given in both Types 304 and 316 stainless steel. These extrapolations were obtained from analytical procedures and do not take into account possible changes in fracture made at long times which could influence the gradients of the stress-rupture curves at times approaching the lifetime of the heat transport system. Current design will proceed on the assumption that the stress-rupture curves given in ASME Code Case 1592 are a reasonable extrapolation to long times. The validity of this assumption is supported by the following observations:

a. Long term creep rupture tests (~6 years) at 1100 and 1200°F have revealed no discontinuities in stress rupture curves for annealed Type 304 stainless steel (References 31, 32, and 33). 23

b. Discontinuities in the stress rupture curves occur most frequently in cold work materials (References 33 and 34). 23

Data on annealed Type 316 stainless steel indicate that at high stresses creep occurs principally by deformation within the grains. At lower 23
stresses (long rupture times) there is an increased tendency to form voids at grain boundaries. This is likely to be a result of the greater incidence of grain boundary sliding at low creep rates. Although there are no available data to predict the fracture mode at times approaching the design life of the heat transfer system it seems likely that the fracture mode will remain intergranular.

Virtually all allowable design stress values specified in the ASME Code were obtained from creep tests in air. Procedures have been developed to account for stress rupture behavior in non-oxidizing sodium
53 environments. Two major environmental factors were considered. These are:

a. The possible decrease in rupture strength which will result from exposure to sodium instead of air.

b. The change in rupture strength which will result from carbon and nitrogen transfer.

5.3.3.10.1.2 Extrapolation of Creep Fatigue Interaction Data

The techniques for evaluating creep fatigue interaction will generally follow the requirements of ASME Code Case 1592, or its latest revision. From Section 5.3.3.10.1.1, above, creep strength in the hotter regions of the heat transport system will be adversely affected by sodium exposure. On the other hand, available data indicate that fatigue life under strain controlled conditions will be increased by high temperature sodium exposure.

5.3.3.10.1.3 Degradation of Short Time Strength Properties

The austenitic stainless steels are unlikely to experience any degradation in strength from thermal aging, neutron irradiation, or prior creep. In fact, significant strengthening is likely from these factors. In the higher temperature regions of the primary and intermediate heat transport system, however, losses of carbon and nitrogen due to sodium exposure will decrease strength. Procedures are currently being developed to determine the rate of interstitial loss and the rate of change in the mechanical strength.

5.3.3.10.1.4 Elevated Temperature Tests

Comprehensive test programs were performed to generate materials property data to verify the design procedures being adopted, or proposed, in high temperature design. Results are provided in References 60 and 61.

5.3.3.10.1.5 Ability to Withstand Thermal Transients

Notch ductility as measured by the Charpy-V-notch test has historically been utilized for ferritic materials to determine the transition from ductile to brittle behavior. The applicability of Charpy-V-notch toughness measurements to austenitic materials and weldment has not been demonstrated. J-integral test and analysis procedures are currently being explored and utilized in the analysis of fracture resistance or toughness of austenitic materials because of the plasticity associated with fracture in this class of materials (References 38 and 39).

Typical J-integral values for Type 304 base metal are between 1680 and 4000 in lb/in^2 over the temperature range of from room temperature to 830°F (Reference 39 and Ref. 2 of Section 1.6). A typical value for Type 316 stainless steel is 3500 in. lb/in^2 at RT (Reference 40). To provide perspective to these values, it is noted that these J-integral values translate to K fracture toughness values in the range of 217 to 385 ksi/in.

In the case of austenitic stainless steel weld metal, the available J-integral values are 660 in. lb/in.^{2} for Type 308 submerged-arc weld deposit (Reference 39) and 640 in. lb/in.^{2} for Type 316 submerged arc weld deposit (Reference 38). These J-integral values translate to K values of approximately 130 ksi/in. though the J-values for the weldments are lower than those for the base metal, it is expected that elastic plastic conditions will prevail for weldments even in thick sections. (Reference 2 of Section 1.6)

Thermal transients during the plant life may result in fatigue crack growth of defects which might not be detected during the non-destructive examination of the welds. The elevated temperature fatigue crack growth behavior of weldments of Type 308 and 316 filler metal has been studied by several investigators. The effects of test temperature (References 3 and 9), thermal aging (Reference 6), crack orientation relative to the weld (References 10 and 7), welding process (Reference 7) and irradiation (Reference 41) have been examined and the general conclusion reached was that the weld metal has resistance to fatigue crack extension that is at least as good as that if the base metal tested under similar conditions.

Research is being continued at both NRL and HEDL to investigate the
toughness of austenitic stainless steel weldments, evaluate and correlate
toughness measurement techniques and to further develop analytic methods
such as the J-integral for engineering application. In addition, a long term 24
thermal aging program on CRBRP prototypic weld and base metal samples in air
53 has been initiated to monitor possible degradation of mechanical properties
such as crack propagation and fracture toughness.

5.3.3.10.2 Austenitic Stainless Steel

Comprehensive cleaning and heat treatment procedures are being developed for heat transport systems materials. In addition to the major system components which will be fabricated from Types 304 and 316 SS a number of other alloys are used for specialized purposes. These include but are not limited to Stellite hard facing materials for valve seats and pump bearings, and SA508 (Class 2) steel for the reactor vessel head forging.

5.3.3.10.2.1 Cleaning and Contamination Protection Procedures

Procedures to avoid contamination which could give rise to intergranular attack or stress corrosion cracking during fabrication, handling, transportation, storage, cleaning, installation and service have been outlined in Section 5.3.2.1.4.

5.3.3.10.2.2 Heat Treatment Requirements

Heat treatment of austenitic stainless steel components are usually performed to either minimize sensitization or to prevent inservice distortion which would arise from internal stresses. Heat treatments are usually recommended at temperatures in excess of 1600°F where sensitized structures are eliminated, followed by rapid cooling. The rate of cooling depends on section thickness and geometry, and must not be so fast that internal stresses are reintroduced during cooling. Precise heat treatments for the various heat transport system components have yet to be finalized.

43 5.3.3.10.2.3 Control of Delta Ferrite

Depending on weld metal composition, hot cracking in the weld region may occur during weld solidification. A reduction in hot cracking susceptibility will be effected by controlling the weld chemical composition to allow optimum levels of delta ferrite to be present. Delta ferrite levels will conform to the requirements of Section III of the 1974 ASME Boiler and Pressure Vessel Code and, at elevated service temperatures, to Code Case 1592. Ferrite levels will be verified with the aid of the Shaeffler diagram. Weld rods will conform to the requirements of RDT Standards M1-1T, March 1975 and M1-2T, July 1975.

Meeting these limits per the code obviates the need for additional testing of welds or weld metals since these limits are consistent with RDT Standards M 1-1-T and M 1-2T weld rod material. Accordingly, the presently funded program referred to in section 5.3.3.10.1.4 does not include any test specimens of welds or weldments.

The welding of austenitic stainless steel is controlled to mitigate the occurrence of microfissuring or hot cracking in the weld. It has been well documented in the technical literature that the presence of delta ferrite is one of the mechanisms for reducing the susceptibility of stainless steel welds to hot cracking. However, there are insufficient data to specify a minimum delta ferrite level below which the material will be prone to hot cracking. It is assumed that such a minimum lies somewhere between 0 and 3 percent delta ferrite.

The scope of these controls discussed herein encompasses welding processes used to join stainless steel parts in components designed, fabricated, or stamped in accordance with ASME B&PV Code, Section III Class 1, 2, 3 and CS components. Delta ferrite control is appropriate for the above welding requirements except where no filler metal is used, or for other exceptions where such control is not applicable. These exceptions include electron beam welding, autogenous gas shielded tungsten arc welding, explosive welding, and welding using fully austenitic welding materials.

The fabrication and installation specifications require welding procedures and welder qualifications in accordance with Section III, and include the delta ferrite determinations for the austenitic stainless steel welding materials that are used for welding qualification testing and for production processing. Specifically, the undiluted weld deposits of the "starting" welding materials are required to contain a minimum of 5 percent delta ferrite* in the case of Section III or between 3 and 9% in the case of Code Case 1592 as determined by chemical analysis and calculation using the appropriate weld metal constitution diagrams in Section III. Since the ASTM requirements for delta ferrite in Section III and Code Case 1592 are different for the temperature range to which they apply, the CRBRP has adopted the range of 5 to 9% delta ferrite for use, in conjunction with applicable RDT standards.

*The equivalent Ferrite Number may be substituted for percent delta ferrite. 25

Combinations of approved heats and lots of "starting" welding materials are used for all welding processes. The welding quality assurance program includes identification and control of welding material by lots and heats as appropriate.

All of the weld processing is monitored according to approved inspection programs which include review of "starting" materials, qualification records and welding parameters. Welding systems are also subject to quality assurance audit including calibration of gages and instruments; identification of "starting" and completed materials; welder and procedure qualifications; availability and use of approved welding and heat treating procedures; and documentary evidence of compliance with materials, welding parameters and inspection requirements. Fabrication and installation welds are inspected using non-destructive examination methods according to Section III rules as supplemented by RDT standards where applicable. Welds made in accordance with these criteria have continually resulted in sound production welds, which are free from detrimental fissuring and consistently conform to non-destructive acceptance standards.

To further assure the reliability of these controls, Westinghouse Power Systems has initiated a verification program. A detailed description of the verification program and a review of data collected to date are reported in WCAP-8324, Control of Delta Ferrite in Austenitic Stainless Steel Weldments submitted by Westinghouse PWR Systems. New data generated from this program have been published periodically as addenda to WCAP-8324. 25

5.3.3.10.3 Compatibility with Coolant

5.3.3.10.3.1 General Interstitial and Mass Transfer Effects

A list of the primary heat transport system components is given in Section 5.3.1.2. The major construction materials in contact with the sodium coolant are Type 304 and Type 316 stainless steel. Generally, these alloys show excellent compatibility with the coolant. Although metallic elements, such as Fe, Cr, Ni, and Mn are transferred from the hotter to the cooler regions of the system, the overall effect is quite small since most of the components are thick in section. A measure of the rate of metal loss may be made using equations 7 and 8 of Section 5.3.2.2.5.

Weld metals, because of their similar chemical composition, would be expected to have similar rates of corrosion in sodium.

In the cooler regions of the sodium system, mass deposition has been shown to occur. This will cause a reduction in the rate of heat transfer in the intermediate heat exchanger. However, a recent study based on the intermediate heat transport system indicates that the overall loss in thermal efficiency in the heat exchanger is not significant (see Reference 2a). 14

A more important effect which will occur in the heat transport system is associated with interstitial transport. Both carbon and nitrogen will be transferred from the hotter regions of the stainless steel and deposited in the cooler regions. Because of the slower rates of diffusion, the depth of carburization and nitridation will be small in the cooler regions of the system. Interstitial transfer will show a maximum effect in thin section components such as intermediate heat exchanger tubing and fuel cladding. For the intermediate heat exchanger, a recent study indicates that in the hottest regions of the intermediate heat exchanger the carbon level will fall below the minimum permissible value of 0.04 weight percent during service. A wall thickness allowance was used to counter resulting strength loss (see Section 5.3.3.5).

Of the heat transport system components only the reactor vessel, vessel head, and guard vessel are likely to experience significant irradiation. Current work will evaluate the effect of neutron dose and spectra on mechanical degradation of these components and sufficient shielding will be used to ensure that the lifetime neutron fluence is maintained with acceptable limits.

5.3.3.10.3.2 Mass Transfer of Radioactive Species

The radioactive aspects of mass transfer in the reactor coolant from the reactor to the Heat Transport System are discussed in detail in Section 11.1. These aspects, in themselves, do not affect the structural integrity of the HTS.

5.3.3.10.4 Compatibility with External Insulation and Environmental Atmosphere

Within the heat transport system the reactor vessel, pumps, and intermediate heat exchangers are enclosed by guard vessels. Between these components and the guard vessels a semi-inert gaseous atmosphere of 0.5-2% oxygen/nitrogen is maintained. The piping and the upper portions of the components containing sodium external to the guard vessels are also insulated to minimize heat loss to the PHTS cells. The thermal insulation consists of alumina silicate blanket
53 material manufactured under controlled conditions to minimize the pickup of halogens and/or moisture. The insulation is protected from halogen pickup during shipping, storage and installation. The insulation has an inner liner and is installed on standoffs to provide an annulus for heaters and leak
59 53 detection equipment and, therefore, does not directly contact piping or components. No field compounded thermal insulation materials are used. This will minimize any potential contamination of the piping by corrosive elements in the insulation. Most piping is also exposed to the 0.5-2% oxygen/nitrogen atmosphere.

Sodium leaks into the guard vessels, should they occur, are unlikely to be self sealing in view of the low oxygen content. Small leakages will be contained within the guard vessel. With respect to the piping (except that which is situated within the guard vessels) any sodium leakage will react with oxygen, nitrogen, and thermal insulation. No comprehensive data appear to be available to evaluate the reaction in detail but available information from experimental sodium loops indicates that the leaking sodium will form a sodium oxide (and very likely sodium nitride) "growth" beneath the insulation at the point of
53 leakage. For temperatures below about 1000°F no self sealing of the leak is usually observed. Studies were conducted to evaluate the nature of sodium leakage through precracked austenitic stainless steel piping into a 1.2 v/o oxygen/98.7 v/o nitrogen atmosphere. Materials of Construction are listed in Tables 5.3-4 thru 5.3-9.

5.3.3.10.5 Chemistry of Reactor Coolant

The heat transport system sodium chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the coolant quality meets the specifications.

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Sodium purification capability is provided through the use of cold traps.

Capabilities are provided both for "in-line" primary and
56| intermediate sodium purity determinations (sodium plugging temperature indicators) and for direct sampling and laboratory analysis to monitor impurities. The systems are described in Section 9.3.2.

5.3.3.11 Protection Against Environmental Factors

Protection of the principal components of the PHTS is provided by the structural integrity of the Reactor Containment Building and the individual cells within the building. Environmental factors to be considered include the following:

Fire Protection - See Section 9.13.

Flooding Protection - See Section 3.4.

Missile Protection - See Section 3.5.

Seismic Protection - See Section 3.7 and 3.8

Accidents - See Section 15.6

5.3.4 Tests and Inspections

5.3.4.1 Provisions for Access to the Reactor Coolant Pressure Boundary

Access hatches (floor plugs) are strategically located over the HTS cells and reactor cavity such that direct access to all HTS equipment is provided. In addition to the access hatches, periscope penetrations are located over the HTS cell for inservice inspections of the HTS equipment. Remote access to the reactor vessel is provided by an enclosed raceway below the head access area floor. Details of the inservice inspection plan are provided in Section 5.3.2.1.3.

5.3.4.2 Equipment for Inservice Inspection

The HTS inservice inspection equipment consist of two periscopes for inspection of the HTS cell and interior of the HTS guard vessels. The periscope for HTS cell inspection is about 8 inches in diameter and shall be capable of viewing during plant operation without requiring deinerting of the HTS cells. The periscope for inspection with the HTS guard vessels is approximately 3 inches in diameter and use will not require awaiting decay of Na^{24} or deinerting of the HTS cell. This equipment is intended to carry out the inservice inspection plan of Section 5.3.2.1.4.

Inspections within the reactor vessel guard vessel are performed by television camera.

5.3.4.3 Recording and Comparing Data

Quality Assurance records will be kept in accordance with Section 17.1 of this PSAR.

5.3.4.4 Coordination of Inspection Equipment with Access Provisions

Coordination of the periscope penetration interfaces over the HTS cells and HTS guard vessel will be controlled by an Interface Control Drawing. The exact location of these penetrations has not been established at this time, however, it is tentatively planned that there will be seven 12 inch diameter penetrations over each HTS cell and seven 6 inch penetrations over the HTS guard vessels in each HTS cell.

The current inservice inspection plan uses two separate periscopes for cell and guard vessel inspections respectively. It is a requirement of both periscopes, that use does not require deinerting of the HTS cell or require decay of Na-24, however, due to temperature limitations of the periscope lenses, guard vessel inspections require loop cooldown to 400°F. HTS cell inspections can be conducted during power operation.

5.3.4.5 Maintenance of a Single Loop

As discussed in Section 3.A.1, each of the three PHTS cells and the reactor cavity has its own recirculating gas cooling system (RGCS). This system circulates gas, either nitrogen or air, through the cell and contains a gas to water cooler where heat is transferred to the Chilled Water System. The PHTS cells and the reactor cavity are separated from each other by concrete walls and the piping penetrations. This arrangement provides the capability to individually inert or deinert a PHTS cell or the reactor cavity.

Each loop can be idled and vented; that is, by opening the high point
53 vents (Figure 5.1-2b) and de-energizing the pony motor, the flow in a particular loop can be stopped and sodium communication between the idled and vented loop and the rest of the primary system can be interrupted. Draining of a primary loop is achieved by draining through the drain line at the bottom of a primary pump and/or the drain connection at the bottom of the IHX primary outlet pipe. (Both drain lines rise above the guard vessel tops and are isolated during normal operation).

Prior to primary system maintenance requiring draining a loop, the following conditions will be satisfied:

1. Reactor is shut down.
2. Control rods are in and unlatched.
3. Main motors of PHTS and IHTS pumps are locked off.
4. Primary system is cooled to the refueling temperature 400°F.
5. Sodium makeup pumps are locked off.
6. Primary system is drained to a maintenance level in the reactor vessel (∿4-1/2 feet below the normal operating level)
7. Pony motor of affected loop is locked off.
8. Siphon is broken in the hot and cold legs of the affected loop.
9. Radioactive argon is decayed or purged from system. 25

There will be no operational modes such as two loop operation when one loop is drained for maintenance. The two unaffected loops will have sodium circulating through them with the pumps in those loops operating at pony motor speed. Two loop operation will not be attempted unless the down loop is completely full of sodium.

With regard to the earliest time for access to a cell, analyses indicate that when considering only the decay of Na^{24}, the dose rate two feet from a drained main coolant piping having a 3 mil sodium film uniformly deposited over the pipe, would be less than 100 mrem/hr 5 days after draining. Analysis also indicates that if the reactor were operated on two loops at 50% power for ten days (with back leakage through the check valve into the down loop at 100 gpm), the dose rate due to Na^{24} would be less than 100 mrem/hr in the affected loop 2.82 days after shutdown even though the affected loop has not been drained. In general, access to a HTS cell will be accompanied by draining of the affected loop and deinerting the cell.

Since the check valve is located at the high point of the cold leg in each loop, when one loop is vented and drained, the sodium in the reactor inlet line downcomer is not in contact with the check valve. Therefore, check valve leakage is not a consideration in the maintenance configuration.

When one loop is vented and drained and the pumps in the unaffected loops are operating at pony motor speed, the level of the sodium in the affected loop downcomer adjacent to the reactor will be higher than the level in the reactor vessel. This difference in level will be equal to the pressure drop associated with pony motor flow of the other two loops. As noted previously, the reactor vessel level is lowered during this operating mode to accommodate these effects. Therefore, no credit is taken for the existence of the check value.

Refilling and restarting of the down loop would, in general, follow the procedures for initial fill. Trace heating may be required to assure metal temperatures of 400°F followed by refilling through the drain lines. Thermal shocks in the system are minimized by maintaining essentially an isothermal condition at $\sim$400°F in the reactor and unaffected loops. 25

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TABLE 5.3-1

THERMAL & HYDRAULIC DESIGN BASIS PARAMETERS

		Parameter	Units	Value
	1.	Power per loop	MWt	325
	2.	Flow per loop Primary	$\#/hr \times 10^6$	13.82
		Int.	$\#/hr \times 10^6$	12.78
	3.	Primary Hot Leg Temp.	°F	995
	4.	Primary Cold Leg. Temp.	°F	730
	5.	Int. Hot Leg Temp.	°F	936
	6.	Int. Cold Leg Temp.	°F	651
	7.	Maximum Primary System Pressure Drop	psi	160.3
	8.	Maximum PHTS Loop ΔP (Exclusive of Reactor)	psi	37.3
	9.	Required Primary Pump Head (THD)	ft	458
36	10.	Maximum Primary Pump Drawdown	ft	8

TABLE 5.3-2

PRIMARY HEAT TRANSPORT SYSTEM COMPONENT DESIGN CONDITIONS

Component	Parameter	Design Conditions Temp. (°F)	Design Conditions Press. (psi)	
1. Pump	Pump tank	1015	*	33
	Hydraulics & Discharge Assembly	1015	200	
2. IHX	Shell Side	1015	200	
	Tube Side	1015	325	
	Primary to Int. Diff. Pressure	775	200	
	Int. to Primary Diff. Pressure	1015	325	
	Primary to Int. Hot End ΔT	75	-	
	Primary to Int. Cold End ΔT	120	-	
3. Check Valve		775	200	
4. Flowmeter		775	200	
5. Piping	Reactor to Pump	1015	30	
	Pump to IHX	1015	200	
	IHX to Check Valve	775	200	
	Check Valve to Reactor	775	200	

* Design pressure is 325 psig @ 775°F in accordance with intermediate pump design for identicality. 33

TABLE 5.3-3

STEADY STATE OPERATING CONDITIONS FOR STRUCTURAL EVALUATION

System Location	Refueling T (°F)	Refueling P psig	Hot Standby T (°F)	Hot Standby P psig	40% Power T (°F)	40% Power P psig	80% Power T (°F)	80% Power P psig	100% Power T (°F)	100% Power P psig
Reactor Outlet Nozzle	400	6.8	600	6.6	910	6.1	974	5.4	1015	4.9
Reactor Inlet Nozzle	400	18.5	600	18.1	610	33	674	95	715	134
Pump Inlet	400	9.1	600	8.9	910	8.2	974	6.9	1015	6.0
Pump Discharge	400	14.4	600	14.3	910	34	974	115	1015	168
IHX Inlet	400	6.5	600	6.4	910	26	974	104	1015	155
Check Valve Inlet	400	1.3	600	1.4	610	19	674	88	715	133

NOTES:

1. "100%" power is actually 115% of rated power or 1121 MWt. The flow rate in each loop = 14.03 x 10^6#/hr = 1.015 times design flow.

2. "80%" power is actually 897 MWt or 115% of rated 80% power (780 MWt). The loop flow = 11.22 x 10^6#/hr.

3. "40%" power is actually 40% of rated power (390 MWt) which is 35% of the 115% power condition. The loop flow is 4.84 x 10^6#/hr.

TABLE 5.3-4

INTERMEDIATE HEAT EXCHANGER MATERIALS SPECIFICATION

	Product Form	RDT Standard*	Grade
	Plate	M5-1 Nov. 1974	304 & 316
	Forgings	M2-2 Dec. 1974	F304
		M2-4 Nov. 1974	F8 & F8 m
	Tubing	M3-2 Dec. 1974	TP 304H
	Pipe	M3-3 Nov. 1974	TP 304
	Bars	M7-3 Nov. 1974	
	Bolting	M6-1 Feb. 1975, M6-3 Feb. 1975	
	Nuts	M6-4 Feb. 1975	
	Springs	M8-1 May 1975	
7	Studs	M6-3 Feb. 1975	B8M
	Diaphragm	M5-1 Nov. 1974	TP 316
	Spool Pieces	M3-6 Apr. 1976	TP 316
57	Flange	M2-4 Nov. 1974	

The following additional chemistry controls apply:

Carbon - 0.04 to 0.08% for material <0.25 in. thick (Types 304 and 316)

- 0.05% minimum for material $\leq$0.25 in. thick (Type 304H)

*RDT Materials Standards apply only to those parts forming portions of the pressure retaining boundaries or that are exposed to liquid sodium or sodium containing environments.

TABLE 5.3-5

PRIMARY PUMP PREFERRED MATERIALS SPECIFICATION

	Product Form	ASME Specifications	RDT Standards Supplements
	Forgings	SA182, SA336, SA654, SA105(1)	*M2-2, *M2-4, M2-1
	Fittings	SA403, SA652	*M2-5
45	Tubing	SA213, SA655	M3-2
	Pipe	SA376, SA655	*M3-3
	Plate	SA240, SA647	*M5-1
47 49	Bolting(2)	SA320, SA193, SA614, SA540(1), SA637	*M6-1, M6-3, M6-5
	Nuts	SA194, SA614	*M6-4
45	Bars	SA479, SA654	*M7-3

*These materials may be used for the containment boundary. This specification does not invoke RDT Material Standards for pump parts which are inside the boundaries of code jurisdiction but not covered by it.

Notes:

49 Materials in contact with liquid sodium or sodium vapor shall be austenitic stainless steel Type 304, Type 316, or Inconel 718, with the exception of areas hard surfaced for wear resistance or antigalling. (This requirement does not apply to the cover gas rotating seals.)

Hard surfacing shall be used only in areas to resist wear and galling. Hard surfacing procedures and qualifications shall be submitted for approval. The following additonal chemistry controls apply:

Carbon - 0.04 to 0.08% for material >0.50 in. thick (Types 304 and 316).
- 0.055% to 0.08% for material <0.50 in. thick (Types 304 and 316).

(1) These materials may be used for code containment boundary members which are external to the sodium vapor or sodium environment consistent with the design temperatures established.

(2) Acceptable materials for bolting inside the pump in contact with the sodium or sodium vapor environment but not part of the containment boundary may include SA637 GR718.

45

TABLE 5.3-6

PRIMARY SYSTEM CHECK VALVE MATERIAL SPECIFICATION

Product Form	RDT Standard	Grade	
Plate	M5-1 Nov. 1974	304	(1)
Forgings	M2-2 Dec. 1974	F304	(1)
Bar	M7-3 Nov. 1974	304	(1)
	M2-18 Dec. 1974	Inconel 718	(2)
	M7-7 April 1975	Stellite 6B	(3)
Casting	M4-3 July 1975	C-1 (Stellite 6)	(3)
Bolting	M6-3 Feb. 1975	B8M	(4)
Springs	M8-1 May 1975	Age Hardenable Ni-Cr-Fe Alloy	(2)
Strip	M5-21 April 1975	Inconel 718 Ni-Cr-Fe Alloy	(2)
27 Wire , AMS 5687, Jan. 1971		Inconel 718 Ni-Cr-Fe Alloy	(2)

(1) The following additional chemistry controls apply: Carbon - 0.04% Min.

(2) Inconel 718 is for nonwelded, high strength applications such as disc pivot pins and dashpot plunger and springs. Material test programs have shown this material to be resistant to self-welding independent of the other material in the wear couple. It will be used with Stellite 6 which has been proven by testing in sodium for FFTF Check Valve Component parts. (Refs. 21 thru 27.)

(3) Machined and centrifugally cast Stellite 6 parts have been established by sodium testing and full-scale component parts under prototypical conditions for FFTF isolation valves. (Refs. 20 and 28)

Stellite 6 hard-faced applications have been widely used in sodium service for valve seat and bearing surface applications. The wear couple has been established by extensive material testing and by full-scale component part testing for FFTF Large Valve applications. (Refs. 20 thru 28)

(4) Bolting materials are not for use in sodium and are for external use only outside of seal welds. 1

TABLE 5.3-7

PRIMARY HEAT TRANSPORT SYSTEM PIPING

MATERIALS SPECIFICATIONS[1]

ITEM	PRODUCT FORM	MATERIAL GRADES COVERED
Pipe	Welded [2] Seamless [2]	Types 316H, 304H, 304, 2 1/4 Cr-1 Mo Types 316H, 304H, 304, 2 1/4 Cr-1 Mo
Fittings	Welded [2] Seamless [2]	Types 316H, 304H, 304, 2 1/4 Cr-1 Mo Types 316H, 304H, 304, 2 1/4 Cr-1 Mo
Branch Connections	Forging	Types 316H, 304H, 304, 2 1/4 Cr-1 Mo
Thermowells	Fabrication	Type 316H, 2 1/4 Cr-1 Mo
Flued Heads, Pipe Support	Forging plus Fabrication	Types 316H, 304H, 304
Hangers	Fabrication	Various - as allowed by ASME Code
Snubbers	Fabrication	Various - as allowed by ASME Code
Clamps	Fabrication	Type 304
Auxiliary Steel & Hardware	Fabrication	Various - as allowed by AISC Code
IHX Vent-Line Flow Restrictor	Fabrication	Type 316H
Shop Fabrication of Pipe Sub-Assemblies (Spools)	Fabrication	Types 316H, 304H, 304, 2 1/4 Cr-1 Mo

(1) The CRBRP Materials Specifications are based on ASME Code Section III and RDT Standards requirements. They are used for all large diameter sodium piping in both Primary and Intermediate Heat Transport Systems.

(2) Welded and seamless products to these specifications are interchangeable for the intended service.

57

TABLE 5.3-8

IHX AND PRIMARY PUMP GUARD VESSEL
MATERIALS SPECIFICATION*

44

Product Form	ASME Specification	Grade
Stainless Steel		
Forgings	SA 336	F8
Plate	SA 240	304
Bolting	SA 540	B22 Class 2

44

*No RTD Materials Standards apply.

44

Amend. 44
April 1978

TABLE 5.3-9

PRIMARY HEAT TRANSPORT SYSTEM COMPONENT
WELD FILLER MATERIALS SPECIFICATIONS

	Weld Material Form	RDT Standards	Filler Material Classification
27	Stainless Steel	M1-1 March 1975	E 308L-15 or 16
	Covered Electrodes		E 308-15 or 16
			E 316L-15 or 16
			E 316-15 or 16
			E 16 - 8 - 2
27	Stainless Steel	M1-2 July 1975	ER 308, ER 308L
	Bare Rods and Electrodes		ER 316, ER 316L
			ER 16 - 8 - 2

WELD MATERIALS FOR GUARD VESSELS

	Stainless Steel Covered Electrodes	None	E 308-15
57	Stainless Steel Bare Rods and Electrodes	None	ER 308

Amend. 57
Nov. 1980

TABLE 5.3-10

PRIMARY REACTOR COOLANT PRESSURE BOUNDARY VALVES AND PUMPS

Component	Status Active/Inactive	Normal Operating Mode*
PHTS Pumps	Active	Running
PHTS Check Valve	Inactive	NO
PHTS Fill and Drain Valves†	Inactive	LC
Reactor Coolant Make-up Pumps†	Inactive	Running
PHTS High Point Vent Valve†	Inactive	LC

*NO - Normally Open
LC - Locked Closed
†These pumps and valves are part of the primary coolant pressure boundary, but are not actually parts of the heat transport system. These components are discussed in Section 9.3 and 9.5.

Amend. 40
July 1977

TABLE 5.3-11

S_t (3×10^{5}h) AND S_m VALUES FOR AUSTENITIC STAINLESS STEELS*

TEMPERATURE (°F)	S_t (ksi)				S_m (ksi)			
	304 L	304 and 304 H	316 L	316 and 316 H	304 L (SA-182)	304 and 304 H	316 L (SA-182)	316 and 316 H
100	N/A	N/A	N/A	N/A	16.6	20.0	16.6	20.0
200	N/A	N/A	N/A	N/A	16.6	20.0	16.6	20.0
300	N/A	N/A	N/A	N/A	16.6	20.0	16.6	20.0
400	N/A	N/A	N/A	N/A	15.7	18.7	15.5	19.2
500	N/A	N/A	N/A	N/A	14.7	17.4	14.4	17.9
600	N/A	N/A	N/A	N/A	13.9	16.4	13.5	17.0
700	N/A	N/A	N/A	N/A	13.4	15.9	12.8	16.3
800	N/A	20.4	N/A	20.8		13.0	15.1	12.315.8
900	N/A	16.0	N/A	19.3	N/A	14.6**	N/A	15.6**
1000	N/A	9.3	N/A	14.0	N/A	14.0*	N/A	15.4**
1100	N/A	5.7	N/A	7.8	N/A	13.2**	N/A	14.8**
1200	N/A	3.4	N/A	4.5	N/A	12.7**	N/A	14.6**

*Data are from ASME Boiler and Pressure Vessel code, Section III, 1974 Edition with Addenda through Summer 1975 and ASME Code Case 1592-7, and apply to Class 1 components. For other editions of the Code, these values may vary somewhat.

**For Type 304 and Type 316 grades only.

N/A = Not applicable.

TABLE 5.3-12

PRIMARY HEAT TRANSPORT SYSTEM CODE
AND SEISMIC CATEGORY MATRIX

Primary System Component	RDT Component Standards[1]	ASME Code Section/Class	Seismic Category	F9-4T Req'ts For Nuclear Components At Elevated Temp.
Primary Pump	E3-2T June 1974	III/1	I	Note 2
IHX	E4-6T April 1975	III/1	I	Note 2
Check Valve	E1-18T May 1975	III/1	I	Note 2
Flow Meter	C4-5T April 1974	III/1	I	-
27 Guard Vessels	E10-2T July 1973	III/2	I	Note 3
Piping & Fittings	N/A	III/1	I	Note 2
Pipe Hangers				
27 Supports & Snubbers	E7-6T May 1972	III/ Subsection NF	I	-
Thermal Insul.	N/A	N/A	III*	-
59 Trace Heating	N/A	N/A	I	-

Notes:

1. Component standards used as guidance in equipment specification preparation, not invoked.

2. This standard, modified only as indicated in the equipment specification, is to be applied in its entirety to all structures in the component.

3. Constructed to rules of Class 1 but not hydrostatically tested or code stamped. An elevated temperature supplement to the equipment specification, equivalent to RDT Standard F9-4 with modifications and code cases, will be used.

*Thermal insulation is functionally seismic Category III, however, it is designed to the requirements of Seismic Category I, utilizing static analysis
59 rather than dynamic analysis.

TABLE 5.3-13

COLD LEG CHECK VALVE CHARACTERISTICS

Requirement	Units	Design Value
Design Flow Rate (at 730°F)	lbs/hr	13.82×10^6
Flow Range at Normal Operating Conditions	% of Design Flow	40 - 100
Max. Shutoff, P Imposed Across Seat		
Steady State	psi	160
Pressure Loss at Design Flow	psi	≤10
Pressure Loss at Pony-Motor Flow Conditions of 2500 gpm at 600°F	psi	≤0.20
Pressure Loss at Natural Circulation Flow Conditions of 670 gpm at 730°F	psi	≤0.03
Temp. at Which Design Flow Pressure Loss is Calculated	°F	730
Allowable Leakage in Reverse Direction at Shutoff at 730°F	gpm	21
Pressure Difference for Allowable Leakage, Reverse Direction	psi	50

Closure Characteristics

The maximum steady state reverse flow allowed by the check valve shall be less than 1100 gpm. The valve shall not require a pressure differential across the
59| disk greater than 1.75 psi to shut. Closing time shall be 12 seconds maximum (after flow reversal) with a resultant pressure surge of less than 50 psi under the specified reverse flow conditions.

56 53 TABLES 5.3-14 thru 5.3-22

HAVE BEEN DELETED

TABLE 5.3-23

PHTS and IHTS Pump Generated Frequencies

	Condition	Frequency ~ hz: Structural Vibration	Frequency ~ hz: Pressure Pulse, Enter Impeller	Frequency ~ hz: Pressure Pulse, Leave Impeller	
	Primary Coolant Pump				
	Pony Motor Flow	1.87	19	56	
	40% Rated Flow	7.4	74	223	
	80% Rated Flow	14.9	149	447	
59	100% Rated Flow	18.6	186	553	
	Intermediate Coolant Pump				
	Pony Motor Flow	1.6	16	48	
	40% Rated Flow	6.4	64	192	
	80% Rated Flow	12.8	123	384	
59	100% Rated Flow	16.0	160	480	25

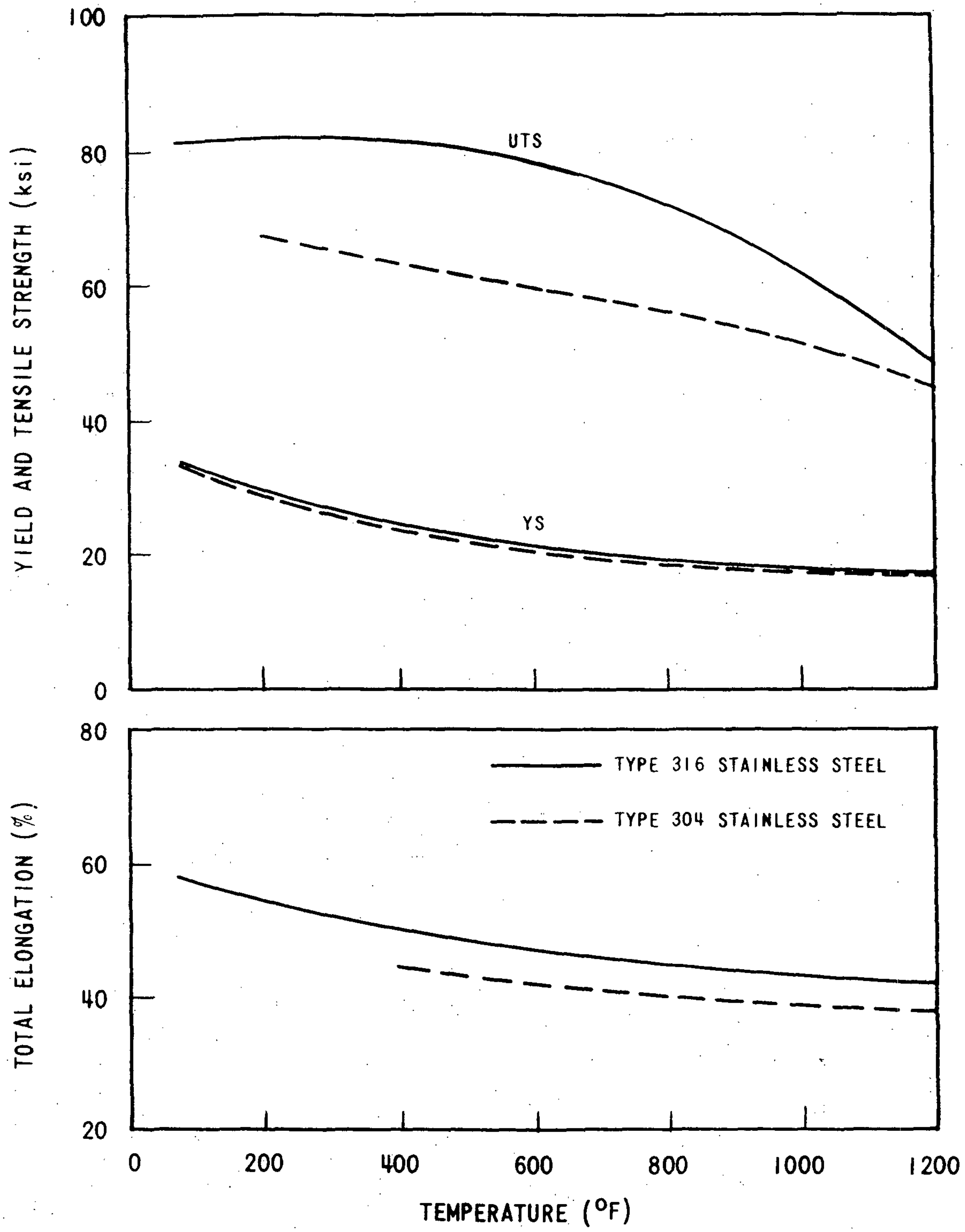


Figure 5.3-1. **Comparison Between the Strengths and Ductilities of Solution Treated Types 304 and 316 Stainless Steel Containing 0.08 Weight Percent (C + N)**

* Taken from Reference 54.

6669-17

Amend. 56
Aug. 1980

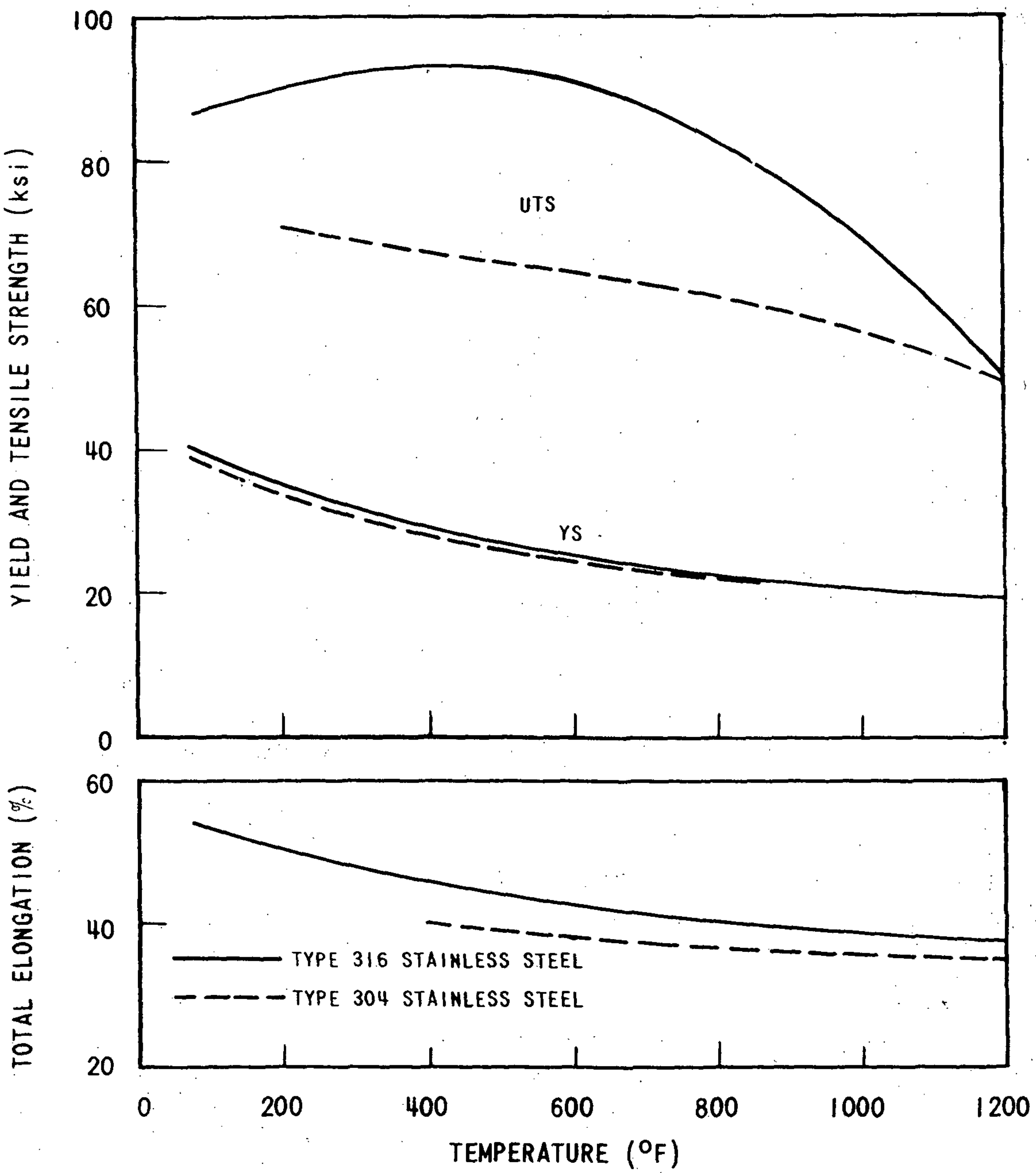


Figure 5.3-2. Comparison Between the Strengths and Ductilities of Solution Treated Types 304 and 316 Stainless Steel Containing 0.12 Weight Percent (C + N)

* Taken from Reference 54.

6669-18

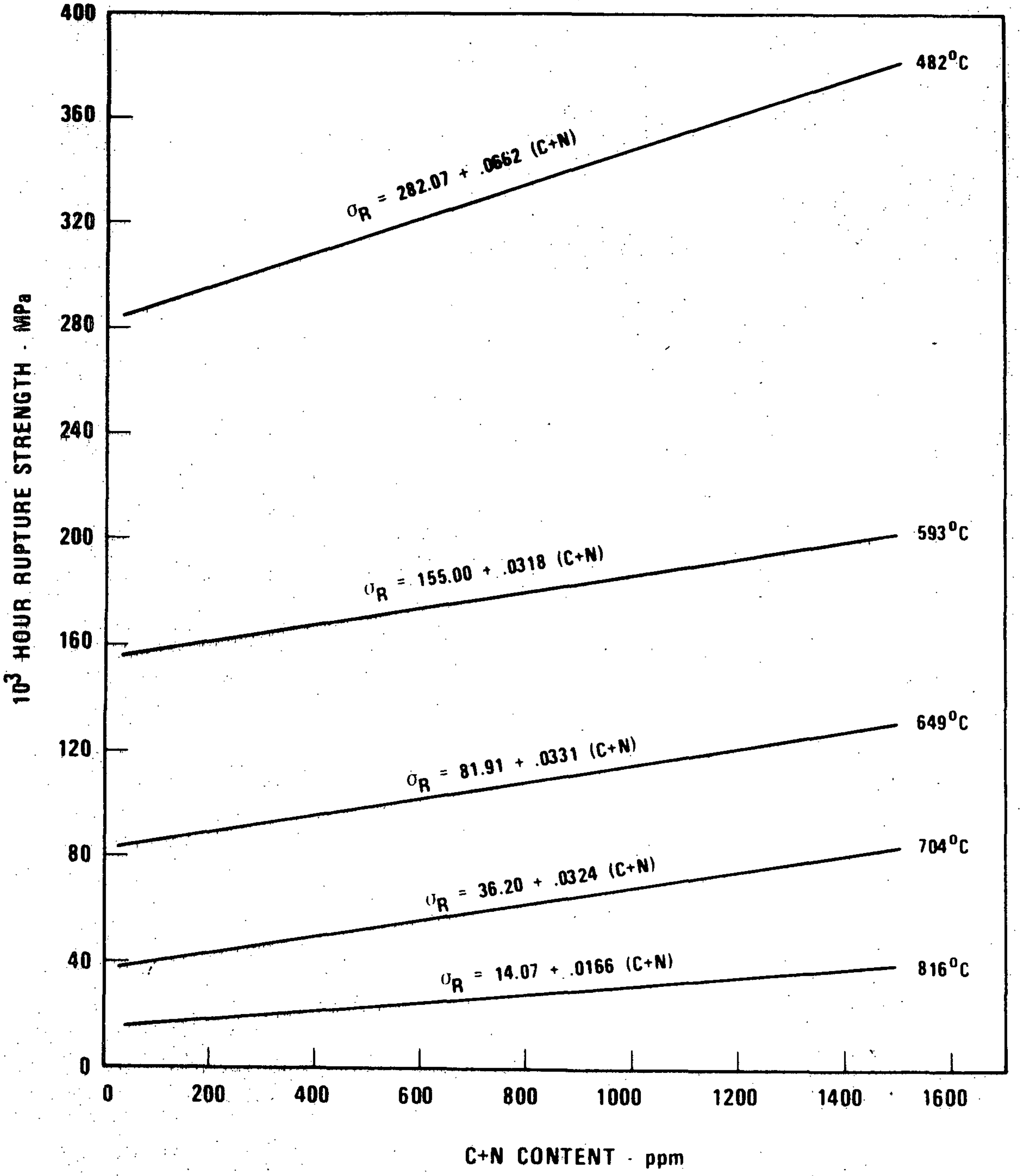


Figure 5.3-3. Effect of C+N Content on the Rupture Strength of Type 304 SS (Taken from Ref. 59)

4425-1

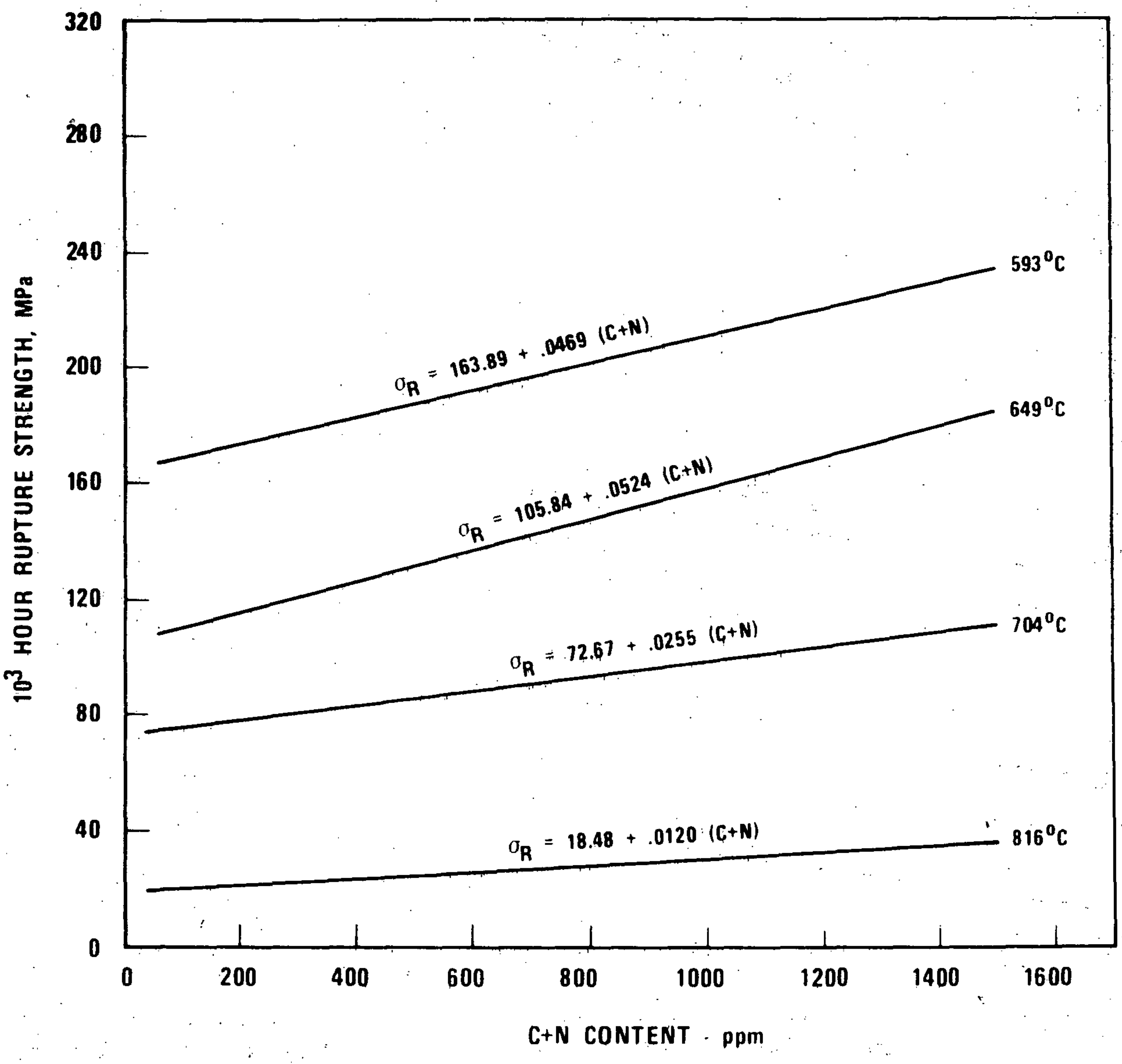


Figure 5.3-4. Effect of C+N on the Rupture Strength of Type 316 SS (Taken from Ref. 59)

4425-2

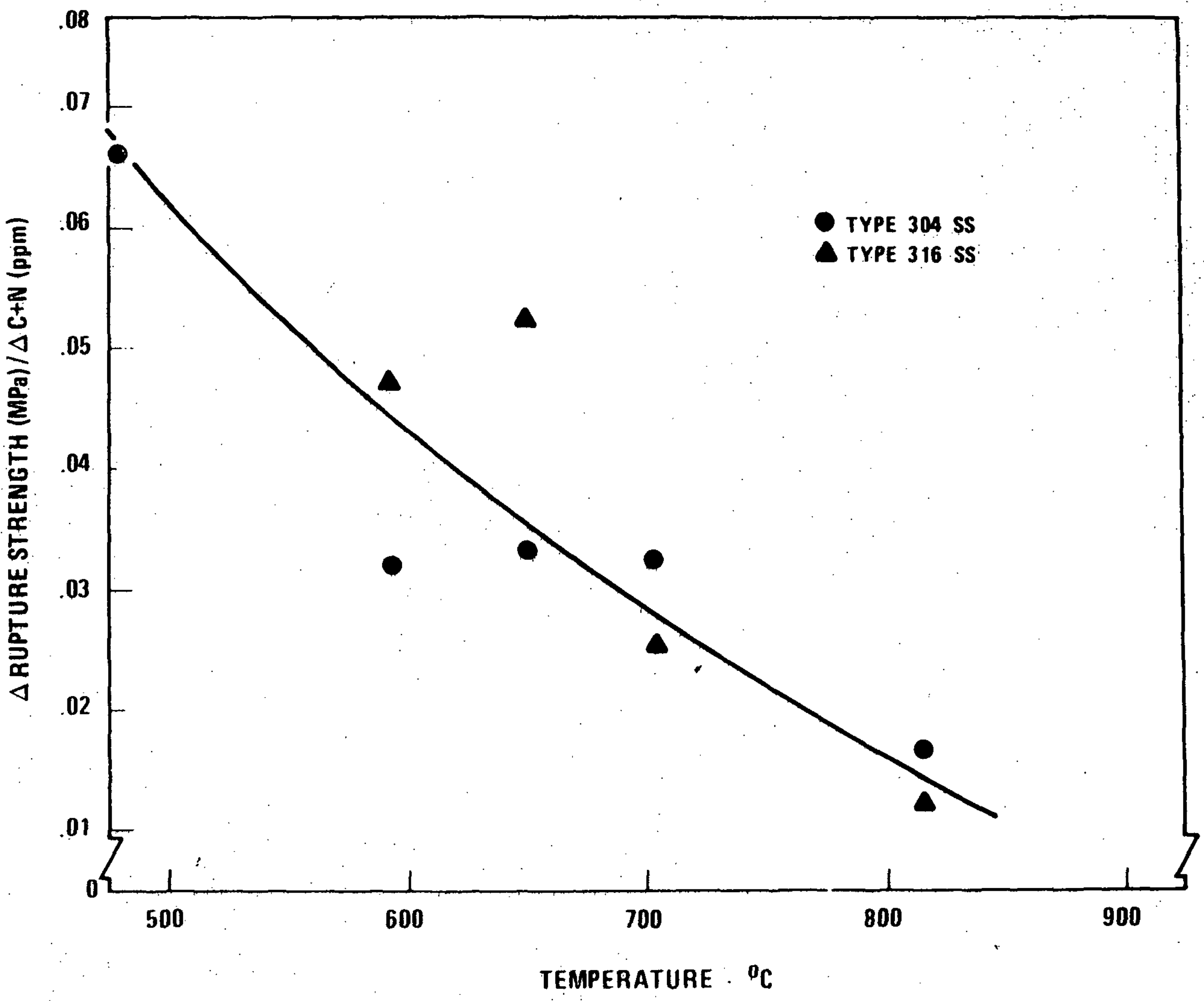


Figure 5.3-5. Effect of C+N Content on the 10^3 Hour Rupture Strength of Types 304 and 316 SS as a Function of Temperature (Taken from Ref. 59)

4425-3

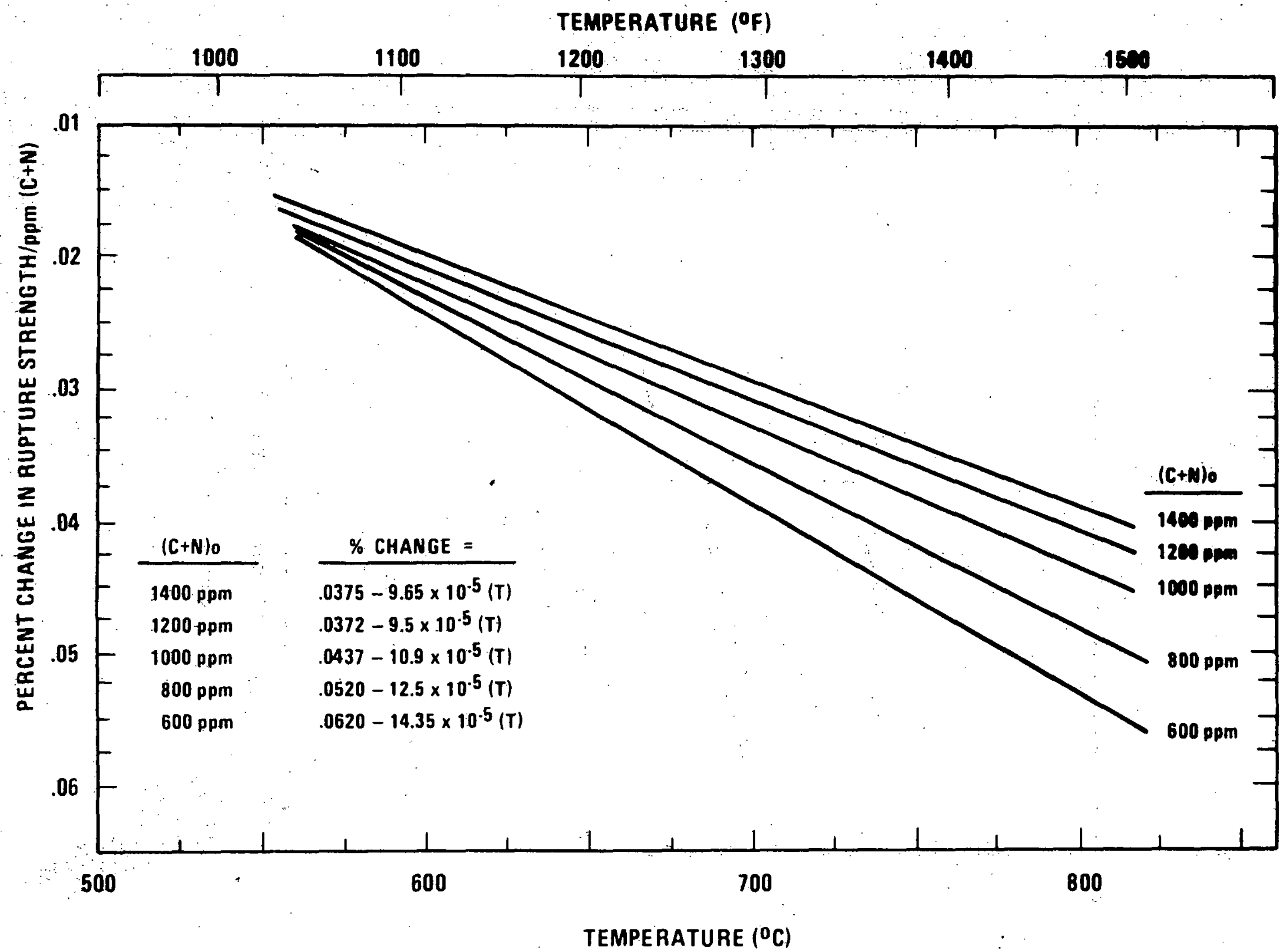


Figure 5.3-6. Change in Rupture Strength of Type 304 SS Due to Change in C+N Content (Taken from Ref. 59)

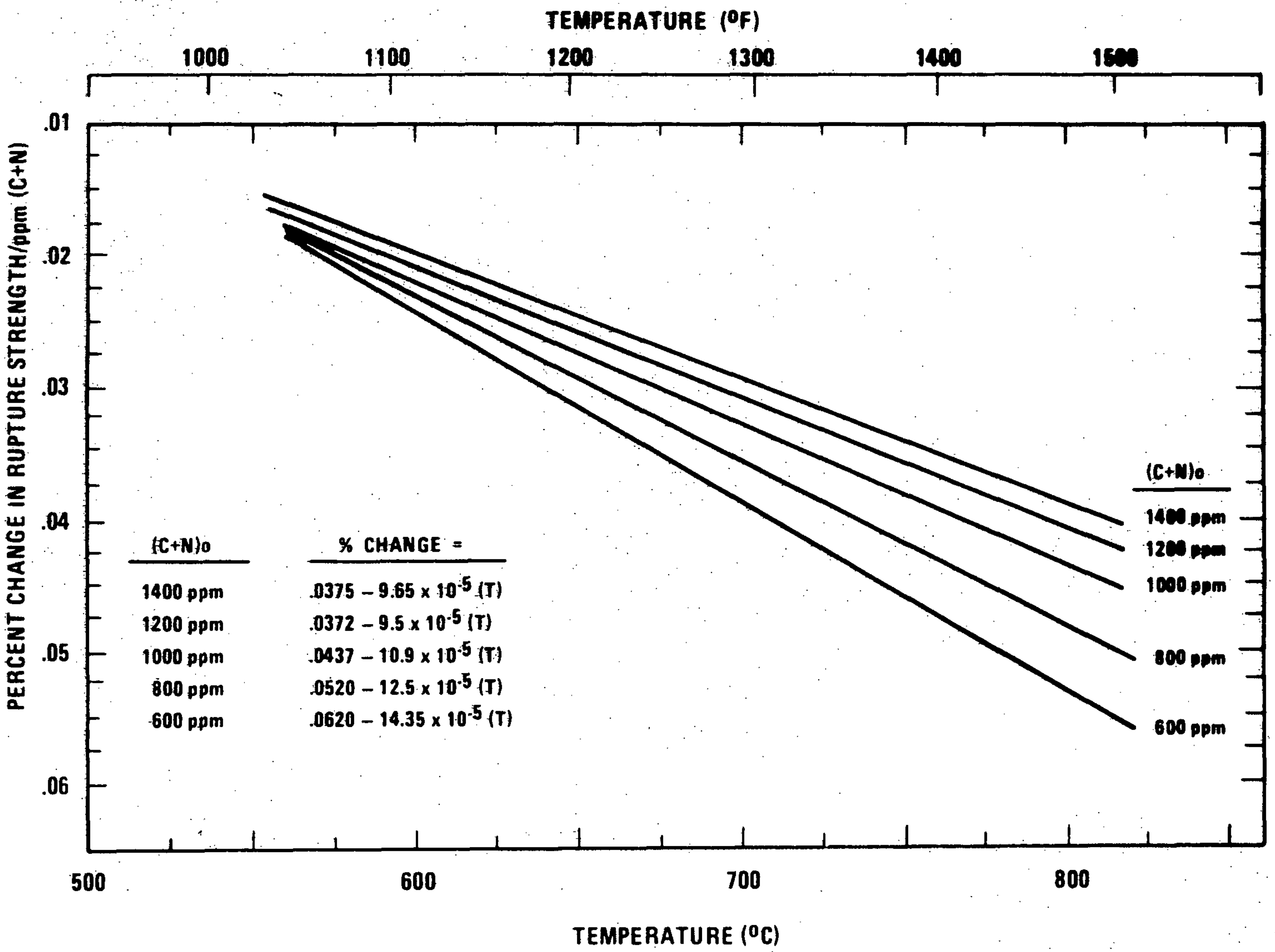


Figure 5.3-7. Change in Rupture Strength of Type 316 SS Due to Change in C+N Content (Taken from Ref. 59)

FIGURE 5.3-8 HAS BEEN DELETED

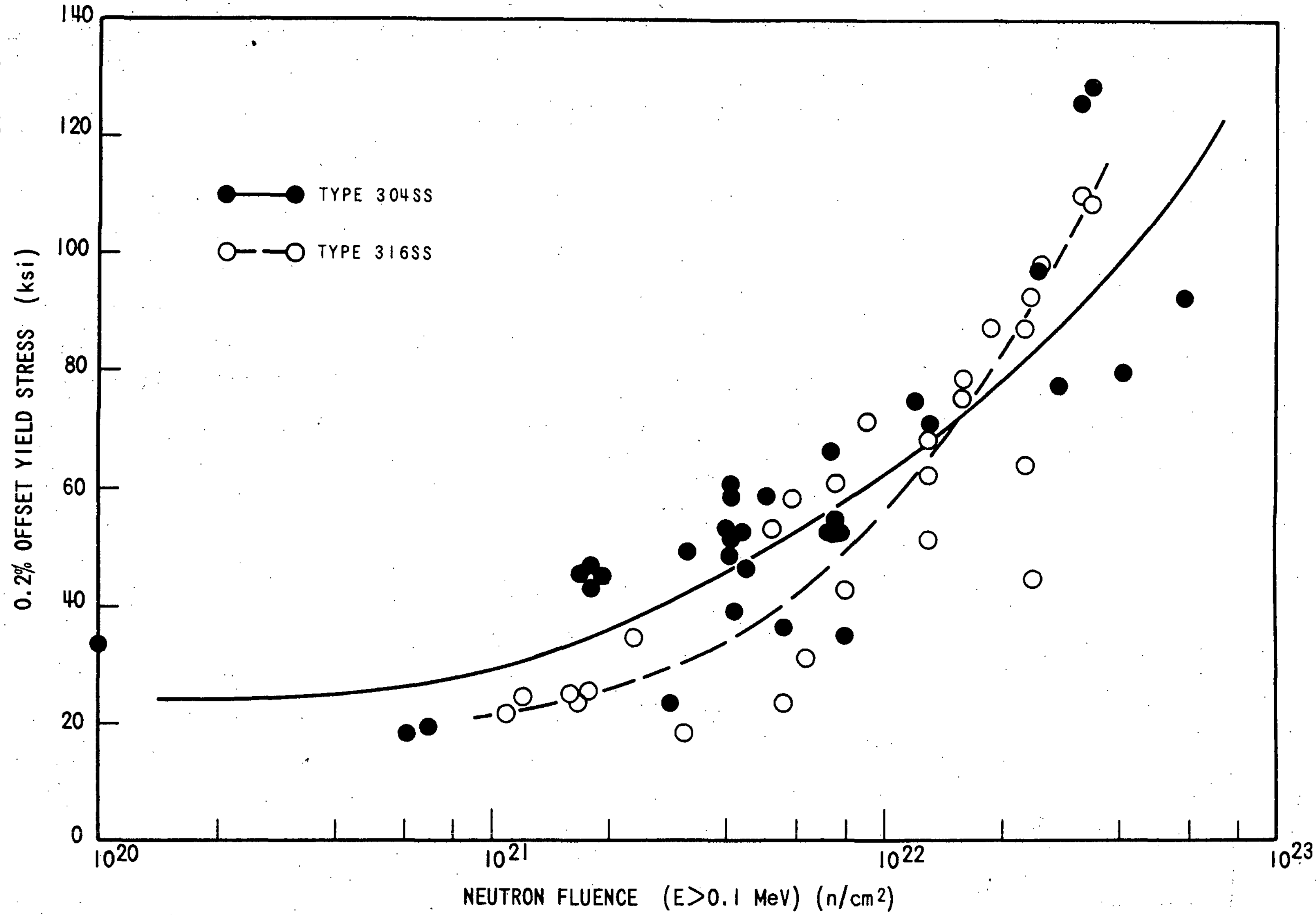


Figure 5.3-9. Yield Strength of Neutron Irradiated Austenitic Stainless Steel at 800°F

* Taken from Reference 57.

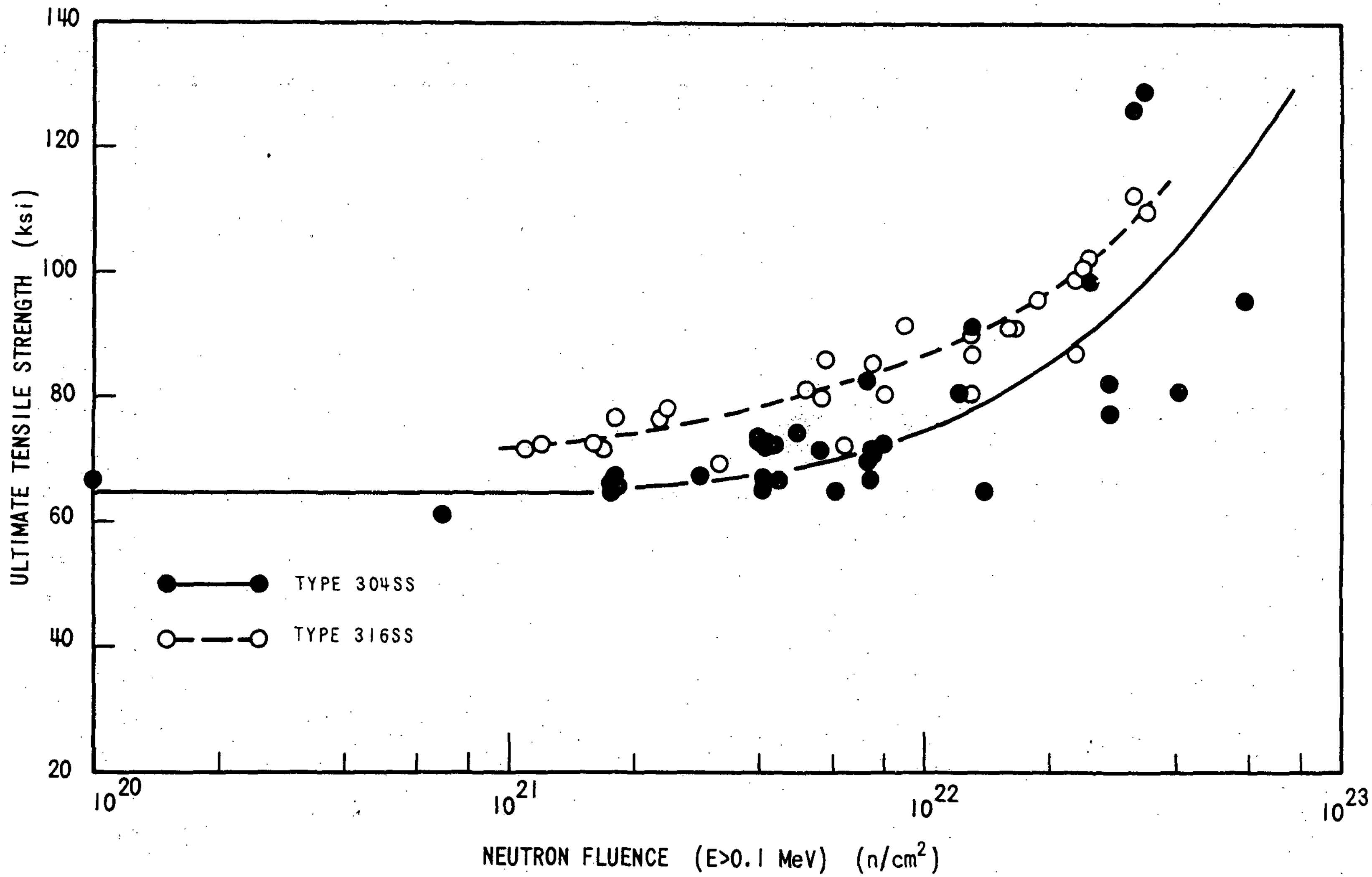


Figure 5.3-10. Ultimate Tensile Strength of Neutron Irradiated Austenitic Stainless Steel at 800°F

* Taken from Reference 57.

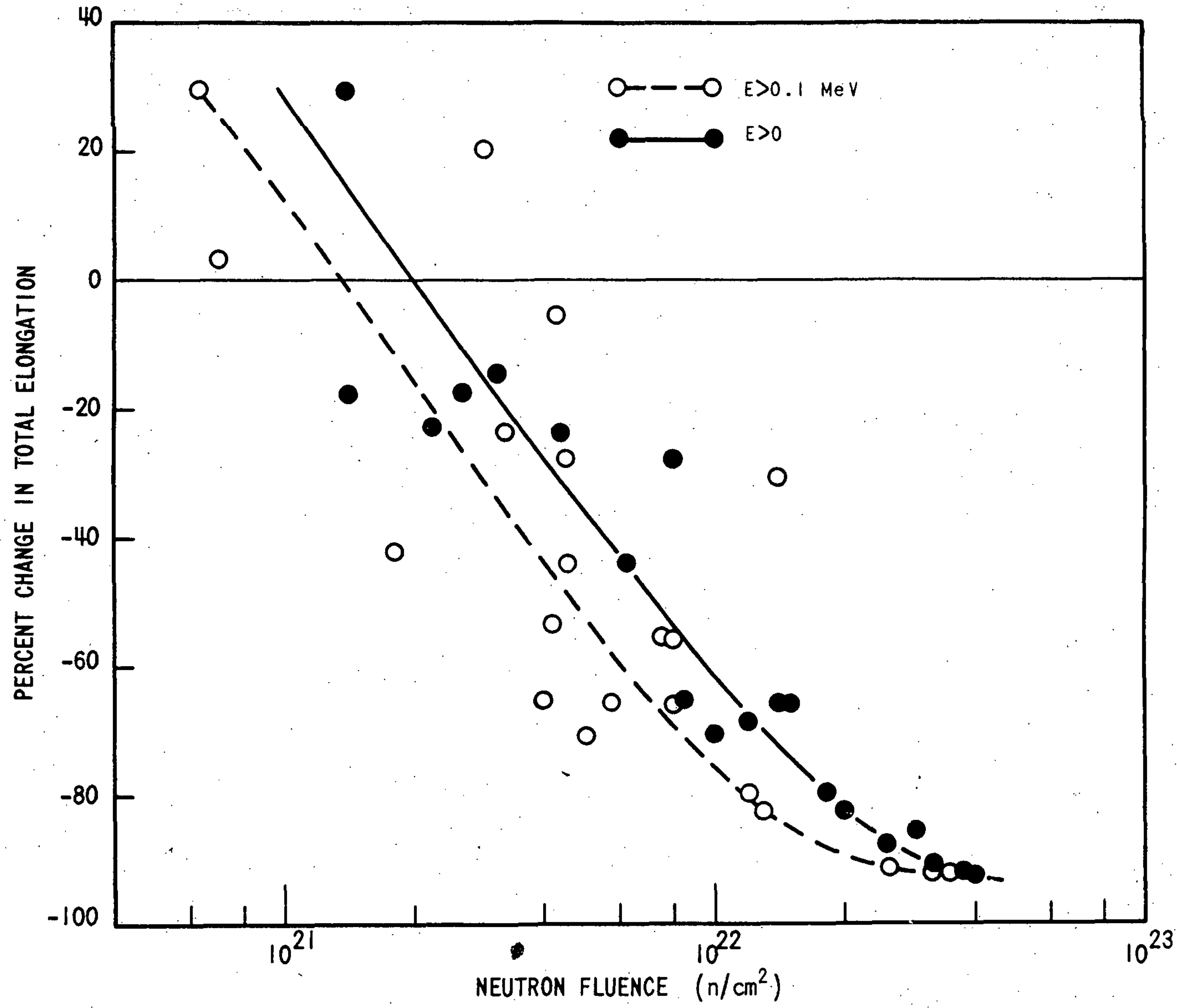


Figure 5.3-11. Irradiation-Induced Ductility Changes in Type 304SS at 800°F

* Taken from Reference 57.

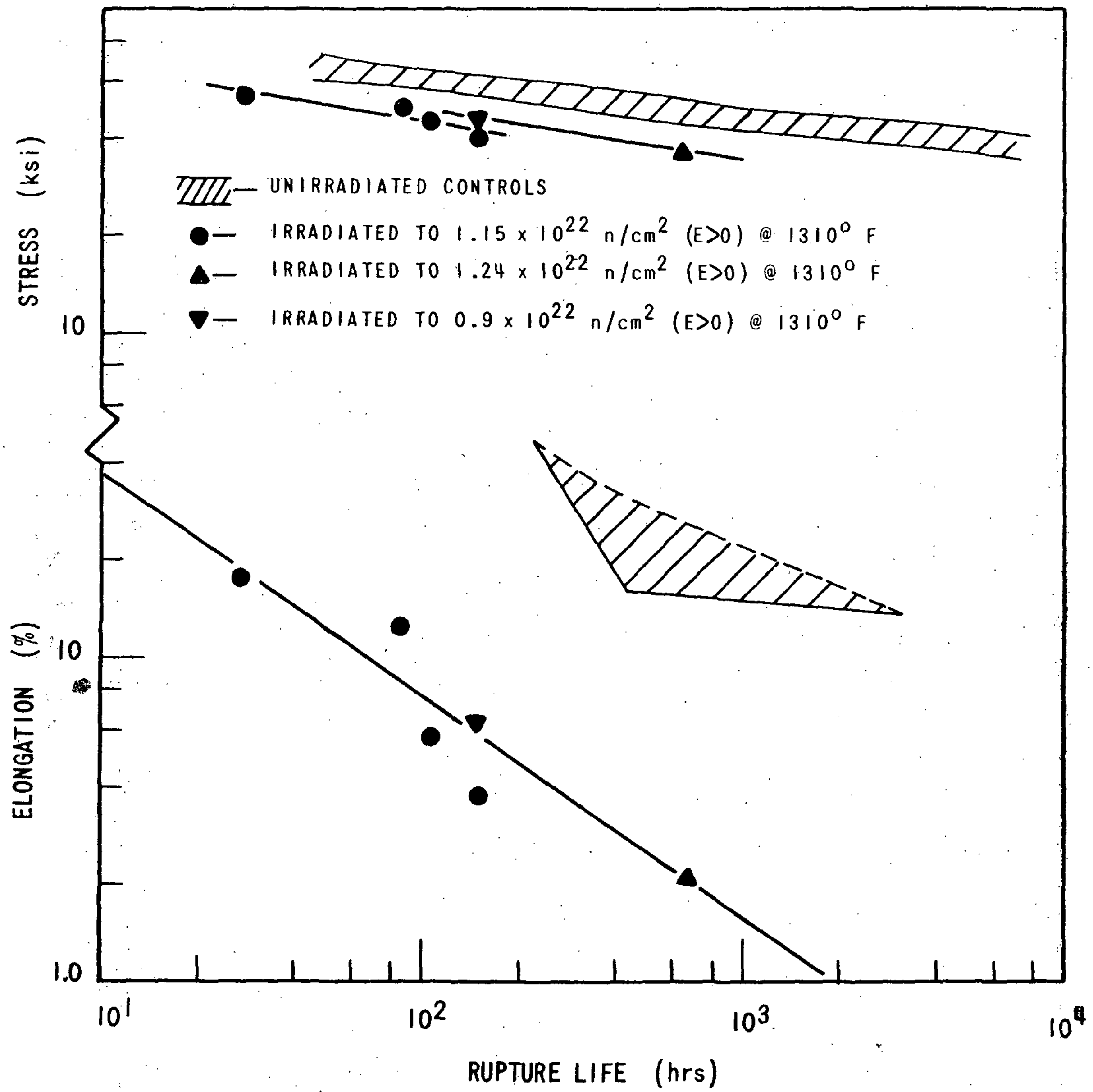


Figure 5.3-12. Creep-Rupture of Irradiated Type 316SS at 1000°F

* Taken from Reference 57.

6669-15

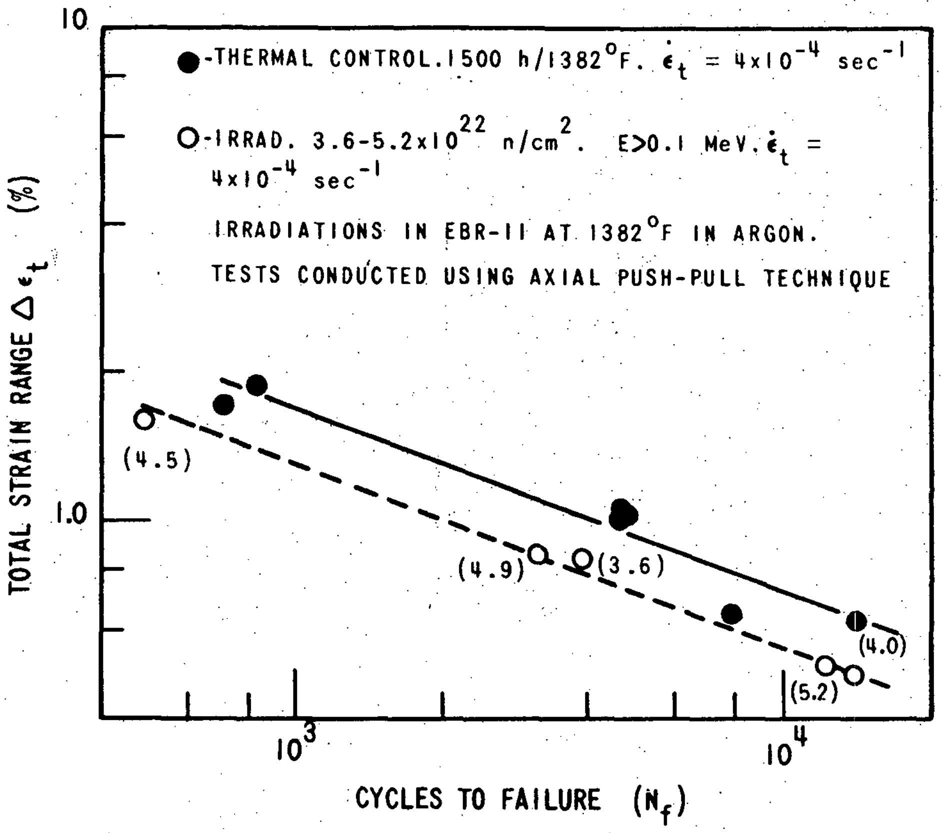


Figure 5.3-13. Effect of Neutron Irradiation on the Fatigue of Type 304SS at 932°F

* Taken from Reference 47. | 25

6669-16

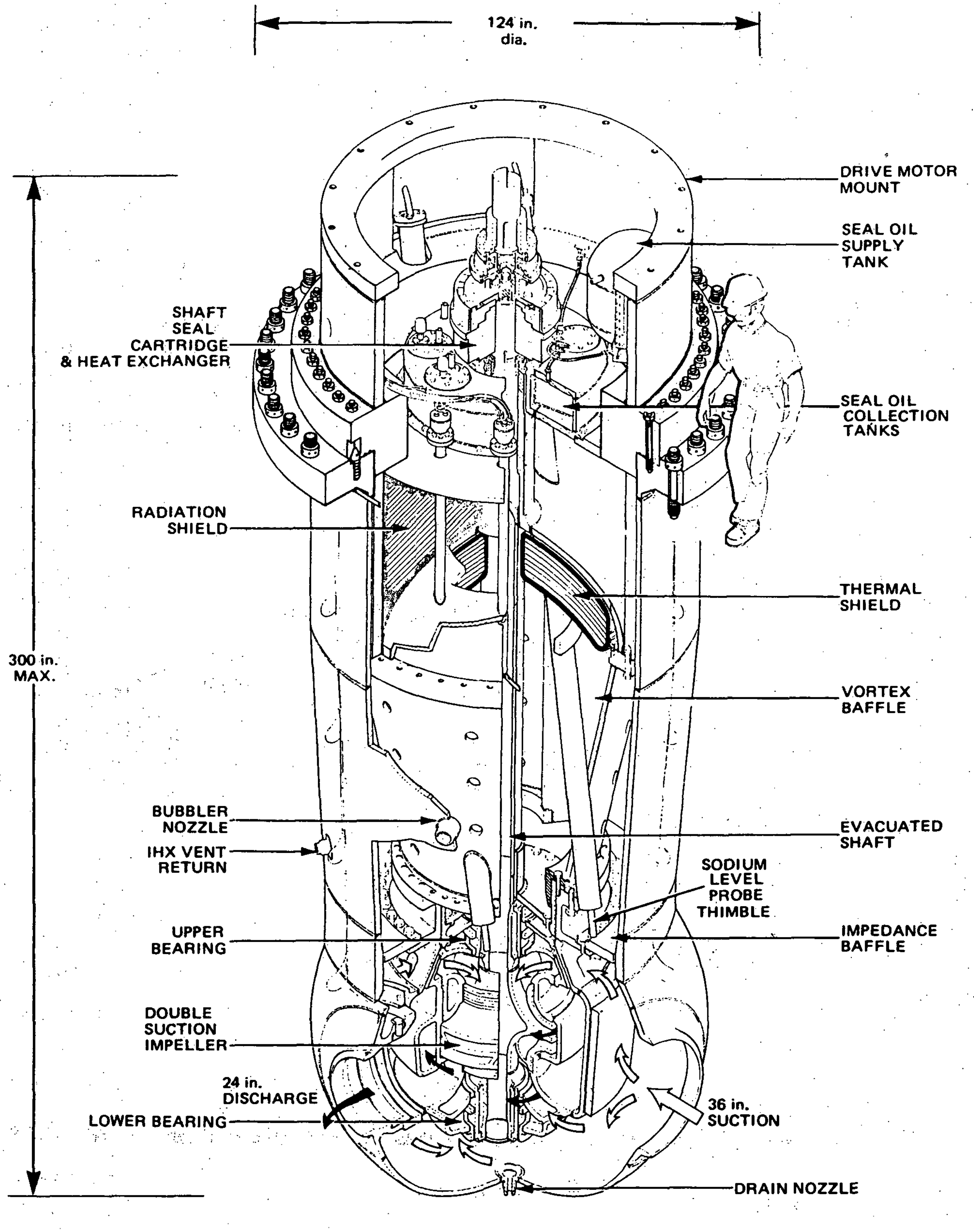


Figure 5.3 - 14 PRIMARY PUMP ISOMETRIC

Amend. 68
May 1982

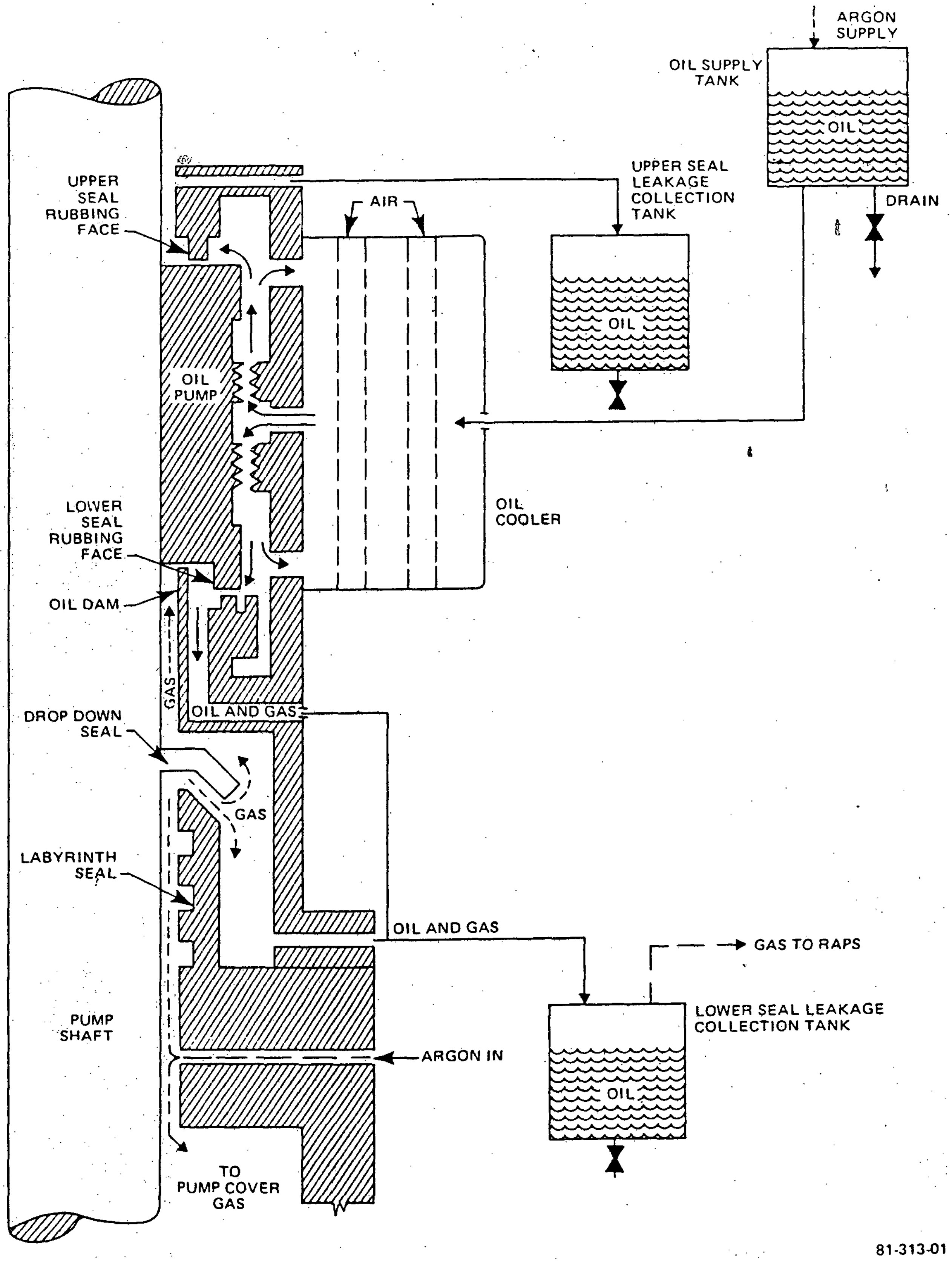


Figure 5.3-14A. Schematic of Typical Seal Oil System

Amend. 62
Nov. 1981

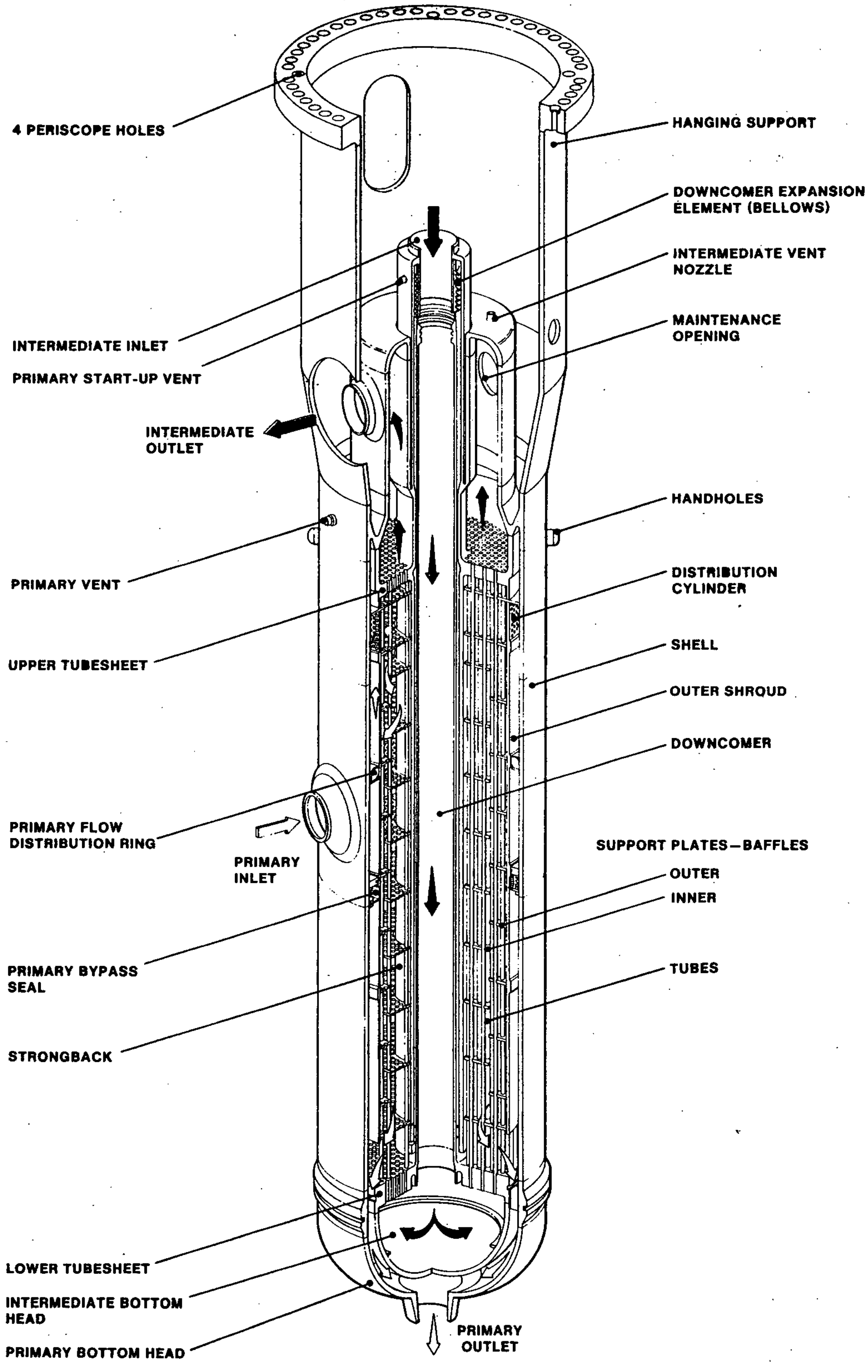


Figure 5.3-15. Intermediate Heat Exchanger (IHX)

4290-1

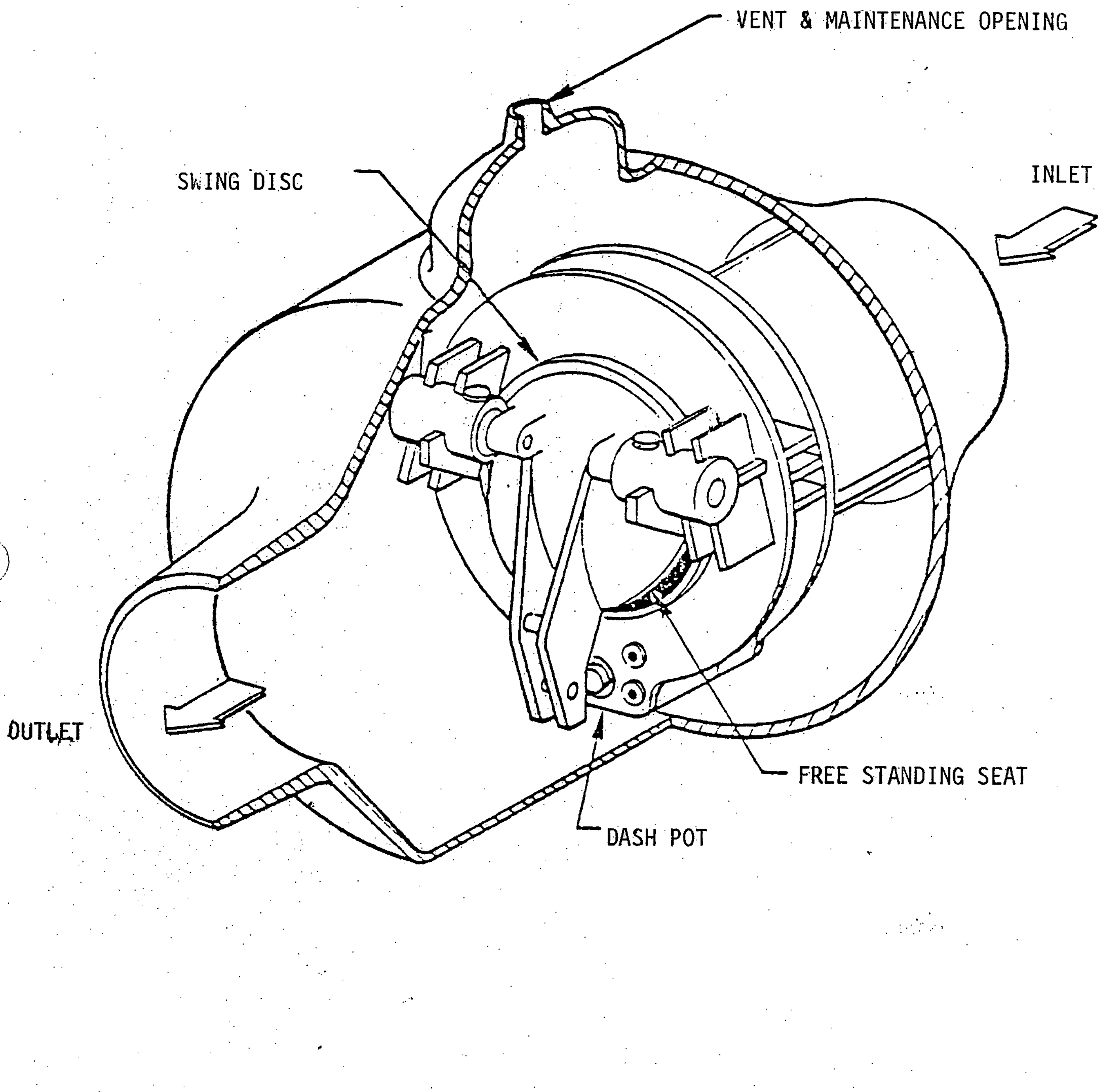


Figure 5.3-16 Preliminary Cold Leg Check Valve Concept

SECTION AREAS CONSIDERED FOR ANALYSIS													
I. ANALYSIS PREPARATION													
A. REVIEW DESIGN SPECIFICATION													
B. REVIEW DESIGN CONCEPT AND MAKE FAILURE MODE ANALYSIS													
C. PREPARE STRUCTURAL EVALUATION PLAN (SEP)													
II. CLASS OF ANALYSIS AND REPORT PREPARATION													
A. PRELIMINARY													
B. INTERIM													
C. FINAL													
D. FORMAL													
III. OPERATING CONDITIONS													
A. DESIGN													
B. NORMAL													
C. UPSET													
D. EMERGENCY													
E. FAULTED													
F. TESTING													
G. SPECIAL													
IV. TYPE OF ANALYSIS													
A. HEAT TRANSFER													
(1) TRANSIENT METAL TEMPERATURE													
(2) STEADY STATE METAL TEMPERATURE													
B. MECHANICAL STRESSES													
(1) PRESSURE LOADS													
(2) DEADWEIGHT													
(3) PIPING LOADS													
(4) EXTERNALLY APPLIED LOADS													
(5) SEISMIC LOADS													
(6) DYNAMIC LOADS													
(7) VIBRATORY LOADS													
C. THERMAL STRESSES													
(1) RADIAL GRADIENT													
(2) LONGITUDINAL GRADIENT													
(3) THERMAL DISCONTINUITY													
D. BUCKLING													
(1) GROSS INSTABILITY													
(2) LOCAL INSTABILITY													
(3) CREEP BUCKLING													
(4) THERMAL WRINKLING													
E. DEFORMATION AND STRAIN LIMITS													
(1) FUNCTIONAL LIMITS													
(2) STRUCTURAL LIMITS													

6669-9

Figure 5.3-17 Sample Structural Design and Analysis Checklist
(Continued on Next Page)

SECTION AREAS CONSIDERED FOR ANALYSIS													
F. CREEP-FATIGUE													
(1) STRESS/STRAIN CONCENTRATIONS													
(2) THERMAL AND LOAD HISTORY													
(3) TIME & CYCLIC CUMULATIVE DAMAGE													
G. STRUCTURAL DYNAMICS													
(1) NATURAL FREQUENCIES & MODAL DATA													
(2) EXCITATION FREQUENCIES													
(3) FLOW INDUCED VIBRATION & FATIGUE													
(4) SEISMIC RESPONSE													
(5) SHOCK													
H. SYSTEM COMPOSITE STRUCTURE													
V METHOD OF ANALYSIS													
A. HAND CALCULATIONS													
B. COMPUTER CODE													
(1) 1-D OR 2-D ELASTIC													
(2) 3-D ELASTIC													
(3) 1-D INELASTIC													
(4) 2-D OR 3-D INELASTIC													
C. EXPERIMENTAL													
VI ADEQUACY OF DESIGN EVALUATION													
A. SECTION III OF ASME BOILER AND BOILER AND PRESSURE VESSEL CODE													
B. CODE CASE 1331													
C. BEIGE CODE													
D. SECTION VIII. DIVISION 2 OF ASME BOILER AND PRESSURE VESSEL CODE													
E. RDT F9-1													
F. RDT F9-4													
G. RDT E15-2T													
VII ENVIRONMENTAL EFFECTS													
A. AIR													
B. SODIUM													
C. ARGON													
D. STEAM													
E. WATER													
F. FLUENCE													
G. TIME AT TEMPERATURE													
H. HELIUM													
I. OTHERS													
VIII DESIGN REVIEWS													
PRELIMINARY													
FINAL													
IX SUPERVISION													

Figure 5.3-17. (Cont.) Sample Structural Design and Analysis Checklist

6669-10

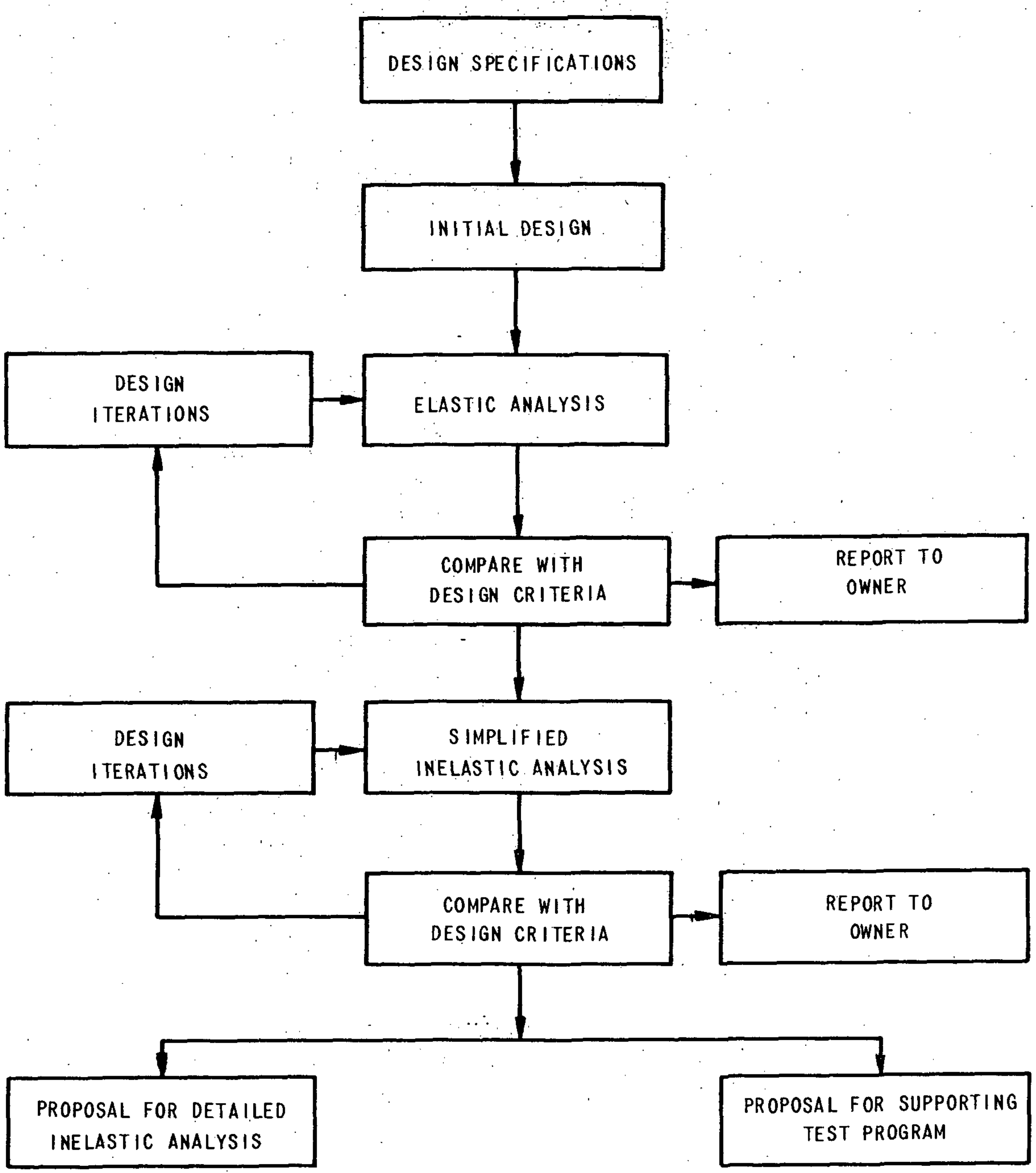


Figure 5.3-18. Sample Screening Analysis

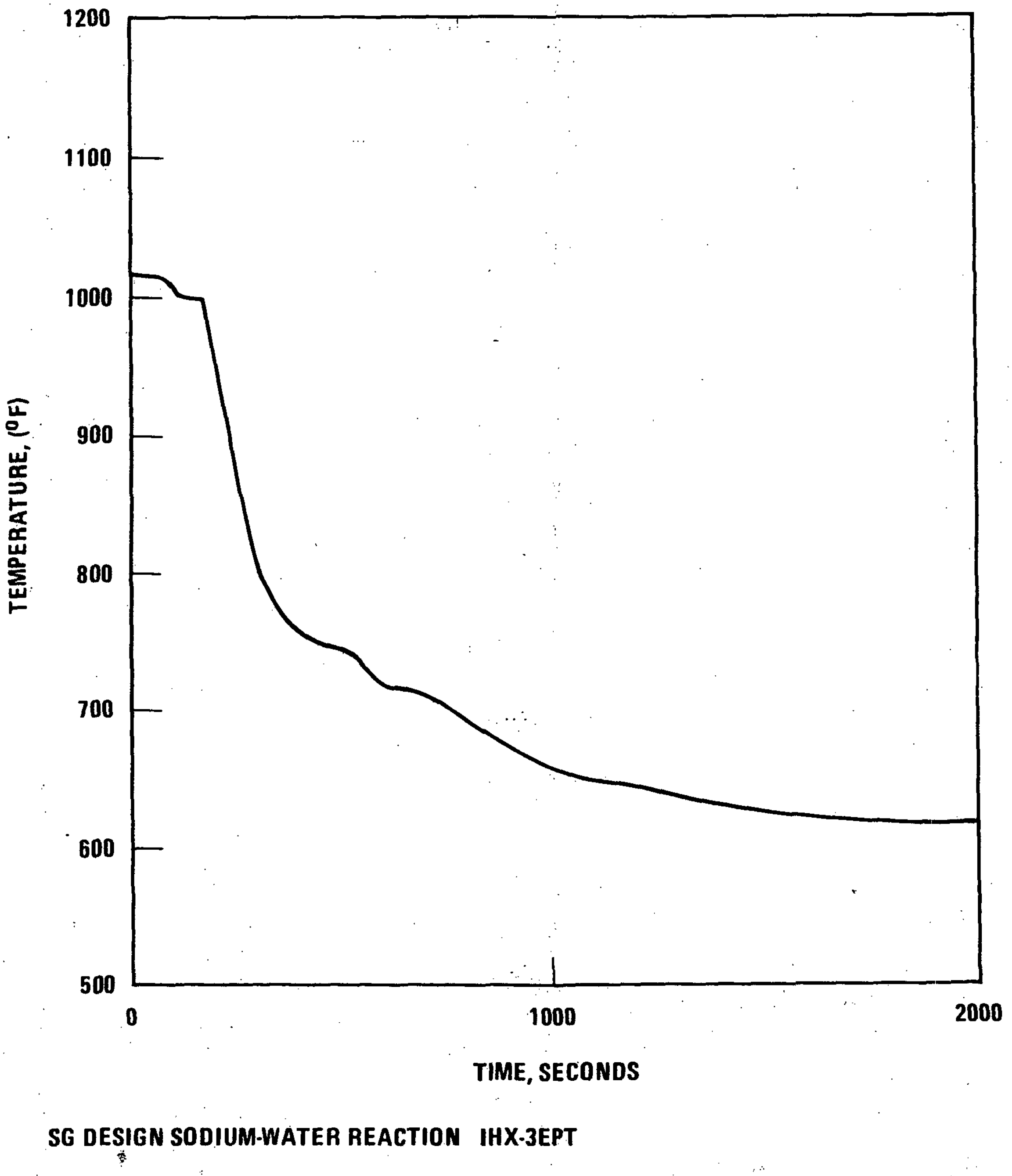


Figure 5.3-18A. Intermediate Heat Exchanger, PRI IHX Inlet Nozzle Temperature-vs-Time

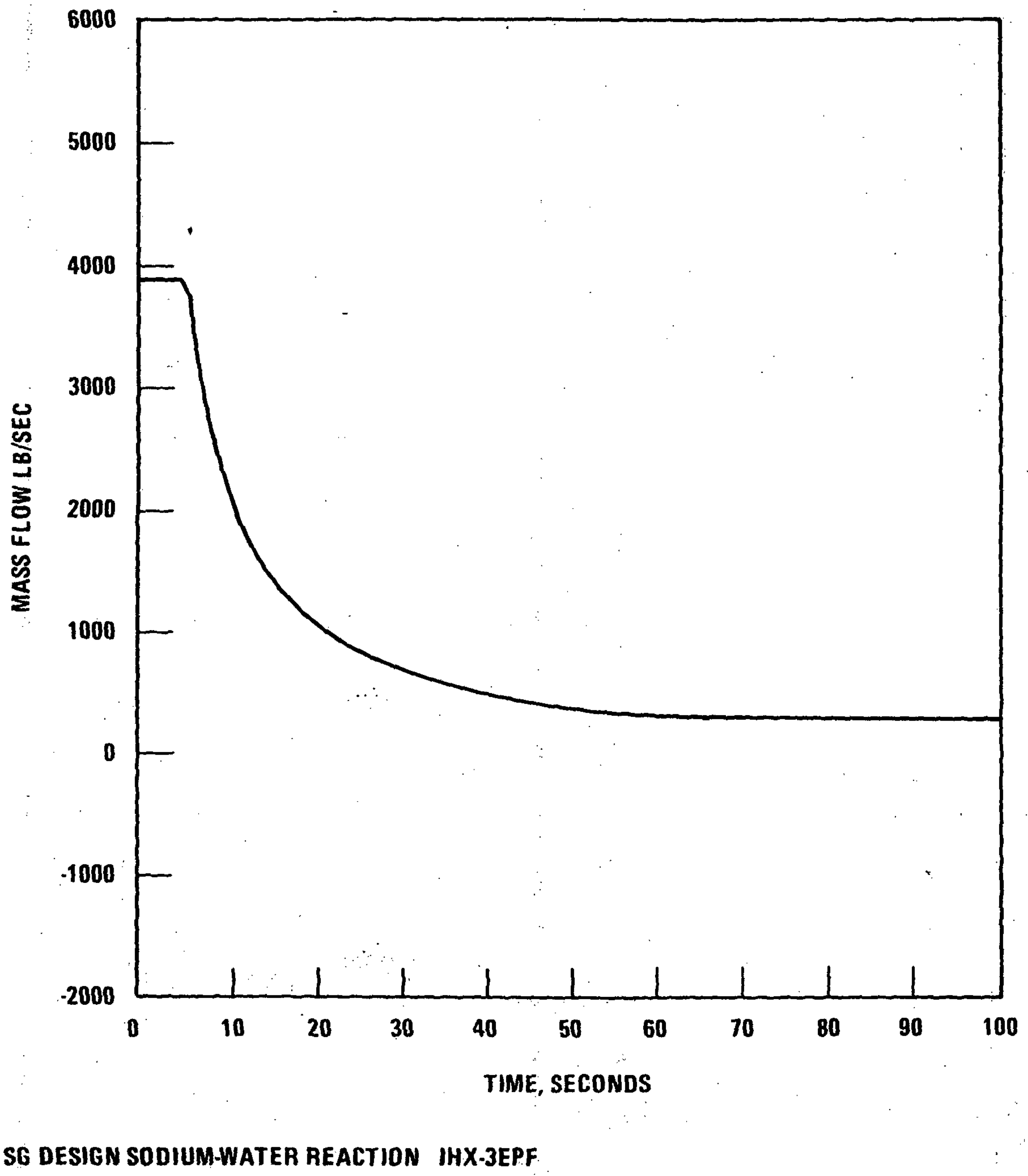
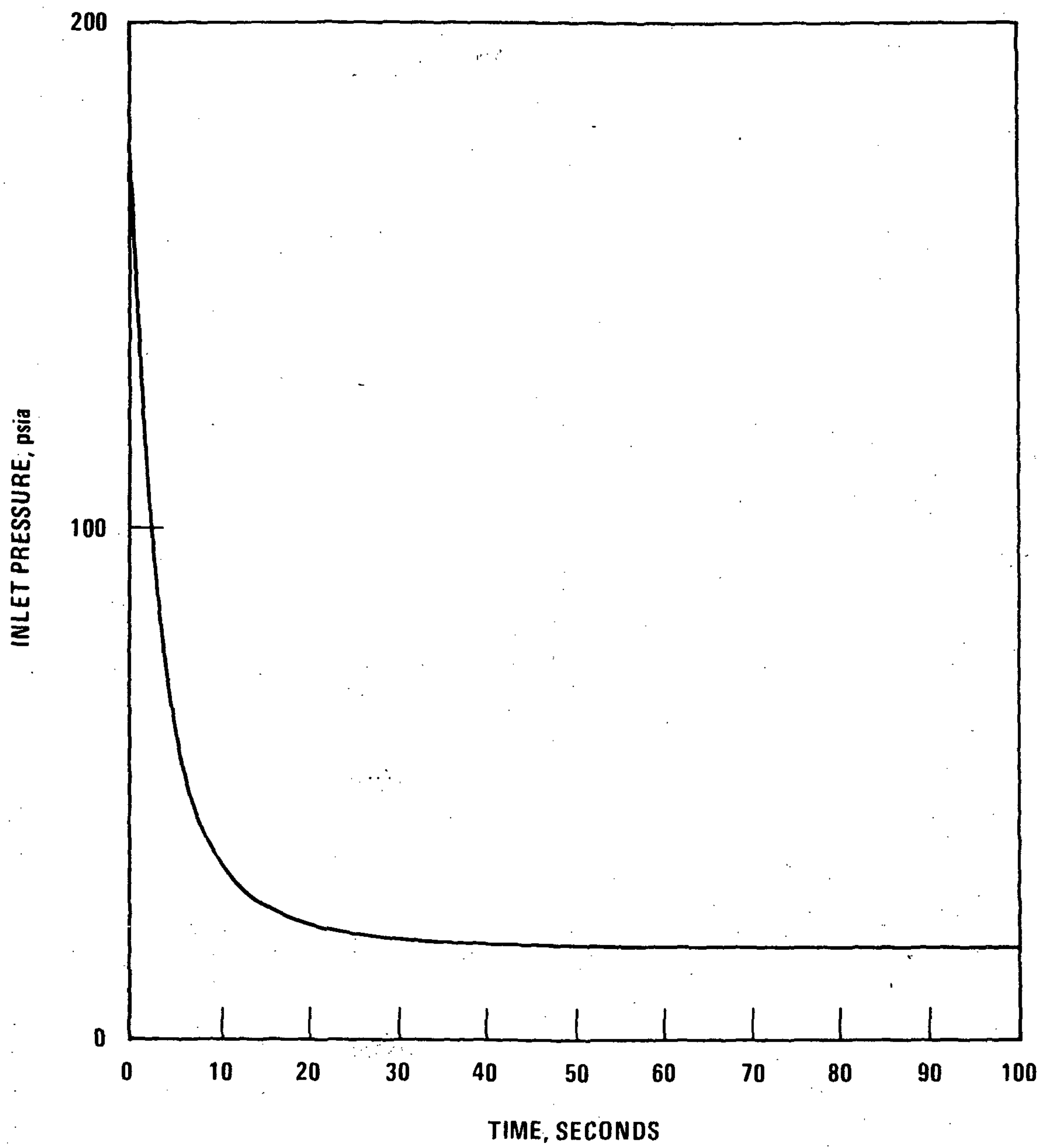


Figure 5.3-18B. Intermediate Heat Exchanger, PRI Pump Ex Massflow-vs-Time



SG DESIGN SODIUM-WATER REACTION IHX-3EPP

Figure 5.3-18C. Intermediate Heat Exchanger, Primary IHX Inlet Pressure-vs-Time

4786-2

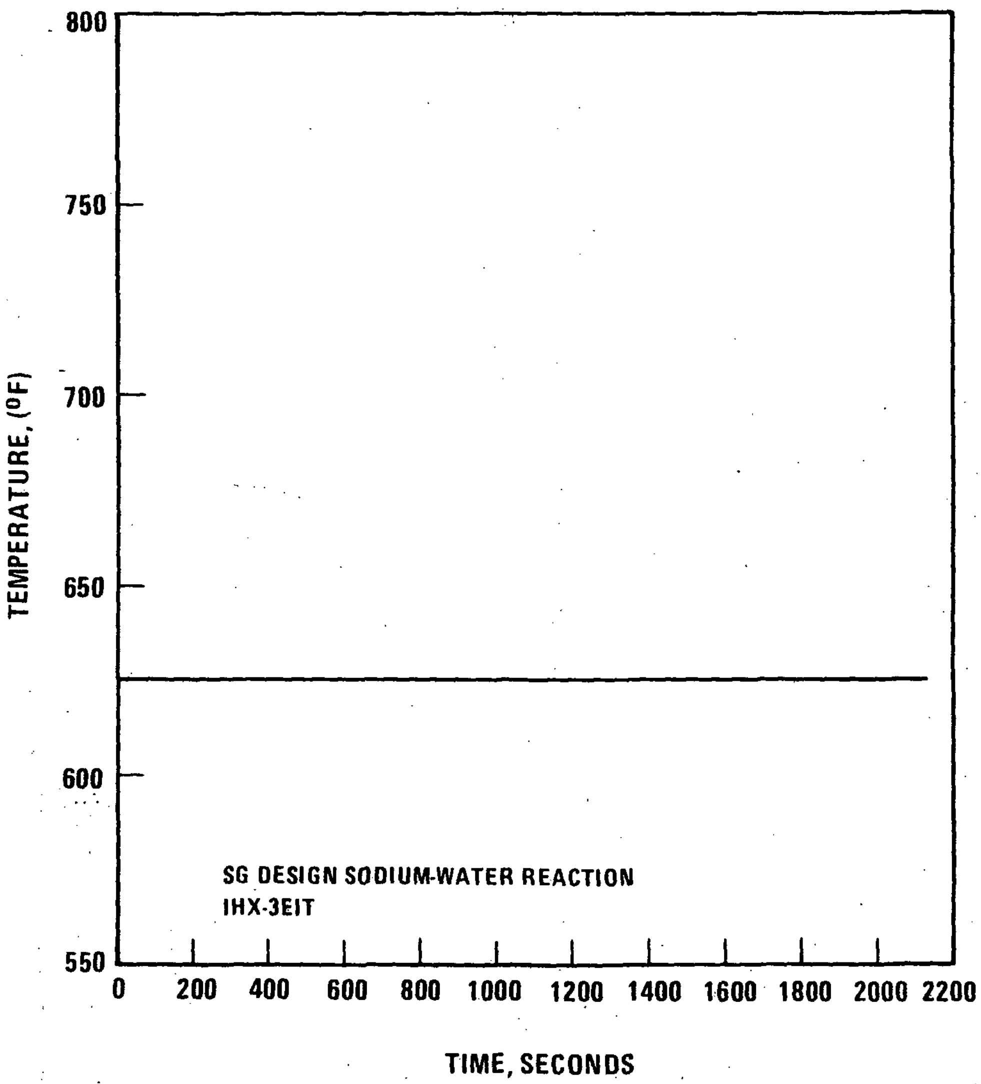


Figure 5.3-18D Intermediate Heat Exchanger, INT IHX Inlet Nozzle Temperature-vs-Time

7683-168

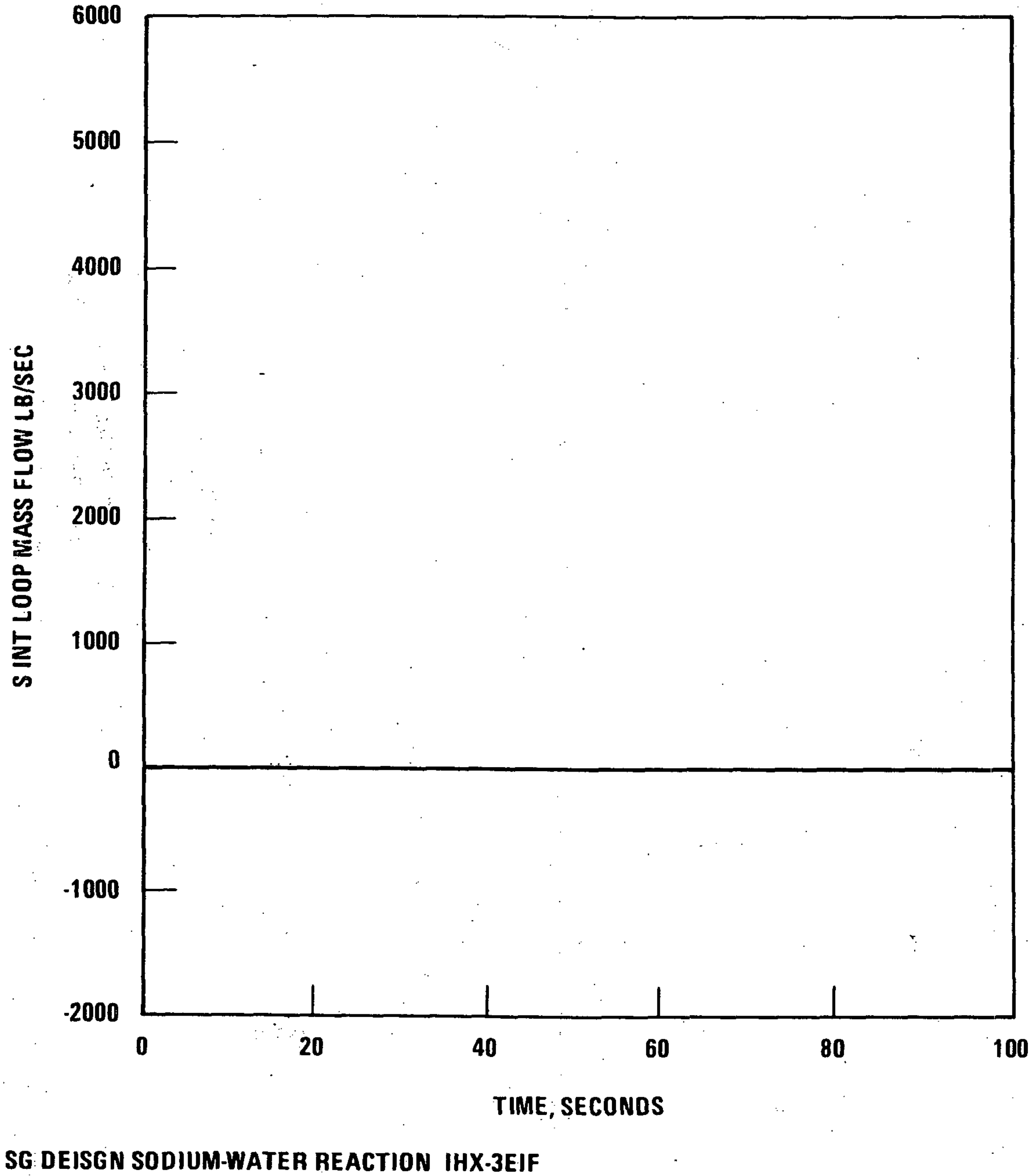


Figure 5.3-18E. Intermediate Heat Exchanger, INT IHX Inlet Nozzle Flow-vs-Time

4786-6

7683-170

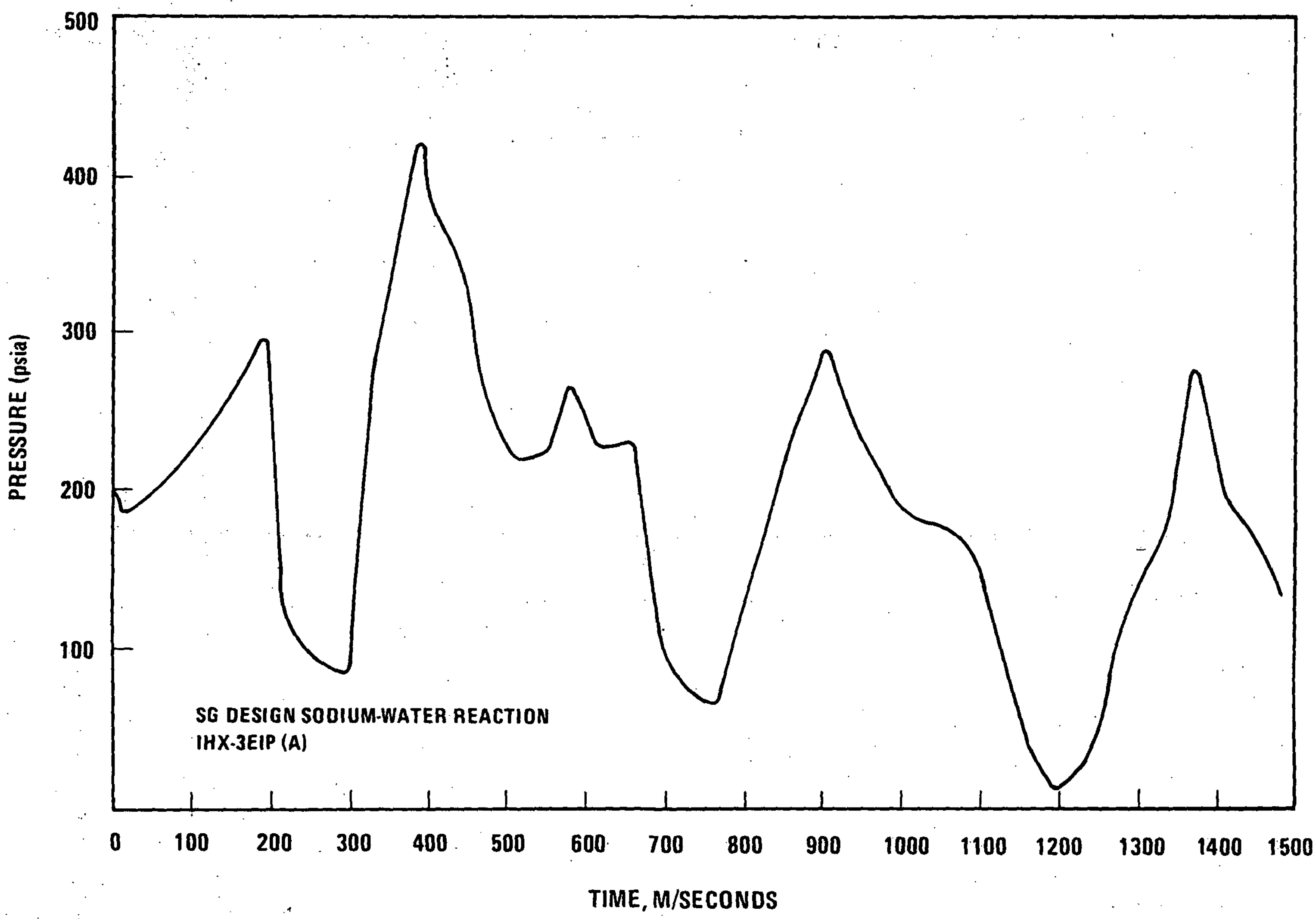


Figure 5.3-18F Intermediate Heat Exchanger, INT IHX Inlet Pressure-vs-Time

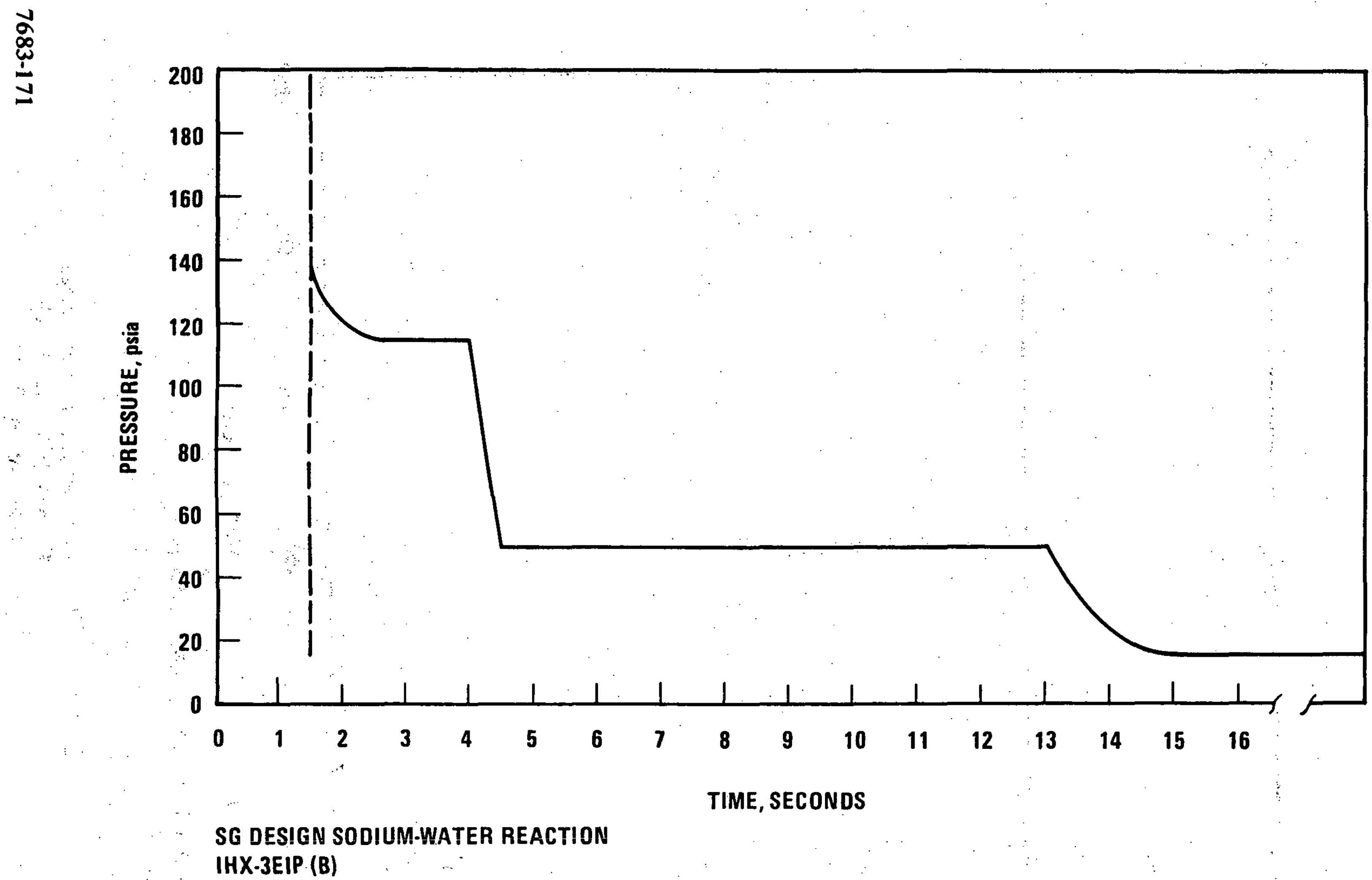
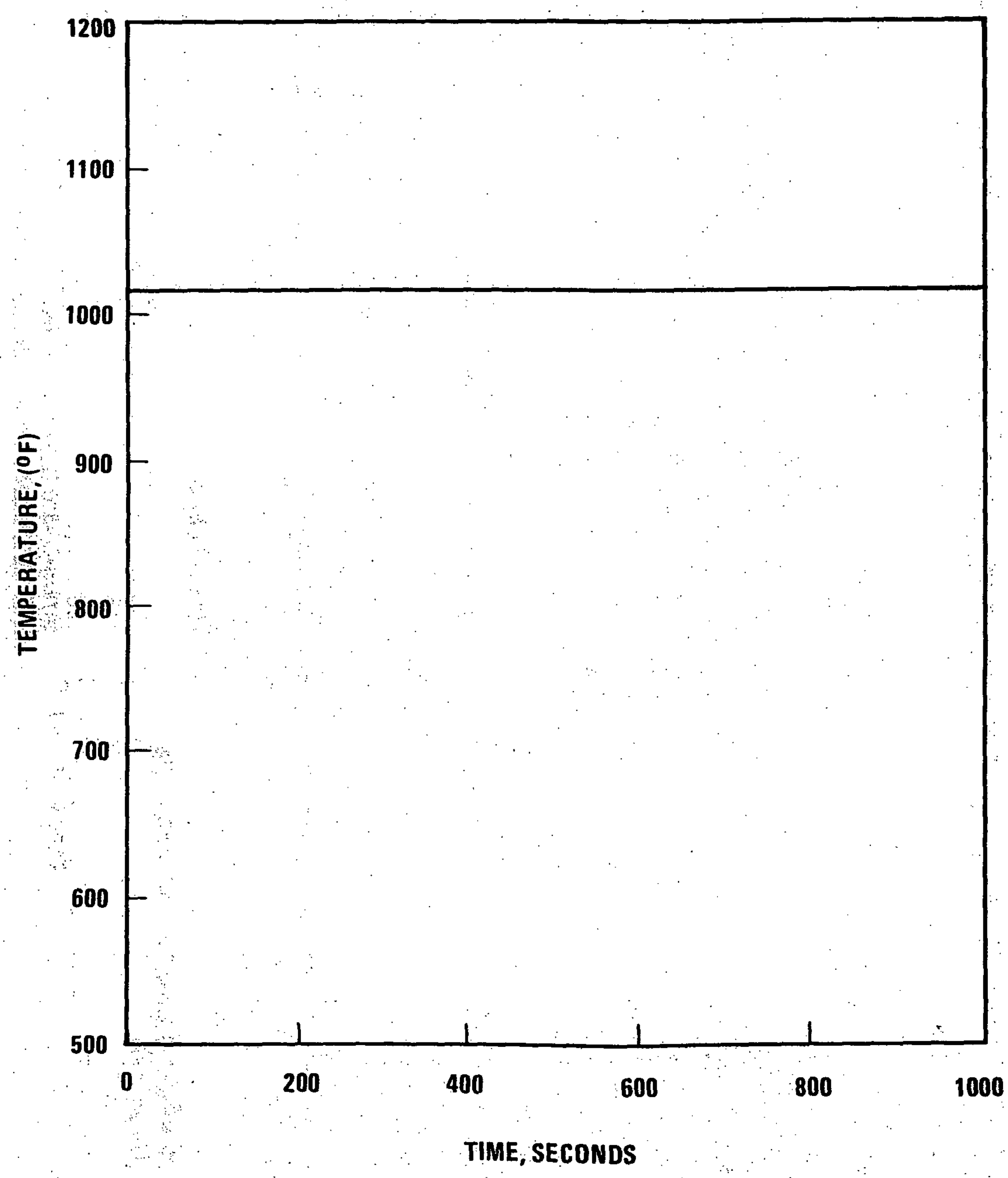


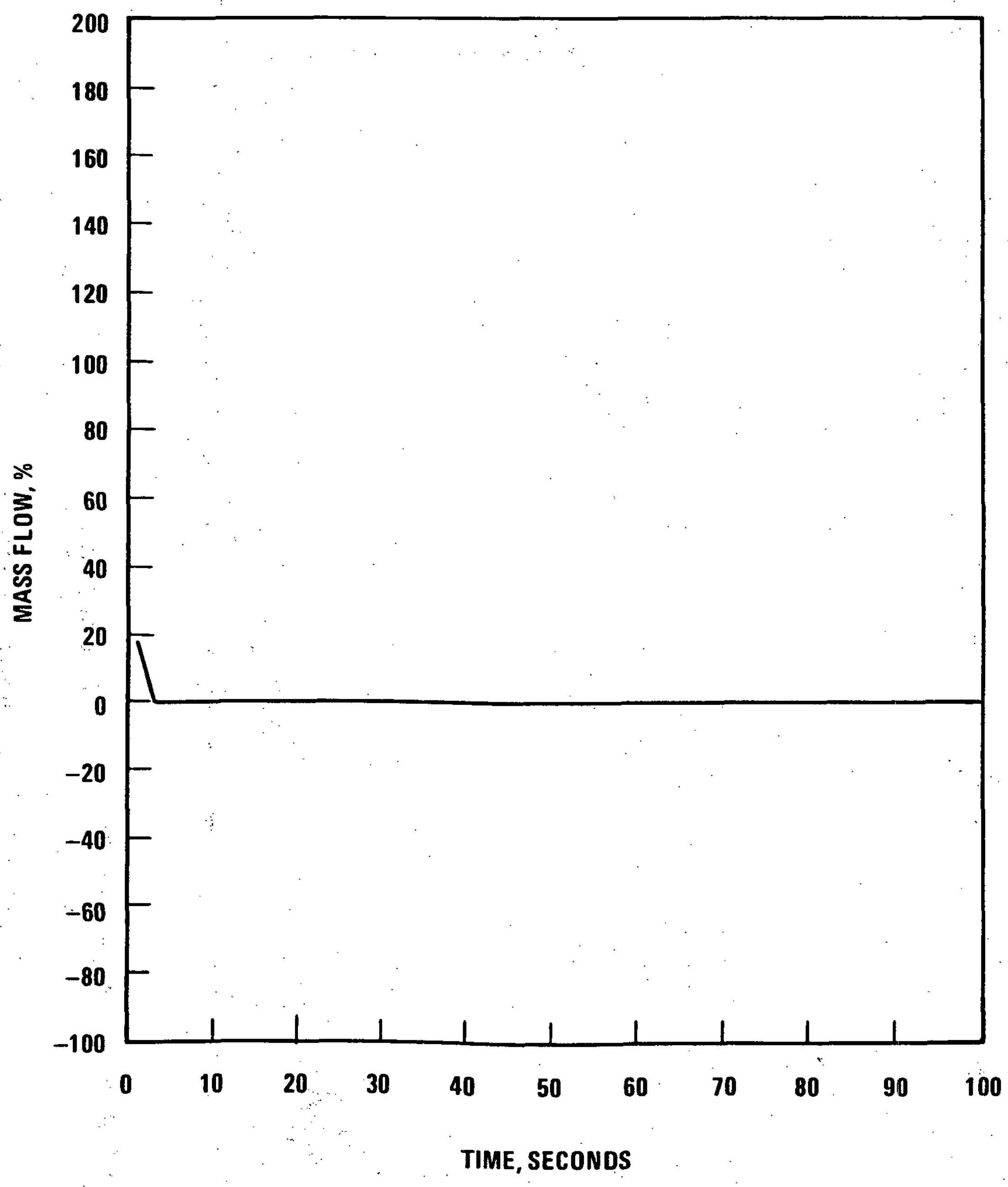
Figure 5.3-18G Intermediate Heat Exchanger,
INT IHX Inlet Nozzle Pressure-vs-Time



E-1 PRIMARY PUMP MECHANICAL FAILURE.
IHX-6EPT

Figure 5.3-18H Intermediate Heat Exchanger, Primary IHX Inlet Nozzle Temperature-vs-Time

7683-172



E-1 PRIMARY PUMP MECHANICAL FAILURE.
IHX-6EPF

Figure 5.3-18I Intermediate Heat Exchanger, Primary Pump Ex Massflow vs Time

7683-173

Amend. 16
Apr. 1976

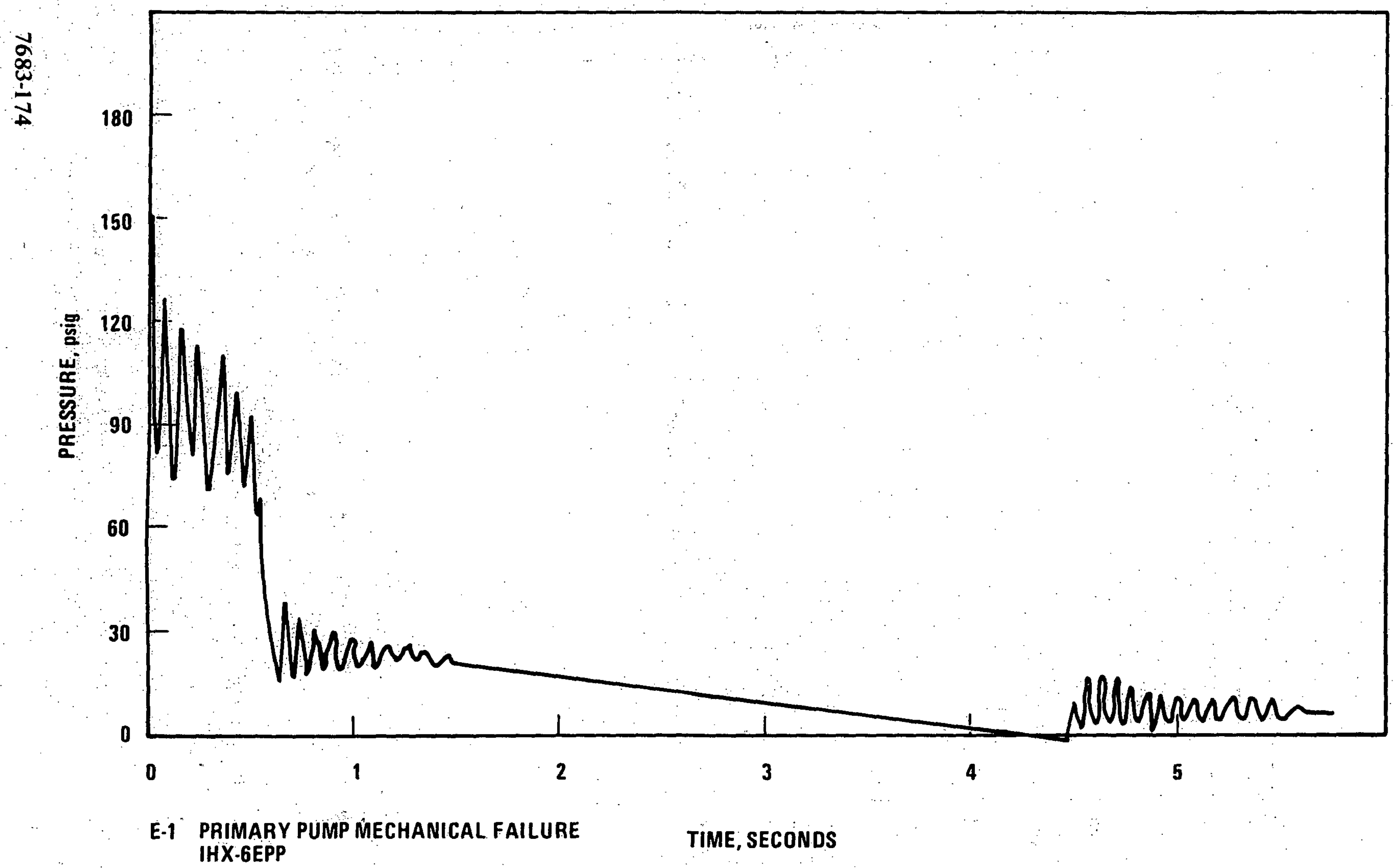
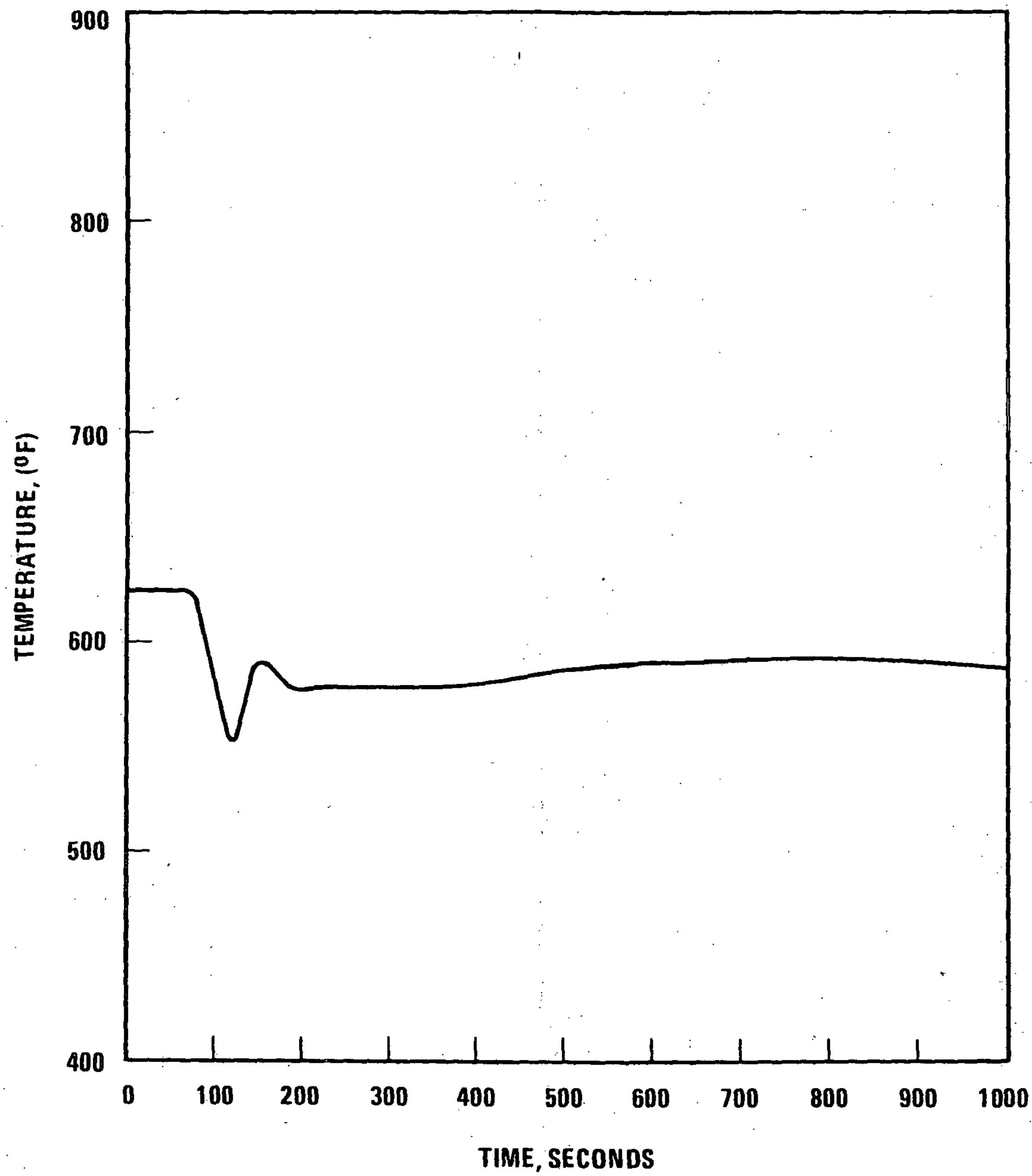


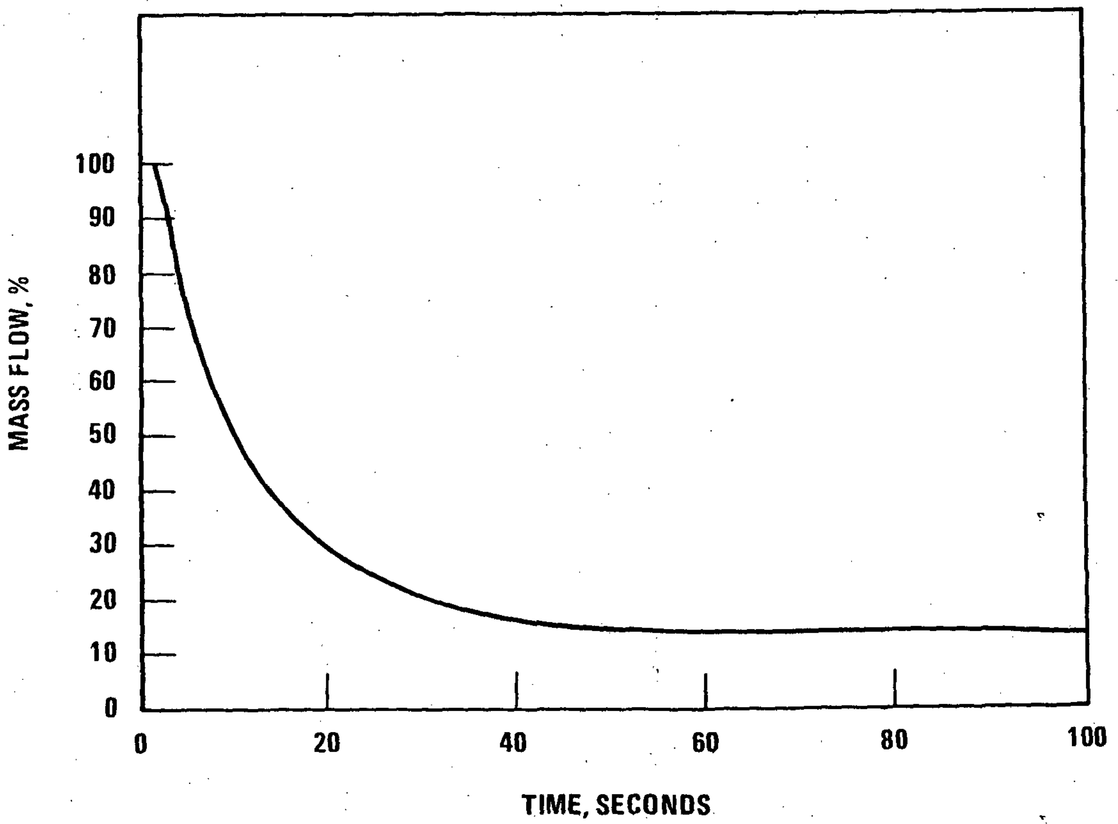
Figure 5.3-18J Intermediate Heat Exchanger, IHX Primary Outlet Pressure-vs-Time



E-1 PRIMARY PUMP MECHANICAL FAILURE.
IHX-6EIT

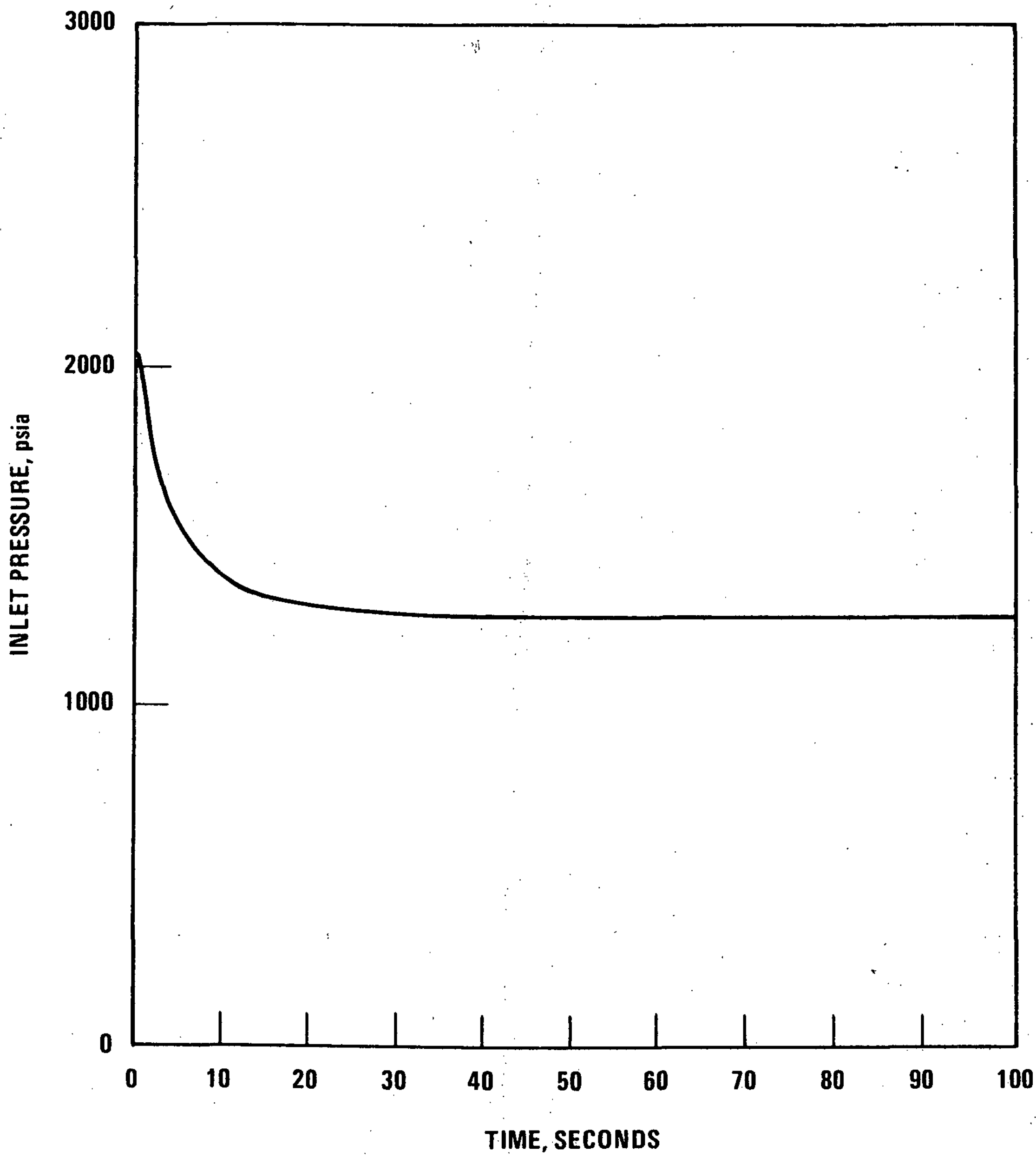
Figure 5.3-18K Intermediate Heat Exchanger, INT IHX Inlet Nozzle Temperature-vs-Time

7683-175



E-1 PRIMARY PUMP MECHANICAL FAILURE. IHX-GEIF

Figure 5.3-18L. Intermediate Heat Exchanger, INT Loop Massflow vs. Time



E-1 PRIMARY PUMP MECHANICAL FAILURE. IHX-GEIP

Figure 5.3-18M. Intermediate Heat Exchanger, INT IHX Pressure vs. Time

4786-5

Table 5.3-18n has been intentionally deleted.

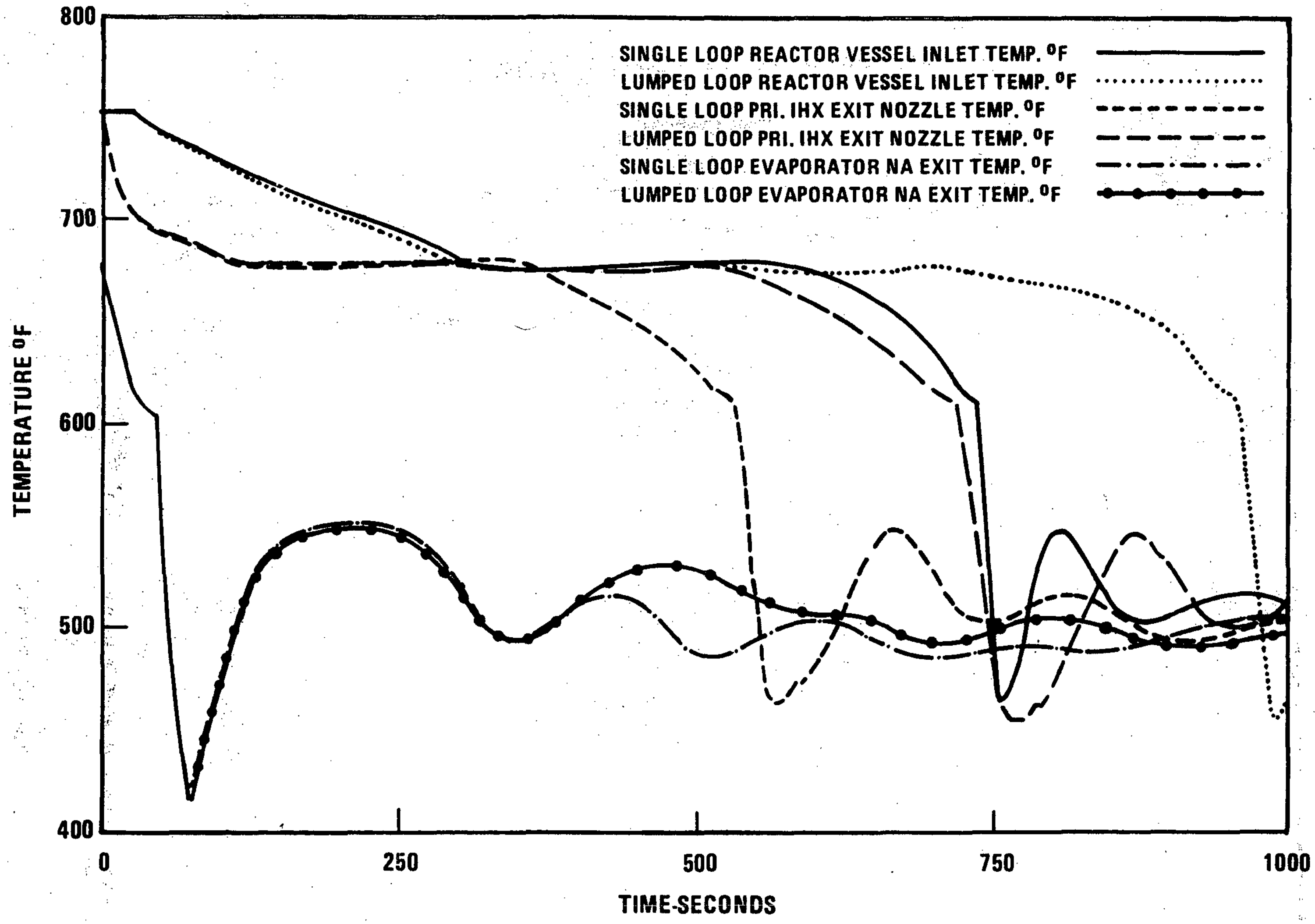


Figure 5.3-18 P. Natural Circulation, 3 To 3 Loop Case, Temperature-vs-Time

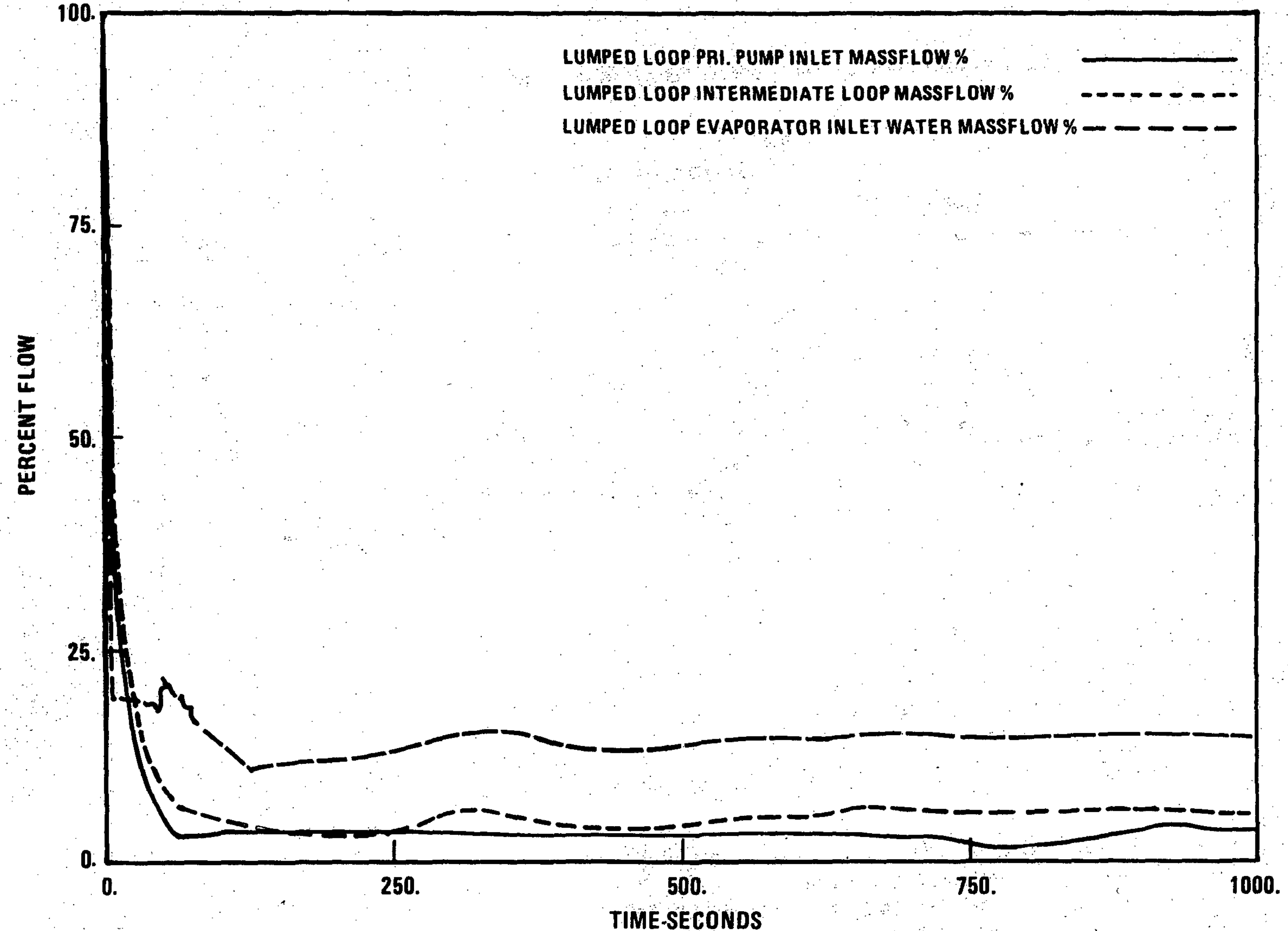


Figure 5.3-18 Q Natural Circulation, 3 To 3 Loop Case, Flow-vs.-Time

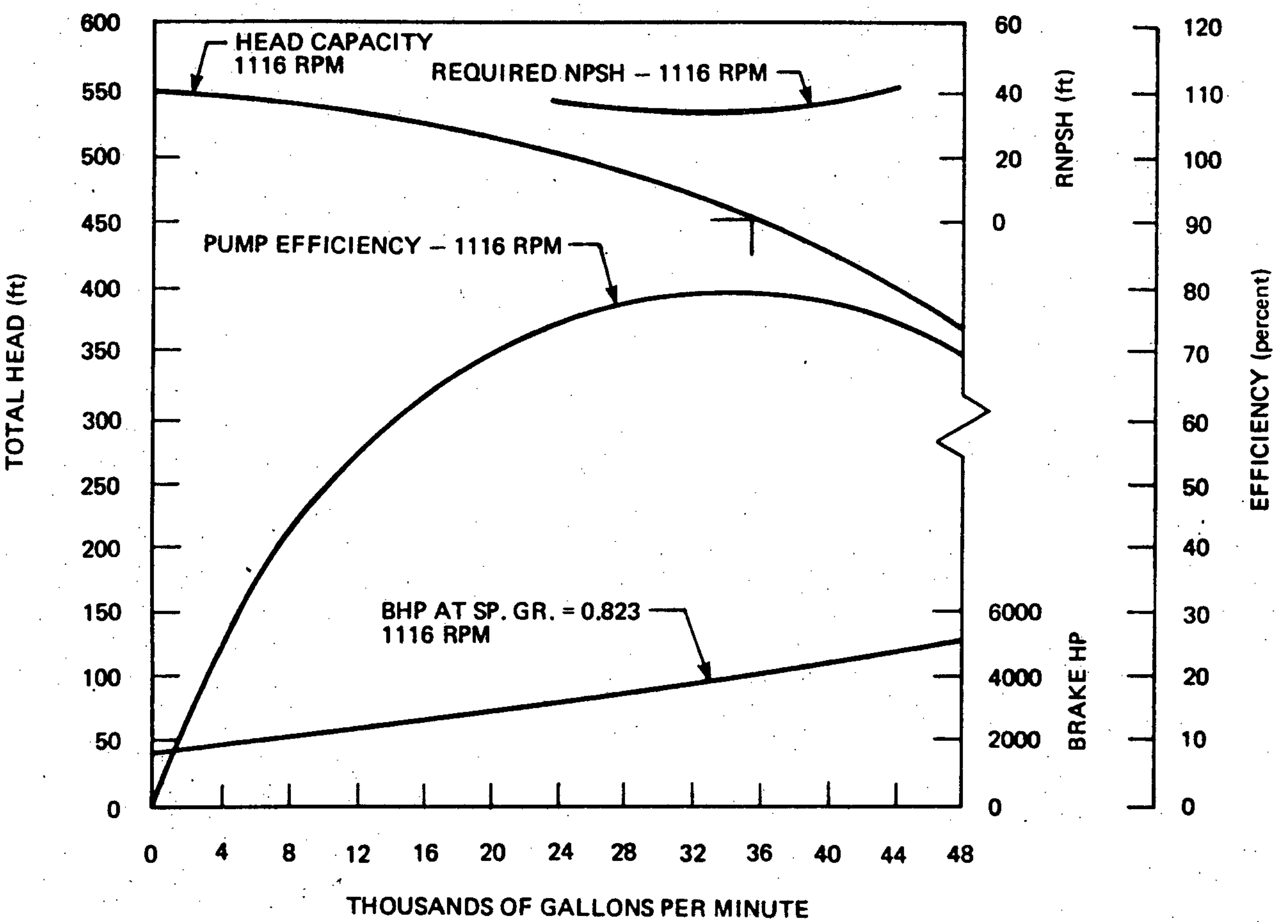


FIGURE 5.3-19 PRIMARY PUMP CHARACTERISTICS

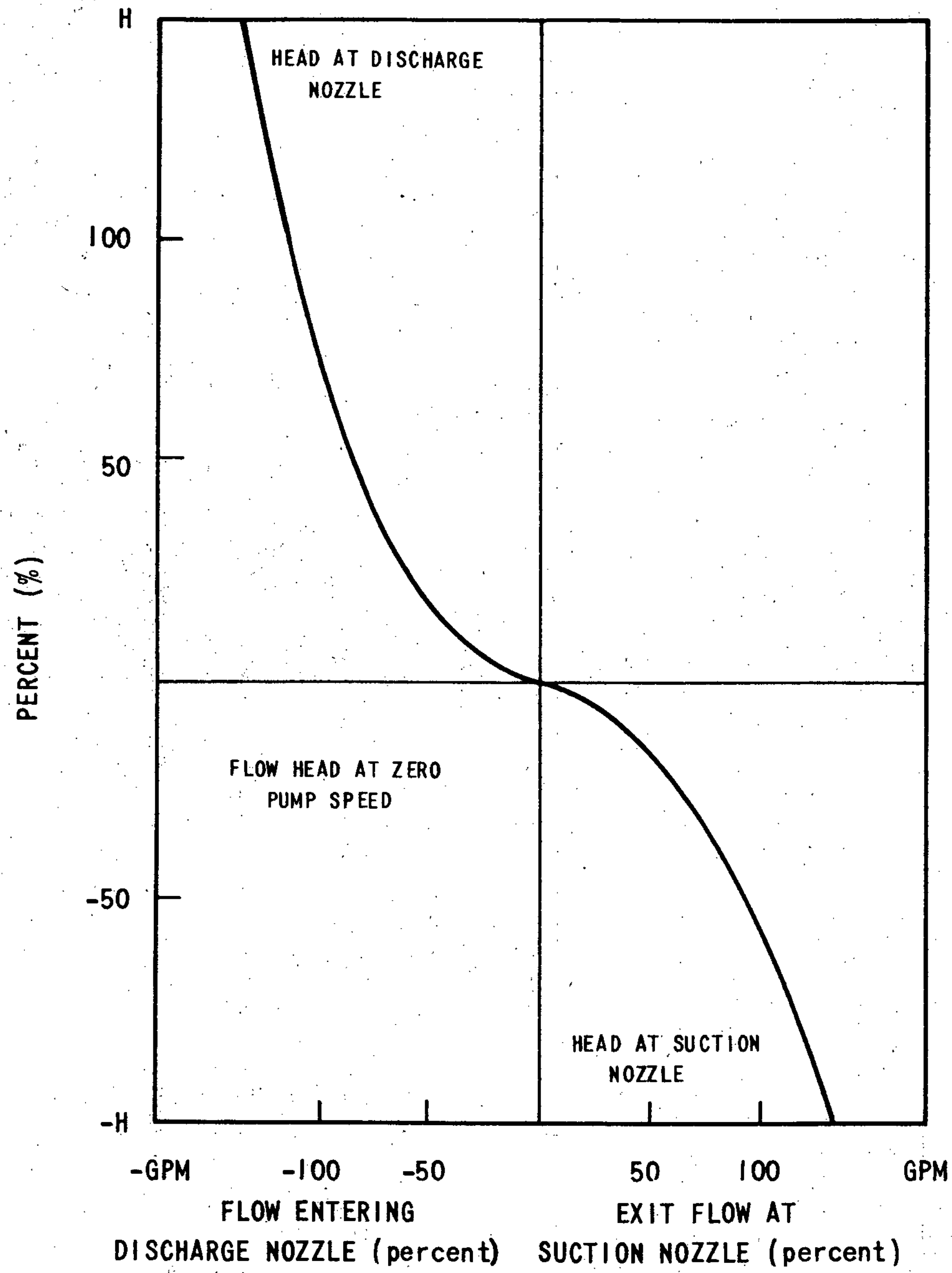


Figure 5.3-20. Pump Pressure Drop Vs. Flow Plotted in Percent of Normal Peak Operation

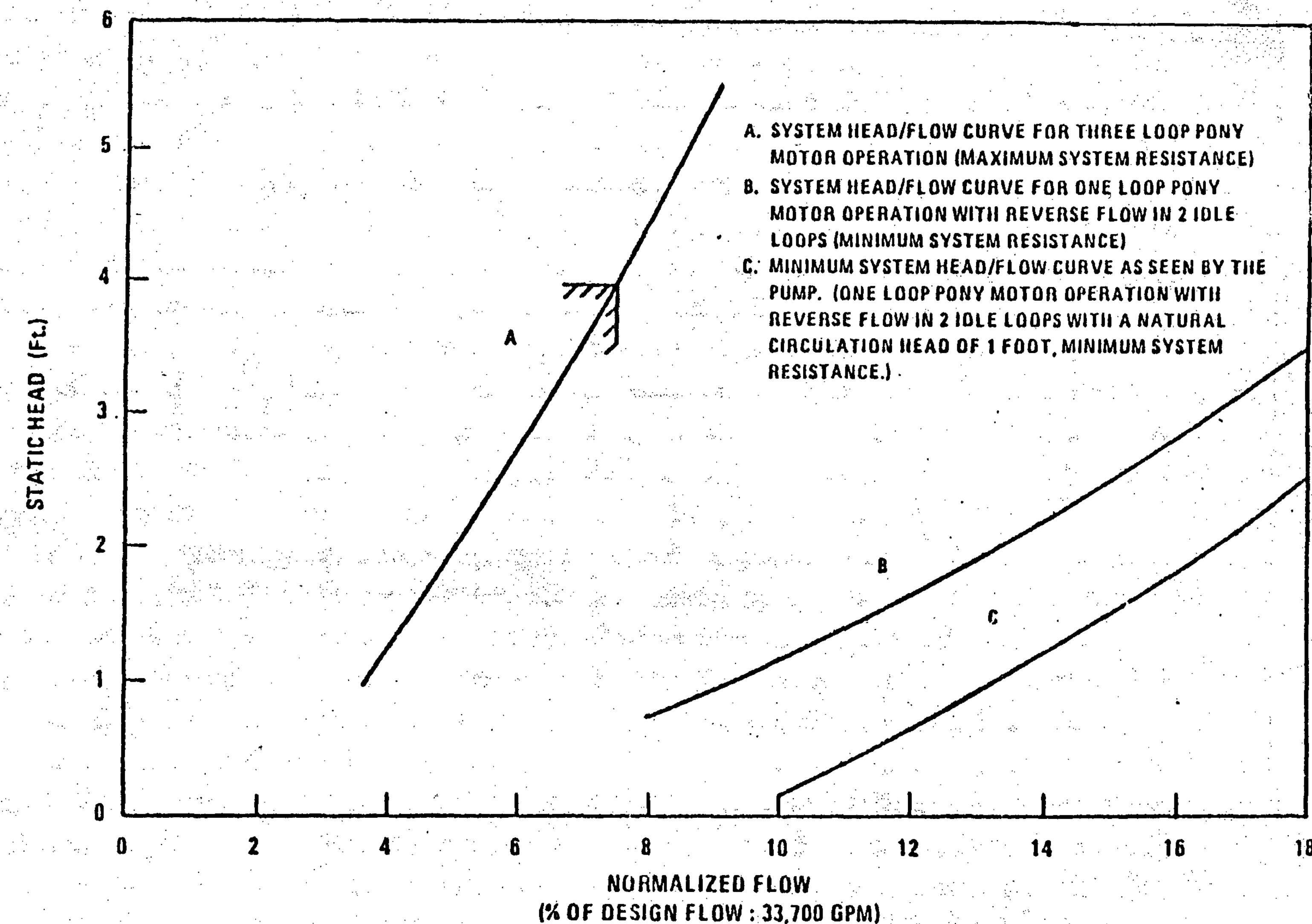


Figure 5.3-21 System Head/Flow Characteristics at Pony Motor Operation

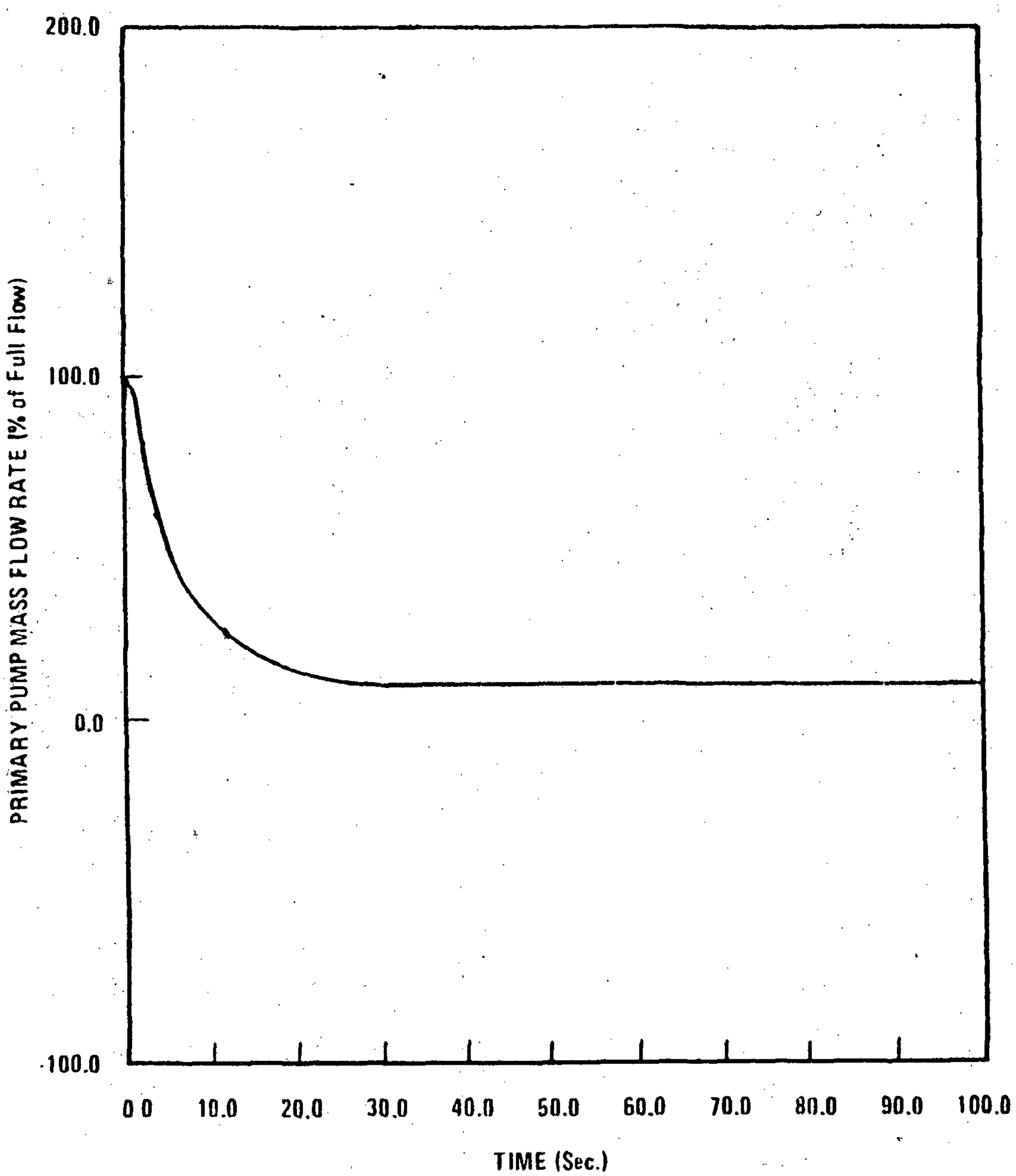


Figure 5.3-22. Pump Coastdown Characteristics

6669-45

43 Figure 5.3-23 has been intentionally deleted.

43 Figure 5.3-24 has been intentionally deleted.

56 FIGURES 5.3-25 thru 5.3-35
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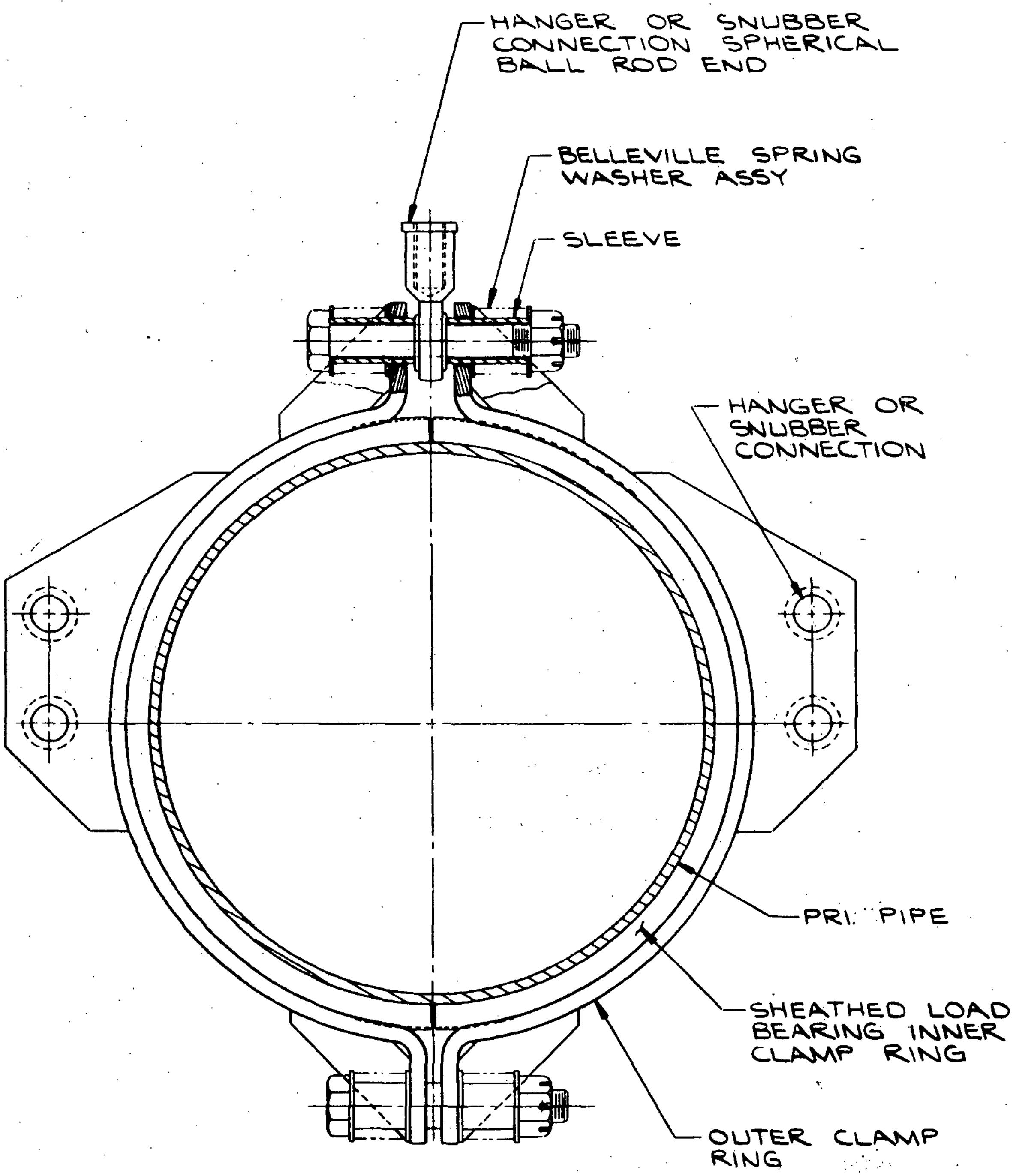

Figure 5.3-36 Pipe Hanger Clamp Assembly

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July 1975

FIGURE 5.3-37A

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FIGURE 5.3-37B

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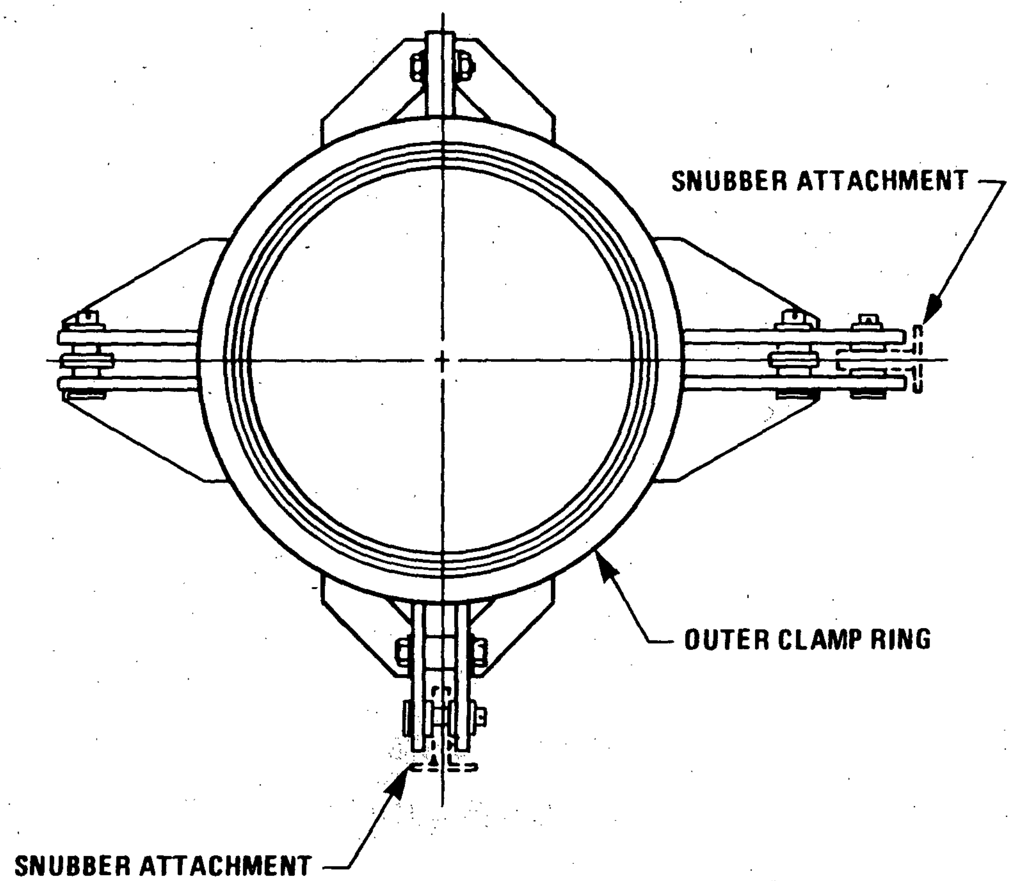

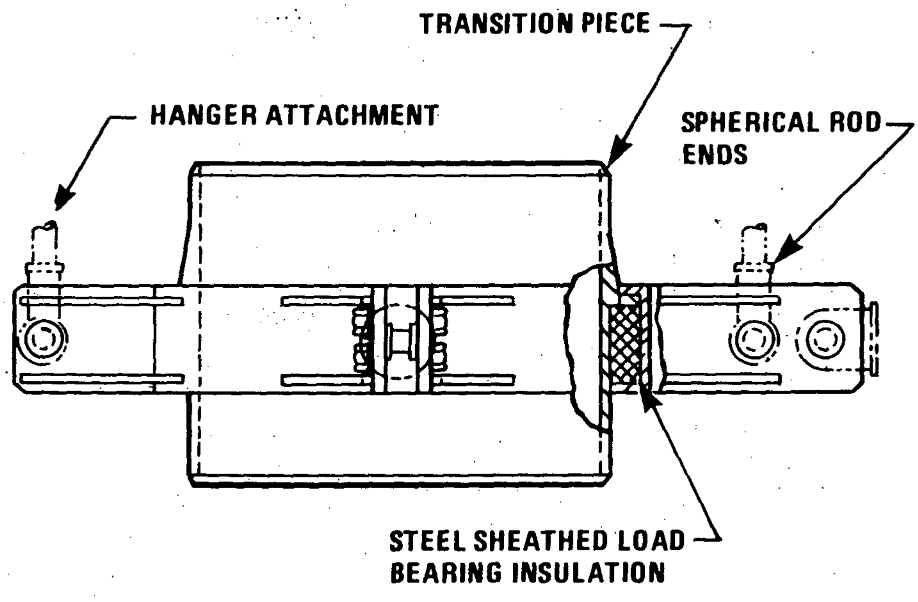

Figure 5.3-38. Vertical Pipe Clamp Assembly

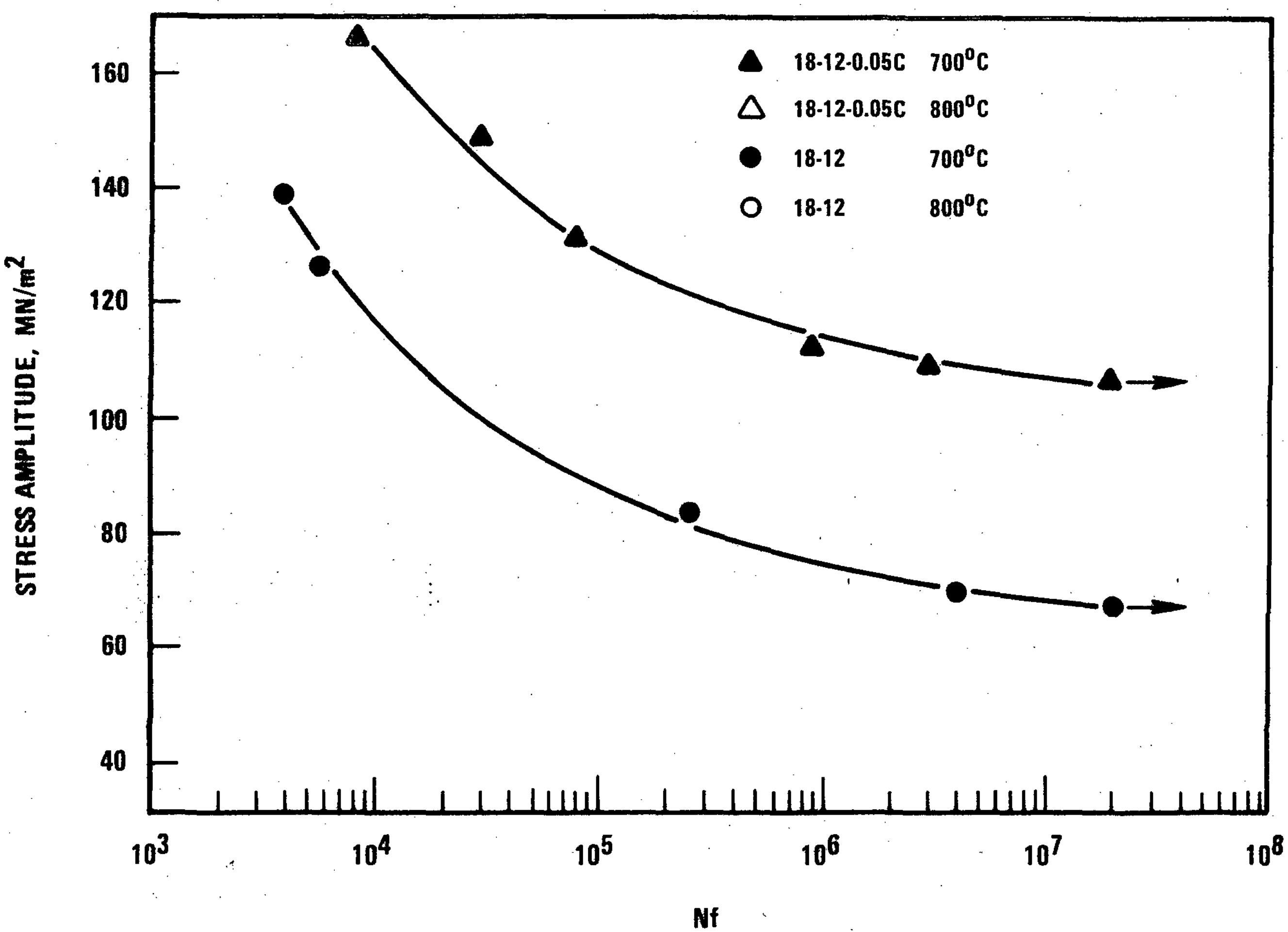

FIGURE 5.3-39 S/N Curves Of 18/12 And 18/12/0.05 C Alloys At 1292°F

56 FIGURE 5.3-40 HAS BEEN DELETED

56 FIGURE 5.3-41 HAS BEEN DELETED

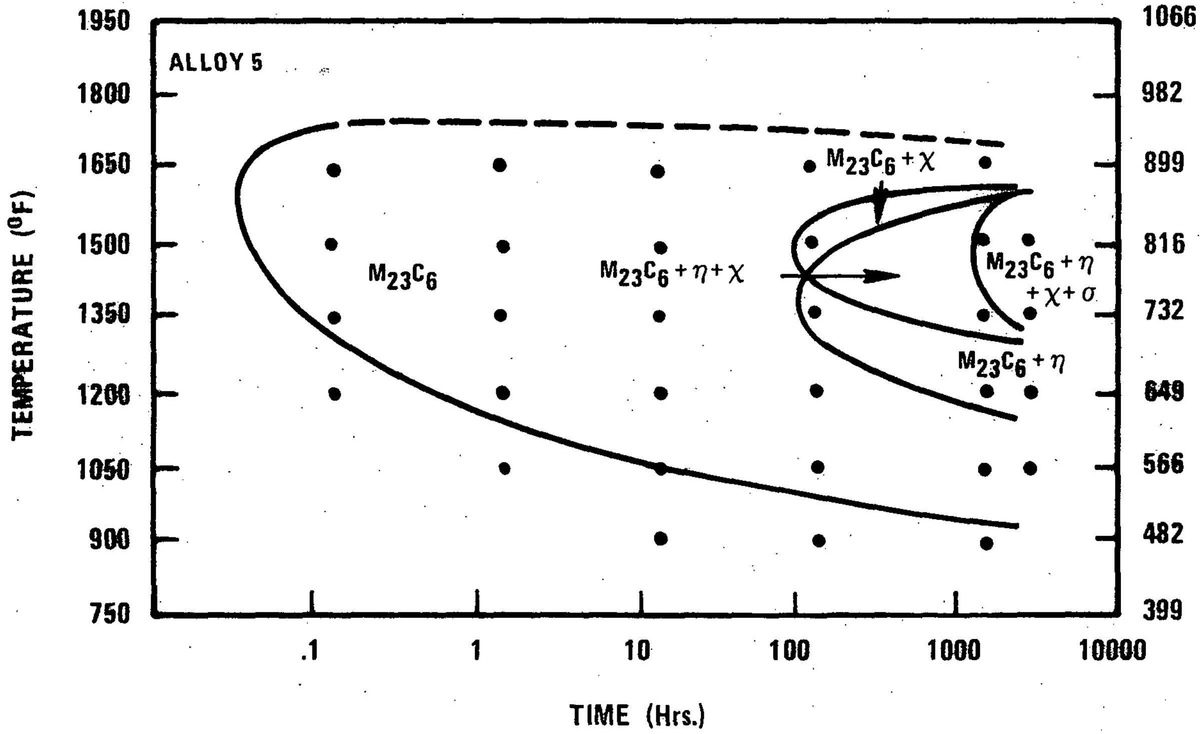

FIGURE 5.3-42 Precipitation Reactions In Type 316 Stainless Steel Solution Treated At 2300°F For 1.5 Hours And Water Quenched.

5.4 INTERMEDIATE HEAT TRANSPORT SYSTEM (IHTS)

5.4.1 Design Bases

5.4.1.1 Performance Objectives

The Intermediate Heat Transport System extracts reactor generated heat from the Intermediate Heat Exchanger and delivers this heat to the steam generator system under all normal and off-normal operating conditions. Specific performance objectives include:

Heat Transport and Flow Performance

a. Transport of reactor generated heat (975 mw_t) through the intermediate coolant system to the Steam Generation System while maintaining an adequate flow rate for controlling reactor temperature conditions within limits which preclude damage to the reactor vessel, fuel and reactor internals.

b. Regulation of heat transport system flow in response to plant process control over the full operating power range of 40 to 100 percent reactor thermal power.

c. Transfer of decay heat to the Steam Generation System under all normal and off-normal conditions including failure of a heat transport system component or loop. Specifically, there will be capability to remove decay heat by pony motor flow or natural circulation with two or three loops following operation at rated power and with one or two loops following operation at approximately two-thirds of rated power.

d. Containment of sodium coolant by providing a boundary for coolant confinement.

e. Transport of reactor generated heat to the Steam Generation System with two-loop operation at nominally two-thirds rated power output.

f. Provide a sodium coolant system which can be easily filled, vented and rapidly drained.

g. Support of operation in a hot stand-by condition - nominally
35 7-1/2 % of full flow at a nominal cold leg temperature of 590^{o}F.

Structural Performance

a. Design, fabrication, erection, and testing of the HTS components which comprise the sodium boundary except the dump lines, shall be in accordance with the ASME Boiler and Pressure Vessel Code,
29 Section III,

Division 1 Class 1, Nuclear Components, 1974 Edition, with all addenda up to and including Winter 1975 (except for the IHTS Mixing Tee fatique analysis which uses the 1977 edition with addenda through Summer 1979) and Applicable Code Cases for elevated temperature components and RDT Standards E15-2T (November 1974 Edition with addenda through June 1975), F9-4T (January 1976 Edition), F6-5T (August 1974 Edition with Addenda through June 1976), F3-6T (October 1975 Edition), F2-2 (August 1973 with addenda I, dated 12/73, addenda II, dated 3/74 and addenda III, dated 7/75), M1-1T (March 1975 Edition) M1-2T (March 1975 Edition), M1-22T (September 1975 Edition), M1-23T (September 1975 Edition with Amendment I dated 10/75), M2-3T (May 1975 Edition), M2-5T (May 1973 Edition with Addenda I, dated 1/74, Addenda II, dated 3/74 and addenda control sheet dated 1/75), M3-3T (April 1976 Edition), M3-7T (April 1975 Edition), M3-11T (May 1975 Edition), M3-12T (April 1976 Edition) as applied in each equipment specification.

b. The effective dates for the sodium pumps are ASME Boiler and Pressure Vessel Code, Section III Division 1, Class 1, Nuclear Power Plant Components, 1974 Edition with all addenda up to and including Winter 1974 and applicable Code Cases for elevated temperature components and RDT Standards E15-2T (November 1974 Edition with addenda through June), F9-4T (September 1974 Edition), F6-5T (August 1974 Edition), addenda-2/75 F3-6T (December 1974 Edition), F2-2 (August 1973 with addenda I dated 12/73, and addenda II, dated 3/74), M1-1T (March 1975 Edition), M1-2T (March 1975 Edition).

c. Effective dates of codes and standards will correspond to those listed above or for components ordered at some future date will use the latest available version at the time the contract is let.

d. The dump/fill and drain lines downstream of the first dump valve and the gas equalization lines downstream of the first valve and rupture disc assembly, shall be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 2.

e. The natural frequencies of all components will, where possible, avoid resonance with all expected pump driving frequencies. Where not possible, the component design shall insure that structural damage will not occur as a result of resonance.

f. Structural design shall provide for dry IHTS piping and component heat up at a rate of 3°F/hr.

g. Structural design shall provide for a system fill under conditions of full vacuum with system components at an average temperature of 400°F and hot spot temperature of 600°F.

h. All IHTS components and piping shall be designed with consideration of the following environmental factors as follows:

   1) Floods - flood protection is provided by ensuring the integrity of the Reactor Containment Building (RCB) and the Steam Generator Building (SGB). (See Section 3.3)

b. The system shall be designed such that a normal or upset event does not adversely affect the useful life of any IHTS component.

c. Following an emergency condition, resumption of operation must be possible following repair and re-inspection of the components, except that the intermediate coolant pumps (damaged or undamaged) must maintain capability to provide pony motor flow following all emergency events except in the affected loop for a pump mechanical failure, steam generator leak, or inadvertent dump of intermediate sodium.

d. Following a faulted condition, the intermediate heat transport system must remain sufficiently intact to be capable of performing its decay heat removal function, including maintenance of intermediate coolant pump pony motor flow.

The structural design parameters of the IHTS and individual components are listed in Table 5.4-1, IHTS Design Parameters. The thermal and hydraulic
59 design parameters are given in Table 5.1-1.

<u>Seismic Loads</u>

The intermediate heat transport system components under the jurisdiction of the ASME Code, Section III, Nuclear Power Plant Components, shall be designed to accommodate the load combinations prescribed therein without producing total combined stresses and strains in excess of those allowed in the Code. No component of an individual loading condition shall be included which would render the combination non-conservative. Transient loadings shall be included as required by the Code. For elevated temperatures, Code Case 1592 supplemented by RDT Standard F9-4T will apply.

Details of seismic loading combinations and analysis are provided in Section 3.7.

5.4.1.2 <u>Applicable Code Criteria and Cases</u>

The IHTS pressure containing components shall be designed, fabricated, erected, constructed, tested, and inspected to the standard(s) listed below:

Component	Applicable Standard and Class
IHTS Pump	ASME III, Class 1
IHTS Expansion Tank	ASME III, Class 1
IHTS Dump Valves	ASME III, Class 1
IHTS Piping	ASME III, Class 1
IHTS Flowmeter	ASME III, Class 1
IHTS Thermowell	ASME III, Class 1
IHTS Pressure Taps	ASME III, Class 1
44 33 29 IHTS Dump/Fill and Drain Lines (Downstream of First Valve)	ASME III, Class 2

All intermediate heat transport system components except the dump lines shall be analyzed as Class 1, nuclear components in accordance with the following rules:

a. The 1974 Edition of the ASME Boiler and Pressure Vessel Code and addenda through and including Winter, 1975, Section III, (except for the IHTS Mixing Tee fatique analysis which uses the 1977 edition with addenda through Summer 1979).

b. ASME Code Case 1592, "Class 1 Nuclear Components in Elevated Temperature Service."

c. RDT E15-2T (Supplement to Section III).

d. RDT F9-4T (Supplement to Code Case 1331-8).

The "Liquid Metal Fast Breeder Reactor Materials Handbook", (Ref. 1), shall be used to obtain material properties data not available from the above sources. As required in RDT F9-4, the use of additional or alternative material properties* shall require the approval of the purchaser. Code Case 1521, "Use of H-Grades of SA-240, SA-479, SA-336, and SA-358, Section III," may be used for H-Grades of Type 304 and 316 austenitic stainless steels. RDT-F9-5 (Section 6) provides alternative procedures for satisfying the strain limits of Code Case 1592 which are acceptable to the purchases. Section 6 of RDT F9-5 also provides time temperature limits below which the primary plus secondary and peak stress limits of Section III may be used in place of the limits of Code Case 1592. The scope of the analysis of Code Case 1592 shall be used even if the limits from Section III are used. For example, the primary plus secondary stress intensity range due to Emergency as well as Normal plus Upset Conditions is limited.

In addition, Code Class 1593, for fabrication and installation of elevated temperature components, 1594 for their examination, 1595 for their testing; and 1596 for their overpressure protection shall apply for the intermediate heat transport system components.

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* The term "alternative material properties" refers to the material property data used alternatively to the data of the same property contained in the authoritative sources of ASME Code (Section III and Code Case 1592), RDT Standards (E15-2 and F9-4), and the LMFBR Materials Handbook. The intent of the statement is to make clear that alternative material property data, which may be more conservative or less conservative than the data supplied in the authoritative sources, cannot be used without the approval from the purchaser. The purchaser will only approve the less conservative property data after obtaining permission to use it from the ASME Code and RDT Standards Committees, and will approve the more conservative property data upon valid justification by the user. The data approved in this manner will be incorporated by the purchaser in the design specification for alternative use.

5.4.1.3 Surveillance Requirements

The need to monitor austenitic stainless steel toughness changes (due to carburization, plastic creep straining and the thermal environment) will be assessed as part of an ongoing program. These studies will be performed in parallel with design.

An inservice inspection program will be implemented for the IHTS in accordance with the intent of ASME Section XI. See subsection 5.4.2.1.3.

If fracture toughness surveillance is determined, by ongoing programs, to be required, then the surveillance program will be designed in accord with the philosophy of Appendix H to CFR Part 50.

5.4.1.4 Material Considerations

5.4.1.4.1 Basis for High Temperature Design Criteria

The basis for high temperature design and analysis of all HTS (PHTS and IHTS) Class 1 components are given in Section 5.3.1.4.

5.4.1.4.2 Materials of Construction

The materials to be used for the intermediate heat transport piping and pump system will be specified the same as the PHTS materials described in Section 5.3.1.4.2 and Tables 5.3-6 and 5.3-7. The valves and expansion tank will utilize specifications shown in Table 5.4-3. The discussion in Section 5.3.1.4.2 on selection of material and specifications apply to all IHTS material.

Selection of alternate materials shall be based on the mechanical properties, metallurgical stability, sodium compatibility and response to radiation under the applicable design and environmental conditions. When recommending the use of alternate materials, the supplier shall document the justification which shall include, as a minimum, a summary of available test or experience data and a discussion of the adequacy of the recommended materials relative to the
36 materials specified in Table 5.4-3.

5.4.1.4.3 Additional Requirements

The additional requirements described in Section 5.3.1.4.3 for the PHTS materials apply to the IHTS materials.

5.4.1.4.4 Welding

The discussion in Section 5.3.1.4.4 concerning the welding of PHTS components shall apply to the IHTS components. The welding filler materials and specifications are given in Table 5.4-4. Welding of the trimetallic transition joints (ferritic - Alloy 800H - austenitic with ErniCr-3 as the weld filler material between ferritic and Alloy 800H, and 16-8-2 stainless steel filler
59 between Alloy 800H and austenitic) is covered in Section 5.5.3.11.2.

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5.4.1.5 Leak Detection Requirement

The IHTS leak detection subsystem (which is part of the Leak Detection System discussed in Section 7.5.5) will provide indication and location information to the operator in the event of a sodium leak from the IHTS to a cell atmosphere.

Leakage from the intermediate system to the primary system will be detected by volume changes within the IHTS as discussed in Section 7.5.5.2. Leakage from the primary system to the intermediate system which may occur under accident conditions will be detected by a radiation monitoring system as discussed in
Section 7.5.5.2. 30

The leak detection system sensitivity requirements are discussed in Section 7.5.5.

5.4.1.6 Instrumentation Requirements

The intermediate system is provided with an instrumentation
system which monitors the process variables within the IHTS and which
49 provides signals for safety action and operational information and control.
The measured variables and instrumentation provided are discussed in
Section 7.5.2.

5.4.2 Design Description

5.4.2.1 Design Methods and Procedures

5.4.2.1.1 Identification of Active and Passive Components which Inhibit Leaks

Table 5.4-5 lists those active and inactive components of the IHTS and their normal operating mode.

The IHTS is composed of three independent and physically separated loops, contained within individual cells. Each cell is designed to contain the maximum expected sodium spill and is designed such that the adjacent loops are
29 not affected by a disabled loop.

In the event of a pipe leak in an operating loop, the IHTS is designed to provide shutdown and decay heat removal capabilities with one of the two remaining loops.

Small pipe leaks in the IHTS can be detected by sodium leak detectors. Leak detection ability allows for operator action to manually shut down the plant and drain the affected loop. The leak detection system is described in Section 7.5.5.

A pipe leak in the IHTS would result in a level decrease in the expansion tank and IHTS pump, which activates an alarm in the control room. The operator can then take appropriate action. In addition, a reactor scram can be initiated by the primary to intermediate flow ratio and pump speed ratio trips and by the high primary cold leg temperature trip dependent upon the location and size of the break. These events are discussed in more detail in Section 15.3 and 5.5.3.6.

5.4.2.1.2 Design of Active Pumps and Valves

The IHTS sodium pumps will be designed, analyzed, manufactured, tested, and shipped as described for the PHTS in Section 5.3.2.1.2.

5.4.2.1.3 Surveillance and Inservice Inspection Program

An inservice inspection program for the IHTS will be implemented and conducted in accordance with the intent of the ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear 1 Reactor Coolant System. The inservice inspection program will include all IHTS components
59 such as pressure vessels, piping, pumps and valves.

To facilitate the inspection program, it is a design goal that all IHTS sodium welds be accessible for inspection after insulation and heater removal. Where necessary, hand held optical aids or remote devices such as periscopes will be used for inspection.

5.4.2.1.4 Protection Against Accelerated Corrosion and Material Degradation

The materials of construction of the IHTS are protected against stress corrosion and intergranular corrosion from purchase through the operating life of the plant using the same methods as employed for the PHTS described in Section 5.3.2.1.5.

5.4.2.1.5 Material Inspection Program

The material inspection program for the IHTS is the same as that identified for the PHTS in Section 5.3.2.1.5.

5.4.2.2 Material Properties

The coolant boundary sections of the intermediate heat transport system are fabricated from unstabilized austenitic stainless steels (Type 304, Type 304H and Type 316H), ferritic steel (2-1/4 Cr - 1 Mo), and Alloy 800H. Properties of the austenitic stainless steels used in the intermediate heat transport system are similar to those described in Section 5.3.2.2 for the primary heat transport system. Properties of the ferritic steel and Alloy 800H used in the intermediate heat transport system are similar to those described in Section 5.5.3.11 for the steam generator system.

5.4.2.3 Component Descriptions

5.4.2.3.1 Intermediate Coolant Pumps

The intermediate sodium pumps are free surface, single stage vertically mounted, drawdown type centrifugal pumps driven by a variable speed 5000 Hp squirrel cage induction motor. An auxiliary 75 Hp pony motor on each pump provides low flow capability (selected by gear set, in the range of 7.5 to 12 percent of design flow) for decay heat removal and other low power, hot standby conditions. Variable pump speed from 22-1/2% to 100% design speed is achieved by the main drive motor supplied with variable frequency power from a fluid coupled MG set. The intermediate pump is identical to the primary pump (Section 5.3.2.3.1) except for the following significant differences in requirements. The maximum pump static head requirements for normal power operations will be established from the IHTS loop with the maximum hydraulic resistance at steam generator system thermal hydraulic conditions. The minimum head requirements will be established from the IHTS loop with the minimum hydraulic resistance and the minimum IHTS flow required at 40% power operation. The nominal pump static head will be at least 330 feet of sodium at IHTS thermal hydraulic design conditions.

The intermediate pump is identified to the primary pump (Section 5.3.2.3.1) except for the following significant differences in requirements.

- o No radiation or thermal shielding is required.
- o Head and flow requirements are different (see Figure 5.4-3).
- o Transition piping is required to connect the 36 inch pump suction nozzle to the 24 inch IHTS cold leg piping.

- The intermediate pump does not require a stand pipe-bubbler for sodium level control.

- The IHX vent line in the primary pump is replaced by a cover gas equalization line.

- The argon and oil vapors are vented from the seal leakage reservoir to the atmosphere rather than to RAPS.

- Journal flow holes are optimzized for IHTS transients (i.e. cold to hot).

The design envelope for the intermediate pump is shown in Figure 5.4-1. It is a design basis that the PHTS pumps operate with 10" W.G. (0.36 psig) cover gas pressure. Since the PHTS and IHTS pumps are being procured as identical units, the IHTS pumps could also operate with essentially atmospheric cover gas pressure without cavitation even though this is not a design basis. This type of IHTS operation is not anticipated, however, because the IHTS pump cover gas pressure requirement is 100 psig during normal operation (10 psig at pony motor flow) based on the requirement to maintain a positive pressure on the IHTS side of the IHX with respect to the PHTS side of the IHX.

Section 5.4.1.1 specified performance requirements of the IHTS following a faulted condition. These are applied to the pumps as follows:

The IHTS coolant pumps are required to remain operable only at pony motor speed following all system-related emergency conditions and the faulted conditions. For pony motor operation, the IHTS pumps require (1) unimpaired pump shaft rotation, (2) a physically intact shaft seal lube oil system (lube oil pump operation not required; however, lube oil containing boundary must retain its integrity, (3) continuous lube oil cover gas supply, and (4) uninterrupted power supply to the pony motors. These functional requirements on the pumps are in addition to the requirement that the IHTS sodium boundary retain its integrity. The decay heat removal function can be accomplished with as few as one of the three IHTS loops operating.

Free surface level changes in the IHTS pumps are significant only for system temperature changes. The minimum sodium level corresponds to the level of sodium in the system after filling at 400°F. As the system is heated to operating temperatures, the sodium level rises about 3' 3". The expanded volume of sodium is contained in both the pump tank and a connecting intermediate sodium expansion tank. An additional free volume of 1' 11" above the normal liquid level is provided to accommodate potential abnormal operating conditions.

The kinetic energy of the total rotating pump mass at 963 rpm following an intermediate pump trip is as follows:

Pump (impeller, bearing, and shaft)	487,000 ft-lbs
Motor	4,100,000
Total estimated kinetic energy	4,587,000 ft-lbs

Design techniques used in the prevention of fracture-type failure and prevention of oil leakage in the Intermediate Coolant Pump are the same as those identified for the Primary Coolant Pump in Section 5.3.2.3.1.

The IHTS sodium pumps are designed to withstand the loadings associated with (1) the extremely unlikely plant condition occurring from a design basis leak in a steam generator, and (2) the Safe Shutdown Earthquake (SSE). The analyses required to demonstrate this treats the intermediate pump and its hydraulic assembly as a Class 1 component in accordance with the rules of the ASME Code Section III and modified RDT Standards as defined in Section 5.4.1.1.

The preliminary design basis sodium-water reaction produces a pressure transient at the IHTS pump suction nozzle. This transient is described in Section 5.5.3.6. The calculated pressure transient arriving at the intermediate pump tank is considered an emergency condition for the purpose of evaluating the pump tank.

Dynamic analyses of the pump tank will be performed to determine the structural response. Simplified models such as beams, frames, and plates will be used if results can be shown to be conservative. Finite element shell models will be used if simplified models cannot be shown to give adequate results.

In the final code analysis for the stress report, if the pressure load is a transient, it and the dynamic load will be treated as time dependent dynamic loadings. This analytical method may amplify or mitigate the stresses. In either case, the dynamic analysis will be used in the final design.

Stress analysis of the pump tank will be performed according to the rules of ASME Code Section III (see Sections 5.4.1.1.1 and 5.3.2.1.2 for applicable classification and Code Cases and RDT F9-4). Stresses due to thermal transients occurring simultaneously will be combined with the primary load set. The IHTS pump tank will be designed to meet the limits of ASME Code Section III, 1974. 19 33

Edition up to and including Winter 1974 addenda, applicable to emergency conditions, for the combined load set. The effects of cyclic pressure loading during the transient will be included.

5.4.2.3.2 Expansion Tank

An expansion tank is provided in each IHTS loop to accommodate the sodium volume change in excess of that accommodated in the IHTS pump tank, due to the thermal expansion associated with normal and off normal conditions. The pump tank and expansion tank cover gas volumes are connected by a gas service line. The expansion tank sodium line is connected to the main IHTS cold leg piping just upstream of the IHTS pump suction nozzle.

5.4.2.3.3 IHTS Piping and Support

The IHTS piping conducts sodium in a continuous loop to transport reactor heat to the Steam Generation System. An isometric drawing of the IHTS piping and components in a typical steam generator cell is shown in Figure 5.4-2.

Each IHTS piping run is provided with the necessary elbows, tees and reducers to provide adequate loops for thermal expansion. While each loop contains the necessary appendage piping and associated fittings, there are no valves in the main sodium flow piping. Each loop has similar components, although the loop piping differs in length and configuration because of the differences in distance between the IHX units and steam generator modules.

The hot leg piping is 24" OD x 1/2" wall Type 316H stainless steel and extends from the IHX outlet nozzle through the reactor containment penetration to the superheater inlet nozzle. From the superheater outlet nozzle, the piping consists of two parallel runs of 18" OD x 9/16" wall 2-1/4 Cr - 1 Mo ferritic steel and extends to each of the two evaporator inlet nozzles. The cold leg piping commences at the evaporator outlet nozzles and consists of two parallel runs of 18" OD x 1/2" wall Type 304H stainless steel pipes. The lines are joined together through expanders at a 24" x 24" x 24" Type 304 H mixing tee and continue as a single run of 24" OD x 1/2" wall Type 304H stainless steel pipe which is connected to the 36" diameter IHTS coolant pump suction through a seven foot long diffuser. The cold leg continues as a 24" OD x 1/2" wall Type 304 H stainless steel pipe from the pump discharge through the reactor containment penetration and completes the loop at the IHX inlet nozzle. The Type 316H and Type 304H stainless steel pipes are joined to the 2-1/4 Cr - 1 Mo ferritic steel superheater and evaporators, respectively, through Alloy 800H transition spool pieces. Transition spool pieces (2-1/4 Cr-1Mo/A800H/316H) are used in the instrumentation and steam generation vent lines.

The auxiliary IHTS piping includes appendage piping for instrumentation, system high point vents and low point drains, fill lines, sodium dump lines and service connections for sodium purification.

The IHTS piping except for the dump lines, will be designed in accordance with Section III, Class 1 of the ASME Boiler and Pressure Vessel Code. The dump/fill and drain lines downstream of the second dump valve and the gas equalization line downstream of the second valve and the rupture disc assembly will be designed in accordance with Section III, Class 2. The material used in piping is in accordance with the applicable code and will be of all welded construction. Transition spool pieces (2-1/4 Cr-1Mo/A800H/316H) are utilized to join lines to the dump tank.

All IHTS piping and components, except the dump/fill and drain piping downstream of the second dump valve in series, the gas equalization line downstream of the second valve in series and the gas equalization bypass line downstream of the rupture disc assembly are required to be Safety Class 2 as a minimum and therefore, ASME Boiler and Pressure Vessel Code, Section III, Class 2, as a minimum; however, they are being optionally upgraded and will be designed, constructed and code stamped in accordance with ASME B&PV Code, Section III, Class I. The dump/fill and drain piping downstream of the second valve in series, the gas equalization line downstream of the second valve in series, and the gas equalization by-pass line downstream of the rupture disc assembly are required to be Safety Class 3 as a minimum, and therefore, ASME B&PV Code, Section III, Class 3 as a minimum; however, they are being optionally upgraded and will be designed, constructed and code stamped in accordance with ASME B&PV Code, Section III, Class 2.

The IHTS piping will be supported from the building structure with constant load support hangers and rigid rods and will be restrained with seismic snubbers. Attachments to the piping for supports will be of the clamp type on the outside of load bearing insulation. If any attachment requires direct support to the pipe full penetration welds will be used. Special requirements and controls on piping supports are identified in Section 3.9.2.6.

Piping penetrations through the Reactor Containment will be a flued head, rigid type seal. Piping penetrations through the Steam Generator Building will not provide leak tight seals.

The piping within the IHTS consists of large sodium containing piping which must be installed per detailed drawings and rigid quality assurance requirements. There is no piping which can be field run.

5.4.2.3.4 Intermediate Heat Exchanger

The CRBRP Intermediate Heat Exchanger (IHX) serves to transfer reactor thermal energy from the radioactive primary sodium to the non-radioactive intermediate sodium. The IHX is a counterflow shell and tube type unit with a vertical orientation in the plant. The design arrangement provides for downflow of the cooled (primary) fluid and upflow of the heated (intermediate) fluid to enhance natural circulation for reactor decay heat removal. A detailed description of the IHX design is given in Section 5.3.2.3.2.

5.4.2.4 Overpressurization Protection

The IHTS has no isolation valves within the normal circulation path so that isolation of individual system pipe sections or components is not possible. If for some reason the system becomes blocked, the intermediate pump will not overpressurize the system as the IHTS structural design is sufficient to withstand the pump shutoff head. In the event of an argon supply valve failure, the system would not be overpressurized as the combination of argon supply pressure of 115 psi and pump shutoff head would not exceed the IHTS design pressure. This is true even if the pump shutoff head associated with the PHTS pump were reached instead of the IHTS pump shutoff head.

The system may be subjected to overpressure in the event of a water or steam leak in the Steam Generation System. For large or intermediate sodium-water reactions, the resulting pressure increase due to the formation of reaction products in the faulted evaporator or superheater module is relieved through rupture disks. (See Section 5.5.2.4).

5.4.2.5 Leak Detection System

5.4.2.5.1 Leak Detection Methods

The methods used to detect Liquid Metal to gas leaks from pipes and components of the IHTS are aerosol detectors, cable detectors, contact detectors and visual inspection with back up from smoke detectors. See Section 7.5.5.1.

A sodium level monitoring system is provided to monitor any leakage
56 between reactor and intermediate coolant occurring in the IHX. The method
is described in Section 7.5.5.2 in detail.

5.4.2.5.2 Indication in Control Room

Audible alarms will be sounded in the control room as described in Sections 7.5.5.1 and 7.5.5.2.

5.4.2.5.3 IHTS Coolant Volume Monitoring

The IHTS coolant volume is monitored by the level indicators in the
IHTS pump tank and in the expansion tank. Details are discussed in
Section 7.5.5.2. These monitors coupled with the sodium temperature measure-
56 ments allow monitoring of the total sodium in the IHTS loops. Small leakages of
sodium from the IHTS can be replaced by use of the sodium fill system.

5.4.2.5.4 Critical Leaks

Critical leaks are discussed in Section 5.3.2.5.4. Detection
29 capability is discussed in Section 7.5.5.

5.4.2.5.5 Sensitivity and Operability Tests

56 Periodic maintenance will provide for checking the operational
readiness of leak detectors. During installation and checkout, the correct
electrical functioning of each leak detector and level detector will be
tested. 28

5.4.2.5.6 Confinement of Leaked Coolant

If there is any leakage from the IHTS in the RCB it will be confined
as described in 5.3.2.5.6. Any leakage from the IHTS in the SGB will be
29 contained in the catch pans and will be detected by the leak detection system. 28
Fires as a result of sodium spills are evaluated in Section 15.6. Leaks
in the IHX are still contained in the passive coolant boundary, and no leakage
56 into the RCB will result.

5.4.2.5.7 Intermediate/Primary Coolant Leakage

Primary to Intermediate coolant leakage is very unlikely due to the
higher operating pressure of the intermediate system. The IHTS pressure
shall be maintained at a minimum of 10 psi higher than the PHTS pressure at
41 all points in the IHX during all normal modes of operation. Intermediate to
primary coolant leakage detection is described in Section 7.5.5.2.

5.4.2.6 Coolant Purification (IHTS)

The IHTS coolant purification is accomplished by six cold traps, two in
each of the three loops. All six traps are normally in operation, however, operation
58 of a single trap per loop will still maintain required Na purity. These cold

traps are air cooled, and process the sodium at a flow of 60 gpm. The single trap is sufficient to limit oxygen and hydrogen to a maximum of 2 and 0.2 ppm respectively, and to limit the tritium concentration in the sodium to 0.062 µCi/gm (corresponds to a transport of tritium to the steam generator
50 water of 0.016 Ci/day). A more detailed description of the Intermediate Sodium Processing System is given in Section 9.3.2.4.

5.4.3 Design Evaluation

5.4.3.1 Analytical Methods and Data

The design of the IHTS is based upon technology attained during the development, design, construction and operation of sodium systems of similar type . IHTS hydraulic and fluid volume changes are calculated and based upon sodium thermodynamic data from "Standard FFTF Values for Physical and Thermophysical Properties of Sodium". (Ref. 2)

The IHTS sodium pressure losses are calculated using Darcy's formula, the general equation for pressure drop of fluids. The factors for equivalent lengths of fittings and pipe friction are based upon "Tube Turns," Bulletin TT725 (Ref. 3) and "Friction Factors for Pipe Flow," by L. F. Moody (Ref. 4), respectively.

5.4.3.1.1 Structural Evaluation Plan (SEP)

The procedure for developing SEP's for the IHTS is the same as described for the PHTS in Section 5.3.3.1.1.

5.4.3.1.2 Stress Analysis Verification

The IHTS stress analysis verification procedures and methods are the same as those described for the PHTS in Section 5.3.3.1.2.

5.4.3.1.3 Compliance with Code Requirements

The design of the IHTS will be in compliance with code requirements as identified for the PHTS in Section 5.3.3.1.3.

5.4.3.1.4 Category I Seismic Design

All IHTS components and suppprting structures required for system operation are designed for Seismic Category I. See Section 3.2 for list of major components classified as Seismic Category 1. Limits of Category 1 are identical to the system limits which are on Figure 5.1-2. The modeling and analysis will be performed in the same fashion as for the PHTS as described in Section 5.3.3.1.4.

5.4.3.1.5 Analytical Methods for Pumps and Piping

26 The analytical methods used for the intermediate coolant pumps and in-containment piping will be the same as those given in Section 5.3.3.1.5.

Ex-Containment Piping

The ex-containment sodium piping is designed and analyzed to a large extent during design iterations by elastic methods and complies with the requirements of ASME Code Section III, Class 1 and elevated temperature Code Case 1592. If all the Code Criteria cannot be complied with in the final iterations performing elastic analysis only, then inelastic analysis (time-dependent as well as time-independent) will be performed.

An inelastic (time-dependent as well as time-independent) flexibility analysis may be required for certain circumstances where the calculation of induced forces from an elastic flexibility analysis is excessively conservative or where the amount of elastic follow-up must be evaluated. A computer program developed for inelastic nonlinear flexibility analysis will be selected and verified before use. The results from the program will be used in making appropriate evaluations of Code Case 1592. These results may also be used for further simplified or detailed inelastic analysis of the piping components critical to meeting the Code requirements. Elastic flexibility analysis results may also conservatively be used for simplified or detailed inelastic analysis of the critical piping components. All the inelastic analyses - flexibility analysis, simplified or detailed analysis - will conform to the requirements of RDT Standard F9-4T and the guidelines of RDT Standard F9-5T.

Causes of component degradation, such as thermal aging, sodium vapor or liquid sodium environment, environments involving combinations of sodium, argon, nitrogen or oxygen are identified in the design specification, and the methods to account for the effects of the identified degradations in the analytical methods or in the criteria for evaluation are also specified.

No structural tests on the piping istalled in the plant are planned in support of the design analysis.

The following failure modes and associated leading conditions were identified:

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1. Ductile rupture from short-term loadings such as pressure (static as well as dynamic), dead weight and seismic loading.

2. Creep rupture from long-term loadings such as pressure, dead weight, thermal expansion loads at sustained high temperature (above 800°F for austenitic and 700°F for ferritic steels) operating conditions.

3. Creep-fatigue failure from cyclic duty histogram involving startup, shutdown and thermal transient conditions with sustained high temperature operating states.

4. Gross distortion or structural integrity damage due to incremental accumulation of inelastic strain or ratchetting coupled with possible elastic follow-up from cyclic duty histogram.

5. Loss of function due to excessive deformation from dynamic pressure loading due to large sodium water-reaction in steam generators or from loadings during faulted operating condition such as safe shutdown earthquake.

6. Buckling due to short term loadings such as internal vacuum, dead weight, thermal expansion loading and seismic loading.

7. Creep buckling (load or deformation controlled) due to long-term loadings at sustained high temperature, such as deadweight and thermal expansion loading with a possibility of elastic follow-up.

The structural strength verification analyses are based upon linear elastic methods. Evaluation of the acceptability of the design is based upon comparison of the calculated and allowable stress intensity limits specified in the Code.

Creep-fatigue evaluation for cyclic stress is performed in accordance with Paragraph T-1400 of Code Case 1592 and/or Article NB-3222.4 of ASME B&PV Code, Section III.

RDT Standard F9-4T is utilized as a supplement to Code Case 1592, particularly for evaluation of strain, deformation and fatigue limits.

Seismic response is evaluated in accordance with the requirements of WARD D-0037 Revision 0 and the Code. The steam generator is designed to withstand the effects of the SSE and remain functional.

Buckling and instability criteria are based upon Paragraph T-1500 of Code Case 1592 and/or Paragraph NB-3033 in Section III of the ASME B&PV Code.

Inelastic analysis shall be performed for design verification analyses only. Initial design iterations are to be based on elastic methods. Areas requiring inelastic analysis shall be identified and technical justification for continuing with the design prior to performing inelastic analysis will be established.

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5.4.3.1.6 <u>Analytical Methods for Evaluation of Pump Shaft and Bearing Integrity</u> 33

The analytical methods for the IHTS pump speed and bearing integrity evaluation are the same as the methods for the PHTS described in Section 5.3.3.1.6.

5.4.3.1.7 <u>Operation of Active Valves Under Transient Loading</u>

There are no active valves in the IHTS.

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5.4.3.1.8 Analytical Method for Component Supports (Vessels, Piping, Pumps, and Valves)

In accordance with the ASME Code, component supports will have the same code classification as the components they support. Design of each component support will comply with the ASME Section III Paragraph NF design rules corresponding to the component support classification. In order to provide assurance that the component support stresses comply with limits specified in paragraph 5.4.3.1.2, analysis of each component support will be performed. The analytical techniques and applicable computer codes discussed in paragraph 5.3.3.1.5 will also apply to detailed analysis of support components. If the component support design temperatures exceed those for which allowable stress values are given by Section III, the rules covered in Code Case 1592 and RDT F9-4 will be satisfied.

5.4.3.2 Natural Circulation

To provide sufficient intermediate system flow following a postulated loss of all pumping power, the IHTS piping and components are arranged to promote natural circulation of the sodium. The hydraulic profile shown in Figure 5.1-3 shows relative elevations of piping and components.

As noted on Figure 5.1-3, the sodium inlet at the evaporators is 35 feet above the geometric center of the IHX under normal operating conditions. For off normal and transient condition the resulting thermal head is sufficient to provide the required post shutdown IHTS flow. 33

With loss of pumping power and start of natural circulation, the thermal center in the IHX will move. The plant transient computer program DEMO is used to calculate the natural circulation flow rates. The IHX model is highly nodalized (45 nodes in the direction of flow path on each side). Heat transfer is individually calculated at each node. As a result, shifts in thermal center elevation are inherently calculated by this mode. Consequently, the transient results presented include changes in coolant circulation due to the shift of the thermal center.

5.4.3.3 Intermediate Coolant Pump Characteristics

Preliminary IHTS pump characteristics under normal operating conditions are presented in Figure 5.4-3.

The information is an approximation and is based on pump vendor test data from similar pump designs and on data from Reference 5.

Since the intermediate pump hydraulics are identical to the hydraulics of the primary pump, the locked rotor characteristics are as shown in Figure 5.3-20. The following expressions for intermediate pump locked rotor impedance are based on Figure 5.3-20:

Forward flow $\quad H = 184\,\bar{\omega}^2$

Reverse flow $\quad H = 235\,\bar{\omega}^2$

where, H = pressure drop through pump, feet

$\bar{\omega}$ = normalized flow, i.e., the ratio of actual system flow to design flow 1

59 The pump drive system incorporates separately powered pony motors for circulating intermediate sodium at low flow rates during startup, testing and stand-by operations following accident conditions. Basically, the requirement 33
is to provide approximately 7.5 percent flow (2210 gpm) at 400°F to 650°F at a
29 head of about 3.0 feet of sodium. Following reactor trips, the pony motors will maintain flow for decay heat removal.

There will be no vortexing or gas entrainment at flows in the operating range of the IHTS pumps. All parts which need to be submerged during operation at
59 pony motor speed or restart to pony motor speed are located below the minimum sodium level.

The oxygen concentration in the intermediate system sodium will be maintained
59 below 2 ppm during normal operation. This level of sodium impurity does not affect the pump operating characteristics.

Pump Integrity

The IHTS pump shaft supporting assembly and inner structure are supported on the pump tank mounting flange at the IHTS pump support structure. Circumferential bolts secure the pump tank head to the supporting assembly and 33
inner structure which is in turn secured in a similar fashion to the pump tank. The pump tank is circumferentially bolted to the support structure.

59 | 43 For the IHTS Pump SSE analysis, a 2% damping value was used. The SSE loading is considered to occur in conjunction with a plant trip. Following the SSE, the Intermediate Heat Transport System, Steam Generation System, and Steam Generator Auxiliary Heat Removal System must provide for removal of stored and
59 decay heat. The IHTS pumps are designed to maintain pony motor flow without loss of structural integrity after SSE. Computer programs, such as SAP IV and
59 ANSYS, were utilized to perform seismic analyses on the intermediate pumps. Descriptions of these computer programs can be found in Appendix A of the PSAR.
59 The intermediate coolant pump and its hydraulic assembly are treated as a Class 1 component for the seismic analysis.

5.4.3.4 Valve Characteristics

There are no valves in the IHTS main sodium piping.

5.4.3.5 Evaluation of Steam Generator Leaks

The large sodium water reaction evaluations covering a range of leak sizes are given in Section 5.5.3.6. Large sodium water reactions generate acoustic pressure pulses which propagate through the IHTS and also result in long term pressurization of the IHTS. The IHX has been identified as the critical component in the IHTS for these overpressure effects since the IHX tubes are the boundary between the primary and intermediate sodium. For the entire range of large leaks considered in Section 5.5.3.6, peak pressures in the IHX do not result in primary stress levels exceeding the stress allowable for emergency conditions as given in the ASME Code, Section III.

5.4.3.6 IHTS Coolant Boundary Integrity

The integrity of the IHTS will be evaluated in a manner similar to that described for the PHTS in Section 5.3.3.6.

The highest quality of engineering, fabrication, installation, and inspection will go into the IHTS piping. The IHTS piping is a Class 1 item and will require a detailed stress analysis as required by the ASME Code.

The design basis and analyses as described in Section 5.3.3.6 for PHTS piping have direct application for the IHTS piping within the inerted HTS cells of the Reactor Containment Building. That is, the analysis methods and computer codes, the fracture mechanics analysis, the corrosion effects and sodium leak detectability will be considered as for the PHTS piping and the coolant
59 boundary integrity.

For the IHTS sodium piping in an air environment within the Reactor Containment Building Intermediate Bay and the Steam Generator Building, the approach for proving pipe integrity as presented in Section 5.3.3.6 is applicable with supplemental consideration of corrosion effects and leak detection capability. Separate discussions on corrosion rates and sodium leak detection for the IHTS piping within the air atmosphere are presented in Sections 1.5, 5.4.3.6.3 and 7.5.5.

5.4.3.6.1 Design and Quality Assurance

5.4.3.6.1.1 Design Assurance

As for the PHTS piping, the main IHTS piping will be designed as an ASME Code Class 1 system and the applicable Code Sections, Addenda, Code Cases, and RDT Standards will be used as the design bases. Detailed loads and resulting stresses will be obtained for each segment and component of the piping system. The detailed results of the stress analysis will enable a comprehensive assessment of the structural capability of the IHTS piping, and will be issued as a formal stress report as required by the ASME Code.

Stress Analysis Methods

The stress analysis procedures to be used in the integrity analysis of the IHTS are those as described in Section 5.3.3.6.1.1 for the PHTS piping. The same or equivalent computer codes will be used to ensure that the stress and strain evaluation of the IHTS piping is of sufficient detail to demonstrate structural integrity.

Piping System Design Basis

The bases and criteria for design of the IHTS piping system will be included in the HTS Piping Design Specification. The Piping Design Specification will be for the complete HTS system which includes both the PHTS and IHTS piping systems. This will ensure that the details of the design criteria, loadings and conditions for design of the IHTS will be of the same detail as for the PHTS piping.

The operating conditions for the IHTS piping are similar to those discussed for the HTS piping in Section 5.3.3.6.1.1. The IHTS piping system is a low pressure system operating in the temperature range of $650^{\circ}F$ to $965^{\circ}F$. Due to the low pressures, (<225 psia) the system is constructed of thin-walled piping operating at low primary (or load-controlled) stresses.

The highest stressed piping components of the intermediate system are in loop 1. The loop 1 hot leg (reactor containment penetration to superheater inlet) has the highest thermal expansion stresses. The elbows, the SWRPRS tee and the 2 1/4 Cr-1 Mo stainless steel transition piece are the critical items in the pipe line. The integrity of this piping during the service lifetime of the plant is ensured by stringent quality assurance, conservative design practice, and leak detection equipment which would identify small through-wall leaks and an inservice inspection program, following the intent of Section XI of the ASME
59 Code, Rules for Inservice Inspection of Nuclear Power Plant Components. 25

The specified loadings on the IHTS piping will be categorized into design, normal, upset, emergency and faulted conditions and organized into a load histogram. The design stress analyses will be performed using these loading categories and histograms. Pressure surge, vibration and temperature fluctuation effects will be assessed with the usual effects of internal and external pressure, deadweight, support reactions, thermal expansion, seismic and thermal transient gradients. As specified by Regulatory Guide 1.48, the IHTS piping will be designed for two levels of seismic acceleration spectra, one for an Operating Basis Earthquake and one for a Safe Shutdown Earthquake.

The piping stress analysis will be performed using verified and documented computer programs. Detailed stress maps for each IHTS piping segment will be obtained using the computer code ELTEMP which is described in Section 5.3.3.6.1.1. These stress maps will be the basis of the fracture mechanics analysis to be carried out on the IHTS piping elbows and components discussed in Section 5.4.3.6.2.

5.4.3.6.1.2 Quality Assurance

The quality assurance program for the IHTS system piping, in conformance with RDT Standard F2-2, is designed to provide the highest possible system integrity. Fabrication, installation, and inspection procedures, in accordance
29 with Class 1 requirements of Section III of the 1975 Edition of the ASME B&PV Code, will be used as well as supplemental requirements as imposed by applicable RDT Standards to assure that the IHTS piping will be able to function properly and safely throughout the projected service life of the plant. A detailed discussion of the quality assurance procedures is provided in Chapter 17.0.

Piping Fabrication

59 The IHTS hot leg (IHX-to-superheater) will be made from 24 inch O.D, 0.50 inch nominal wall thickness, Type 316H stainless steel, ASME Class 1 welded pipe, according to design specifications and the supplementary requirements imposed by RDT Standards M3-7T and M2-5T. In accordance with these two standards, the weld filler material will conform to RDT Standards M1-1T and M1-2T.

The IHTS cold leg will be made from 18 inch O.D. (evaporators-to-mixing tee) and 24 inch O.D. (mixing tee-to-intermediate pump-to-IHX), 0.50 inch nominal wall thickness, Type 304H stainless steel, ASME Class 1 welded pipe. The applicable RDT Standards for the hot leg piping will be applied to the cold piping also.

The piping between the superheater and evaporator modules is 18 inch O.P., 0.562 inch nominal wall thickness, 2 1/4 Cr-1MO ferritic steel, ASME Class 1. 29
Other aspects of the piping fabrication discussed in Section 5.3.3.6.1.2 for the PHTS piping will be the same for the IHTS piping system.

Mixing Tee Fabrication

The IHTS mixing tee joins the outlet sodium flows from the two evaporators before the flow enters the IHTS pump. The tee geometry was tested at ANL and
59 ORNL. The development program was aimed at optimizing the design of the tee to 29
assure long life and reliable service. The basic design is a tee 24 x 24 with two 18" - 24" expanders located upstream from the tee in the 18" diameter lines coming from the evaporators. The material is 304HSS. Development work was 29
59 required for the mixing tee since under transient operation, outlet temperatures may vary between the two evaporators. The mixing tee must be able to accommodate this temperature fluctuation within fatigue limits. 2

Elbow Fabrication

The elbows for the IHTS piping system will be procured to RDT Standard M2-51 and will be of welded, stainless steel, Types WP304 and WP316 materials. Other 29
aspects of the elbow fabrication discussed in Section 5.3.3.6.1.2 for the PHTS elbows will also be applied to the IHTS piping elbows.

Pipe to Elbow Fabrication

See Section 5.3.3.6.1.2 (direct application)

Comparison of CRBRP and ASME Code Inspection Criteria

See Section 5.3.3.6.1.2 (direct application)

Hydrostatic Pressure Test and Code Stamp Application

See Section 5.3.3.6.1.2 (direct application)

Nonconformance Reports, Waiver Requests and Trouble Records

See Section 5.3.3.6.1.2 (direct application)

Installation

See Section 5.3.3.6.1.2 (direct application)

Installation Inspections and Tests

See Section 5.3.3.6.1.2 (direct application)

In-Service Inspection

In-service inspection shall be performed on the components of the IHTS to provide a continuing assurance that these components can perform their functions safely. The examinations will include continuous monitoring by liquid metal-to-gas leak detection systems and visual inspections of the IHTS components. Dissimilar metal welds will be inspected by volumetric methods. Hangers and snubbers will be visually inspected. Snubbers will also be
59 functionally tested to assure continued operability.

5.4.3.6.2 IHTS Piping Failure Potential Due to Fatigue Crack Growth

In this section a postulated defect much larger than the QA allowable size was analyzed for fatigue growth. This analysis employed the materials properties and applied stresses typical of those in the IHTS piping system along with the postulated crack size. The defect was assumed to exist in a highly stressed elbow in the hot leg of IHTS piping. A preliminary estimate was made of the fatigue crack extension of the hypothetical flaw. As the analysis of the IHTS piping system progresses, other piping locations will be examined in detail using the methods presented here.

Along with crack growth considerations, preliminary calculations were made to determine the critical crack size for elbows in each leg of the IHTS piping system. These crack sizes were calculated assuming through-the-wall flaws with operating and design pressures. The purpose was to show that starting with a flaw that is much larger than any flaw allowed by the RDT Standards and applying the cyclic load history, the flaw will not grow to the critical crack size.

Based on preliminary results obtained from the analysis of the one elbow using the analytical procedures described in 5.3.3.6, the crack extension is negligible and the critical crack length will not be. It is expected that the analyses to be done on the other sections of IHTS piping will also result in the conclusion that the growth of a postulated large defect would be negligible and that the potential for pipe failure is negligibly small. Details of the crack growth calculations for the IHTS elbow are given below.

The crack growth calculations for the IHTS elbow are representative of the IHTS austenitic steels (304, 304H and 316H) only and are not necessarily applicable to the ferritic steel (2-1/4 Cr - 1 Mo) and Incoloy materials. Crack growth
35 will be investigated for these additional materials.

Material Behavior Considerations

The materials properties data given in Figures 5.3-26 thru 5.3-28 are representative of the behavior in a sodium or air environment of the IHTS piping material, as discussed in Section 5.3.3.6 for the PHTS piping, and may be used for IHTS crack growth calculations. This data can be conservatively used for both IHTS Types 304 and 316 stainless steel.

Stress Analysis Considerations

The loadings to be considered in the final crack growth analyses for all the sections of the IHTS piping will include all anticipated normal, upset and emergency events of stress magnitude sufficient to cause fatigue-crack extension. The preliminary estimate of crack propagation for the most-highly-loaded hot leg elbow is based on the loadings for the reactor scram event applied for the total number of expected upset and emergency events expected during the life of the plant. This event is described in Figure 5.4-6 and consists of a transition from refueling at 400°F to normal operating temperature (965°F) followed by a scram which returns the system to 400°F. This event was selected for the preliminary crack growth analysis, since it is one of the most severe upset events and it makes up more than 400 of the 600 events to be considered. The final analyses for crack growth calculations for the IHTS piping system will consider each loading cycle separately for each highly stressed region in the IHTS piping system.

Analyses for the load cycle of Figure 5.4-6 have identified the hot leg elbow adjacent to the HTS cell penetration to be the region of highest stress from a fatigue standpoint. Circumferential and axial stress components were calculated for the inside surface, midplane and outside surfaces at various points around the circumference. The maximum values are given in Table 5.4-6 (also see Figure 5.3-30). These values are used for a preliminary estimate of crack propagation.

In general, it was found that the circumferential stress components are greater than the axial components. Therefore, the analysis assumed the existence of an axial crack (acted upon by circumferential stresses) on the inside elbow surface at the location of maximum circumferential (hoop) surface stress in the middle of the analyzed elbow. The stress values used in the calculation of the crack growth are given in Table 5.4-6.

Other aspects of the stress analysis considerations for crack growth predictions are discussed in Section 5.3.3.6 for the PHTS piping system and are directly applicable to the IHTS piping system.

The preliminary tensile stresses of Table 5.4-6 are below the 0.2% offset yield strength (approximately 17,200 psi for Type 316SS at 965°F). Therefore, the use of linear-elastic fracture mechanics in this case is justified.

Initial Crack Size Considerations

The postulated initial defect was hypothesized to have a length (2c) of 1.5 in. and a depth (a) of 0.125 in. This is considerably larger than the size of the largest crack that is estimated might be missed by non-destructive evaluation which has 2c = 1.0 in. and a = 0.015 in. The postulated crack is illustrated schematically in Figure 5.3-31 and the ratio of crack depth to nominal wall thickness (a/t) is 0.250.

Crack Propagation Calculations

The procedures discussed in Section 5.3.3.6.2 for the PHTS piping is directly applicable and is used for determining crack growth in the most-highly-loaded IHTS piping hot leg.

The postulated semi-elliptical crack assumed in the elbow is shown in Figure 5.3-31. The K-solution (stress intensity) for such a crack in a flat plate subjected to a linearly varying stress is given by:

$$K = \frac{\Pi a}{\Phi} (M_k \sigma_t + M_B \sigma_b) \qquad (1)$$

where all of the terms are as defined in Section 5.3.3.6.2.

Applying a shell curvature correction factor and adding a term to account for the effect of pressure acting upon the crack faces, the approximately K - solution for the cracked elbow becomes:

$$K = [M_k \sigma_t + M_B \sigma_b + 1.13P] \sqrt{\pi a} \;\; f(\lambda) \qquad (2)$$

where

$$f(\lambda) = 1 + (0.481\,\lambda + 0.386\, e^{-1.25\lambda} - 0.386)\,(a/t) \qquad (3)$$

$$\lambda = [12\,(1 - \nu^2)]^{1/4} \;\; \left(\frac{c}{\rho t}\right) \qquad (4)$$

$2c = 1.50$ in.

$\rho = 12.0$ in. (outside radius of pipe)

$t = 0.4375$ in. (the nominal wall thickness of 0.500 in. has been reduced 12.5% (spec. allowable) to account for dimensional tolerances.)

$\nu = 0.29$ for Type 316 SS

$P = 181$ psi

Using the values above and substituting into Equations (3) and (4) gives the following:

$$\lambda = 0.596$$

$$f(\lambda) = 1.02$$

Equation (2) is used for the stress intensity factor calculation involving surface cracks on the analyzed elbow. This equation applies only to Mode I or "opening mode" cracking (see Reference 6 for definitions of cracking modes). The potential for Mode II cracking is small because the shear stresses are low relative to normal stresses.

Because of the stress ratio effect (discussed in Section 5.3.3.6.2) the effective stress intensity factor, K_{eff}, provides better correlation of crack growth data than does K as determined from Equation (2). The effective stress intensity is given by:

$$K_{eff} = K_{max} [1 - R]^m \tag{5}$$

where:

$R = K_{min}/K_{max}$

$m = 0.5$

The data presented in Figures 5.3-26 thru 5.3-28 are plotted as a function of K_{eff}.

For the case of circumferential (hoop) stress cycling between the scram transient and refueling condition, the following results from Table 5.4-6, $\theta = 37.5°$;

Refueling Condition:

σ_{circ} = 7415 psi (inside surface)

σ_{circ} = 2331 psi (outside surface)

Scram plus Normal Operating Condition:

σ_{circ} = 15,980 psi (inside surface)

σ_{circ} = -6394 psi (outside surface)

The stress gradients across the wall at any point may be resolved into a uniform tension component and a linear bending component. Hence,

Refueling Condition:

Circ. Stress; σ_t = 4873 psi, σ_b = ±2542 psi

Scram plus Normal Operating Condition:

Circ. Stress; σ_t = 4793 psi, σ_b = ±11,187 psi

Using Equation (2) and assuming P = 181 psig, we obtain for the Refueling Condition,

$$K_{min} = 5028 \text{ psi } \sqrt{\text{in.}}$$

Similarly, for the scram plus normal operating condition,

$$K_{max} = 9529 \text{ psi } \sqrt{\text{in.}}$$

From Equation (5), an effective stress intensity factor is found as follows:

$$K_{eff} = 9529 \left[1 - \frac{5028}{9529}\right]^{0.5} = 6549 \text{ psi } \sqrt{\text{in.}}$$

Using an upper-bound of the crack growth rate for Type 304 SS at 800°F from Figure 5.3-28*, the crack growth rate associated with a K_{eff} of 6549 psi $\sqrt{\text{in.}}$ is;

$$\frac{da}{dN} = 1 \times 10^{-7} \text{ in./cycle for air environment}$$

$$= 1 \times 10^{-9} \text{ in./cycle for sodium or nitrogen environment}$$

There are a total of 600 significant transient events during the service life. Therefore, a conservative estimate of the crack extension during the life of the plant due to these transients is:

(a) for sodium environment or exterior of pipe in inert atmosphere:

$$\Delta a = \frac{da}{dN} (N)$$

$$= (1.0 \times 10^{-9}) (600) = 6 \times 10^{-7} \text{ inches (negligible)}$$

(b) for air environment (exterior of IHTS pipe in SGB)

$$\Delta a = \frac{da}{dN} (N)$$

$$= (1.0 \times 10^{-7}) (600) = 6 \times 10^{-5} \text{ inches (negligible)}$$

Based upon these preliminary analysis results, the total fatigue crack growth extension from all loading events is expected to be negligible for the IHTS piping within the inerted or air atmospheres.

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*Figure 5.3-28 is assumed to apply to Type 316 SS at 965°F.

Safe Life Margin

The previous section illustrates the negligible amount of fatigue crack growth expected for the service life of an elbow in the IHTS hot leg piping. This was shown to be the case even for a flaw 1.50 inches long and 0.125 inches deep which is considerably larger than the largest flaw that could be missed by inspections.

In this section, consideration is given to the size of a through-the-wall crack which under given loading conditions would exhibit considerable plastic deformation resulting in a bulging open of the crack (rupture). Extensive bulging would increase sodium leakage significantly. Critical crack size is defined in this discussion as that crack size for which the crack will bulge under operating pressure stresses. A complete discussion of critical crack determination for the PHTS piping is presented in Section 5.3.3.6.2 and the methods are directly applicable to the IHTS piping. The discussions and test results justify making calculations of the critical crack size in the IHTS as well as the PHTS piping systems with considerable confidence.

The following parameters are used in the calculation of critical through-the-wall crack sizes for the IHTS piping system;

A. Hot Leg Piping

	18-in. O.D.	24-in. O.D.
Material	Type 316 SS	Type 316 SS
Temperature	865°F	965°F
Yield Strength	17,400	17,200
Ult. Strength	59,100	56,000
Flow Stress	30,600	28,280
Operating Press.	127 psi	181 psi
Design Press.	325 psi	325 psi
Pipe Thickness	0.4375 (Min.)	0.4375 (Min.)
Pipe Mean Radius	8.75 in.	11.75 in.

B. Cold Leg Piping

	18-in. O.D.	24-in. O.D.
Material	Type 304 SS	Type 304 SS
Temperature	671°F	671°F
Yield Strength	17,800 psi	17,800 psi

	18-in. O.D.	24-in. O.D.
Ult. Strength	54,000 psi	54,000 psi
Flow Stress	28,720 psi	28,720 psi
Operating Pressure	127 psi	225 psi
Design Pressure	325 psi	325 psi
Pipe Thickness	0.4375 (Min.)	0.4375 (Min.)
Pipe Mean Radius	8.75 in.	11.75 in.

Using the parameters listed above, assuming an axial crack, and using the model results of Figure 5.3-36, critical through-the-wall crack sizes were determined for the hot and cold legs of the main coolant IHTS piping at operating pressure and the design pressure, respectively. These critical crack sizes (a_{cr}) are listed in Table 5.4-7.

The margin of safe life for the IHTS piping can be evaluated by taking the assumed crack size (established by either inspection limits or hypothesized), adding on the expected crack growth and comparing it to the critical crack size. In addition, crack extension due to sodium-leakage-induced corrosion should be considered. In equation form, this may be stated as follows;

Initial Crack Size

$$\left.\begin{array}{l}\text{(a) Determined by inspection limits} \\ \quad\quad\text{or} \\ \text{(b) Hypothetical size}\end{array}\right\} + \text{Fatigue Crack Growth} + \text{Crack extension due to corrosion} \le a_{cr}$$

For the case of an inert atmosphere the margin of safe life for piping is very large because, (1) the critical through-the-wall crack is several times larger than the postulated crack which was shown to grow a negligible amount during the service life and (2) if a through-the-wall crack or leak could develop, the leaking sodium would be detected by the sensitive leak detection system before the crack could reach critical crack size (see Section 7.5.5 for a discussion on the leak detection system).

For the case of the IHTS piping in the air environment, crack extension due to sodium-leakage-induced corrosion must be considered. Section 5.4.3.6.3 discusses sodium induced corrosion in an air environment and the testing planned to quantify it. In any case, the margin of safe life for the IHTS piping in the air environment has been shown to be large because (1) the postulated crack would not grow to a through-the-wall crack during the plant service life and (2) the critical crack size is so large that a through-the-wall crack, enlarged by sodium, will produce such a large sodium spill that the leak detection system will detect the leakage before further significant crack growth could occur.

5.4.3.6.3 Sodium-Leakage-Induced Corrosion

If it is postulated that a through-the-wall crack could occur in the IHTS piping system, then sodium leakage could occur. Section 7.5.5.1 describes the several diverse leak detection systems that will provide alarms for a wide range of leakage rates. Experimental investigations characterizing sodium leaks to inerted gas and air environments are also discussed in that section.

In order to demonstrate that a small sodium leak does not create a safety related problem and to show that there is a large margin in the design against potential adverse consequences of small leaks, experiments as described in Section 1.5 are being conducted to investigate the consequences of small leaks in an inerted gas environment and an air atmosphere. A major goal of these experiments is the measurement of the rate of corrosion of austenitic stainless steel pipe due to the reaction products of sodium, gas and air environments.

The experiments to date have been carried out at FFTF operating conditions. Section 1.5.2 discusses the experiments being done at CRBRP operating conditions.

The available data are discussed in Section 5.3.3.6.3. For the case of sodium leaking into an inerted gas environment, the resulting corrosion rate of the stainless steel pipe at FFTF operating conditions is small amounting to the order of one-thousandth of an inch per month. It is believed that similar corrosion rates will be demonstrated for CRBRP conditions for the case of the PHTS and IHTS piping in the inerted gas environment.

Additional corrosion tests identified in Section 1.5.2 are in progress to provide additional data for both the cases of the CRBRP inert gas environment and an air atmosphere to better define the processes involved.

Based on the data described in Section 5.3.3.6.3, the sensitivity of the leak detection system discussed in 7.5.5.1, and the magnitude of critical crack sizes determined in the previous section, it concluded, subject to verification by the tests described in Section 1.5.2, that more than one of the several leak detection methods for monitoring the IHTS piping system would detect the leak before it could grow to a large size.

The corrosion data for austenitic stainless steels are not necessarily applicable to the ferritic steel (2-1/4 Cr - 1 Mo) and Alloy 800H material used
35 in the IHTS. Corrosion will be investigated for these additional materials.

5.4.3.7 Inadvertent Operation of Valves

The major IHTS piping contains no valves. The valves listed in Table 5.4-5 are discussed in other sections as noted in the table.

5.4.3.8 Performance of Pressure-Relief Devices

Devices used to protect the IHTS from overpressurization are evaluated in Section 5.5.3.9.

5.4.3.9 Operation Characteristics - Design Transients

The overall plant duty cycle list, which includes the event classification according to the ASME Section III categories of normal, upset, emergency faulted, and the event frequencies are given in Appendix B. Events in the plant duty cycle list with similar thermal transient characteristics for the IHTS components have been grouped under a reduced number of "umbrella" transients to reduce the amount of structural analysis required. The umbrella transients for these IHTS components will be used to construct component histograms for use in the structural analysis.

The design loading combinations and the associated stress or deformation limits to be used are those specified in the ASME Section III, Nuclear Component Code in the following figures:

A. Figure NB-3222-1, Stress Categories and Limits of Stress Intensity for Normal and Upset Operating Conditions.

B. Figure NB-3224-1, Stress Categories and Limits of Stress Intensity for Emergency Conditions.

5.4.3.10 Material Considerations

The material considerations for the austenitic stainless steel portion of the IHTS are the same as those for the PHTS described in Section 5.3.3.10 except for design temperatures. The highest design temperatures for the IHTS are 775°F for Types 304 and 304H stainless steels and 1015°F for Type 316H stainless steel. Creep is not significant for Types 304 or 304H stainless steels at 775°F.

59 The material considerations for the 2-1/4 Cr - 1 Mo ferritic steel and Alloy
800H used in the IHTS are the same as those for the steam generator system
35 described in Section 5.5.3.11.

5.4.3.11 Protection Against Environmental Factors

Protection for the principal components of the IHTS against environmental factors is provided by the structural integrity of the Reactor Containment and Steam Generator Building. Environmental factors to be considered include the following:

Fire Protection - See Section 9.13.

Flooding Protection - See Section 3.4.

Missile Protection - See Section 3.5.

Seismic Protection - See Section 3.7.

Accidents - See Section 15.6.

References to Section 5.4

26 * 1. Hanford Engineering Development Laboratory, Liquid Metal Fast Breeder Reactor Materials Handbook", HEDL-TME-71-32.

26 * 2. W. H. Yunker, "Standard FFTF Values for the Physical and Thermophysical Properties of Sodium", WHAN-D-3, July 6, 1970.

3. "Flow of Fluids, Tube Turns", Bulletin TT725, Louisville, Kentucky.

26 * 4. L. F. Moody, "Friction Factors for Pipe Flow," Transactions of the American Society of Mechanical Engineers, Volume 66, November, 1944; pages 671 to 678.

26 * 5. Stepanoff, A. J., "Centrifugal and Axial Flow Pumps", 2nd Edition, John Wiley and Sons, Inc., New York, New York, 1975.

6. P. C. Paris and G. C. Sih, "Stress Analysis of Cracks" in <u>Fracture Toughness Testing and its Applications</u>, pp. 30-83, ASTM, Philadelphia, Pa., 1965.

26 * References annotated with an asterisk support conclusions in the Section. Other references are provided as background information.

TABLE 5.4-1

IHTS DESIGN PARAMETERS

COMPONENT	STRUCTURAL DESIGN PARAMETERS PRESSURE (psig)	TEMP. (^{o}F)
Main Loop Piping		
IHX to Superheater	325	965
Superheater to Evaporator	325	885
Evaporator to Pump	325	775
Pump to IHX	325	775
Components		
IHX Tube Side	325	1015
Venturi Flowmeters	325	885
Transition Joint, Inst. Press. Tap	325	885
Pump	325	775
Expansion Tank	325	775
Mixing Tee, Cold Leg	325	775
Transition Spool Piece, Superheater Inlet, Vent & Dump	325	965
Transition Spool Piece, Evap. Outlet, Vent & Dump	325	775
Dump Valves	325	965
Transition Joint, Leak Detector	325	885
Dump Piping (Upstream of 1st Valve)		
Superheater Line	325	965
Evaporator Line	325	775
Pump Discharge to IHX Low Point Line	325	775
IHX to Superheater, Low Point Line	325	965
Gas Equalization Line	325	775
Dump Piping (Between Valves)	325	800
Dump Piping (Downstream of 2nd Valve)	50	800
Transition Spool Pieces at Dump Tank	50	800
Fill/Drain Piping & Header	325	800

TABLE 5.4-1 (Continued)

IHTS DESIGN PARAMETERS

COMPONENT	STRUCTURAL DESIGN PARAMETERS	
	PRESSURE (psig)	TEMP. (^{O}F)
Vent Piping		
Superheater to Expansion Tank	325	965
Evaporator to Expansion Tank	325	775
44 Pump to Expansion Tank Piping	325	775

Amend. 44
April 1978

TABLE 5.4-2

59 HAS BEEN DELETED

TABLE 5.4-3

IHTS MATERIAL SPECIFICATIONS

PRODUCT FORM	MATERIAL	ASME	RDT STD	SPECIFICATION
Pipe	304H SS 316H SS	SA-358	M3-7T	Electric-Fusion-Welded Austenitic Chromium - Nickel Alloy Steel Pipe for High Temperature Services
		SA-376	M3-3T	Seamless Austenitic Steel Pipe for High Temperature Central-Station Service
	2-1/4 Cr-1 Mo	SA-155	M3-11T	Electric Fusion-Welded Pipe for High-Pressure Service
		SA-335	M3-12T	Seamless Ferritic Alloy Steel Pipe for High-Temperature Service
Fittings	304H SS 316H SS	SA-403	M2-5T	Wrought Austenitic Stainless Steel Pipe Fittings
	2-1/4 Cr-1 Mo.	SA-234	M2-3T	Pipe Fittings of Wrought Carbon Steel and Alloy Steel for Moderate and Elevated Temperatures
Plate	304 SS 304H SS 316 SS 316H SS	SA-240	M5-1T	Heat Resisting Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels
	2-1/4 Cr - 1 Mo	SA-387	M5-22T	Pressure Vessel Plates, Alloy Steel, Chromium-Molybdenum
Bars and Shapes	304 SS 304H SS 316 SS 316H SS	SA-479	M7-3T	Stainless and Heat-Resisting Steel bars and Shapes for use in Boilers and Other Pressure Vessels.
Castings	304 SS	SA-351	M4-2T	Ferritic and Austenitic Steel Castings for High-Temperature Service.

TABLE 5.4-3 (Cont)

IHTS MATERIAL SPECIFICATIONS

PRODUCT FORM	MATERIAL	ASME	RDT STD	SPECIFICATION
Forgings	304 SS 304H SS 316 SS 316H SS 2-1/4 Cr - 1 Mo	SA-182	M2-2T	Forged or Rolled Alloy-Steel Pipe Flanges, Forged, Fittings, and Valves and Parts for High-Temperature Service
	2-1/4 Cr - 1 Mo	SA-336	M2-4T	Alloy Steel Forgings for Seamless Drum, Heads, and Other Pressure Vessels
9 35	Ni-Fe-Cr (Alloy 800H)	SB-407	M3-9T	Nickel Alloy for Transition Joints

TABLE 5.4-4

WELD FILLER MATERIAL SPECIFICATIONS

MATERIAL	CLASSIFICATION	ASME	RDT STD	SPECIFICATION
304 SS 304H SS 316 SS 316H SS	E308 E316-15 E316-16 E16-8-2-15 E16-8-2-16	SFA-5.4	M1-1T	Specification for Corrosion Resisting Chromium and Chromium-Nickel Steel Covered Welding Electrodes
	ER308 ER316 ER-16-8-2	SFA-5.9	M1-2T	Specification for Corrosion Resisting Chromium and Chromium-Nickel Steel Welding Rods and Bare Electrodes
2-1/4 Cr - 1Mo	Composition per RDT M1-22T	SFA-5.17	M1-22T	Bare Mild Steel Electrodes and Fluxes for Submerged Arc Welding
	Composition per RDT M1-23T	SFA-5.18	M1-23T	Mild Steel Electrodes for Gas Metal Arc Welding
Inconel 82	ER-Ni-Cr-3	SFA-5.14	M1-11T	Nickel and Nickel-Alloy Bare Welding Rods and Electrodes

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TABLE 5.4-5

59 CLASSIFICATION OF IHTS PUMPS AND VALVES

	Component	Classification	Normal Operating Mode +
	IHTS Pump	Active	100% of Rating
1	Sodium Dump Valves		
	Pump Discharge	Inactive	Isolation - NC
	Superheater Inlet Line	Inactive	Isolation - NC
59	Superheater Outet Line	Inactive	Isolation - NC
	Evaporator Outlet	Inactive	Isolation - NC
49	Exp/Dump Tank Eq. Valve	Inactive	Isolation - NC
1	Hydrogen Detector Valves*	Inactive	Isolation - NO
	IHX Vent Valves*	Inactive	Isolation - LC

+ NC = Normally Closed

NO = Normally Open

LC = Locked Closed

*These valves form part of the intermediate coolant boundary, but are not actually parts of the IHTS. The valves are discussed in Sections 5.5, 9.3, and 9.5.

TABLE 5.4-6

SUMMARY OF STRESSES IN MOST HIGHLY LOADED IHTS HOT LEG ELBOW

Condition	Location ϕ	Hoop Stresses			Axial Stresses		
		Inside	Mid	Outside	Inside	Mid	Outside
Refueling	0°	-410	4827	10,062	1950	3521	3506
	7.5°	709	4787	8865	3437	4661	5884
	37.5°	7415	4873	2331	4969	4207	3444
	90.0°	5335	4657	3979	2545	2342	2138
Normal Plus Scram	0°	-4958	3990	12,940	5603	4023	2970
	7.5°	-2498	4025	10,548	9022	7241	5460
	37.5°	15,980	4793	-6394	14,350	7256	162
	90.0°	11,847	4657	-2533	7812	1917	-3978

TABLE 5.4-7

CRITICAL CRACK SIZES FOR THE IHTS PIPING SYSTEM

Piping Loop	Critical Crack Sizes (Through-the-Wall)~Inches	
	@ Operating Pressure	@ Design Pressure
24-in. O.D. Hot Leg	16.32	9.30
18-in. O.D. Hot Leg	25.44	11.74
18-in. O.D. Cold Leg	25.44	10.95
24-in. O.D. Cold Leg	13.60	9.07

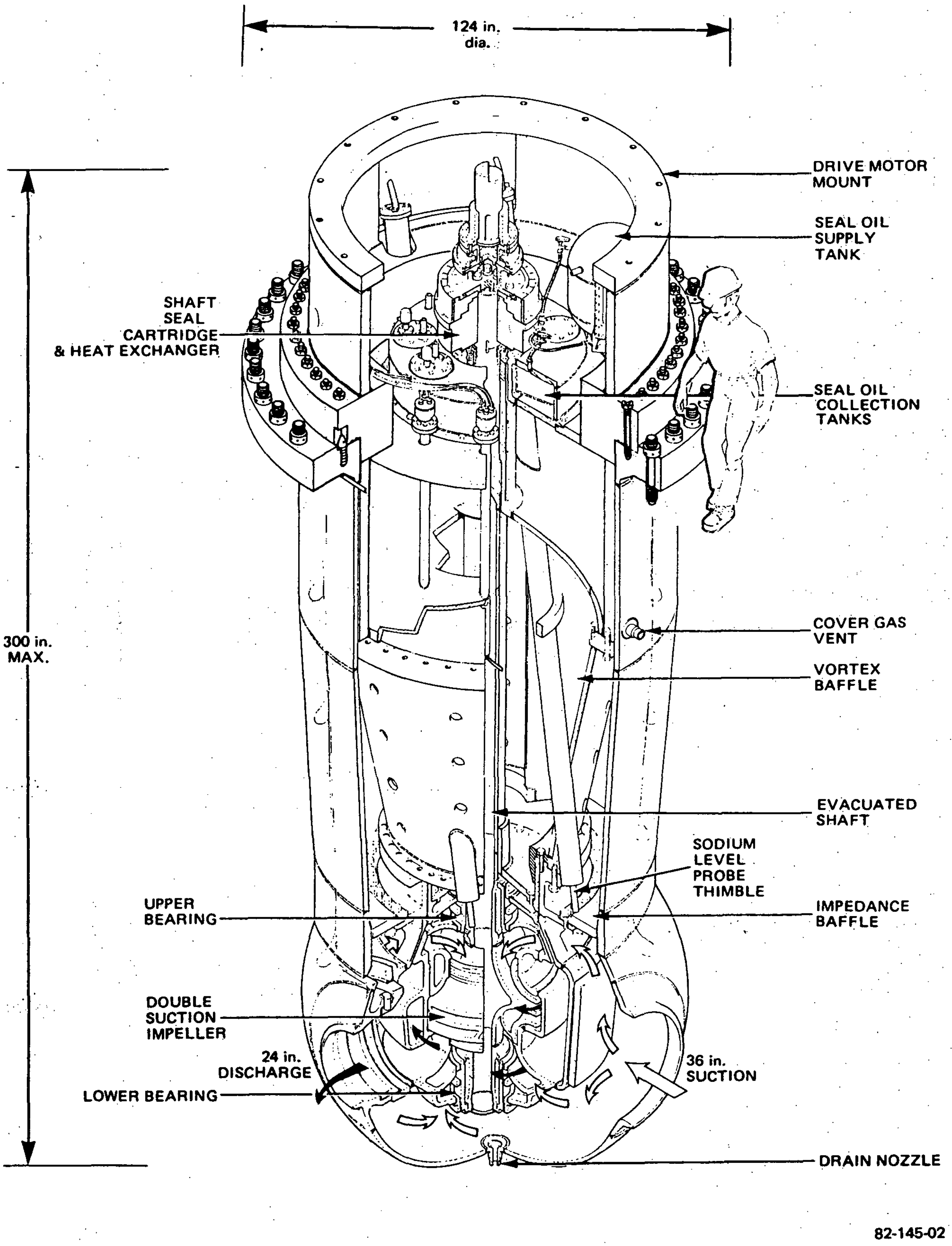

Figure 5.4 - 1 INTERMEDIATE PUMP ISOMETRIC

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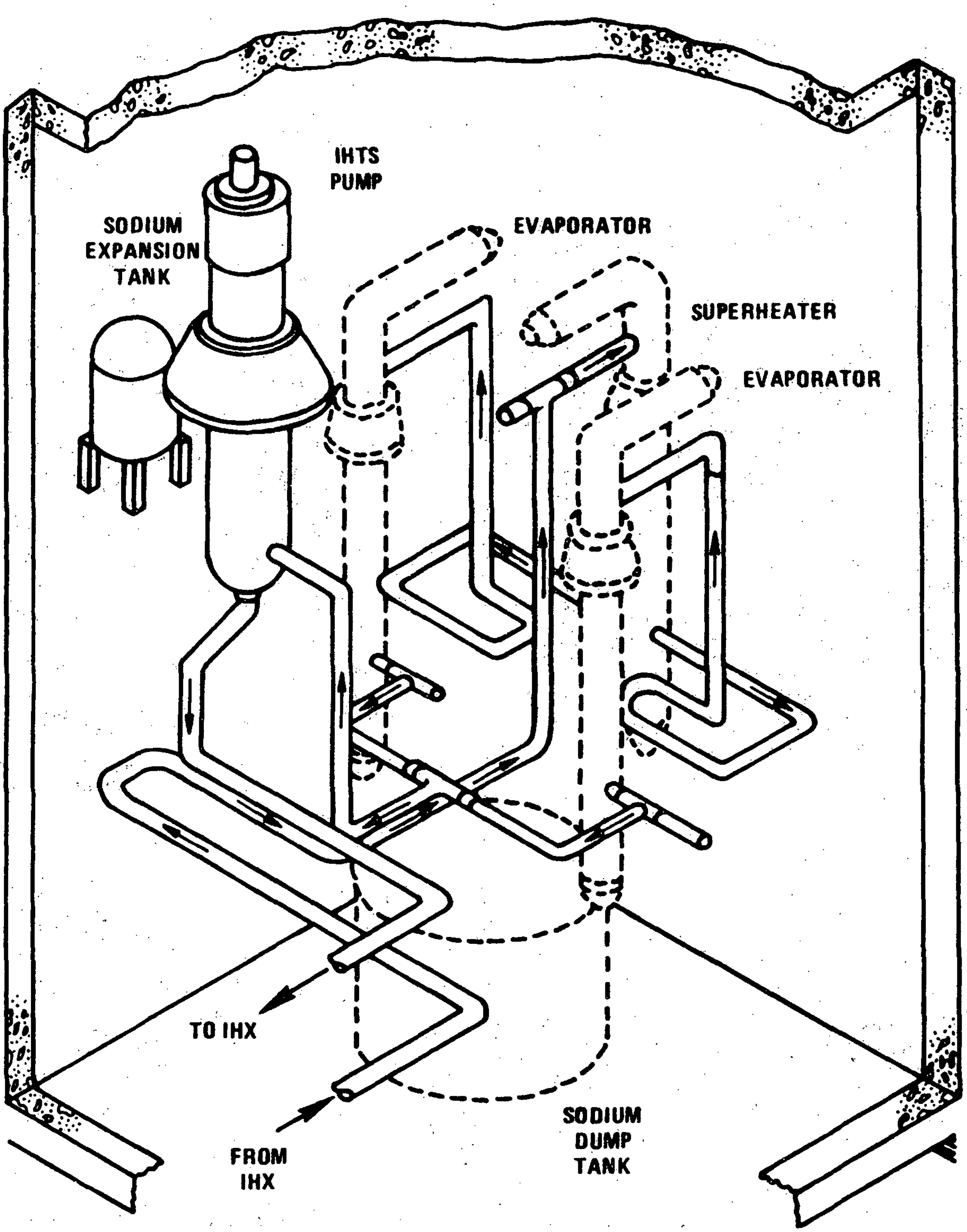

Figure 5.4-2 IHTS Components and Piping(Steam Generator Cell)

6670-16

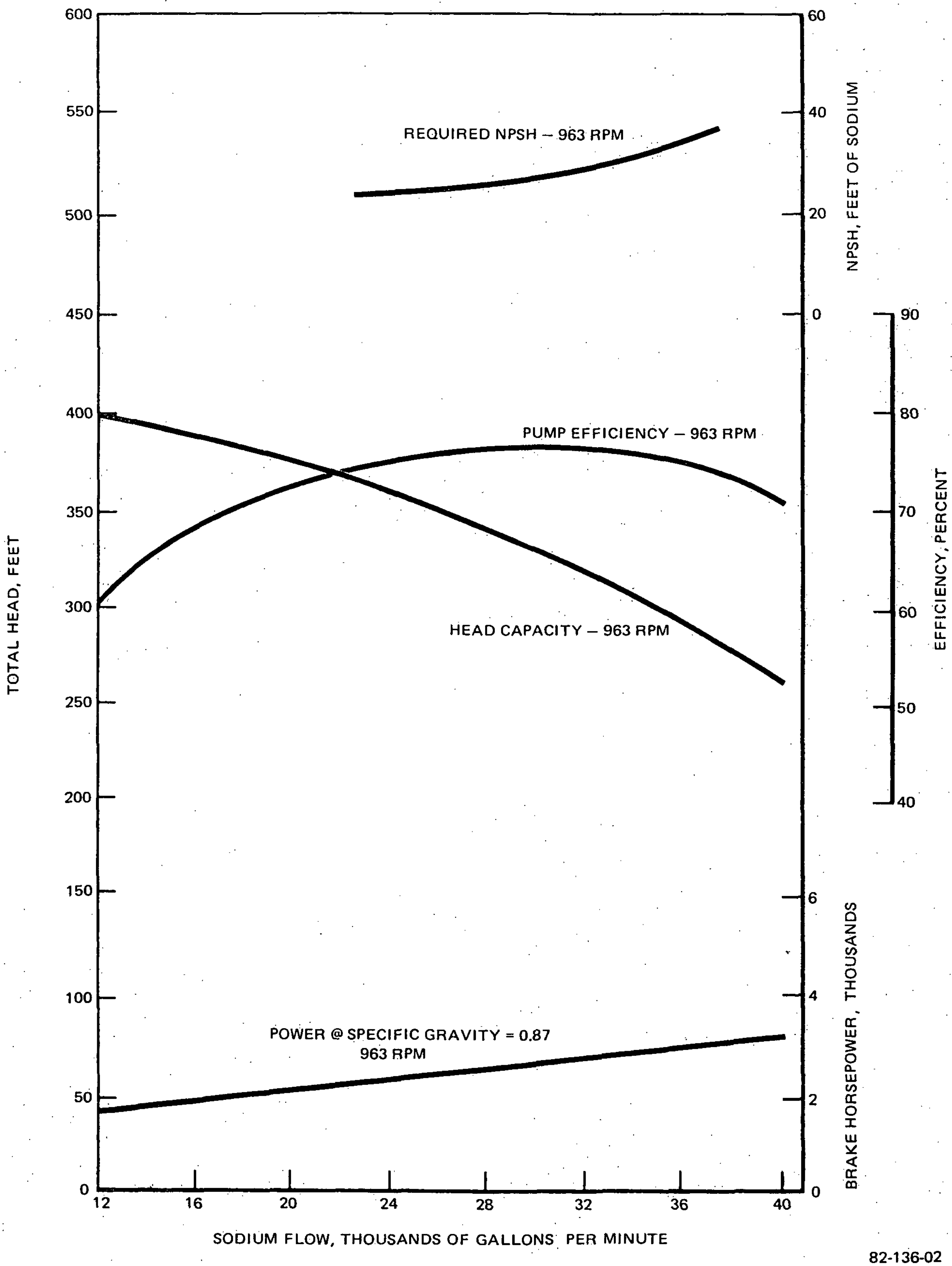

Figure 5.4-3. INTERMEDIATE PUMP CHARACTERISTICS

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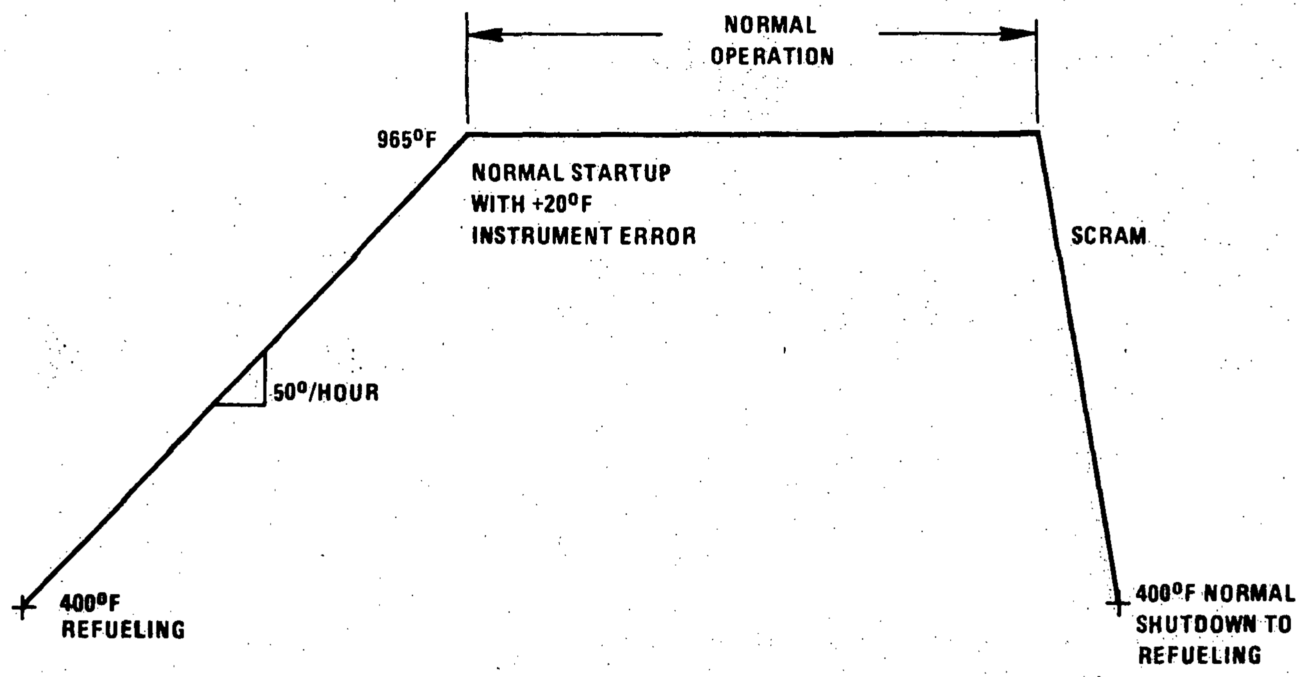
43 Figure 5.4-4 has been intentionally deleted.

43 Figure 5.4-5 has been intentionally deleted.



NOTES: 1) 600 CYCLES (Approximate Number Of Upset And Emergency Events)

2) INCLUDES WEIGHT EFFECTS

Figure 5.4-6. Loadings Considered in Crack Growth Analysis

5.5 STEAM GENERATION SYSTEM (SGS)

5.5.1 Design Basis

5.5.1.1 Performance Objectives

Steam Generator and Water Steam Subsystem

The objectives of the steam generators and water steam subsystems are to:

a. Transfer reactor generated heat (975 MWt) from the intermediate sodium to the water/steam in the steam generator modules at the rate required
41 | to maintain the intermediate sodium cold leg temperature within its operating limits.

b. Provide superheated steam at the temperature, pressure, and flow rate required by the turbine.

41 | c. Regulate the feedwater flow rate in response to plant process control over the full operating power range of 40 to 100 percent reactor thermal power.

d. Remove plant sensible and reactor decay heat from the IHTS (1) following reactor shutdown from rated power with one, two or three
59 | 41 | loops using pony motor flow on the sodium side and forced or natural circulation on the water/steam side; or with two or three loops using natural circulation on the sodium side and forced or natural circulation on the water/steam side; (2) following reactor shutdown from approximately 2/3 rated power with one or two loops using forced or natural circulation on either or both the sodium and water/steam sides.

e. Contain intermediate sodium and maintain a safe boundary between the sodium and the water/steam in the steam generator modules.

f. Prevent the water or steam side pressure from exceeding a safe value.

g. Transfer reactor generated heat from the intermediate sodium to the water/steam in the steam generator modules with two loops at nominally two-thirds rated power output.

Sodium Water Reaction Pressure Relief Subsystem (SWRPRS)

The objectives of the SWRPRS are to:

a. Prevent the pressure generated by a sodium water reaction within a steam generator module from reaching a value which would violate the integrity of the IHX primary to intermediate boundary.

b. Control the solid, liquid, and gaseous products of a large sodium-water reaction so that the solids and liquids are contained in appropriate vessels and the gas (hydrogen) is released to the atmosphere in a safe manner.

59 The SWRPRS is to be designed to accommodate the sodium-water reaction which would result from the design basis leak for a steam generator module. This event is assumed to start with an equivalent of a full guillotine rupture of a heat transfer tube in the most unfavorable location in the unit, which causes the equivalent of two additional guillotine tube ruptures. Justification for the selected design basis leak is given in Section 5.5.3.6. Water or steam flow is injected into the sodium mass from both ends of the tubes, causing high pressures and acoustic waves in the intermediate sodium system. The pressures will burst the rupture discs to the SWRPRS, permitting sodium, sodium-water reaction products, and steam and water to be ejected into the SWRPRS. 9

Leak Detection Subsystem

The objective of the leak detection subsystem is to detect and alert the operators of water-to-sodium and steam-to-sodium leaks in the steam generator modules and to identify the module in which the leak has occurred. See Section 7.5.5.3 for further details.

Sodium Dump Subsystem

59 The objective of the Sodium Dump subsystem is to provide drainage capability and storage for the IHTS sodium, which may be contaminated with sodium-water reaction products following a sodium-water reaction within a steam generator.

The Sodium Dump Subsystem for each Intermediate Heat Transport System (IHTS) shall meet the following design basis:

a. Each subsystem shall accommodate and store all IHTS sodium during
59 drainage.

b. Design shall provide capabilities for the Intermediate Sodium Service System to clean the sodium from the Sodium Dump Subsystem and transfer or fill the IHTS loop, if necessary.

During a rapid dump, the sodium from the steam generators is drained through the dump lines in less than twenty minutes. When the IHTS sodium is dumped in this fashion, the IHX remains essentially filled with sodium because drainage is through the IHTS piping which enters and leaves at the high point of the IHX. 9

59 Water Dump Subsystem

The Water Dump Subsystem for each SGS loop shall meet the following design bases:

a. The Water Dump Subsystem in combination with the power relief valves will be designed to reduce the evaporator operating pressure to about
59 35 300 psig within 30 seconds.

b. Each Water Dump Subsystem shall have sufficient capacity to store the
41 liquid water dumped after isolation of both evaporators in one loop.

c. The water from the Water Dump Subsystem will be drained for reuse or disposal.

d. No single failure of the isolation and dump equipment shall cause the loss of shutdown heat removal capability. In addition, no single failure shall cause the two water dump valves in the same dump path to
59 open.

SGS Design Parameters

Table 5.5-1 lists the structural design temperature, pressure and minimum test pressure for the SGS components. The specified operating transients for SGS components are given in Appendix B of the PSAR. Refer to Section 3.7 for discussion of the input criteria for seismic design of Category I structures, systems and components.

Design requirements for the SWRPRS rupture disks are to assure that the disks will rupture at differential pressures low enough to prevent a loss of integrity of the IHTS and IHX as a result of over-pressures produced by a large sodium-water reaction, and high enough to maintain system operability under all other normal, upset, emergency and faulted plant conditions.

The features which protect the principal components of the SGS against environmental effects are discussed in Section 5.5.3.11.

5.5.1.2 Applicable Code Criteria and Cases

The Steam Generation System will be constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code and applicable code cases. The
59 steam generator modules are classified as Class 2 components but shall be
91 constructed to Class 1 rules, while the remainder of the SGS Components and
59 Piping shall be designed as Class 2 or 3. Class 2 and 3 components will use supplemental Code Cases 1606 and 1607 as applicable. Steam Generation System Piping Classifications are given in Section 5.5.2.3.3.

Construction of all ASME Section III components will be supplemented with appropriate sections of RDT standards (see Sections 5.5.2.3.4, 5.5.3.1.5, and 5.5.3.11.2). Mandatory application of these RDT standards to the steam
41 generator module will be limited to the water/steam to sodium boundary materials.

The SGS modules have normal operating temperatures exceeding those for which
59 allowable stress values are given by Section III. For these high temperature components, the design rules provided in several code cases will be made mandatory. A list of the mandatory code cases is given in Table 5.5-2 with a brief description of their applications. If the rules provided in these code cases are not complied with, justification for using less stringent rules will be provided.

5.5.1.3 Surveillance Requirements

Surveillance requirements for the SGS are defined by the appropriate Section III (Class 1, 2 or 3) of the ASME Code.

5.5.1.4 Material Considerations

High Temperature Design Criteria

The high temperature design bases for the steam generator modules of the Steam
41 Generation System are the same as those identified for the Primary Heat Transport System in Section 5.3.1.4.

Material Specifications

A list of material specifications for the SGS vessels, piping, pumps and valves is given in Table 5.5-3. Corresponding weld material specifications are also listed in Table 5.5-4.

5.5.1.5 Leak Detection Requirements

The steam generator leak detection system provides early detection of possible water to sodium leaks in the steam generator modules. For small leaks, operator corrective action is taken, for large leaks, automatic corrective action is taken.

For further details, see Section 7.5.5.3.

5.5.1.6 Instrumentation Requirements

Section 7.5.2 provides the instrumentation requirements for the Steam Generation System.

Control Systems

Feedwater Flow Control

Instrumentation and control equipment is provided for the automatic regulation of feedwater flow to the Steam Generation System. Details of this control system are discussed in Section 7.7.1.5.

Recirculating Water Flow Control

The recirculation pump runs at constant speed for the full range of power operation and no automatic control system is required. See Section 7.7.1.6.

Sodium Dump and Water Dump Interlocks

Suitable interlocks will be provided to minimize the possibility of erroneous dumps.

Sodium Dump Tank Pressure Control

Instrumentation and control equipment is provided to maintain the argon cover gas pressure over the sodium dump tank within prescribed limits. Details of this control system are covered in Section 7.7.1.7.

Sodium-Water Reaction Pressure Relief Control

In the event a large or intermediate water-to-sodium leak occurs in one of the steam generators, instrumentation and control equipment is provided for (1) automatic isolation of the steam generator modules, (2) venting of water and steam, and inerting the system with nitrogen. Details of this system are discussed in Section 7.5.6.

5.5.2 Design Description

5.5.2.1 Design Methods and Procedures

5.5.2.1.1 Identification of Active Pumps and Valves

Pumps and valves within the Steam Generation System are classified as active or inactive in Table 5.5-5.

Leaktightness capability requirements for all active valves shall be included in the applicable valve specifications. Valve parts forming the pressure boundary shall be pressure tested per the requirements of NC-6100 of ASME Section III. The maximum allowable leak rate past valve seats shall be as specified in the particular valve specification.

Table 5.5-12 provides the maximum allowable leak rate through pressure boundary valves. The bases for the leak rates are:

(1) limit the steam and water leakage from the SGS during SGAHRS operation to conserve the auxiliary feedwater supply

(2) minimize steam and water losses through relief and water dump isolation valves during normal plant operation

(3) limit the amount of steam and water available for sodium-water reaction after failure of the sodium/water boundary in a steam generator module

The values listed for the isolation valves are leak rates with a 30 psi pressure differential across the valve seat. Since the pressure is applied over the valve seat, the leakage will decrease with increased differential pressure across the seat.

The values for the relief valves are with the pressure at the valve inlet at 90% of the valve set pressure.

5.5.2.1.2 <u>Design of Active Pumps and Valves</u>

In order to assure the functional performance of active components of the SGS, the active ASME Section III Class 2 or 3 valves will be designed according to the specifications of Reference 12, PSAR Section 1.6.

44 19

5.5.2.1.3 Surveillance and In-Service Inspection Program

Design of the steam generator system piping and components will include provisions for anticipated in-service inspection requirements of critical pressure boundaries and welds in these boundaries. Section XI of the ASME Boiler and Pressure Vessel Code currently covers requirements for water moderated reactor systems, but a code committee has been formed for the purpose of developing in-service inspection requirements for liquid metal cooled fast reactor systems. The requirements for ISI of the SGS will be based upon the information available from the committee.

Emphasis will be placed on certain critical pressure boundary welds of the intermediate sodium system. These will include the material transition pieces in the sodium piping where piping material is changed from stainless to chrome-moly steel, and certain welds in critical locations in the shell of the steam generator units. The material transition pieces will be located in straight sections of piping which are readily accessible to permit relatively frequent inspection. In addition, some of the heat transfer tubes of the steam generators will be inspected for wear and corrosion during periodic in-service inspections in a program similar to that which will be established for inspection of the reactor coolant pressure boundary welds. The inspections will probably be made from the water side of the tubes by means of an inspection probe which can be passed through the tubes.

The in-service inspection program for the steam drum, recirculation system and water-steam, ASME Code Class III, piping and valving will be according to the inspection requirements of Section XI of the ASME code.

5.5.2.1.4 Protection Against Accelerated Corrosion and Material Degradation

A discussion of corrosion protection is given in Section 5.5.3.11.

5.5.2.1.5 Material Inspection Program

The SGS material inspection program will be based on the requirements of the ASME Code, Section XI, for carbon steel and 2 1/4 Cr-1 Mo steel. Piping of stainless steel will be inspected as set forth in Appendix G.

5.5.2.2 Material Properties

The materials used in the SGS are discussed in Sections 5.5.3.11.

5.5.2.3 Component Descriptions

5.5.2.3.1 Valves

Line valves furnished shall be standard manufactured types, designed and constructed in accordance with the requirements of ASME Section III for Class 2 and 3 valves. All materials, exclusive of seals and packing, shall be designed for a 30-year plant life under the environmental conditions applicable to the particular system.

Power operators shall be sized to operate successfully under the maximum differential pressure delineated in the design specification.

The main steam isolation valves (superheater outlet isolation valves and superheater bypass valves) are automatically closed to stop the venting of steam into the steam generator or turbine buildings in case of a steam line pipe break downstream of the isolation valves. The maximum steam flow rate is expected from a steam line break immediately downstream of the isolation valve; thus the disc and stem of each valve has been designed to withstand the forces produced when closing the valve under choked flow conditions. These are active valves, designed for quick closing upon interruption of the electrical control signal.

Figure 5.5-2A shows a main steam isolation valve. It is a conventional gate valve to provide a minimum resistance flow path when the valve is wide open. A schematic of the quick-closing electro-hydraulic actuator used on the main steam isolation valves is provided in Figure 5.5-2B. Upon interruption of electrical power to the fail-position solenoid (SV3), accumulator pressure opens a pilot-operated check valve to pressurize the top of the actuator cylinder to close the valve. The hydraulic oil below the actuator piston returns to the reservoir through a pilot-operated check valve and the pressure-compensated flow control valve. The flow control valve ensures uniform valve closing times over wide variations in load, and the moving parts are decelerated at the end of the closing stroke by a hydraulic damper which provides soft seating of the gate valve discs. Position switches are provided to indicate gate position remotely for instrumentation and control purposes. The valve is normally repositioned by energizing the hydraulic pump motor and controlling the direction solenoid (SV2). The pump motor is tripped by pressure switch PS1 or PS2 when the proper opening or closing pressure is attained. The accumulator is maintained at pressure by automatic energization of the pump motor and recharge solenoid (SV1) when the pressure switch (PS3) senses low accumulator pressure.

Each valve used in the SGS was evaluated as to its performance relative to plant safety and mode of operation in the event of electrical power failure (fail open, fail closed, or fail as-is, etc.). As part of these evaluations, the need for a pneumatic accumulator adjacent to a valve and solenoid uninterruptible power requirements for emergency operation were determined.

<u>Tests and Inspections</u>

Line valves will be shop tested by the manufacturer for performance according to the design specifications for leakage past seating surfaces and for integrity of the pressure retaining parts. Line valves will be manually operated during loop shutdown periods to assure operability.

5.5.2.3.2 Recirculation Pumps

The recirculation pump is a single stage, centrifugal type, driven by a constant speed, 4.0 KV, 1000 HP motor. It takes suction from the steam drum, and provides 2.22×10^6 pounds of water per hour to the evaporators.

The pump and its support was designed and fabricated per ASME Section III, Class 3 as shown in Table 5.5-6.

5.5.2.3.3 Steam Generation System Piping

Design Basis

Steam Generation System (SGS) piping for each loop shall meet the following design bases:

1. The design shall accommodate operational stresses, such as internal pressures and safe shutdown earthquake loads without failure.
2. The piping will accommodate the worst possible loading from the duty cycle (Appendix B) according to the design requirements for the water/steam side SGS piping given in Table 5.5-7.
3. The design of sodium piping shall be the same as for the Intermediate Heat Transport System as described in Section 5.4.2.3.4.

4. Piping shall be designed with suitable access to permit in-service testing and inspection.

5. All "horizontal" piping shall be sloped. Steam traps and drain valves shall be located at the low points to permit complete draining of the piping.

6. Piping sizes shall be chosen such that average fluid velocities at the 100% plant power condition will not exceed the following values:

   a. water 25 fps

   b. water-steam mixture 50 fps

   c. saturated steam 125 fps

   d. superheated steam 175 fps

System Description

All Steam Generation System piping is shown in Figure 5.1-4. The design characteristics and ASME Code classifications are presented in Table 5.5-7.

The only field run piping planned for the steam generator system is non-safety class piping. The internal diameter of the piping will be 2 inches or less and is used for drain lines from steam traps. The design pressure would not exceed 100 psia and the design temperature would be less than 300°F.

The Seismic Category I design requirements are placed on the Steam Generation System's steam-water piping. Superheater and evaporator modules and the steam drum are provided with quick acting isolation valves. Design pressures of all piping are nominally 110% of the operating pressure at rated power.

The use and location of rigid-type supports, variable or constant spring-type supports, and anchors or guides will be determined by flexibility and stress analysis. Piping support elements will be as recommended by the manufacturers and will meet applicable code requirements. Direct weldment to thin wall piping will be avoided where possible.

Attachment and penetrations shall be designed and fabricated according to the ASME Code requirements.

Design loading used for flexibility and seismic analysis for the determination of adequate piping supports will include all expected transient loading conditions. Spring-type supports will be provided for the initial dead weight loading during hydrostatic testing of steam systems to prevent damage to piping supports.

Test and Inspection

In-service inspection is considered in the design of the main steamwater and feedwater supply piping. This consideration assures adequate working space and access for the inspection of selected pipe segments.

After completion of the installation of a support system, all hanger elements will be visually examined to assure that they are correctly adjusted to their cold setting position. Upon hot start-up operations, thermal growth will be observed to confirm that spring-type hangers are functioning properly. Final adjustment capability will be provided for all hanger or support types.

5.5.2.3.4 Steam Generator Module

The steam generator module shown in Figure 5.5-2 is a shell and tube heat exchanger with fixed tubesheets. Flow is counter-current, with sodium on the shell side and water/steam on the tube side. The evaporator modules transfer heat from the sodium and generate 50 percent quality steam from the subcooled recirculation water. The steam-water mixture exiting from the evaporator is separated into saturated water and saturated steam in a steam drum. The superheater modules transfer heat from the sodium to superheat the saturated steam to the temperature required for admission to the turbine.

The Atomics International - Modular Steam Generator (MSG) was a 32.1 Mwt maximum power, hockey stick designed unit used as the basis for the CRBRP Steam Generator design. The salient features of the MSG unit are as follows:

• Maximum Power	32.1 Mwt
• Temperature	930°F
• Pressure	2550 psig
• Startup/Shutdown	37 Cycles
• Tube Design	158 Tubes 5/8 in. O.D. x 109 mil. wall
• Length	66 ft
• Material	100% Ferritic Steel - 2 1/4 Cr-1 Mo

For further details see Reference 4.

Evaporator and superheater modules are identical in all respects except for the inlet orifices that may be added to the evaporator tubes at the lower tubesheet to increase the evaporator water flow stability margin. Each module consists of a 53 1/2 inch O.D. shell containing a tube bundle with locations for 739 5/8 in. O.D. x 0.109-inch wall tubes. The design employs autogeneous buttwelded, tube-to-tubesheet joints. The shell and tube material is 2-1/4 Cr-1 Mo steel. There are two evaporator modules and one superheater module per loop. The evaporator modules operate with a recirculating ratio of 2:1.

The steam generator design requires that each loop (two evaporators and one superheater) develop 325 MWT at rated full load. The design life of the steam generator module is 30 years with items that cannot be reasonably expected to last 30 years being replaceable during in-service inspection periods. The steam generator is designed to withstand the normal, upset, emergency and faulted operating conditions in accordance with CRBRP Criterion 26. The steam generator module is also designed to withstand the loading combinations indicated in Section 3.9.2.2.

The materials used in design of the steam generator module major components are as follow:

Pressure Boundary (2-1/4 CR-1 Mo-Ref. 3, Vol. 1, Section 2.2):

Shell Forgings	-	SA 336, Class F22A
Shell Plate	-	SA 387, Grade 22, Class 1
Tubesheet Forgings	-	RDT M2-19 with optional provision 2 and Code Case 1557-2
Tubing	-	RDT M3-33 as modified to limit silicon and carbon content for weldability and carburization considerations (Reference 5).

The steam generator module supplier will provide procedures for welding and heat treating in accordance with the requirements specified in the Code as modified by RDT E15-2NB (see Section 5.5.1.2). Welding qualification is controlled by the Code as modified by RDT F6-5 and RDT E15-2NB (see Section 5.5.1.2).

The weld method employed in the tube-to-tubesheet welds of the CRBRP Steam Generators is an in-bore butt weld. It was selected to avoid the crevices which exist if a front face fillet weld would be used. The weld method as well as the welding equipment has been utilized before. A development program was aimed at the improvement of the weld quality and dependable repeatability of the process. The measures taken to this end are described as follows:

For the steam generator tube-to-tubesheet welds, the ASME Code requirements (NB-4000 and NB-5000) were supplemented requirements of RDT E15-2 and additional requirements. Requirements imposed on the tube-to-tubesheet welds above those of the Code include:

- o Vacuum-Arc Remelt or Electroslag Remelt - material is specified to reduce impurities and improve properties for tubesheet forgings and tubes.
- o Post weld heat treatment range defined to optimize resistance to caustic stress corrosion cracking.
- o Helium leak test.

- o Penetrant test requirement limiting defect size to much less than that of the Code.
- o Weld geometry requirement limiting concavity, convexity and wall thinning.
- o Micro-focus radiographic examination - developed to radiograph tube-to-tubesheet welds with improved resolution.

Material integrity prior to placing the steam generators in service will be assured by complying with the ASME Code Section III which requires weld radiography, tubing ultrasonic testing, plate ultrasonic testing, tubing hydraulic testing, component pressure testing and helium leak testing.

Material considerations are indicated in Sections 5.5.1.4 and 5.5.3.11. Section 5.5.3.1.5 indicates the tests being conducted to support the steam generator design. It is not anticipated that back-up materials will be required.

The following is a description of the steam generator modules as shown in Figure 5.5-1.

a. Tube-Bundle Arrangement

The tube size was selected as 5/8 inch OD x 0.109 inch wall tubing based on optimization studies performed as part of the prototype design program. The tubes are positioned on a 1.22 inch triangular pitch. The effective tube heat transfer length is 46 feet. The tubes are supported along their length by Inconel 718 tube spacer plates in the active heat transfer region and by Inconel 718 tube support bars in the elbow region. Axial spans between the tube spacers and the tube support bars were determined by analysis and tests to keep the tube vibration levels significantly below acceptable levels.

The tube bundle is enclosed in a 36.5-inch I.D. shroud along the active heat transfer length to insure proper flow distribution. An elbow shroud is provided in the hockey stick region to protect the outer shell from the direct effects of wastage in the event of a water/steam to sodium leak. Sodium flow enters the module through the upper sodium nozzle, and turns up into the flow distributing annulus. The sodium then enters the tube bundle over the top of the shroud. The sodium then flows into and down through the tube bundle. At the lower end of the module, the sodium exits the tube bundle below the shroud, turns up and flows through the annulus, and flows out the exit nozzle(s).

b. Other Internals

Thermal baffling covers the face of the lower tubesheet. These baffles serve several purposes: 1) they protect the lower tubesheet from sodium outlet temperature transients, 2) they partially fill the volume below the level of the 6-inch drain nozzle, and 3) they act as wastage baffles in the area of the tube to tube-sheet welds. The baffling is attached to the sodium side of the tubesheet in a manner which allows lateral motion of the baffling to occur without placing loads upon the tubes.

Clearance between the tube and baffle tube hole is large enough to vent the products of a sodium-water reaction.

The effects of sodium-water reaction to the wastage baffles have been tested (Reference 17). One of the primary objectives of the test series was to verify that the reaction products escape from the narrow annulus between the plates and tubes. The report concludes, "there is a clear indication of flow away from the leak site and no severe building of solid reaction products occurred in the gap or the outside of the plates".

At the upper tubesheet, wastage baffles are also provided which serve the same purpose as the lower wastage baffle.

Plates and seals in the annular region serve to protect the hockey stick region from thermal eddies induced by the sodium inlet flow. This arrangement also provides an adequate relief path for sodium-water reaction pressure that would occur in the hockey stick region in the unlikely event of a large tube rupture. Plates seal the exit annulus at the lower end of the active region.

An upper header thermal liner and an inlet nozzle thermal liner are provided to mitigate the effects of system sodium transients.

c. Shell Arrangement

(1) Major Components of Shell

The shell connects to an upper and lower tubesheet, and consists of two reducers, an elbow, an inlet header "tee" section, an outlet header "cross" section, a main support section and a main shell section. These components have been sized structurally to contain postulated maximum large leak SWR conditions as well as meet design operating conditions.

(2) Shell Penetrations

Each superheater and evaporator module is fitted with one inlet sodium nozzle and two outlet sodium nozzles. Present intermediate sodium loop arrangement drawings show both superheater outlet nozzles being used, while only one of the two outlet nozzles is used on each of the two evaporator units. The spare evaporator exit nozzles are capped. The inlet sodium nozzle is a 30-inch nozzle that attaches to the 4 1/4-inch thick inlet sodium header in the direction of the hockey stick. The 30-inch nozzle is reduced to a 26-inch, 1-inch thick wall pipe, which will be mated to the loop piping. The two outlet sodium nozzles are 22-inch nozzles that attach at 90° to the direction of the hockey stick to the 4 1/4-inch thick outlet sodium header. The 22-inch nozzles reduce to 18-inch, schedule-60 pipes, which will be mated to the loop piping. The purpose of the oversized nozzles in regard to the piping size is to provide space in the nozzles for thermal liners and to reduce flow velocities in the inlet/outlet regions.

Two 8-inch sweepolets are attached to the reducers located at both tubesheets. These serve as ports to inspect the final closure welds. Also, one of the ports on the lower reducer is attached to a 6-inch schedule-80 pipe by a transition section to provide for rapid drainage of the lower stagnant end of the modules, should it be required. Again, the purpose of the transition section is to provide for possible lining of the nozzles. A one-inch drain is also provided through the lower tubesheet to drain the lower thermal baffle region. A three-inch sodium bleed vent is provided in the hockey stick end of the module to provide for: 1) venting during initial filling of the shell side, and 2) a small sampling flow to a hydrogen detector to allow detection of any small leak in that region during operation.

(3) Steam/Water Heads

The steam/water heads are integrally welded to the tubesheets. The steam piping is in turn welded to the steam heads. An integral steam head provides an enhanced maintenance capability since 1) the heads are not removed for in-service inspections, 2) drainage of the module is not required since the integral steam head will serve as the tank to contain the water medium and 3) the air/water exposure of the steam tubes will be minimized. The welded steam head also significantly reduces potential steam water leakage by exchanging a large diameter steam head seal for a smaller diameter manway seal which is relatively insensitive to distortion and leakage during normal transients.

Steam Generator Inspection

Access to the heat transfer tubes of the steam generator is readily obtained by removal of the manway nuts and removal of the manway cover. The steamhead is basically a 32 inch radius sphere which provides larger stress margin than the alternate bolted design. The manway is a standard 16 inch diameter port. The 57 inch ID sperical head provides adequate space and headroom for inspections and maintenance and tube plugging as required. The upper steamhead also serves as the water tank for in-service inspection (ISI).

The inner diameter of the heat transfer tube is readily available for inspection by ultrasonics, eddy current and/or other suitable means which will be determined acceptable at the conclusion of a development program (now in progress). The outer surface of the heat transfer tubes cannot be readily inspected since the shell of the steam generator is a fully welded assembly. However, it is expected that the above tube inspection techniques will give sufficient information on the condition of the tubes to provide assurance of integrity of the sodium/water boundary.

5.5.2.3.5 Steam Drum

The steam drum, shown in Figure 5.5-4, is a horizontally mounted 82 inch O.D., 35 ft. long cylinder with hemispherical heads (42 ft. overall length). Most of the major nozzles are located in a vertical plane through the steam drum centerline. These consists of one 12 inch steam outlet nozzle located at vessel midpoint and directed vertically upward, two 16 inch riser nozzles (evaporator return) located at approximately cylinder quarter points and directed downward, four 10 inch downcomer nozzles (recirculation pump suction) spaced evenly along the cylinder and directed downward, one 6 inch continuous drain nozzle located in one head and directed downward normal to the head at a 45° angle to the vertical, and one 10 inch feedwater inlet nozzle located in the opposite head and directed downward normal to the head at a 45° angle to the vertical. The only nozzle that is not coplanar with the vessel centerline is the auxiliary feedwater nozzle. This is a 4 inch nozzle located on the same head as the main feedwater inlet nozzle in a vertical plane rotated 45° from the vessel centerline; the nozzle is directed downward normal to the head at a 45° angle to the vertical.

Functionally, the drum receives a saturated water/steam mixture from the evaporators and subcooled feedwater and produces saturated steam of low moisture content for the superheater and subcooled water of low steam content for the recirculation pump. The water/steam mixture from the evaporators enters the drum through the water/steam nozzles and flows into an annular volume along the sides of the inner drum wall created by a girth baffle extending along the side of the drum for the length of the cylinder. Centrifugal steam separators mounted along the length of the drum draw from this annular volume, separate the mixture into phases, and direct the steam upward and the water downward into the inner volume of the drum. The main feedwater enters the drum through a single nozzle which feeds two distribution pipes through a "Y" connection inside the drum. The feedwater is distributed along the length of the drum by rows of orifice holes in the two pipes which are located along each side of the drum beneath the steam separators. The auxiliary feedwater enters through a separate nozzle and is distributed along the length of the drum by two rows of spray nozzles in a single distribution pipe located above the water level in the drum. To preclude waterhammer due to injection of highly subcooled water interfacing with saturated steam within a closed volume, the spray line is vented by the nozzles plus 18 7/32" OD holes. These vents ensure that the feed line will remain full of water at the temperature of the steam drum inventory. Verification will be performed during pre-operational and startup testing that these features operate as intended and are effective in precluding waterhammer due to highly subcooled water interfacing with saturated steam in the steam drum. Other waterhammer loads (e.g., sudden reduction in feedwater velocity) have been evaluated and found to be acceptable. Feedwater mixes with the water from the separators and is drawn downward and out through the water outlet nozzles by the recirculation pump. The steam passes upward through chevron type dryers in the upper portion of the drum and out through the steam outlet nozzles to the superheater. The dryers remove all but the last fractional percent of the moisture from the steam and drain this moisture back to mix with the resident drum water. Drum drain piping, located along either side of the drum in the region where the water from the separators enters the drum inner volume, draws water of high impurity concentration from the drum.

5.5.2.4 Overpressure Protection

Location of Pressure Relief Devices

Safety/power relief valves are located in the steam generation system to:

1. Prevent a sustained pressure rise of more than 10 percent above system design pressure at the design temperature within the pressure boundary of the system protected by the valve under any pressure transients anticipated; and

2. Provide steam generator module blowdown capability.

Installation of the valves will comply with the requirements as specified in Section 3.9.2.5. Safety/power relief valves are installed on the outlet lines from each evaporator to provide venting capability and a portion of the required safety/relief capability. Safety valves are installed on the steam drum to provide the remainder of the safety capability for the recirculation loop. Additional safety/power relief valves are installed on the steam exit line from the superheater because the steam lines to and from the superheater have isolation valves. The P&ID for the Steam Generation System, Figure 5.1-4 shows the locations of these safety/power relief valves. The power operation features of the relief valves is fail closed to assure continued integrity of the system. In addition, an acoustic sensor is located on the outlet of each valve to inform the operator that the valves are not opening or not closing. Additional details of sizes and pressure rating are given in Table 5.5-8.

Pressure Relief Devices

Water/Steam Side

Each safety relief valve on the evaporator outlet piping provides a saturated steam (100% quality) relief capacity of 430,000 lb/hr, or 39% of the rated steam generating capacity of the recirculation loop. Each safety relief valve on the steam drum provides a saturated steam relief capacity of 410,000 lb/hr, or 37% of the rated steam generating capacity of the recirculation loop. The difference in rated capacity of these valves is due to the difference in the valve set pressure. The combined relief capacity of the six valves for the recirculation loop is therefore 230% of the rated steam generation capacity. This generous margin is provided for two reasons: (1) the capacity required to relieve most of the overpressure transients in the recirculation loop can be satisfied by opening one or both of the steam drum valves, relieving the system with dry steam rather than wet steam; (2) the capacity of the evaporator relief valves is based on the capacity required to achieve rapid blowdown of the evaporator modules following a water to sodium leak.

Three safety/power relief valves installed on the exit line from the superheater provide a relief capacity of 75% of rated superheater steam flow at a pressure of approximately 1800 psig and temperature of 900°F. The remaining 25% of rated flow is relieved by the steam drum valves.

Settings for the safety/power relief valves are in accordance with Code requirements. Setting presently selected are shown in Table 5.5-8.

The capacity requirements stated for the safety/power relief valves are based
upon the steady-state plant ratings. During the plant design period, it was
59 determined that the relieving capabilities required by the steady-state
conditions are adequate for all plant unbalanced or transient conditions.

The safety/power relief valves and the piping for these valves will be
subjected to sizable impulse forces during operation of these valves. The
relief/safety valve reaction forces, due to dead weight, seismic events,
discharging fluid, thermal expansion, and the dynamic effects of valve opening
or closure, will be accommodated with the use of supports on the valve
discharge piping and proper sizing of the inlet nozzle to limit the stress in
the valve and nozzle within Code allowables. As details of the installation of 25
these valves and the piping were established, the magnitudes of these loads
were determined and suitable restraints for the valves and piping were
provided. The discharge from the valves are routed to a suitable location on
the roof where plant personnel will not be exposed to hazards from the
59 discharge.

Sodium Side

The IHTS is protected by pressure-relief devices in the steam generator IHTS
sodium piping and in the expansion tank/dump tank gas equalizer line. Each
device consists of two rupture discs installed in series that relieve to the
41 Reaction Products Separation Tank and sodium dump tank respectively. Between
each pair of rupture discs a sodium leak detector is installed so as to give an
alarm in the event the sodium-containing rupture disc develops a small leak. A
leak developed by the downstream rupture disc would not affect the safety
function of the pressure-relief assembly. Three pressure sensors are located
41 immediately downstream of the IHTS sodium piping rupture disc assemblies. A
coincident signal from any two of the three pressure sensors initiates an
automatic loop shutdown and water/steam side pressure reduction. See Section
7.5.6 for plant control actions following rupture of a rupture disc.

Rupture disc assemblies are located on the 24-inch sodium inlet piping to the
superheater, each evaporator 18-inch outlet piping, and in the IHTS expansion
41 tank to sodium dump tank gas equalizer line. Their capacity and the magnitude
and application of any reactive forces generated on the system or its
components will be included in the description of the Steam Generation System.

5.5.2.5 Leak Detection System

The steam generator leak detection system description is provided in Section 7.5.5.3.

5.5.2.6 Sodium Water Reaction Pressure Relief System (SWRPRS)

35 The SWRPRS includes piping, reaction products separator tanks, rupture disc
assemblies, stacks and non-return valves, and ignitors. The system for each of
54 the three heat transfer loops is identical.

Dual rupture disc assemblies are installed on the intermediate sodium piping
adjacent to the steam generator units and in the IHTS expansion tank to sodium
dump tank gas equalizer line. For large sodium-water reactions excessive
42 pressure in the intermediate sodium system will burst the rupture discs,
59 41 dumping sodium and reaction products to the separator tanks. Gaseous reaction
products and any entrained sodium and liquid or solid particulate where the
major portion of the extrained particulate matter will be removed. The gaseous
41 reaction products are then directed to the flare stack where an igniter is
installed. The dual rupture disc configuration is used to avoid complete
dumping of the intermediate sodium system into the SWRPRS in case of a leak in
the first disc.

Intermediate size steam or water leaks in the steam generator modules up to
approximately 2 lbs/sec can be accommodated without failure of the SWRPRS main
rupture discs. This is accomplished by relieving IHTS system pressure at the
expansion tank through the expansion tank/dump tank gas equalization line
41 rupture disc to the sodium dump tank.

A large sodium-water reaction may inject slugs of sodium into the piping from the rupture discs to the separator tank. These slugs can be accelerated rapidly by the hydrogen gas pressure resulting from the sodium-water reaction, resulting in high velocities for these slugs while in the pipes and high reaction forces in the piping. The results of the TRANSWRAP analyses for various configurations of piping and sodium-water reactions will be used to optimize the piping sizes and system from the rupture disc assemblies to the separator tank.

Inconel 600 was selected as the rupture disc material based on an extensive
review of material properties and past experience. The rationale for its
35 selection is that at the design temperature, I-600 has: 1) high creep
41 strength; 2) mechanical properties are not significantly changed with long term
aging; 3) demonstrated good resistance to sodium corrosion; 4) minimal change
in strength with increasing temperature in the temperature range of interest
35 and 5) expected low carburization rate in sodium (See Ref. 3 & 4). In
41 addition, I-600 was used as the disc material for the rupture disc on the
Modular Steam Generator (MSG) during the testing of that unit in the Sodium
Components Test Installation (SCTI) at the Liquid Metal Engineering Center
(LMEC). The discs performed satisfactorily.

The basis for selecting carbon steel as the SWRPRS piping and equipment
material was to minimize the potential for caustic stress corrosion failure
from exposure to sodium hydroxide following a postulated large sodium-water
59 41 reaction in a steam generator module. Austenitic stainless steels are known to
be very susceptible to caustic stress corrosion failures. 19

54 The reaction products separator tanks (two per loop) are used to separate most of the liquid sodium and solid sodium-water reaction products from the gases, hydrogen and steam, which are ejecting the sodium from the failed unit. The relief piping enters the separator tank tangentially, near the top of the tank, causing rotation of the materials in the tank. This rotation creates a centrifugal force which assists in the separation of the liquid and solid
54 materials from the gases. The tanks are sized to hold the total volume of sodium in the three steam generator modules, the volume of sodium in the
59 intermediate sodium pump and expansion tank, and the volume of sodium which would be expected to spill from the remaining secondary sodium system into the superheater unit because of the momentum of the fluid in the loop. The superheater sodium inlet and intermediate sodium pump inlet are high points in the intermediate sodium system, such that sodium will not drain by gravity from the remainder of the system into the reaction products separator tanks. The gases separated from the other materials in the reaction products separator
54 tanks are discharged through central nozzles at the top of each tank.

Gases, which will be primarily nitrogen used to inert the SWRPRS and hydrogen resulting from the sodium-water reaction, but may include some steam, are piped to a stack where the hydrogen, when of a combustible concentration, will be
35 ignited and burned. Burning the hydrogen prevents the postulated creation of a hydrogen "bubble" in the atmosphere which could mix with oxygen and create a potentially explosive region in the atmosphere. A non-return valve is located in the line between the tanks and the stack to reduce backflow of air into the
54 system.

The entire SWRPRS is normally filled with an inert gas to avoid the possibility of a hydrogen explosion in the system after activation of the system. The inert gas is maintained above atmospheric pressure so that oxygen will not leak into the system and to verify that the inert gas atmosphere is being maintained in the SWRPRS. To maintain an inert gas in the system, a low pressure rupture disc is required in the line to the stack. This rupture disc is quickly broken or ejected after a small pressure increase in the SWRPRS.

A study of the effect of the vent piping diameter upon pressures in the IHTS was made early in the evaluation of SWRPRS. Analyses made with a 24" diameter pipe from the rupture disc assembled to the Reaction Products Separator Tanks showed that the pressures within the IHTS following the postulated design basis sodium-water reaction were within the specified limits.

The Reaction Products Separator Tanks are sized to accommodate the maximum amount of sodium and liquid or solid sodium reaction products which can be ejected or drained from the IHTS into SWRPRS during and following a sodium-water reaction. Regions of the IHTS and components in that system which might drain into SWRPRS during and following a sodium-water reaction were determined by study of the hydraulic profile of the IHTS. It was assumed that all of the rupture discs in the main rupture disc assembled would be broken by the sodium-water reaction. An additional capacity of about 25% above that determined by evaluation of the hydraulic profile of the IHTS was then added to establish the capacity of the Reaction Products Separator Tanks.

SWRPRS piping and equipment internal to the SGB are seismic Category 1. The SWRPRS piping and equipment external to the steam generator building are designed as seismic Category 3. If a major seismic event should damage this portion of the SWRPRS, such that a vent path to the atmosphere is not available, overpressure protection is maintained by the internal SWRPRS volume. The maximum system pressure following a steam generator DBL would be maintained at less than 100 psig by the automatic isolation and venting actions initiated with SWRPRS actuation. The minimum design pressure for SWRPRS is 125 psig.

Tests and Inspections

During plant operation, inert gas pressure will be maintained above atmospheric in the system and observation of the pressure in the system will verify the leak-tightness.

5.5.2.7 Sodium Dump Subsystem

The Steam Generator System (SGS) provides one sodium dump subsystem for each of the three parallel independent Intermediate Heat Transport System (IHTS) circuits: each sodium dump subsystem consists of a sodium dump tank located at the lowest building level beneath the evaporator and superheater modules.

A sodium dump tank having a usable capacity of approximately 9480 cubic feet at 930°F is provided within each sodium dump subsystem which is large enough to store all of the sodium from the IHTS and auxiliary systems to serve as a:

a. Sodium dump for sodium at operating temperatures

b. Intermediate sodium storage

c. Sodium fill

d. Removal or clean-up of Na-H_2O reaction products.

The Sodium Dump Subsystem will be used for normal drainage or for drainage of sodium from each IHTS circuit after the sodium has been contaminated by sodium-water reaction products from a large sodium-water reaction occurring within an evaporator or superheater module. Upon manual initiation, rapid drainage of IHTS sodium and Na-$H_2$0 reaction products will be accomplished by lines which are connected at five different locations within the IHTS sodium circuit. These piping runs provide gravity drains from the sodium expansion tank, the superheater, the evaporators, and the low points in the hot and cold legs of the IHTS main piping to the dump tank, as shown in Figure 5.1-2a. Each piping run contains a pair of isolation valves in series, with independent controls. To prevent inadvertent draining of an IHTS loop, interlock features are incorporated into the operation of the valves.

The IHTS sodium will be cooled to a bulk average temperature of less than 800°F prior to opening the sodium dump valves following duty cycle events which increase the IHTS average bulk sodium temperature above that associated with full load steady state operating conditions. The sodium dump tank will accommodate the average bulk sodium temperature associated with the IHTS at full load steady state operating conditions.

In the event that a sodium-water reaction occurs, sodium contaminated with sodium-water reaction products would be dumped into the dump tank and maintained in a molten state by trace heaters on the tank. Later, the contaminated sodium in the dump tank will be cleaned by circulation through the

Intermediate Sodium Processing System (see Section 9.3) and then transferred to the IHTS sodium loop. The dump tank could then be cleaned and the Sodium Dump Subsystem be made ready for use again.

In the event of a large sodium-water reaction event within a Steam Generator Module, sodium and sodium-water reaction products will be discharged through connecting piping to the Reaction Products
41 Separator Tank, where most of the sodium and liquid or solid reaction
35 products will be accumulated. The detailed methods and processes to be used to remove the reaction products have not been chosen, however, studies are underway to evaluate potential alternatives. It has been recognized that in addition to removal of the affected Steam Generator Module, replacement of the Reaction Product Separator Tanks may be necessary following a large sodium-water reaction event. Since most of the liquid and solid reaction products will end up in the tanks, replacement of the tanks will remove most of the reaction products from the system.

The Reaction Products Separator Tanks provide sufficient capacity so that the solid reaction products from a sodium-water reaction may be left in the tank following the incident if the tanks have not sustained mechanical damage from the incident which would render them unsuitable for further service. Sodium would be melted and removed from the tank, leaving the solidified reaction products in the tank.

Contamination of the remainder of the IHTS with sodium-water reaction is expected to be limited to the piping between steam generator modules and the piping between the evaporator modules and the sodium pump. The sodium-water reaction products are not expected to reach the pump. (See Section 5.5.3.6.2).

As a result of a water to sodium leak, caustic stress cracking can occur under certain reasonably well defined conditions of stress, temperature, and concentration of aqueous sodium hydroxide. It is important to note that the sodium hydroxide must be aqueous to cause cracking and that the lifetime of aqueous sodium hydroxide in sodium is very short. The backflow of sodium from the pump is expected to transport the reaction products in the IHTS piping upstream of the pump to the SWRPRS relief lines and RPST's. Therefore, the residence time of the reaction products in the IHTS piping and the potential for caustic stress corrosion are minimized. Any reaction products contained in the IHTS sodium remaining in the loop will be drained into the sodium dump tank following water side depressurization of the Steam Generator Modules to 300 psig. Solid reaction products, if any, will be removed from the IHTS sodium contained in the dump tank by circulation through the Auxiliary Liquid Metal System. 19

Each dump tank will store the IHTS sodium during a schedule and/or controlled drainage of the IHTS loop. Sodium shall be maintained molten and cleaned, if necessary. When the Intermediate Heat Transport System loop is ready for operation again, the IHTS sodium loop will be refilled from the Sodium Dump Subsystem and the Subsystem will be maintained in a condition for future reuse.

The sodium dump tank and its associated components have a design life of thirty years. To assure this life a corrosion allowance of .065" has been provided. Since most of the time the tank is exposed to hot sodium, an allowance was made for thirty year exposure based on data from Reference 5. Sodium (regardless of the amount) mixed with oxygen at a concentration of 100 ppm by weight (conservative amount of oxygen) will corrode only .010 inch in the 30 year period at 700° F. This is based on equation

$$\frac{\Delta W}{t} = .0264 O_x^{.89} \exp\left(\frac{-9700}{RT}\right) \quad \text{(Reference 5)}$$

where $\frac{\Delta W}{t}$ = corrosion rate in CM per year

O_x = oxygen concentration in PPM

T = Temperature in °K

This equation is for flowing sodium at high velocity and the equation is conservative relative to the stagnant dump tank.

Since the primary recipient of the sodium-water reaction products is the Reaction Products Separator Tank, a conservative concen-
41 tration of 10% NaOH in sodium was assumed to enter the sodium dump system. This amount of reaction products results in a corrosion prediction of .009" based on being exposed to reaction products for 6 intervals of 18 days per interval.

Allowance for decarburization was calculated at .027 inch in 30 years.

Summing the above allowances, the total expected degradation amounts to .046 inch in 30 years of exposure versus the allow-
ance of .065 inch originally provided. 2

The Sodium Dump Subsystem components are designed in accordance with ASME Code Section III Nuclear Power Plant Components.

Sodium dump tank is designed to satisfy the requirements of Seismic Category I requirements. (See Section 3.7.)

Tests and Inspections

The sodium dump tank shall be designed such that all normal inspection, maintenance, and repair can be performed during normal shutdown periods, or after the removal of sodium and/or sodium-water reaction products following an IHTS sodium dump. Access ports will be provided for inspection purposes. The inspection of dump tank, tank internals, tank supports, burst discs and other instruments will be conducted in accordance with the manufacturing instructions. Sodium level indication, sodium and other system component temperatures will be available or displayed in the control room.

5.5.2.8 Water Dump Subsystem

One Water Dump Subsystem is provided for each of the three parallel and independent Steam Generation Systems (SGS). Each Water Dump Subsystem consists of water dump piping and a water dump tank. The water dump tank for each loop is located in the Steam Generator Building near the SGS recirculation pumps.

The dump tank is provided in each SGS loop to accept and store the water from the evaporator when rapid depressurization of the evaporator modules is required.

Blowdown of an evaporator module is accelerated by the quick dump valves, which are part of the Steam Generator and Water Steam Subsystem. The quick dump valves will be located between the recirculation pump discharge and the inlet of each evaporator module. Additional blowdown capacity is provided by the power relief/safety valves located at the outlet of each evaporator module. The water dump valves exhaust to the water dump piping, which directs the water/steam mixture to the water dump tank, where the water is temporarily stored and the flashed steam is vented to the atmosphere.

59 Later, when the HTS loop is ready for operation, the water from the dump tank will be drained, either for reuse or for disposal, thus making the Water Dump Subsystem available for future use.

The Water Dump Subsystem components are designed in accordance with ASME Code Section III Nuclear Power Plant components.

The dump tank and its associated piping are designed to satisfy the
59 41 requirements of Seismic Category 2. (See Section 3.7.)

41 The water dump tank is relieved to the atmosphere through an open pipe to limit the maximum internal pressure and to assure the integrity of its components.

Tests and Inspection

The water dump tank shall be designed such that all normal inspection, maintenance, and repair can be performed during normal shutdown periods or after the removal of water following an evaporator blowdown.

59 41 The inspection of the dump tank, tank supports, and other components will be conducted in accordance with manufacturer's instructions. Water level indication, water and other system component temperatures and pressures will be available or displayed in the control room.

5.5.3 Design Evaluation

5.5.3.1 Analytical Methods and Data

The design of the SGS relative to sodium flow and water or steam flow in the interconnecting piping is based upon technology attained during the development, design, construction, and operation of sodium and water or steam systems of similar type. The design of the steam generator components is based
41 on available technology, testing of similar components, model testing, and development testing. The design of the Sodium-Water Reaction Pressure Relief Subsystem (SWRPRS) is based on experimental data and analysis.

Sodium properties are based on the report "Standard FFTF Values for the Physical and Thermo-physical Properties of Sodium". (Ref. 1)

Water or steam properties are based on the "1967 ASME Steam Tables".

The pressure losses are calculated using established equations and values available in textbooks and piping catalogs.

41 49 5.5.3.1.1 Structural Evaluation Plan (SEP)

The procedure for developing SEP's for SGS components is the same as that
41 described for the PHTS in Section 5.3.3.1.1.

5.5.3.1.2 Stress Analysis Verification

The analytical requirements given in the ASME Section III Code for Class 1, 2, and 3 components under normal, upset, emergency, and faulted operating
59 conditions were used as design limits for the SGS. The appropriate figures and sections in Section III, that contain the analytical requirements corresponding to the appropriate design class and operating conditions, are given below. Accidents classified as faulted will satisfy the code requirements referenced below for faulted operating conditions.

For components in which temperatures exceed those provided for in Section III, the rules of Code Case 1592 provide analytical requirements for Class 1
41 components.

5.5.3.1.3 Compliance with Code Requirements

Regulatory Guide 1.48 delineates design limits and appropriate combinations of loading associated with normal operation, postulated accidents, and specified seismic events for the design of Seismic Category I fluid system components. The SGS will be treated as a Seismic Category I component and will comply with the intent of Regulatory Guide 1.48 as described in Section 3.9.

5.5.3.1.4 Category I Seismic Design

The limits of required Category I design for the SGS are shown in Figure 5.1-4. The components and supporting structure for the major components will also comply with the Category I requirements. (See Table 3.3-2). Diagrams of preliminary seismic models of SGS components are provided in Figures 5.5-6 17
through 5.5-9.

5.5.3.1.5 Analytical Methods for Pumps, Valves, and Heat Exchangers

59 Standard text book techniques along with finite element computer methods were used to evaluate the stresses in the SGS. For these components, elastic, inelastic, and dynamic analysis methods were utilized. A brief description of
59 the types of analysis, along with a list of computer codes used, are given below. Analysis of a particular component was made by appropriate selection of
59 the computer codes listed below.

59 5.5.3.1.5.1 Steam Generator

The steam generator is a Class 2 vessel but is designed as a Class 1 vessel in accordance with Section III of the ASME B&PV, RDT E-15-2NB and Code Case 1592 as supplemented by RDT F9-4 (see Section 5.5.1.2). Non-pressure boundary
41 components are designed to the intent of Code Case 1592.

The failure modes which are expected to dominate the steam generator design are identified below. The loadings which could contribute to such failures are also identified.

a. Rupture of steam generator tubing. The design recognizes that the tubing is subject to an internal, high water/steam pressure loading, augmented by steady and cyclical temperature gradients through the tubing wall and by point loadings at support plate contact points.

b. Water/steam leak through tubing or tube-to-tubesheet weld. This failure mode is identified more with material degradation situations than with loading conditions. It can result from water/steam leakage paths caused by weld inclusions or porosity, waterside metal corrosion on cyclic fatigue, sodiumside metal wear at support points, sodiumside metal corrosion by adjacent tube leak.

c. Sodium-to-air boundary rupture. To preclude this eventuality, the design recognizes large sodium/water reaction pressure pulse loadings resulting from rupture of water/steam tubing, low pressure loadings from the sodium system and transient thermal gradients during duty cycle events.

Tests which have been performed are planned to support the steam generator design are:

a. Modular Steam Generator (MSG) Tests, (1972-1974)

Objectives

The objectives of the MSG tests were to confirm both the thermal-hydraulic and structural performance of the hockey stick steam generator design concept. The program tested the MSG unit to it's designed power rating of 32 Mwt maximum and included steady state tests, transient tests and an extended endurance test. The MSG unit had 158 tubes and was operated in a once through mode.

Results

The modular Steam Generator test unit received a total of 9300 hours of sodium exposure with 4000 hours of steaming. Tests were carried out to cover a wide range of operating parameters which enveloped most CRBRP conditions, even though the unit was not operated in the recirculation mode. The test confirmed the design concept, including such design features as the tube/tubesheet welds, use of shroud and inlet thermal liners, and floating spacer plates. The test also confirmed the basic material choice (2 1/4 CR - 2 Mo), the performance capability and analytical prediction techniques. No sodium to water boundary leak indications, cracks or corrosion were discovered during post-test examinations, or subsequent use of the unit for sodium water reaction tests. Overall, the unit performed in a very "well-behaved" manner.

b. Hydraulic Test Model (HTM), (1969-1976)

Objectives

The objective of the HTM test program was to determine the response of sodium side design features to prototypic sodium flows. The test used water as the working fluid in a full scale, but shortened, model of the CRBRP Prototype hockey stick steam generator design.

The HTM unit contained the full internal details of the prototype steam generator including a complete complement of tubes, and the test facility duplicated CRBR inlet and outlet piping configurations.

<u>Results</u>

The tests confirmed the hydraulic acceptability of the prototype steam generator sodium (shell) side. Information on pressure drops, velocities and flow fields was obtained. It also confirmed that the prototype configuration did not have any flow-induced vibration problems. In fact, the test showed that the unit had very minimal tube vibrations. These results confirmed the choices of spacer plate locations, spacer plate flow and tube hold sizing, thermal liner/shroud flow passage size, inlet window sizing, outlet region window and passage sizing and internal clearances.

c. Sodium to Water Boundry Leak Tests (1974 - Present)

<u>Objectives</u>

The objective of these tests was to characterize sodium to water boundary leaks through the use of both small and large scale leak tests. The test program includes investigation of leak initiation, leak propagation tube to tube, leak enlargement within a tube, leak detection both chemical and acoustic, leak mitigation by automatic action, large sodium/water reaction dynamics, large leak damage predictions, and post leak investigations. In addition, the program is intended to support the establishment of adequate design and operational methods to accommodate large sodium-water reactions, and to help develop inservice inspection equipment and techniques.

The test article for these tests is the Modular Steam Generator (MSG) unit discussed in Section 5.5.3.1.5.1.a which was modified to incorporate leak injection tubes.

<u>Results</u>

The results and data obtained from the Large Leak Test Rig (LLTR) Series 1 tests demonstrate that the analytical methods used to predict the pressures and velocities resulting from large SWR.

Events are conservative, and thus confirmed that the design includes loading estimates well in excess of those actually produced. The effects of sodium-water reactions were characterized including leak reaction mechanisms and dynamics, large leak dynamic effects and resulting system pressure pulses, leak propagation mechanisms, and pressure relief system response. Leak detection capabilities were determined test inspection techniques were refined. Inservice inspection equipment was developed which could detect tube degradation from leak growth, leak propagation, tube wear, or tube corrosion. Very good resolution and accuracy of wall thickness and flaw measurements has been demonstrated. For more detail concerning test procedures and results see sections 1.5.1.4.2 and 5.5.3.6.

d. Few Tube Tests (FTT), (1978)

Objectives

The objectives of the Few Tube Test Program were to (1) conduct endurance tests of the Few Tube Test Models (FTTM) to evaluate tube/tube support° wear and the reliability of tube-tubesheet welds under long-term operating conditions, and (2) obtain performance data for operating conditions ranging from natural circulation to full power. Including both steady state and transient operations.

Component Characteristics

The physical configuration of the FFTM's were similar to the reference hockey stick design of the CRBRP steam generator evaporator/superheater configuration. The evaporator employed 7 active tubes and the superheater employed 3 active tubes (4 tubes were plugged during fabrication). The tube-to-tube support interface details were designed to be representative of the CRBRP units. The length and radii of the FTTM tubes were selected to be the same as the CRBRP steam generator shortest row of tubes on the basis that this condition represented the worst combination of tube-to-tube support movement and side forces.

Results

During the tests, the models were exposed to transients of a severity which damaged the internals to a degree that testing had to be discontinued. Post test examinations revealed the following:

1. Thermal performance prior to the transients was as expected but test data was insufficient for thermal analysis studies.

2. It was found that the design of the tube supports and tolerances imposed on the parts did not permit the tube bundle to thermally expand/contract readily and led to binding or jamming of the tubes, resulting in tube buckling and mechanical failure of the shroud support.

3. The Sodium/Water pressure boundary remained intact even under the extreme transient condition and the severe mechanical loads caused by the transients.

The test resulted in a redesign of the plant unit internals, particularly in the geometry and location of the tube supports and prompted a change in the material selection to reduce friction. The test also increased confidence in the tube design and the tube to tubesheet weld design, since both successfully withstood conditions which were considerably more severe than required by plant operating conditions.

e. Department from Nucleate Boiling (DNB) tests, (1975-1976)

Objective

The overall objective of the test program was to verify that excessive damage to the CRBRP evaporator tubes will not be produced by operation with departure from nucleate boiling (DNB) or liquid film dryout in conjunction with maximum specified CRBRP water chemistry conditions.

Test conditions were selected which represented the worst case conditions to which the CRBRP evaporator tubes could be subjected: (a) Maximum (T) or tube wall temperature oscillations and (b) Maximum sodium hydroxide conditions in the evaporator water.

The test program was comprised of the following phases:

- Initial thermal conditioning run of 400 hours for establishing proper film condition inside the tube. During this period, thermal/hydraulic test data were taken to establish the endurance test condition.
- Endurance testing of about 3000 hrs with in-situ nondestructive examinations after about 1000 hrs and after test completion.
- Post-Test Destructive Examination.

Results

A total of 2820 hours were accumulated at endurance test conditions. In order to achieve these test hours, the test section was exposed to 4181 hours of steaming with CRBRP water chemistry and with thermal/hydraulic parameters varying $\pm$ 10% from the nominal.

The principal findings from the destructive post-test examination were:

1) No localized damage or accelerated corrosion attributable to DNB operation was found in the DNB region nor in any other region of the steam tube.

2) The corrosion of the entire tube (including nucleate boiling, DNB, and film boiling) was found to be essentially uniform. The observed maximum loss of wall thickness was found to be 0.40 mils (10.2 pm) for the period of testing which translates to about 0.8 mil/year (20 pm). By conservative extrapolation of the test data, a long-term life in the order of 30 years would be expected for the CRBRP evaporator tubes.

3) Deposition/fouling on the tube wall was minimal and was characterized by nickel alloying of the magnetite scale. Only a small fraction of the corrosion products present in the recirculation water deposited in the steam tube.

Complete test program results are presented in Reference 22.

f. Friction and Wear Tests (1973-1979)

Objectives

As a subtask of the National Friction and Wear Test Program, tests were conducted using Steam Generator tubing and support plates to verify that selected material couples will meet the wear allowance at end-of-life under simulated operating conditions. Tests were used to select the proper material couple to assure tube wear due to differential thermal expansion meets the design limit of .004" and that the wear does not result in galling between the tube and spacer plate.

Several different tests were performed including pin/plate wear samples and simulated tube/space plate geometrics.

Results

Tests performed by two independent organizations produced the following conclusions.

1) The material couple between Inconel 718 and 2 1/4 Cr-1 Mo has the best wear characteristics of the couples tested.

2) When subjected to simulated CRBR conditions, the selected material couple produced wear less than the .004" allowance.

3) For the selected couple, the wear is abrasive and not adhesive.

4) The couple of Inconel 718 and 2 1/4 Cr-1Mo meets the .004" wear allowance with margin for tube sides loads of less than 25 lbs. The couple may be satisfactory for side loads of up to 100 lbs. as indicated by limited test results from one organization.

g. Single-tube Performance, Stability and Interaction Tests (1976-1977)

Objectives

The objectives of these tests were to establish single tube heat transfer correlations, hydraulic stability data and the effect of tube to support interactions on the structural vibration of a tube.

The program utilized single tube tests in sodium and air. The tests in sodium included prototypic temperatures, heat fluxes and flows for a single tube. The tests in air parametrically studied structural vibration of tubes.

Results

The tests provided heat transfer correlations for use in all three heat transfer regimes of an evaporator. These included applicable sodium-side heat transfer information.

Stability results were encouraging but not conclusive. As a result, inlet orifices were added to the evaporator tubes which raise tube differential pressures and thus ensure stability.

The prototype steam generator test program will provide conclusive data which it is anticipated will show that these orifices are not required.

The effects of tube orientation, tube/support misalignment, fluid medium, tube/support-hold clearance, support thickness, exciting force amplitude, and support spacing on the vibrational characteristics and displacement response amplitude of a multi-span tube were determined. The test results were compared with the analytical results based on the multi-span beam with "knife-edge" supports. The experimental results showed a small variation of resonant frequencies due to tube orientation, tube/support-hold clearance, support thickness, tube-support spacing and excitation force amplitude. Measured frequencies were close to the calculated natural frequencies. The tube/support-hole clearance was found to be the most sensitive parameter for response amplitude. A significant variation in displacement amplitude was observed for the tube/support hole clearances greater than 20 mils. Short spans placed at both sides of the excitation span reduced the overall displacement amplitude significantly, despite the fact that the lowest resonant frequency is not maximized.

h. Tube to Tubesheet Welds Tests (1976 - 1980)

Objectives

Mechanical testing of tube to tubesheet weld specimens prototypic of CRBRP steam generators was performed to determine the specific effects of microstructure, composition, environment and stress/strain on the failure susceptibility of the 2 1/4 Cr - 1 Mo steel welds. The study investigated the most probable mechanisms likely to cause failure of CRBRP tube to tubesheet welds. Predominantly this work has evaluated "standard" good quality welds, a few lower quality welds have also been tested, since these were considered most likely to have the highest failure susceptibility. The tests included weld and/or HAZ-notched tensile, impact and bend specimens of the tube to tubesheet weld region.

Results

Weld failure susceptibility was not observed during testing at ambient temperature under conditions of high uniform stress and strain, high local stress and strain (all above yield point) and high strain rates. Both microscopic and macroscopic ductility of the weld area was retained under these testing conditions, and specimen rupture occurred only in the base materials. Using the design as employed in the CRBRP steam generator, the following specific results were found with respect to:

1) Caustic stress corrosion (CSC) - Tests at 232°C with 10% and 20% NaOH in pure water indicate that post weld heat treatment (PWHT) at 727°C (1340°F) for twenty (20) minutes or longer, provided a high degree of resistance to CSC of welds.

2) Biaxial stress-rupture - Tests at 510 and 566°C (950 and 1050°F) of tube to tubesheet weld specimens with and without a one hour, 727°C PWHT were performed. Up to 10,600 hours at 566°C and 14 Ksi failures of material with or without PWHT occurred in the annealed tubing in a ductile manner. For durations of greater than 10,600 hours, the failure times were not reduced relative to rupture times for unwelded base material.

3) Four point bend flexural fatigue - Tests at 510 and 566°C of tube to tubesheet weld specimens both with and without PWHT, and with PWHT plus 1,000 hour, 510°C aging, resulted in base metal failure, with secondary HAZ cracking. Tests of welds with induced outside diameter concavity of varying depths showed that the PWHT'd welds crack preferentially in the weld if the concavity is 0.25 mm (0.010 in) deep or greater. Cantilever bend tests (applicable to the upper tubesheet) resulted in tubesheet spigot failure. Weld failures were associated with shorter fatigue lifetimes than for annealed material.

Tests were used to determine the proper post weld heat treatment procedure and weld geometry with respect to concavity and convexity. Procedures have been adopted that limit concavity to .010 inch to preclude failure in fatigue in the base material.

I. Mechanical Properties Tests (1968 - 1981)

Objectives

The overall objective is to verify and supplement ASME Code and RDT standards methods and design information for assuring the structural adequacy of the steam generator.

Results

Properties required to characterize the material in the CRBRP high temperature environment have been obtained. The CRBRP use of 2 1/4 Cr - 1 Mo has involved inelastic analysis and thus a full creep equation has been developed. Recent data of high quality on five heats were used to develop the creep equation. In some time/temperature/stress domains non-classical behaviors were observed and captured.

Fabrication and environmental effects upon the properties of the material have been established. Tests of Post Weld Heat Treated (PWHT'ed) material disclosed that the tensile strength of 2 1/4 Cr - 1 Mo can be reduced by extremely long time post weld heat treatments (40 hours at 1340°F). The loss of carbon (by N_a transport) can reduce the stress-rupture life of 2 1/4 Cr-1Mo.

Extensive further testing of 2 1/4 Cr - 1 Mo has accompanied the CRBR steam generator program. Since ASME Code Case N-47 does not contain fatigue data for 2 1/4 Cr - 1 Mo there have been a number of further fatigue and creep-fatigue tests. Data from in-sodium tests at the Argonne National Laboratory reveal that the lack of air (and thus lack of severe exfoliation) removes the compressive hold time damage. These data are now being evaluated in order to provide an appropriate creep-fatigue limit for 2 1/4 Cr - 1 Mo in each environment of the CRBR Steam Generator.

The design limits of the CRBR Steam Generator were modified to reflect the effects of the service specific environment as outlined below:

1) The design limits were reduced to account for the potential effect of PWHT on mechanical properties.

2) In those areas where significant carbon loss could occur the stress-rupture based limits were adjusted accordingly.

3) A design fatigue curve is in use that reflects these data. In most cases fatigue itself is not a concern; the stress-rupture damage dominates.

j. Scale Hydraulic Model Feature Tests (1980 - Present)

Objectives

Tests of 1/6 scale models of various regions of the steam generator will be carried out to support the final design effort. These tests were designed to confirm and refine the sodium-side internals structures design. Features and phenomena modeled are evaporator sodium outlet thermal striping, superheater sodium inlet thermal striping, sodium inlet region flow distribution, sodium side mixing in elbow region, elbow region thermal striping and sodium inlet nozzle inlet liner seal effectiveness.

Results

Thermal striping has been determined to be acceptable at the outlet and is being investigated for the inlet and elbow regions. Flow distributions in the inlet plenum have been obtained. Elbow region flows have been characterized and the elbow shroud design requirements will be determined. The sodium inlet nozzle seal behavior has been confirmed.

Boundary conditions have been established for thermal/hydraulic analysis models.

Design refinements will be tested as necessary.

Thermal/hydraulic analysis boundary conditions and analytical models will be correlated to test data. A disk type sodium inlet nozzle seal has been included in the design to restrict flows and transients on the shell at the inlet nozzle. Detailed modifications will be made to tube spacer flow hole patterns to improve inlet/elbow flow mixing. Thermal striping data to be used in structural analysis will be correlated to a test data base. Performance of current design features in outlet, inlet and elbow regions will be verified.

k. Flow Induced Vibration (FIV) Tests (To begin in 1983)

Objectives

The objective of the test program being planned is to confirm the absense of any degrading flow induced vibration effects in the CRBRP plant unit design. The test program consists of three (3) phases.

Phase I uses low temperature water ($\leq 180^{o}F$) as the test fluid in a 0.42 scale model with fluid on the shell side only. Test conditions will conservatively envelope all critical velocities and forces expected in the plant with sufficient over testing of simulated flow conditions (approximately 25%) to show that no nearby thresholds exist where problems may develop.

The tests and assessment of data are scheduled to be available prior to the start of significant plant unit fabrication. Although this is a confirmation test, it's schedule will allow results to be incorporated into the plant unit design if required.

Phase II uses low temperature water as the test fluid in an externally instrumented spare plant unit steam generator with fluid on the shell side only. These tests will also run to 125% of simulated full power flow and will conservatively envelope plant unit operating conditions. Phase II is scheduled for 1985.

Phase III uses instrumentation installed on one of the CRBRP Superheaters during startup and pre-operational testing to provide final confirmation of the absense of any degrading flow induced vibration effects in the CRBR steam generators under actual operating conditions.

I. Prototype Steam Generator Tests (To begin in 1982)

<u>Objectives</u>

The objective of the prototype steam generator test is to verify certain performance characteristics. By testing the full-scale SG at high temperatures under steady-state conditions, the effects of numerous design and operating parameters can be determined. This test program supports the overall verification of the CRBRP steam generator units for plant use. The test program includes operation of a single plant prototype unit under steady-state conditions over a range of power from 1 to 70 MW_t to obtain the data necessary for determining steam generator performance characteristics. Operating experience will be obtained to provide input for establishing plant start up and operating procedures.

Steady-state thermal hydraulic performance data will be obtained at power levels representative of steam generator operation under part power and plant emergency decay heat removal conditions. Data will be obtained to evaluate steam side two phase flow stability and sodium side temperature stratification characteristics. Steady-state thermal-hydraulic performance data will be obtained over a range of recirculation ratios. Tests will also be performed to establish Na/H_2O chemical leak detection system response, to measure steam generator acoustic characteristics and demonstrate acoustic leak detection system performance, to measure fouling, to measure tube vibration and to measure system natural circulation capability. Fouling tests will be performed in parallel with the performance tests.

The Prototype test data will also be used to confirm the applicability and accuracy of the analytical models used to prodict the units performance. Because of the nearly identical heat transfer designs of the Prototype test module and the ten plant units, the test data and verified analytical models can be directly applied to predicting plant unit thermal-hydraulic performance over the entire CRBRP operating range.

m. In-Situ Evaporator Performance Tests

Objectives

These tests are intended to provide a final check of the steam generator performance through data acquired from instruments built into one of the steam generator modules installed as an evaporator in CRBR. The required performance data will be obtained during plant pre-operationnal and start-up testing.

5.5.3.1.5.2 Valves

The steam generator system control valves shall be designed to the alternative rules defined in ND 3512 of the ASME Code, Section III. In addition, thermal transient stress analysis, transient pressure analyses, and seismic response analyses were performed for appropriate valve components and, as applicable, for the valve operators. The analyses, which demonstrated that the valve assembly will function as designed and in accordance with the criteria specified in the ASME Code and the valve equipment specification, were provided by the valve manufacturer, after review and approval of their analytical methods.

Inelastic methods were used by the manufacturers to verify that stresses were within code allowables.

Included in the failure modes to be considered were:

9 1. Rupture of valve assembly components from short term loadings such as pressure (static or dynamic), and seismic loading. 26

Group I tests will be on Electroslag Remelted (ESR) base metal tubing to determine rupture life at 510°C (950°F). Approximately twenty more base metal tests will follow for Group 2 to characterize actual CRBRP ESR tubing. A complete range of stresses will be studied for both decarburizing and control tests. Approximately 16 elevated temperature tensile tests will be run to determine the influence of pre-exposure time to decarburizing sodium on tensile properties at 510°C (950°F). Analytical work to assess decarburization profiles will be performed. Decarburized layers will be machined from specimen surfaces and analyzed for carbon by combustion and wet chemistry methods. Results will be compared with probe profiles. Specimens will be exposed at 510°C (950°F) for 2000 hours. A facility to determine ratchetting in the presence of a surface carbon gradient (decarburization) will be prepared, and tests will be performed.

The objective of these experiments is to develop an analytical model to predict the mechanical behavior as a function of the time and temperature of sodium exposure.

4. <u>Effects of IHTS Sodium Environment on Mechanical Behavior of Transition Metal Joints</u>

Uniaxial creep tests and fatigue crack growth tests identical to those described in item 3 will be performed on 2 1/4 Cr - 1 Mo/182/A800 and A800/16-8-2/316SS weld joints. Experience has indicated that the 2 1/4 Cr - 1 Mo/182 transition is the more critical of the four transitions and thus, will receive the highest priority in the testing schedule. Testing will be performed at 510°C (950°F) to simulate IHTS conditions.

<u>Schedule</u>

Preliminary data from this program will be available for the CRBRP Steam Generator Final Design Review and Final Fabrication Release. The program,
36 as planned, will be essentially completed by 1980.

43 41 Computer codes; MARC, TAP-4F, and DRIPS are used in the design of the steam generator and are described in Appendix A of the PSAR.

5.5.3.1.5.2 <u>Valves</u>

The steam generator system control valves shall be designed to the alternative rules defined in ND 3512 of the ASME Code, Section III. In addition, thermal transient stress analysis, transient pressure analyses, and seismic response analyses were performed for appropriate valve components and, as applicable, for the valve operators. The analyses, which demonstrated that the valve assembly will function as designed and in accordance with the criteria specified in the ASME Code and the valve equipment specification, were provided by the valve manufacturer, after review and approval of their analytical 26
59 methods.

Inelastic methods were used by the manufacturers to verify that stresses were within code allowables.

Included in the failure modes to be considered were:

59 1. Rupture of valve assembly components from short term loadings such as pressure (static or dynamic), and seismic loading. 26

2. Loss of valve function due to excessive deformation of the valve assembly components, including air supply connections, because of seismic loading and/or thermal distortion.

3. Structural integrity damage from cyclic loadings.

4. Stem deformation due to excessive loading by the valve operator.

59 No structural tests of the valves were required to support the design analysis.

59 Computer programs used to verify vendor analysis, are those identified below.

59 The steam generator system pressure relief valves were designed to two different criteria, depending on the valve location in the steam generator system. For the valves located in the recirculation loop and on the steam drum, the design requirements of the ASME Code, Section III, ND 3511 and ND
59 7000 were used. For the relief valves subjected to elevated temperatures on the superheated steam line, the design requirements of the ASME Code, Section III, ND 3511 and ND 7000 were used. In addition, thermal transient stress
59 analysis, transient pressure analyses, and seismic response analysis were provided for the appropriate valve components and the valve operator, as applicable. As with the control valves, all of the analysis was provided by the valve manufacturer after approval of the analytical methods, which included
41 59 those methods outlined below. Also, where it was necessary to use inelastic analyses, they conformed to the guidelines of RDT Standard F9-4T and RDT Standard F9-5T, as applicable to Class 3 components at elevated temperatures.

59 41 Included in the failure modes considered for all relief valves were:

59 1. Rupture of valve assembly components from short term loadings such as seismic loadings; system pressure (static and dynamic), and loading due to discharging fluid.

2. Loss of the safety function due to excessive deformation of the valve assembly components under seismic loading.

3. Structural integrity, valve stem, and valve spring damage from cyclic loadings.

59 4. Loss of function due to cyclic loading on the pilot valve spring. 26

59

No structural tests were required to support the design analysis. 26

Isolation valves which are required to close under normal as well as pipe rupture fluid conditions (identified per BTP APCSB 3-1 as Essential Components) were analyzed for the resulting stresses. The valve velocity at closure was limited to a value which will result in negligible impact effects, based on past tests and/or analysis of similar valve configurations. To maintain a relatively constant valve closing velocity regardless of the closing forces due to normal and pipe rupture conditions, hydraulic flow control device is provided on the valve operator. The stresses in the valve components, at valve closure, determined using applicable static loads and the design is such that structural response will be within the elastic limits for the material thereby satisfying functional deformation requirements. For the stem, the actuator thrust/force was used. For the disk, valve seat, and valve body, the load due to the differential pressure and actuator thrust was used.

30

Elastic Analysis

Standard text book methods (theory of elasticity, strain energy theory, etc.), including the finite element methods used in the computer codes listed below, were used for elastic analysis.

Inelastic Analysis

Inelastic analyses were required in some cases to demonstrate conformance with ASME and RDT Standards. RDT F9-5T gives a description of

59 acceptable methods for time-independent elastic-plastic analysis. Some of the computer programs listed below have inelastic capabilities, and were used where applicable.

Dynamic Analysis

59 Dynamic analyses were required for evaluation of the seismic response of the SGS components. The computer programs listed below employ several acceptable finite element methods used exclusively for the dynamic analyses of the SGS.

Computer Codes

SAP IV

44 41 MARC

41 ELTEMP

Abstracts of these codes are provided in Appendix A of this PSAR.

59 5.5.3.1.5.3 Analytical Methods for Evaluation of the Recirculation Pump

The analytical methods and computer software discussed in paragraph 5.5.3.1.5 were used to evaluate the SGS pumps during transient conditions. Standard techniques were used to evaluate the stresses in the recirculation pumps. Elastic and dynamic analyses methods were utilized. Brief descriptions of the 59 types of analyses used are given below:

Elastic Analysis

59 Standard methods (theory of elasticity, strain energy theory, etc.) were used for elastic analysis.

Dynamic Analysis

59 Dynamic analysis was required for evaluation of the dynamics and the seismic response of the recirculation pumps. SAP-IV and ANSYS are computer programs which employ several acceptable finite element methods used for the dynamic analyses of the pumps.

See Appendix A for Code Abstracts.

59 5.5.3.1.6 Operation of Active Valves Under Transient Loading

Active SGS valves relied upon for operation under transient loading are the feedwater flow control valves, the superheater outlet isolation valve, the
59 feedwater inlet isolation valve, the safety/relief valves, and the drum drain valves. To ensure proper valve operation during their design life, an in-service testing program will be implemented for the SGS.

59 After a valve and/or its control system has either been replaced, repaired or undergone maintenance that could affect its performance, and prior to the time it is returned to service, it will be tested as necessary to demonstrate that the performance characteristics are within acceptable limits.

The remote position indicators will be visually calibrated at the same frequency as scheduled refueling outages, but not less than one observation every two years, to confirm that remote valve indications accurately, reflect valve operation.

The SGS active valves will be moved through their full-stroke during each
59 shutdown. During plant operation the valves will be exercised by part stroke operation at specified surveillance intervals.

Feedwater Flow Control Valves

The performance of the feedwater flow control valves has been demonstrated in other plants using similar valves under nearly identical feedwater system
59 operating conditions.

44 Superheater Outlet and Feedwater Inlet Isolation Valves

These isolation valves, which are required to operate during transient
59 conditions, and whose functional capabilities may be affected by the abnormal ambient pressure and temperature associated with the transient, will be tested as part of the vendor's qualification test program. Functional requirements will be verified throughout the test sequence. Components tested will be fully representative of production components.

Safety Relief Valves

The safety relief valves will be subjected to tests that simulate conditions experienced during service life.

Drum Drain Valves

Drum drain valves (which permit continuous blowdown flow during operation) will
59 be tested by full stroke operation at specified surveillance intervals.

Valve Requirements

To assure operability of active valves under the transient loadings to be experienced during plant service life, design specifications include the following requirements:

1. Valve bodies will be qualified to withstand seismic forces, and those
59 valves with extended proportions will have minimum frequency-of-vibration requirements.

2. Valve operators will be sized to open or close under the maximum differential pressure across the valve seat that is dictated by the transient service conditions.

3. Valves will be fully cycled at the vendor's shop before delivery to substantiate the vendor's guarantee that they will operate under actual service pressure conditions.

59

5.5.3.1.7 Analytical Method for Component Supports (Vessels, Piping, Pumps and Valves)

In accordance with the ASME Code, component supports have the same code classification as the components they support. Design of each component
41 support complies with the ASME Section III, Subsection NF design rules
59 corresponding to the component support classification. In order to provide assurance that the component support stresses comply with limits specified in
59 paragraph 5.5.3.1.2, analysis of each component support was performed. The analytical techniques and applicable computer codes discussed in paragraph
59 5.5.3.1.5 also apply to detailed analysis of support components.

5.5.3.2 Natural Circulation

The sodium side of the steam generators is vented to the IHTS Expansion Tank. Following a normal scram the main sodium pump motors will coast down to pony motor speed, where a clutch engages and drives the pump to produce about ten percent of design flow. This flowrate occurs approximately 30 seconds after scram initiation. The vent lines characteristic flow curves are similar to those of the main system and as such will result in a similar decrease to ten percent flow at pony motor speeds.

If the pony motor should fail to operate in a given loop the system flow will decrease to the natural circulation level of approximately 6% of full design flow. The vent line flow will also follow a similar characteristic curve.

Since the total vent flow from the modules is only 1/30 of the main loop flow, the vent flow will not significantly affect either pony motor or natural circulation main loop behavior.

During either natural circulation or pony motor operation, flow in the vent lines will be maintained and the vent lines will remain submerged in the expansion tank, preventing the cover gas from entering and inhibiting flow in 25
the vent lines.

Calculated operating values for the important parameters applicable to natural circulation conditions have shown that the SGS will function satisfactorily in the natural circulation operating mode. The anticipated range of sodium flows is from 2 to 15% for SGS inlet temperatures of 650-850°F. For this same temperature range, the recirculation system natural circulation flow is expected to be in the range of 9 to 14.5%. The hydraulic profiles for water and steam side of the SGS and for the IHTS are shown in Figures 5.5-1 and 5.1-3, respectively.

59

5.5.3.3 Pump Characteristics

The recirculation pump in each loop draws water from the steam drum and forces the water through the two evaporators and back to the steam drum. The pump is a single-stage centrifugal pump, which delivers 2.22 X 10^6 pounds per hour of water to the two evaporators. The pump provides a pressure head of about 148 psi at the plant design conditions. This flow provides a quality of 50% at the exit from the steam generator evaporator units at the plant design power. Since the pumps are constant speed, the evaporator exit steam quality decreases as plant power is reduced.

During normal plant operation, feedwater is mixed with saturated water in the steam drum providing subcooled water to the recirculation system. This subcooled water results in an adequate net positive suction head (NPSH) for the pump during normal operation. Under certain operating conditions, however, the recirculation pumps will be operating when there is essentially no feedwater being delivered to the steam drum, such as during start-up before there is any significant power from the reactor, and, in this case, an adequate NPSH for the recirculation pump is provided by the elevation difference between the steam drum and the pump.

The shaft seal of the recirculation pump is externally supplied with filtered injection water from the main feedwater system which cools the seal and minimizes accumulation of corrosion particulates which could otherwise cause accelerated seal wear and increased leakage. In addition, the recirculation pump has an internal auxiliary impeller and cooler to provide seal cooling in the event that the external seal injection system is lost. However, the internal seal cooling system only functions when the recirculation pump is running. With external seal injection operating, 3.5 gpm of injection water is added to each recirculation loop. With external seal injection secured, there is a net loss of 1.5 gpm of recirculation water lost from each loop through the recirculation pump seal staging valve.

In the unlikely event that the recirculation pump trips and external seal injection is lost simultaneously, the pump shaft seal will overheat and could fail sufficiently to result in a large increase in seal leakage. If not controlled and if the SGAHRS system is initiated, this leakge could potentially compromise the Protected Water Storage Tank (PWST) 30 days capacity. Accordingly, a valved bypass line is provided around the recirculation pump and its inlet and outlet isolation valves. The bypass line is designed to provide hydraulic resistance equivalent to that of the stalled recirculation pump and its connected piping and valves inside the bypass connections. Therefore, the flow of water under natural circulation conditions to the evaporators through the bypass line will be essentially the same as through the normal path.

In the event of a loss of offsite power, both the recirculation pump and main feed pumps will trip. In this case, the recirculation pump will be isolated and bypassed automatically because the arrangement of the valve actuator power supplies causes the isolation and bypass valves to realign to their fail safe positions. For other SGAHRS events, the recirculation pump trips and seal injection flow stops, the operator will attempt to restore seal cooling (e.g., by starting the startup feed pump) before the recirculation pump seal overheats. If these attempts are not successful, the operator will activate the recirculation pump bypass line from the main control panel by opening the bypass valves and shutting the recirculation pump isolation valves within one hour of the loss of seal cooling. The 30 day PWST water supply is therefore ensured since PWST inventory is sufficient to allow up to one and a half hours for operator action. For SGAHRS events where all the recirculation pumps continue to operate but seal injection flow can not be provided, the operator has over ten hours to activate the recirculation pump bypass lines to ensure PWST inventory is not jeopardized as a result of recirculation pump seal staging flow.

5.5.3.4 Valve Characteristics

A number of isolation valves are used in the steam generation system to permit equipment to be isolated from the system for maintenance. These valves are conventional valves which are normally open during operation and are closed only to permit maintenance or repair of the equipment.

Quick-operating isolation valves are located on the inlet lines of the evaporator units and the superheater unit to permit isolation of a unit from the rest of the system in case of a failure in the water or steam to sodium barrier, resulting in a sodium-water reaction. These isolation valves will close in about four seconds after an automatic or manual signal. Isolation of a failed unit will limit the quantity of water or steam available for the sodium-water reaction and, in the case of evaporator units, permit dumping of water and steam to a dump system to further reduce the magnitude of the sodium-water reaction.

The evaporator water dump valve is a quick opening shut-off valve located on the inlet line of each evaporator unit between the evaporator and the evaporator inlet isolation valve. This valve will open at the same time the evaporator isolation valves are being closed. This valve dumps water and steam from the evaporator unit to reduce the total quantity of water available for a sodium-water reaction and to reduce the pressure in the unit. This valve is held open until the water-steam pressure in the unit drops to approximately 250 psi, when the valve is automatically closed. Inert gas is then supplied to the unit to reduce the possibility of sodium and sodium-water reaction products entering the water-steam side of the steam generation system.

Safety/power relief valves are installed on the outlet line of the evaporator units, on the steam drum and on the outlet line from the superheater. These valves all meet the requirements of Section III of the ASME Boiler and Pressure Vessel Code for protection against overpressure. Table 5.5-8 indicated design pressures and valve settings for the steam generator safety/relief valves. Additional valve data is provided in Table 5.5-8A.

5.5.3.5 Steam Generator Module Characteristics

Each evaporator module will produce 1.11×10^6 lb/hr of 50% quality steam from subcooled water. Each superheater module will produce 1.11×10^6 lb/hr of superheated steam from saturated steam. The thermal hydraulic normal design operating conditions are given in Table 5.5-9.

The steam generator modules will supply the turbine with steam at design conditions over a 40% to 100% thermal power operating range for both clean and fouled conditions. The steam generator modules are also capable of removing reactor decay heat with the natural convection in both the intermediate sodium loop and the recirculaton water loop.

This hockey stick unit is of the same basic design as that of the Atomics International-Modular Steam Generator (AI-MSG) unit which was tested in a test program carried out at the Sodium Component Test Installation. The AI-MSG employed a 158-tube module with an overall length of 66 feet, as compared to the 757-tube CRBRP Steam Generator which has an overall length of 65 feet. The AI-MSG heat exchanger was operated for a total of 4,000 hours including operation both as an evaporator (slightly superheated steam out) and as a once through evaporator-superheater (from sub-cooled liquid to completely superheated steam).

The AI-MSG served as a proof test of the AI prototype hockey-stick steam generator design. The unit was operated for 4,000 hours under steaming conditions; all of these 4,000 hours, the unit was at the same temperature level at which the prototype will operate, with a steam pressure equal to or greater than prototype conditions. Table 5.5-9A compares various design operating conditions for the CRBRP Units to the AI-MSG, and lists the number of hours which the AI-MSG operated under respective conditions. The AI-MSG operated at steam pressures equal to or greater than the CRBRP Units for essentially the whole 4,000 hrs., and at CRBRP superheater inlet temperature for 750 hrs.

Since the AI-MSG unit was operated in the once-through mod, simultaneous simulation of both inlet and outlet CRBRP conditions for the separate CRBRP evaporator and superheater units was not achieved, but operation over the CRBRP temperature and pressure range was achieved on both the sodium and steam conditions for significant portions of the test.

Safety Evaluation

The steam generators are essential to remove reactor decay heat.
However, since there are three independent loops with each loop containing
two evaporator modules and one superheater module, the loss of one loop
would not preclude removal of reactor decay power. The steam generators
41 are Safety Class 2, but shall be constructed to Class 1 rules.

Design transients for normal, upset, emergency and faulted conditions are discussed in Section 5.7.3 and Appendix B.

Methods for detecting internal leakage between sodium and the water or steam, the margin in tube walls for thinning and time dependence of tube wastage to effect adjacent tubes are discussed under Steam Generator System Leakage Detection System, Section 7.5.5.

The rationale for the selection of any given number of failed tubes to establish an overpressure design for the IHTS is discussed under Evaluation of Steam Generator Leaks, Section 5.5.3.6.

Tests and Inspections

In-service inspection of the steam generator modules is discussed under In-Service Inspection Program, Section 5.5.2.1.3.

Part Load Operation

Part load operation curves over the range of steam flows from 40 to 100 percent are presented in Section 5.7.2.

Design module heat transfer length were used with nominal values of sodium, water or steam, and tube heat transfer correlations for purposes of this analysis. This implied excess area, therefore, results in sodium operating temperatures in the evaporator lower than those used for design. Design heat transfer areas are determined by adding sufficient margin to the module length to permit operation with fouled tubes at 100% power for nominal sodium conditions. The margin calculated is 10% and is arrived at considering the error-band in heat transfer coefficients and tube wall thickness. Also, included in the 10% margin is a 5% surface allowance made for tube plugging.

The steam flow rate is defined by turbine conditions, power level, and feedwater temperature.

The water or steam side temperature and flow rate are essentially the same for both clean operation and fouled operation. However, the presence of fouling will cause an increase in the required sodium operating temperatures and flow compared to clean operation.

For power levels below about 40 percent a good portion of the inlet sodium end of the superheater and the outlet sodium and of the evaporator will operate close to isothermal temperatures with small sodium to water temperature differences. This is because most of the heat transfer takes place in other portions of the modules.

5.5.3.6 Evaluation of Steam Generator Leaks

A primary design objective for the steam generators is that they be of sufficiently high quality that leaks in the sodium/water boundary will not occur. Careful design and close quality control of materials and manufacturing processes are expected to yield units which are free of common defects, and the probability of a leak in a steam generator tube is expected to be quite small. A Steam Generator Leak Detection System, described in Section 7.5.5, has been provided to allow operator action to limit the consequences of a leak. The leak detection system will alert the operator to the existence of a leak rate as low as 2 x 10^{-5} lb water/sec, which will allow sufficient time for operator action to prevent a significant increase in the leak rate for a broad spectrum
59 of leak rates.

As a final level of protection against tube leaks in a steam generator, the steam generators and the IHTS are being designed to withstand the effects of a large sodium water reaction (SWR). The ASME Code categories being applied in the design of the steam generators and IHTS piping and components for the large SWR event are given in Table 5.5-10.

The design basis leak (DBL) for the CRBRP was selected based upon examination of the physical processes which exist for leak initiation and growth. Two types of tests have been reported which provide information on the leak growth mechanism - small scale tests which model effects of a SWR on materials, and large scale tests which model a large water leak in a model of a steam generator. Smaller scale sodium-water reaction tests have been done to develop an understanding of the effect of a SWR on neighboring tubes in a steam generator. Three mechanisms have been identified for leak growth: self-wastage, impingement, and overheating (mechanical damage from pipe whip, although extremely unlikely, could be considered another mechanism, as discussed later in this section). Self-wastage has been shown to occur for very small leaks in the range of 10^{-6} to 10^{-3} lb/sec (Ref. 13). The process is depicted in Figure 15.3.3.3-1. The result of this process is a leak size of the order of 10^{-3} to 10^{-2} lb/sec. which can produce wastage on another tube in the vicinity of the leaking tube.

Wastage can occur on the outside of a steam generator tube from a leak in another tube in the vicinity. Tests of this mechanism have typically been done by using a water jet directed through sodium to a target material sample. Water injection rates of approximately 10^{-4} lb/sec to 1 lb/sec have been tested. The wastage mechanism results in erosion of the target material at maximum rates of 0.001 to 0.007 inches per second (Ref. 14, 29). The wastage rate is found to be a function of the water injection rate, tube spacing, sodium temperature and leak geometry. Wastage occurring on the surface of a CRBRP steam generator tube at these rates could cause a secondary water leak from tube penetration. However, this would require at least 20 seconds to penetrate the 0.109 inch thick tube wall assuming an initiating leak of the proper characteristics to produce maximum wastage.

The size of a secondary water leak resulting from wastage is difficult to quantify since wastage tests are typically done on materials samples rather than pressurized tubes. The wastage areas observed in tests have ranged from 0.1 in^2 to 1.5 in^2. Failure areas corresponding to the highest observed wastage areas would result in water leak rates corresponding to that of a double-ended guillotine tube failure. However, the entire wastage area would not be expected to blow out. The wasted areas are typically pit-shaped with the area of the pit decreasing with depth. It would be expected that the small area at the bottom of the pit would fail, yielding a return water leak which halts the wastage. Therefore, while the size of a secondary failure caused by wastage is difficult to predict, it is expected to be smaller than the leak rate corresponding to a double-ended guillotine failure.

The third mechanism for failing a tube is overheating from the thermal effects of a SWR caused by a leak in an adjacent tube. This could cause heatup in the adjacent tube and a decrease in tube strength until the tube bursts from the internal pressure. The time for tube failure was analytically investigated using boundary conditions of a temperature of 2700°F (the adiabatic SWR temperature) and heat transfer coefficients as high as 10,000 Btu/hr-ft^2°F. The minimum computed time for failure was 0.4 seconds for an evaporator tube and 0.3 seconds for a superheater tube. Measurements during large SWR tests show peak temperatures of 2300°F and heat transfer coefficients of approximately 2000 Btu/hr-ft^2°F (Ref. 15). These measured conditions, if applied uniformly around the circumference of a tube and over a significant longitudinal area, would require longer times than computed above to produce an overheating failure. Further, establishment of the conditions that could cause such a secondary failure require a large initiating leak.

It is concluded that all three leak growth mechanisms require time to develop; hours for small leak self-wastage, tens of seconds to minutes for wastage, and tenths of seconds to seconds for overheating. The secondary failures that could potentially result from these mechanisms are expected to yield water leaks considerably less than that of a single double-ended guillotine failure. An estimate of the worst plausible leak development sequence would be as follows. A small leak (less than 10^{-3} lb/sec) is postulated to develop. This leak will be assumed to not be detected by the operators through readings from the hydrogen leak detectors. This leak then grows by self-wastage to a leak of 10^{-2} or 10^{-3} lb/sec (a leak size that gives maximum wastage rates on an adjacent tube). This leak continues, if no mitigating action is taken by the operators, until the adjacent tube wall is penetrated. The second leak is assumed to be small enough to not activate the SWRPRS by blowing out rupture disc, but large enough to yield a SWR which overheats an adjacent tube to the point of failure. This final overheating failure is assumed to activate the SWRPRS. The result of this worst plausible sequence is leaks for which the total leak flow rate is not expected to exceed that of a single double-ended guillotine failure.

The Design Basis Leak is not intended to represent a realistic sequence, but, rather, to provide a conservative basis for computing design loads. The DBL is defined as a equivalent double ended guillotine (EDEG) failure of a steam generator tube which is followed by two additional single EDEG failures, spaced at 1.0 second intervals, to a total of 3 EDEG. This sequence is superimposed on a system which has been pressurized by an undetected moderate sized leak to just below the rupture disk burst pressure.

The design basis leak for the CRBRP steam generators contains safety margins in the timing and also the magnitude of the assumed failures. Experimental data (Ref. 15, 16, 17, 24) on the consequences of a sudden, single tube rupture event, justify the design basis leak postulated for the CRBRP steam generator. Although much of the data is not strictly prototypic of the CRBRP steam generator modules, the results demonstrate the effects of SWR reactions in LMFBR steam generators using similar materials, tube wall thickness, pressures, water injection rates and sodium temperatures. Japanese, German and US large leak SWR tests have produced no secondary failures.

The Japanese have conducted seven large leak SWR tests ranging from seven to n seconds. The Germans have conducted five large leak tests of durations 4 to 9 seconds. Six large leak tests (in near-prototype configurations) have been conducted in the U.S. The U.S. test have ranged from 3 to 40 seconds in duration. Significant wastage was observed in only one U.S. test in which one tube in the leaksite region exhibited a 0.016 inch reduction in wall thickness. This corresponded to a wastage rate of 0.016 inch/sec.

A full-scale leak progression test has been conducted in a steam generator of prototypic dimensions and materials. This test, Large Leak Test Rig (LLTR) Series II Test A-3 (Ref. 27), was initiated by rapidly pulling apart a pre-notched tube to expose an injection tube containing a pre-drilled 0.040 inch diameter hole. This hole, representing the self-wastage leak depicted in Step 5 of Figure 15.3.3.3-1, was aimed a a target tube two rows away. The aiming and spacing had been previously determined by bench scale experiments to yield the maximum wastage rate on the target tube. Observed secondary failure sequence is tabulated below. For reference, an EDEG failure area (two cross sections) is 0.26 in^2.

TIME-SECONDS	SECONDARY FAILURE	FAILURE AREA-IN.^{2}
0	Injection Tube	Pre-drilled 0.0013 in.^{2}
60	1	0.017
72	2	
95	3	0.125
108	4	0.029
114	RUPTURE DISC OPENED	
114-120	5	0.11
114-120	6	0.200
114-120	7	0.170

The first secondary required one minute to develop. Further secondary failures occurred at 6 to 12 second spacing. The size of each secondary was less than of a EDEG. These results are a conservative representation of how an acutal leak would progress because (1) the initial leak was aimed and spaced to produce maximum wastage on the target tube (wastage is observed in bench scale tests to be sensitive to configuration), (2) the third secondary failure occurred in the injection tube itself which was of non-prototypic (thin) wall thickness, (3) the tubes contained initially subcooled water which was static; as the test progressed the water flashed and was expelled into the supply system - the tubes were thus undercooled, and (4) the sodium was static. The CRBRP Design Basis Leak is conservative in both the magnitude of and timing of these secondary leaks.

British and Russian tests have demonstrated that an initial tube failure can produce secondary tube failures. In the 15 tests of the NOAH series (Ref. 20) there were three tests in which pressurized Cr-Mo secondary tubes failed. No more than one tube failed in any test. Times to failure varied from 2.5 to 4.5 seconds. In three Super-NOAH tests (Ref. 21) there was one secondary failure attributed to wastage. Time to failure was 14.5 seconds. Dumm's (Ref. 15) sodium-water reaction test number 5 had pressurized secondary tubes none of which failed.

The Russians (Ref. 17) report a sodium heated steam generator leak test in which a small leak (13.2×10^{-3} lb water/sec injected through a 0.030" hole) caused two tubes to fail in a period of 100 seconds. The first failure was attributed to direct wastage (from the jet impinging on the tube). The second (large) leak caused a blowout type (overheating) failure of the second tube.

In every case, the mechanism producing failures has been of the type which requires a substantial time lag between the occurrence of the primary event and the initiation of secondary SWR, and the magnitude of the secondary failure was less than the equivalent of one double-ended guillotine failure. This delay or lack of coherence in propagation and the limited extent of secondary tube failures has been a significant finding. These results demonstrate that water/steam injection from any secondary failures occurs at a time when potential pressure or shock effects are mitigated by the sequence of conditions resulting from the primary rupture event, i.e., by the production of a large volume of reaction product gas within the steam generator shell, and by actuation of the SWRPRS rupture discs. Also, while high frequency (1000 Hz) acoustic pressure pulses typically occur in the first few milliseconds of a guillotine-type rupture event, it has been determined that they produce no shell loading problems due to their low energy content (Ref. 15). Therefore, on the basis of large SWR experience to date, no mechanism has been found effective in causing even a single secondary failure on a time scale rapid enough to contribute to substantial IHX loadings. In addition, the number of equivalent double-ended guillotine secondary tube failures that have occurred in any test (approximately one - Ref. 16) is significantly less than the two assumed for the DBL.

In the case of small primary (initiating) leaks, the leak growth mechanisms identified through tests do not cause instantaneous secondary failures and do not cause secondary water leaks equivalent to a double-ended guillotine tube break. Any delay in time to fail the additional tubes would reduce the pressures resulting from this event. Based on existing data and analyses, the design basis leak of a 1-tube double-ended guillotine failure followed at 1.0 second intervals by single EDEG failures to a total of 3 EDEG failures will result in a conservative IHTS and SGS design. Analysis (Ref. 25) of data from LLTR test series (Ref. 24) has verified analysis methods used in assessing the conservatism of the DBL.

59 The three tube event is not intended to represent a realistic, mechanistic sequence, but rather it provides a basis for calculating loads for the design of components and piping which are believed to be conservative for the large number of mechanistic sequences involving secondary failures which can be postulated.

59 Preliminary analyses have indicated that adjacent tubes would not fail when subjected to the peak calculated pressures for the double ended guillotine failure of one tube. Mechanical failure of adjacent tubes due to whipping of the initially failed tube is also considered unlikely. In a series of tests it was demonstrated

that, for the range of pipe sizes and schedules considered, that whipping pipes will not shear off other stationary pipes of at least the same size and 9
schedule. A tube whip analysis is presented in Section 5.5.3.6.1.

The method of analysis and the calculated consequences of the design basis leak are given in Section 5.5.3.6.2. The same sequences are also postulated for the 9
59 superheater.

5.5.3.6.1 <u>Steam Generator Tube Whip Analysis</u>

Tests were performed using a 2.275 inch OD pipe in the configuration shown in Figure 5.5-5 as reported in Reference 10. The tests were for a more severe case of piping whip in which a steam propelled pipe whipped crosswise onto an anchored pipe. Test conditions included:

a) Steam reservoir pressure of 2000 psia

b) Steam discharge at 90° angle to axis of test section of whipping pipe.

c) The whipping pipe was designed to impact a stationary pipe.

In none of the cases of blowdown-induced dynamic bending did either the whipping pipe or the target pipe show any evidence of wall penetration. The tests showed that a whipping pipe is flexible at impact and absorbs energy in both gross bending and local deformation. In addition, the degree of distortion of the target pipes was consistently less than that of the corresponding whipping pipe. Although the range of pipe sizes and schedules in the test did not include steam generator tubes, theoretical extrapolation technique enable extrapolation of results to include the CRBRP steam generator tubes.

An analytical model was developed from the above tests:

$$\frac{KE}{V_{IP}} = (1.8\quad \theta)\,\frac{P\bar{R}_{IP}}{h_{IP}}\left(\frac{\bar{R}_{WP}}{\bar{R}_{IP}}\right)^{2}$$

where: KE = Kinetic energy for unit length of whipping pipe measured at linear velocity of pipe end 9

V_{IP} = Volume of impacted pipe per unit length

θ = Angular displacement of whipping tube.

P = Steam reservoir pressure

$\bar{R}_{IP}$ = Mean radius of impacted pipe

$\bar{R}_{WP}$ = Mean radius of whipping pipe

h_{IP} = Thickness of impacted pipe

Applying the above equation to a whipping tube in the steam generator is as follows: The first term, angle of displacement of the ruptured pipe in the steam generator, is expected to be small, since the steam generator tubes are essentially parallel during impact. The second term is a constant since only the pressure is a variable and both the impacted and whipping tubes are equal. The third term is unity. Therefore, the kinetic energy, which is a measure of the damage potential of a whipping pipe, will be small. It is concluded that a whipping pipe in the steam generator will not damage adjacent pipes. ANSI N176, "Design Basis for Protection Against Pipe Whip" (Reference 11) provides 9
additional support for this conclusion by stating "The energy level in a whipping tube may be considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

59 The LLTR test completed in 1978 (Ref. 24) have strengthened the above conclusion.

5.5.3.6.2 Analysis of Effects and Consequences

Methods, Assumptions and Conditions

The large SWR events have been analyzed using the TRANSWRAP (Transient Sodium Water Reaction Analysis Program) code. The following set of events is modeled by TRANSWRAP. Immediately following the sudden introduction of water into sodium via a large leak (e.g., a full-guillotine tube rupture), accoustic waves of relatively large amplitude and short duration are generated. These acoustic waves may contain sufficient energy to activate the relief system rupture discs. The acoustic waves are generated within the first few milliseconds following the tube rupture and through reflection and reinforcement can cause peak pressures in the sodium that are above the steam-side pressure. In a time frame of milliseconds after the initial tube rupture, the sodium-water reaction creates an expanding hydrogen bubble which begins to eject the sodium from the faulted unit through the burst rupture discs and into the SWRPRS piping: Water/steam continues to enter the hydrogen bubble, and the sodium/water reaction front propagates with the surface of the hydrogen bubble. Liquid sodium is ejected through the relief lines to the reaction products separator
41 tank.

The TRANSWRAP code has been applied to the intermediate loop and the SWRPRS providing a transient model of the evaporators, superheaters, pump, IHX, rupture discs, and associated piping. The primary objective of the code is to calculate pressure readings on the steam generators and associated components in the IHTS and the SWRPRS due to sodium water reaction effects following the failure of one or more tubes in a steam generator module.

The code also predicts velocities throughout the system, times at which rupture discs are actuated, the sodium-inert gas interface locations in the SWRPRS and the reaction products bubble-sodium interfaces. The water injection rate is coupled to the reaction products bubble pressure which is in turn coupled to the sodium pressures at the bubble-sodium interface.

Some of the specific features and capabilities of TRANSWRAP include:

a. Sodium compressivity is treated.

b. Containment wall elasticity is taken into account through Young's Modulus.

c. Pressure and volume in cover gas spaces are computed.

d. Rupture disc bursting and subsequent flow of sodium into the relief system are computed.

e. Pressure pulse reflections and resultant reinforcements and rarefactions are treated.

f. The motion of the bubble-sodium interface is computed.

g. Friction effects are included for all components and piping. Resultant acoustic wave attenuation is computed.

h. Pressure pulse attenuation due to flow into tanks and reservoirs is computed.

Important conditions and assumptions built into TRANSWRAP include:

a. Instantaneous conversion of 65% of the injected water to hydrogen gas. The 65% yield has been determined (Ref. 25) to be conservative through analysis of U.S. large leak test data (Ref. 24). The British (Ref. 20) and Germans (Ref. 15) have inferred, respectively, 55% and 50% hydrogen yields from their large leak tests.

b. The Na_2O remains in the H_2 bubble but has no effect on pressure or volume of the bubble.

c. The reaction products are in thermal equilibrium.

d. The effective hydrogen bubble temperature is 1700°F. This has been determined (Ref. 25) to be conservative through analysis of U.S. large leak test data (Ref. 24).

e. The pump cover gas experiences isentropic compression (expansion) as a perfect gas.

f. The rupture discs are represented by dynamic models which have been conservatively calibrated against prototype rupture disc tests (Refs. 26, 27).

The TRANSWRAP code has been validated through analysis (Ref. 25) of the LLTR Series I test data (Ref. 24).

The RELAP code (Ref. 9) has been calibrated (Ref. 25) against the LLTR Series I large leak test data (Ref. 24) and used to compute a maximum water injection rate of 37.5 lb/sec. This corresponds to 12.5 lb/sec from each of three double ended tube failures. The expanding hydrogen bubble is treated within the TRANSWRAP code as a continuum in which there are no gradients. The mass of hydrogen contained in the bubble increases as the reaction proceeds and the bubble interacts hydrodynamically with the flowing sodium at its boundaries. The bubble is considered to be a perfect gas. The sodium-water reaction is assumed to be instantaneous with a 65% yield of hydrogen plus 1.2 lb/sec from the precursor leak.

Prediction of local hot spots is not within the scope of the TRANSWRAP Code. The tube over-heating analysis presented earlier in this section is a conservative bounding estimate for local hot spot effects. The probability of local hot spots is small considering the excessive amounts of high conductivity sodium and the high degree of turbulence which can be expected at the leak site.

Analytical Model

Each component in the IHTS is represented in the TRANSWRAP model as an equivalent circular cross section pipe (or a combination of equivalent pipes) of specified length, diameter, elasticity, resistance, and associated initial conditions of flow rate and pressure distribution. The model employed for the large SWR analysis is represented in Figure 5.5-3.

The method of characteristics as developed in References 6 and 7 is incorporated in the TRANSWRAP Code. The equations solved between nodal points on the characteristic grid are (p. 23, Reference 6).

$$\frac{g}{a}\frac{dH}{dt} + \frac{dV}{dt} + \frac{fV|V|}{2D} = 0$$

$$\frac{dX}{dt} = a$$

$$-\frac{g}{a}\frac{dH}{dt} + \frac{dV}{dt} + \frac{fV|V|}{2D} = 0$$

$$\frac{dX}{dt} = -a$$

where H = pressure head; g = gravitational constant
V = fluid velocity; a = acoustic velocities
f = friction factor; D = pipe diameter
t = time; X = distance (axial)

Attenuation of head or acoustic pulse between nodal points, i.e., through straight runs of pipe, is thus implicitly recognized through the friction factor f, which is input for each calculational segment within the model. Three types of junction points or joints connecting straight runs of pipe are available within TRANSWRAP; butt joints, tee joints and end joints (dead ends). Attenuation of acoustic pulses in passing through each type of junction is incorporated through the boundary conditions imposed on the above equations. The boundary conditions, as derived in Reference 7, for each junction type are given below.

(1) Head due to reflections at a dead end:

$$H = 2F(t - \frac{L}{a})$$

F = travelling wave function evaluated from boundary conditions

L = length from disturbance to dead end

The meaning of this expression is that at a dead end, the reflected wave is equal to the incident wave and is of the same sign.

(2) At a butt joint with change of diameter, both reflection and transmission are recognized through the factors:

$$\text{Reflection Factor} = \frac{(A_1/a_1) - (A_2/a_2)}{(A_1/a_1) + (A_2/a_2)}$$

$$\text{Transmission Factor} = \frac{2A_1/a_1}{(A_1/a_1)+(A_2/a_2)}$$

where A is cross-sectional flow area and a is acoustic velocity.

(3) Transmission and reflection factors at a tee joint are:

$$\text{Transmission Factor} = \frac{2(A_1/a_1)}{(A_1/a_1)+(A_2/a_2)+(A_3/a_3)}$$

$$\text{Reflection Factor} = \frac{(A_1/a_1)-(A_2/a_2)-A_3/a_3)}{(A_1/a_1)+(A_2/a_2)+(A_3/a_3)}$$

This treatment of attenuation through friction factors and boundary conditions is generally conservative in that the one-dimensional model cannot account for multi-dimensional attenuation-effects, e.g., in the transmission of an acoustic wave around an elbow. Preliminary measurements in water (Reference 8) indicated a reduction in acoustic wave magnitude of from 10 to 30% in passing through an elbow. This is about an order-of-magnitude higher than what is computed in TRANSWRAP using one-dimensional fraction factors.

Results

TRANSWRAP calculations have been performed for both evaporator and superheater tube failures for the Design Basis Leak. Calculated peak pressures in the major components are shown in Table 5.5-11. The predicted pressure history in the IHX for an evaporator DBL is shown in Figure 5.5-4A. The pressure remains at the steady state value until the first pressure pulse arrives. Thereafter, the pressure is affected by reinforcements and rarefactions. The initial disturbance occurs at 90 milliseconds after initiation of a moderate-sized leak which fails to burst the SWRPRS rupture discs. The delay corresponds to the transit time from the leaksite to the IHX via the cold leg. Between 90 and 480 ms the pressure builds up as the leak continues. The pressure decreased between 480 and 700 ms as the burst of the superheater rupture disc and the venting through both the evaporator and superheater Sodium-Water Reaction Pressure Relief System (SWRPRS) relief lines depressurized the system. As seen from Figure 5.5-4b (Leaksite Pressure History for SWR DBL in Evaporator), the two additional EDEG (at 1.3 and 2.3 seconds) have little effect on the IHX pressure since by 1.3 seconds the leaksite is cushioned by about 120 ft^3 of reaction product gas.

For the Design Basis Leak in the superheater, the peak pressures in all the major components are lower than for the DBL in the evaporator. The lower pressures are the result of the lower water mass flowrate throughout the transient.

The steady-state sodium flow rates and pressure drops throughout the IHTS prior to the steam generator tube rupture are represented in the TRANSWRAP model. Following the tube rupture, the expanding bubble of sodium/water reaction products is treated as a continuum of perfect gas which interacts hydrodynamically with the flowing sodium. For the Design Basis Leak in an evaporator, the bubble is predicted to expand away from the leak in both directions, i.e., the sodium which at steady state flows into the evaporator is reversed while the sodium flowing out of the evaporator is accelerated. As the sodium originally within the evaporator below the leak site is displaced by the reaction products, it is driven through the evaporator outlet tee. The bulk of the sodium and reaction products are expelled through the interconnected SWRPRS relief line. However, a portion is also predicted to flow towards the pump. The SWRPRS relief line is cleared of liquid sodium after about 4 seconds. Gas blowdown through the cleared relief line decreases the bubble pressure. Peak system pressures occur during the first second of the event. It is expected that the sodium flow in the pump suction line will reverse before the gas bubble reaches the pump. The sodium will then drain back toward the relief line (low point in the system). Loop draining will be completed by manual opening of the sodium dump valves.

In the unlikely event that the flow in the IHTS does not reverse and the gas bubble reaches the pump, no damage to the coolant boundary of the pump is expected. It is conservatively assumed that the sodium/gas interface reaches the pump inlet about 8 seconds after the SWR is initiated. However, all PHTS and IHTS pumps are tripped by the PPS by approximately 1 second after SWR initiation. Per the specified pump transient, by seven seconds the pump inlet pressure is reduced to the order of 50 psi, and the pump speed will be reduced to the order of 40% full speed.

Since the pump main motor is tripped long before the bubble could arrive at the pump inlet, there is no possibility of pump overspeed and subsequent missile generation. Uneven hydraulic loads and loss of sodium would eventually result in bearing damage and seizure of the pump.

For the Design Basis Leak in the superheater, the reaction products bubble is predicted to expand away from the leak site in both directions also. Reaction products above the leak site and sodium which normally flows into the superheater are expelled through the SWRPRS relief line connected to the superheater inlet tee. Since flow reversal in the superheater sodium inlet line does not occur because of continued flow from the pump, reaction products cannot enter the IHTS hot leg. Reaction products below the leak site are predicted to accelerate downstream toward evaporator relief lines. The sodium drains from the
loop through the relief lines and the sodium dump lines similar to that described 14
above for the evaporator event.

Conclusions

Based on the information provided above, it is concluded that systems
43 and components designed to the ASME Section III categories given in Table 3.2-5
6 using the loadings given in Table 5.5-11 will maintain their

integrity for a large SWR event. Therefore, the integrity of the barrier separating the primary radioactive sodium from the non-radioactive intermediate sodium is maintained preventing the release of radioactive material to the environment.

5.5.3.7 Pipe Leaks

Pipe leaks are categorized into two categories. Category 1 is labeled identified leakage and includes all leakages into a closed system. Identified leakage includes piping seal or valve packing leaks that are directed to a collection tank or a pump. These leaks occur in a system where it is not practical to make the components 100% leaktight. The existence of identified leakage is known in advance and is provided for in the system design. Category 2 is labeled unidentified leakage and encompasses all leakage that is not into a closed system. CRBRP piping systems are designed to preclude unidentified leaks, in that, the piping is of all welded construction and all welds are
59 inspected and leak tested prior to putting the piping into service. Pipe leaks were postulated in each piping run of the Steam Generator System and 13
investigated to identify those areas where unexpected loading and possible damage may result. Section 3.6 presents a discussion of the criteria for postulating leaks, their effects on nearby equipment and the design measures to be used to prevent propagation of damage. Section 15.3 provides discussion of
15 the effects of such leaks on the core cooling function. Utilizing three independent loops in separate cells precludes propagation of a pipe leak failure between loops; each loop alone is adequate for decay and sensible heat rejection. The common points of the loops, the main steam header and the feedwater header, are beyond isolation valves in each loop (see Figures 10.3-1 and 10.4-4); hence a single failure cannot cause a loss of function in more
15 than one loop.

Small unidentified leaks present no special problems to the steam plant. Water or steam leaks will be detected visually during routine inspection of the plant. In addition, experience with sodium and steam piping has resulted in expected failure rates (external leakage) on the order of 1 per 10^6 hr per loop for steam piping. These failure rates are based upon data reported in 25
References 5.3-18 and 19.

5.5.3.8 Inadvertent Operation of Valves

A discussion of the design basis events and their appropriate limits for this plant is given in Section 15.3. As described in Appendix B of this report, the events in Chapter 15 have been selected to envelope the most severe change in critical parameters from events which have been postulated to occur during planned operation.

5.5.3.9 Performance of Pressure-Relief Devices

The safety-relief valves installed in the water-steam system of the steam generation system will be as used in conventional water-steam service. The service conditions are not expected to be significantly different from those encountered in a fossil-fired power plant with similar pressures and temperatures. In case of a major steam generator failure which results in a large sodium-water reaction and activation of the evaporator water dump system, inert gas pressure is introduced into the failed unit as the pressure drops toward the sodium pressure. This inert gas is at a pressure higher than the sodium pressure to prevent entry of sodium into the water-steam system of the evaporator. In addition, the failed steam generator is isolated from the rest of the steam generation water-steam system during such an incident so that, even if the inert gas is not fully able to prevent entry of sodium into the water-steam side of the failed evaporator unit, the sodium cannot get into the rest of the water-steam system loop.

To ensure proper performance of the pressure-relief devices protecting the IHTS, several methods are utilized to minimize the effects of oxide concentrations, including:

a. The pressure-relief device consists of two rupture discs in series, eliminating potential malfunction of moving parts.

b. The upstream pressure relief device is located in liquid sodium and is not exposed to sodium vapor. The downstream relief device is located in an inert atmosphere free of sodium vapor.

c. The pressure relief devices are located near the main high temperature sodium flow stream and trace heating will be provided for the devices and adjacent piping.

A more detailed description of these devices is contained in Section 5.5.2.4.

5.5.3.10 Operational Characteristics - Design Transients

The overall plant duty cycle list, event classification according to the ASME Section III categories of normal, upset, emergency and faulted, and the event frequencies, is given in Appendix B of this document. Further information as to application of the transients to component design is contained in Section 5.7.3.

5.5.3.11 Material Considerations

5.5.3.11.1 Structural Materials for Elevated Temperature Service

Extrapolation of Creep-Rupture Data

The data base used for extrapolation of creep-rupture properties is that
employed in ASME Code Case 1592. For the Steam Generator Modules materials,
59 | 35 | the maximum design temperatures are 965°F for 2-1/4 Cr-1 Mo steel, and 925°F
for 304 SS. The 304 SS components for which design requirements have been
established are associated with the Na-H_2O leak detection subsystem and operate
at low pressures. Therefore, creep is not a major consideration for 304 SS in
the SGS.
59 | 35

Appropriate allowable stress values for 2-1/4 Cr-1 Mo are available in the Code Case 1592 out to 300,000 hours. The allowable stresses for decarburized 2-1/4 Cr-1 Mo have been modified to account for lower carbon content based upon test data. These allowables provide a conservative basis of design for the steam generator modules.

Extrapolation of Creep-Fatigue Interaction Data

A program is in progress at ORNL to gather additional data on 2-1/4 Cr-1 Mo from which design extrapolations can conveniently be made.

Degradation of Short Time Strength Properties

The short time strength properties used for evaluation of faulted condition capability should not degrade as a result of prolonged service under load at the design temperatures. The ability of 2-1/4 Cr-1 Mo to
41 respond to faulted condition loading after partial decarburization should not decrease significantly because the decarburized layer is relatively thin and the strength of the decarburized material is not substantially below that of the base metal.

Elevated Temperature Tests

Tests that are to be conducted for verification of the design data extrapolation are as planned in the ORNL program plan and review for 2-1/4 Cr-1 Mo steel.

Surveillance and In-service Inspection Program

Critical components have been identified as the steam generator tubes and the dissimilar metal welds in the sodium piping. The dissimilar metal welds will be located in such positions that they can easily be inspected and replaced if necessary.

An in-service inspection and surveillance program will be implemented for the SGS in accordance with the intent of ASME Section XI.

5.5.3.11.2 Fracture Toughness

To demonstrate adequate margins of safety for the components of the steam generator coolant boundary, particularly those that are made of ferritic materials, the following information is provided:

Fracture Control Procedures

Fracture control procedures will be guided by paragraph 3241 of Code Case 1592 and Appendix G of Section III for the steam generator
42 modules (See Section 5.5.1.2). ASME Code Case 1592 applies more specifically to
36 elevated temperature components and paragraph 3241 of the case shows
41 special requirements for the prevention of nonductile fracture. These requirements apply to austenitic stainless steels, as well as ferritic steels, when low energy fracture becomes a plausible failure mode.

Time-Dependent Operational Limitations

Various embrittling phenomena may occur in steels as they age at certain temperatures. The most notable are strain aging in carbon steels, temper embrittlement in 2-1/4 Cr-1 Mo steel, creep embrittlement in 2-1/4 Cr-1 Mo steel and possible austenitic stainless steels, and sigma phase formation in high Cr stainless steels.

Stress relieving of the SGS after fabrication will preclude strain aging in carbon steel components. Embrittlement of 2-1/4 Cr-1 Mo will be minimized by heat treatment. Specifically, heating the steel to a temperature greater than 1300°F and holding for at least one hur. Atmospheric cooling follows the heating.

Development programs, in particular for CRBRP, are under way at ORNL and GE to minimize, if not eliminate, the possible degradation of transition welds joining ferritic and austenitic alloys when exposed to long-term, high temperature sodium environment and thermal cycling. The following is an outline of the approach taken for the transition weld development:

1. Screen filler metals based on thermal expansion coefficient, resistance to carburization from the ferritic base metal and inhibition of oxidation notches at the ferritic/weld metal interface.

2. Determine an optimum joint configuration based on elastic stress analysis (and inelastic if required) for the materials under consideration. This analysis will determine the effect of weld geometry and materials combination on the stresses near the fusion line.

3. Characterize the mechanical properties of the base and weld metals and of the transition joint region resulting from fabrication and service.

4. Develop welding procedures, process controls, and equipment that will prevent microfissures, transition-zone cracking, and oxide penetration. Carburization and dilution of the weld metal at the fusion line will be minimized to obtain more predictable properties.

5. Develop initial and in-service NDE methods.

6. Qualify the welding procedures for the joints.

7. Construct and test several transition joints to verify or develop the evaluation of the anticipated service life based on stress analysis and the design basis mechanical properties.

8. Specify the final welding and finishing procedures. 17

From steps 1 and 2, the transition configuration evolved presently is ferritic - Alloy 800 - austenitic steel with ER Ni Cr-3 as the weld material between
59 ferritic/Alloy 800 and 16-8-2 stainless steel filler between Alloy 800/ stainless steel. Step 3 is being taken at ORNL. The hot-wire Gas Tungsten Arc welding technique with necessary equipment is being developed at GE as part of satisfying the steps 4, 5 and 6. For step 7, two test programs are planned:

(a) Creep-rupture properties of the specimens containing transition welds will be determined and compared with those of the base and weld materials. One group of specimens will include ferritic/ER Ni Cr-3
59 82/Incoloy 800 joints and the other Alloy 800/16-8-2 SS Filler/ austenitic stainless steel and

(b) a test plan is being developed to test a large size (about 10 inches) spool with the chosen transition configuration under thermal cycling and long time high temperature heat conditions. Under Steps 5 through
41 8 the transition welds will be given at least two independent (i.e., radiographic and ultrasonic) examinations and the transition spools will be finished machined on the inside and outside to eliminate notches and other discontinuities due to shrinkage and distortion. The welding and finishing procedures will be carefully developed and incorporated in the transition joint Equipment Specification.

The ferritic/austenitic joints located presently in each loop are as follows:

(1) Main Sodium Piping

(a) Hot Leg - A 316SS/2 1/4 Cr - 1 Mo joint is located upstream of Superheater Inlet nozzle, has size of 26 inches O.D. and is ASME Class 1.

(b) Cold Leg - Two 2 1/4 Cr - 1 Mo/304SS joints are located downstream of each of evaporator outlet nozzles, have size of 18 inches O.D. and are ASME Class 1.

(2) Auxiliary Sodium Piping (Size 6 inches or less O.D.)

(a) Cover gas equalization line from expansion tank to sodium dump tank - One 304SS/carbon steel joint is located near the dump tank nozzle and is ASME Class 2.

(b) Dump line from hot leg to sodium dump tank - One 316SS/carbon steel joint is located near the dump tank nozzle and is ASME Class 2.

(c) Dump line from cold leg to sodium dump tank - One 304SS/carbon steel joint is located near the dump tank nozzle and is ASME Class 2. 17

(d) Dump line from superheater dump nozzle to sodium dump tank - One 2 1/4 Cr - 1 Mo/304SS joint is located near the Superheater dump nozzle and is Class 1; another 304SS/carbon steel joint is located near the dump tank nozzle and is ASME Class 2.

(e) Dump line from two evaporator dump nozzles to sodium dump tank - Two 2 1/4 Cr - 1 Mo/304SS joints are located near evaporator dump nozzles and are ASME Class 1; and one 304SS/carbon steel joint is located near the dump tank nozzle and is ASME Class 2. The two evaporators are connected to a tee which connects to a dump tank nozzle, the connecting piping being 304SS.

(f) Vent line from two evaporators to expansion tank - One 2 1/4 Cr - 1 Mo/304SS joint is located near the expansion tank nozzle and is ASME Class 1.

(g) Vent line from superheater to expansion tank - One 2 1/4 Cr - 1 Mo/316SS joint is located near the superheater vent nozzle and is ASME Class 1. 17

5.5.3.11.3 Austenitic Stainless Steel

The materials presently expected to be used in the SGS are: Type 304 austenitic stainless steel, 2-1/4 Cr-1 Mo steel, and carbon steels. The material of which each component is fabricated is shown in Table 5.5-3.

Cleaning and Contamination Protection Procedures

The steam generators will not be chemically cleaned once fabrication is started. Only local wiping with a rag and solvents such as acetone is approved. The 2-1/4 Cr-1 Mo ferritic components in the completed steam generators will be protected against rusting by inerting the tube side and shell side with dry nitrogen gas at a positive pressure after fabrication and hydrostatic testing, and during shipment to the site.

Rust prevention methods for components prior to assembly will be provided following review and approval of supplier proposed techniques. The methods under consideration for the steam generator tubing are as follows:

a. Sealing individual tubes in plastic bags with dry nitrogen.

b. Vapor phase inhibitor (VPI) inserted within the tubing.

c. Generation of a protective oxide layer by a final furnace treatment. 14

Solution Heat Treatment Requirements

Solution heat treatment requirements for the unstabilized austenitic stainless steel to be employed in the SGS are as described in 5.3.3.10.2.

Control of Delta Ferrite

Control of delta ferrite content in austenitic stainless steel welds will be as described in 5.3.3.10.2.3 of this PSAR, in compliance with ASME Code Case 1592. 25

5.5.3.11.4 Compatibility with Coolants

The decarburization kinetics of 2-1/4 Cr-1 Mo base metal and subsequent strength loss are slow enough so that the selection of 2-1/4 Cr-1 Mo steel as steam generator tubing can be made with only a small design stress adjustment. 2-1/4 Cr-1 Mo steel creep rupture properties are insensitive to carbon content until the level drops below 0.03% carbon. This low carbon level will not be reached during the thirty-year tubing design life. The bulk carbon loss is predicted to be 0.04% at the 965°F design temperature based on experimental decarburization data (Reference 5). Since the initial carbon content is greater than 0.07% and less than or equal to 0.11%, the carbon content will not drop below 0.03%.

A minimum wall thickness of 0.109 inches has been specified for the CRBRP steam generator tubes. Of this wall thickness, a minimum thickness of 0.077 inches has been specified for strength with the balance of 0.032 inches allocated for total corrosion allowance of the sodium and water sides. The steam generator turbine material is 2-1/4 Cr-1Mo steel.

The possibility that wastage from a small sodium water reaction (or, more likely, rapid pressure rupture) would cause propagation of a leak to adjacent tubes has been considered in the definition of the design basis steam generator leak event. (See PSAR Section 15.3.3.3)

A small sodium water reaction event that does not result in replacement of the affected steam generator module may also produce wastage of steam generator tubes. The wastage near the leak site (combined with other corrosion) may be equal to or greater than the 0.32 inch corrosion allowance so that tubes will have to be plugged.

Based on the results of the Large Leak Test Rig (LLTR) tests (Reference 29), this amount of wastage is expected to affect only those eighteen tubes within 2 rows of the leaking tube and only a portion of those tubes. Based on the same test series, wastage of tubes beyond the 2nd row is expected to be significantly less than the 0.032 inch corrosion allowance. This is due to the large excess of sodium present which dilutes the SWR reaction products.

After each LLTR/SWR test the steam generator tube wastage was measured using a boreside ultrasonic testing (UT) device. Following the LLTR Series I program of 5 SWR tests and 1 inert gas test, the test steam generator was destructively examined and the wastage directly measured. Excellent agreement was found between these post-test measurements and the inter-test UT measurements. Similar UT equipment will be employed for CRBRP steam tube inspection and would be sensitive down to 4 mils wastage on a routine basis.

The plant will be shutdown on the basis of a confirmed leak. At shutdown the affected unit will be examined, using helium sniffing techniques to locate the leaking tube(s) and UT techniques to determine amount of wall thinning by wastage in surrounding tubes. Leaky tube(s) will be plugged; plugging of adjacent tubes will depend on the extent of wall thinning as determined by UT. The crib for tube plugging will be specified in the FSAR.

Because of the conclusive test results mentioned above and because tubes will be volumetrically inspected and plugged based on actual wastage, quantitative analysis of wastage as a function of temperature or other variables has not been performed; nor will it be usd as a basis for design or operating specifications.

No specific protection is required for protecting Type 304 SS or 2-1/4 CR-1 Mo steels against intergranular attack, stress-corrosion or general corrosion, provided that specified sodium purity is maintained.

In water or steam, carbon steel and 2-1/4 Cr-1 Mo steel are susceptible to caustic gouging and possibly caustic stress corrosion cracking. Maintaining the feedwater and steam drum purity levels as stated below will prevent these forms of localized attack. For normal operation other than start-up conditions, the feedwater and steam drum purity will be specified as follows:

Feedwater Impurities		Feedwater	Steam Drum
Suspended Solids	PPM	--	0.1
Dissolved Oxygen	PPM	.005	-
Silica	PPM	--	0.1
Iron as Fe	PPM	.01	-
Copper as Cu	PPM	.002	-
Hydrazine	PPM	.005-.015	-
Chlorides	PPM	--	.015
Sodium	PPM	.001	.006
Sulfate	PPM	--	.015
pH @ 77°F		8.8-9.2	8.8-9.2
Conductivity (After Cation Removal) @ 77°F micro-mho/cm		0.2	1.0

Limited duration operation with impurity levels above specified limits is allowable for periods not to exceed 24 hours in special instances. These special instances are defined to include condensate polishing system perturbations, such as those immediately associated with a termination of regeneration.

Corrosion impurities may enter the feedwater system through condenser leakage and/or poor makeup water. To guard against damage from such sources, the feedwater and steam drum water are maintained at levels within stated limits by full flow demineralization and continuous steam drum drainflow (blowdown) at a nominal rate of 10% of full power steam flow (See Section 10.4.7).

To determine the feedwater quality, continuous analysers with alarms are provided to sample conductivity, dissolved oxygen, hydrazine, turbidity, pH. sodium, chloride and silica. Continuous samples of steam drum downcomer water and periodic samples of drum drain (blowdown) water are monitored for conductivity, sodium, silica and pH. The downcomer continuous sample monitors are also alarmed if out of specification conditions occur. The condenser hotwell is monitored for conductivity and sodium ions to guard against condenser leakage. The demineralizer effluent is guarded against impurities break-through by in-line measurements of silica, conductivity and sodium. Finally, the feedwater train is monitored downstream of the deaerator for pH and oxygen content to prevent potential corrosion of this portion of the steam system. An alarm is coupled with the most critical in-line measurements to signal departure from specified levels.

5.5.3.11.5 Compatibility with External Insulation and Environmental Atmosphere

Compatibility of austenitic stainless steel with external insulation is assured as set forth in 5.3.3.10.4. Strict control of halide contents in insulation materials is required. Carbon steels and 2-1/4 CR-1 Mo are compatible with
59 external insulation during normal operation in the absence of excessive moisture. Excessive moisture is prevented by quality controlled installation and operating procedures.

5.5.3.12 Protection Against Environmental Factors

Protection for the principal components of the SGS against environmental factors is provided by the structural integrity of the Steam Generator Building. Environmental factors to be considered include the following:

Fire Protection - See Section 9.13.
Flooding Protection - See Section 3.4.
Missile Protection - See Section 3.5.
Seismic Protection - See Section 3.7 and 3.8.
Accidents - See Section 15.6.

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*References annotated with an asterisk support conclusions in the Section. Other references are provided as background information.

TABLE 5.5-1

STRUCTURAL DESIGN TEMPERATURE, PRESSURE AND MINIMUM TEST

PRESSURE FOR SGS COMPONENTS

COMPONENT	DESIGN TEMP (^{o}F)	DESIGN PRESSURE (PSIG)	MINIMUM HYDRAULIC TEST PRESSURE (PSIG)
Evaporator Module			
Shell	885	325	406
Tubes	885	2400	3000
Superheater Module			
Shell	965	325	406
Tubes	965	2200	2750
Steam Drum	650	2200	3300
SGS Water/Steam Piping			
Feedwater Inlet to Drum Iso. Valve	500	3350	5025
Drum Iso. Valve to Drum	650	2200	3300
Drum to Pump	650	2200	3300
Pump to Evaporator	650	2450	3675
Evaporator to Drum	650	2200	3300
Drum to Superheater	650	2200	3300
Superheater to Isolation Valve	935	1900	2850
Superheater Bypass Inlet	650	2200	3300
Superheater Bypass outlet	935	1900	2850
Recirculation Pump	650	2450	3675
Leak Detection Subsystem Piping	985	325	487
Sodium Dump Tank and Piping	700	55	83
Water Dump Tank Piping	420	300	450
Reaction Products Separation Tank	800*	125*	188

* Design pressures and temperatures shown are not coincident in time.

TABLE 5.5-1 (Continued)

COMPONENT	DESIGN TEMP. (°F)	DESIGN PRESSURE (PSIG)	MINIMUM HYDRAULIC TEST PRESSURE (PSIG)
SWRPRS Piping			
Steam Generator to Reaction Products Separation Tank	800*	300*	450
Reaction Products Separation Tank to Centrifugal Separator	800* max.	125*	188
41 Centrifugal Separator to Vent Stack	200* max.	100*	150

*Design pressures and temperatures shown are not coincident in time.

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Oct. 1977

TABLE 5.5-2

MANDATORY CODE CASES FOR SGS AS APPLICABLE

Code Case	
1473-1	Short Time High Temperature Service for Section VIII, Division 2 -Modifications to Section VIII, Division 2, are provided for vessels which are to operate during part of their service life (less than 2500 Hrs.) at temperatures above those now provided for in Section VIII, Division 2.
1481	Elevated Temperature Design of Class 2 and 3 Nuclear Components -Modifications to Section III are provided for Class 2 and 3 components with normal operating temperatures above those provided for Section III.
1592	Components in Elevated Temperature Service Section III, Class 1.
1593	Fabrication and Installation of Elevated Temperature Components, Section III, Class 1.
1594	Examination of Elevated Temperature Nuclear Components, Section III, Class 1.
1595	Testing of Elevated Temperature Nuclear Components, Section III, Class 1.
1596	Protection Against Overpressure of Elevated Temperature Components Section III, Class 1.
1606	Stress Criteria Section III Classes 2 and 3, Piping Subject to Upset, Emergency, and Faulted Operating Conditions. -Design criteria are provided for Class 2 and 3 piping subject to upset, emergency, and faulted conditions.
1607	Stress Criteria Section III, Class 2 and 3, Vessels Subject to Upset, Emergency, and Faulted Operating Conditions. -Design criteria are provided for Class 2 and 3 vessels subject to upset, emergency, and faulted conditions.

TABLE 5.5-3

SGS EQUIPMENT LIST AND MATERIAL SPECIFICATIONS

Component	Material
Major Components:	
Superheaters	2-1/4 Cr-1Mo
Evaporators	2-1/4 Cr-1Mo
Steam Drums	SA-516CS/SA299
Recirc. Pumps	Carbon Steel
Major Subsystem Components:	
Sodium Rupture Discs	Inconel 600
Reaction Prod. Sep. Tank	Carbon Steel
Centrifugal Separator and	Carbon Steel
Drain Tank	Carbon Steel
Flare Stack w/Igniter	Carbon Steel
SWRPRS Vent Line Rup. Disc	Carbon Steel
Evap. Water Dump Tank	Carbon Steel
Sodium Dump Tank	Carbon Steel
Sodium Dump Tank Rupture Disc	*
Piping and Headers:	
S.G. Subsystem and Feed-water subsystem:	
SGB Wall to Main Fdwtr. Drum Isolation Valve	SA-106, Gr B
Main Fdwtr. Drum Isolation Valve to Steam Drum	SA-106, Gr B
Startup Feedwater Control Valve Piping	SA-106, Gr B
Drum to Pump Suction Header	SA-106, Gr B
Pump Suction Header	SA-106, Gr B
Pump Suction Header to Pump	SA-106, Gr B
Pump to Pump Discharge Tee	SA-106, Gr B
Pump Discharge Tee	SA-106, Gr B
Pump Discharge Tee to Evap. Inlet Isolation Valve	SA-106, Gr B
Evap. Inlet Isolation Valve to Evap.	SA-106, Gr B
Evap. to Drum	SA-106, Gr B
Drum to Superheater	SA-106, Gr B
S. H. to S. H. Isolation Valve	1-1/4 Cr-1/2Mo
S. H. Isolation Valve to SGB Wall	2-1/4 Cr-1 Mo
S. H. Bypass Inlet Piping	SA-106, Gr B
S. H. Bypass Outlet Piping	1-1/4 Cr-1/2 Mo
Recirculation Pump Bypass	SA-106, Gr B

TABLE 5.5-3 (Continued)

	Component	Material
	SWRPRS:	
	Sodium Rupture Discs Discharge Lines to Reaction Products Separation Tank	Carbon Steel
	Sep. Tank to Centrif. Sep.	Carbon Steel
	Centrif. Sep. to Flare Stack	Carbon Steel
	Reaction Prod. Sep. Tank Equilizer Line	Carbon Steel
41	Cent. Sep. Drain Line	Carbon Steel
42	Sodium Dump Subsystem (Piping):	
	Sod. Dump Tank Vent Line to Stack	SA-106, Gr B
	Equilizer Gas Line to Expansion Tank	304 SS
41		
	Water Dump Subsystem and Relief Lines (Piping):	
	Stm. Drum Relief Valve Inlet	SA-106, Gr B
41	Stm. Drum Relief Valve Discharge	SA-106, Gr B
	S.H. Relief Valve Inlet	1-1/4 Cr-1/2Mo
46	S.H. Relief Valve Discharge	1-1/4 Cr-1/2Mo
	Evap. Relief Valve Inlet	SA-106, Gr B
	Evap. Water Dump Valve Inlet	SA-106, Gr B
	Evap. Water Dump Valve Discharge	SA-106, Gr B
	Evap. Relief Valve Discharge	SA-106, Gr B
41	Water Dump Tank Discharge	SA-106, Gr B
	Na-H_2O Leak Detection Subsystem (Piping):	
47 42	Na Supply and Return Piping	304 SS
	Drum Blowdown (Piping):	
41	Drum to SGB Wall	SA-106, Gr B
	SGAHRS (Piping):	
	Stm. Supply Valves to SGAHRS F.W. Pump	SA-106, Gr B
	Stm, Supply Valve to SGAHRS HX	SA-106, Gr B
41	Water Return Valve from SGAHRS HX	SA-106, Gr B
	AFW Supply Valve to Main Fedwtr. Line	SA-106, Gr B
42		

TABLE 5.5-3 (Continued)

Valves:

S.G. Subsystem and Feedwater:

S.H. Inlet Isolation Valve	Carbon Steel
S.H. Outlet Isolation Valve	2-1/4Cr-1Mo
S.H. Outlet Check Valve	2-1/4Cr-1Mo
Recirc. Pump Bypass Isolation Valve	Carbon Steel
Evap. Inlet Isolation Valve	Carbon Steel
Evap. Outlet Check Valve	Carbon Steel
Pump Suction Isolation Valve	Carbon Steel
Main F.W. Control Valve	Carbon Steel
Startup F.W. Control Valve	Carbon Steel
Main F.W. SGB Isolation Valve	Carbon Steel
Main F.W. Check Valve	Carbon Steel
Main F.W. Isol. Valve Drum Isol. Valve	Carbon Steel
Superheater Bypass Valve	Carbon Steel

Safety Relief Valves:

Steam Drum (Safety Function Only)	Carbon Steel
Evaporator	Carbon Steel
Superheater	2-1/4Cr-1Mo

SWRPRS:

SWRPRS Vent Line Check Valve	Carbon Steel
SWRPRS Atmospheric Seal Bypass Valve	*

Water Dump Subsystem:

Evap. Water Dump Valve	Carbon Steel
Water Dump Tank Drain Valve	Carbon Steel

Sodium Dump Subsystem:

Sodium Dump Tank Cover-Gas Relief Valve	*

Drum Drain:

Drum Drain Isolation Valve	SA-106, Gr B

TABLE 5.5-3 (Continued)

Component	Material
Inert Gas Supply:	
Evap. Nitrogen System Isol. Valves	Carbon Steel
S.H. Nitrogen System Isol. Valves	Carbon Steel
Sod. Dump Tank Argon Supply Valve	Carbon Steel
Sod. Dump Tank Argon Exhaust Valve	Carbon Steel
SGAHRS:	
Stm. Supply Valve to SGAHRS F.W. Pump	Carbon Steel
Water Supply Valve to SGAHRS F.W. Heater	Carbon Steel
F.W. Valve from SGAHRS F.W. Heater	Carbon Steel
Stm. Supply Valve to SGAHRS HX	Carbon Steel
Water Return Valve from SGAHRS HX	Carbon Steel
Drum Blowdown:	
D. B. Control Valve	Carbon Steel
D. B. Isolation Valve	Carbon Steel
D. B. Bypass Valve	Carbon Steel
$Na-H_2O$ Leak Detection Subsystem:	
Isolation Valve	304 SS
Heat Exchanger	304 SS
Heater	N/A
47 Electromagnetic Pump	*
Rupture Disc - Sodium Dump Tank/ 41 Expansion Tank Equalizer Line	Stainless Steel

* Detail design and evaluation, including material selection has not been completed.

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Nov. 1978

TABLE 5.5-4

SGS WELD FILLER METAL SPECIFICATIONS

Base Material	ASME Section II	Specification
304 SS	SFA - 5.4	Specification for corrosion resisting chromium and chromium-nickel steel core welding electrodes
	SFA - 5.9	Specification for corrosion-resisting chromium and chromium-nickel steel welding rods and bare electrodes
2-1/4Cr-1Mo	SFA - 5.5	Specification for low-alloy steel covered arc-welding electrodes
Carbon Steel	SFA - 5.1	Specification for mild steel covered arc-welding electrodes

TABLE 5.5-5

SGS PUMP AND VALVE DESCRIPTION

PUMPS	ACTIVE	INACTIVE	ACTUATING SIGNAL
Recirculation Pump		X	N/A
VALVES			
Pump Suction Isolation		X	Manual (Remote)
Evaporator Inlet Isolation		X	SWRPRS
Evaporator Inlet Water Dump		X	SWRPRS
Evaporator Outlet Relief	X		SWRPRS**, High Pressure Evaporator (Steam)
Steam Drum Relief	X		High Pressure - Steam Drum
Superheater Inlet Isolation	X		SWRPRS
Superheater Relief	X		SWRPRS**,High Pressure Superheater (Steam)
Superheater Outlet Isolation	X		SWRPRS**, OSIS/SGAHRS or Low Super-heater Outlet Pressure
Superheater Bypass Valve	X		SWRPRS**, OSIS/SGAHRS, or Low Super-heater Outlet Pressure
Steam to SGAHRS HX		X	Manual (L.O.)*
Water from SGAHRS HX		X	Manual (L.O.)*
Steam to SGAHRS Auxiliary FW Pump		X	Manual
Feedwater from SGAHRS		X	Manual (L.O.)*
Main Feedwater SGB Isolation	X		SWRPRS**, High Steam Drum Level, Low Steam Drum
Pressure,			Cell Temp and Humidity
Main Feedwater Drum Isolation		X	High Steam Drum Level
Main Feedwater Check Valve		X	Simple Check
Main Feedwater Control	X		High Steam Drum Level, Cell Temp and Humidity
Startup Feedwater Control	X		High Steam Drum Level, Cell Temp and Humidity
Evaporator Outlet Check Valve		X	Check Valve
Superheater Outlet Check Valve		X	Check Valve
Steam Drum Drain Isolation	X		SWRPRS**, SGAHRS Initiation, Low Steam Drum Pressure

* L.O. - Locked open

** This function is not safety active

TABLE 5.5-5 (Continued)

	Valves	ACTIVE	INACTIVE	ACTUATING SIGNAL
	SWRPRS Stack Check Valve		X	Check Valve
	SWRPRS Atmospheric Seal Bypass		X	Manual
41	Sodium Dump Tank Pressure Relief		X	High Sodium Dump Tank Pressure
59	Evaporator Water Dump Tank Drain		X	Manual

42

TABLE 5.5-6

SGS LOADING CONDITIONS

35 ASME III Code Class 3 system components will be designed considering the following load combinations:

1 Pumps (Recirculation Loop)

Operating Condition	Component	Load	Stress Limit
See Note 1	Pump Case	Design Pressure	Section III
		Design Temperature	Allowable Stress
	Cover	Design Pressure	Section III
	Bolting	Design Temperature	Allowable Stress
		Safe Shutdown Earthquake	
		Pump Thrust	
		Weight	
		Gasket Loads	

Note 1: Design pressures and temperatures of the recirculation system components are established using pressures and temperatures occurring during emergency and faulted transients. The design temperature is not exceeded during these transients. The design pressure may be exceeded by not more than 10% during these transients. Normal and upset conditions are not controlling.

2 Valves (Recirculation Loop and Main Water/Steam)

35 The valve pressure retaining parts designed to ASME - III Class 3 will withstand seismic forces and pipe loads of the SSE as well as design pressure and temperatures. On other parts, if earthquake needs are to be considered, the following applies:

Operating Condition	Loads
Upset	1. Normal Operating
	2. OBE
Faulted	1. Normal Operating
	2. SSE

TABLE 5.5-7

SGS PIPING AND THEIR DESIGN CHARACTERISTICS

PIPING AND HEADERS	COMPONENT SIZE	NO. PER LOOP	NO. PER PLANT	ASME CODE SEC. III CLASS	DESIGN REQUIREMENTS
1. Steam Generator Subsystem & Feedwater Subsystem					
SGB Wall to Drum Feedwater Isolation Valve	10", sch. 160	1	3	3	3000 psig, 500 F
Feedwater Drum Isolation Valve to Steam Drum	10", sch. 140	1	3	3	2200 psig, 650 F
Drum to Pump Inlet Header	10", sch. 140	4	12	3	2200 psig, 650 F
Pump Headers (Inlet)	18", sch. 140	1	3	3	2200 psig, 650 F
Pump Inlet Header to Pump	18", sch. 140	1	3	3	2200 psig, 650 F
Pump to Pump Discharge Tee	12", sch. 160	1	3	3	2450 psig, 650 F
Pump Discharge Tee	12", sch. 160	1	3	3	2450 psig, 650 F
Pump Discharge Tee to Evaporator Isolation Valve	10", sch. 160	2	6	3	2450 psig, 650 F
Evaporator Isolation Valve to Evaporator	10", sch. 160	2	6	3	2400 psig, 650 F
Recirculation Pump Bypass	8", sch. 160	1	3	3	2400 psig, 650 F
Evaporator to Drum	16", sch. 140	2	6	3	2200 psig, 650 F
Drum to S.H.	12", sch. 140	1	3	3	2200 psig, 650 F
S.H. to Isolation Valve	16", sch. 160	1	3	3	1900 psig, 935 F
S.H. Bypass Inlet Line	4", sch. 160	1	3	3	2200 psig, 650 F
S.H. Bypass Outlet Line	4", sch. 160	1	3	3	1900 psig, 935 F
Isolation Valve to SGB Wall	16", sch. 160	1	3	3	1900 psig, 935 F
Startup Feedwater Control Valve Piping	4", sch. 160	1	3	3	3000 psig, 500 F
2. SWRPRS					
Sodium Rupture Disc Discharge Lines to Separator Tanks	18" nom., var. wall, 24" nom., var. wall, 26.46" nom., 2.2" wall	3	9	3	300 psig, 800 F
Separator Tanks to SGB Roof	16", sch. 40	1	3	3	125 psig, 200/800 F
SGB Roof to Flare Tip	16", sch. 40	1	3	ANSI B31.1	125 psig, 100°F
Recirculation Pump Bypass	8", sch. 160	1	3	3	2400 psig, 650F

Table 5.5-7 (Continued)

SGS PIPING AND THEIR DESIGN CHARACTERISTICS

PIPING AND HEADERS	COMPONENT SIZE	NO. PER LOOP	NO. PER PLANT	ASME CODE SEC. III CLASS	DESIGN REQUIREMENTS
54 Separator Tank Equilizer	24", sch XS	1	3	3	125 psig, 800 F
3. Sodium Dump Subsystem					
Sodium Dump Tank Inlet Piping f/Dump Valve	4", sch. 40	5	15	2	50 psig, 965 F
Sodium Dump Tank Vent Line to Stack	8", sch. 40	1	3	3	*
Sodium Dump Tank Equilizer Gas Line to Isol. Valve	6", sch. 40	1	3	2	50 psig, 700 F
4. Water Dump Subsystem & Relief Lines					
Steam Drum Relief Valve Inlet Piping	6", sch. 120	2	6	3	900 psig, 535 F
	8", sch. 40	2	6	3	300 psig, 420 F
Steam Drum Relief Valve Discharge Piping	6", sch. 80	3	9	3	900 psig, 840 F
Evaporator Relief Valve Discharge Piping	10", sch. 40	2	6	3	300 psig, 420 F
	8", sch. 40	2	6	3	300 psig, 420 F
	6", sch. 80	4	12	3	900 psig, 535 F
61 Evaporator Water Dump Valve Inlet Piping	4", sch. XXS	4	12	3	2400 psig, 650 F
Evaporator Water Dump Valve Discharge Piping	6", sch. 80	2	6	3	900 psig, 535 F
	10", sch. 80	2	6	3	900 psig, 535 F
Water Dump Tank Discharge Piping	6", sch. 80	1	3	3	300 psig, 420 F
Water Dump Tank Inlet Piping	10", sch. 80	1	3	3	900 psig, 535 F

59 41 * Design requirements will be provided after final evaluation of transients.

TABLE 5.5-7 (Continued)

SGS PIPING AND THEIR DESIGN CHARACTERISTICS

	PIPING AND HEADERS	COMPONENT SIZE	NO. PER LOOP	NO. PER PLANT	ASME CODE SEC. III CLASS	DESIGN REQUIREMENTS
59	5. Drum Blowdown Drum to SGB Wall	6", sch. 160	1	3	3	2200 psig, 650 F
	6. Na-H_2O Leak Detectors					
59 47	IHTS to Leak Detection Sodium Isolation Valves	3/4"	4	12	2*	325 psig, 985 F
47	Isolation Valves to Leak Detection Modules	1/2"	4	12	3	325 psig, 985 F
41						

* Designated ASME III, Class 2, optionally upgraded to Class 1.

TABLE 5.5-8

SGS SAFETY RELIEF VALVES

	VALVE LOCATION	SIZE (Effective Flow Area)	NO. PER LOOP	NO. PER PLANT	DESIGN REQUIREMENTS	SAFETY RELIEF SETTING
46	Steam Drum *	4.18 in^2	2	6	2200 psig, 650° F	2170, 2220 psig
	Evaporator	4.18 in^2	4	12	2200 psig, 650° F	2250, 2280 psig
59	Superheater	4.18 in^2	3	9	1900 psig, 935° F	1800, 1850, 1900 psig

41

46 * Safety Function Only

TABLE 5.5-8A

VALVE DATA SUMMARY

(a) VALVE IDENTIFICATION STEAM GENERATOR SYSTEM	(b) MAX FLOW lb/hr	(c) TYPE	(d) SIZE INCHES	(e,f) ASME SECTION III DIVISION	(g) CLOSURE TIME SEC	(h) ACTUATOR MECHANISM	(i) POWER SOURCE
Superheater Outlet (53SGV012)	1.11×10^6	Gate	16	Class 3	3 max.	Electro-Hydraulic	1E Electric *
Superheater Bypass (53SGV016)	3.41×10^4	Flow Control	4	Class 3	3 max.	Electro-Hydraulic	1E Electric *
Superheater Inlet (53SGV011)	1.11×10^6	Gate	12	Class 3	3 max.	Electro-Hydraulic	1E Electric *
Evaporator Inlet (53SGV008)	1.11×10^6	Gate	10	Class 3	3 max.	Electro-Hydraulic	1E Electric
Steam Generator Bldg. Feedwater Inlet Isolation (53SGV001)	1.22×10^6	Gate	10	Class 3	3	Electro-Hydraulic	1E Electric *
Main Feed Water Inlet (53SGV002)	1.22×10^6	Flow Control	10	Class 3	5	Air Diaphram	Instrument Air
Start-up Feedwater Inlet (53SGV003)	2.44×10^5	Flow Control	4	Class 3	5	Air Diaphram	Instrument Air
Steam Drum Drain Valves (53SGV014,15)	1.1×10^5	Gate	6	Class 3	3	Electro-Hydraulic	1E Electric *

*Active Function (Safe Position) is 1E Electric

TABLE 5.5-9

35 THERMAL/HYDRAULIC DESIGN OPERATING CONDITIONS

	Sodium	
41	Superheater Inlet Temperature, ^{o}F	922
41	Evaporator Outlet Temperature, ^{o}F	651
35	Flow per Loop, lbm/hr	13.49×10^{6}
	Water/Steam	
	Turbine Steam Pressure, psia	1465
	Turbine Steam Temperature, ^{o}F	900
	Turbine Steam Flow Per Loop, lbm/hr	1.11×10^{6}
35	Feedwater Temperature, ^{o}F	468
	Recirculation Ratio	2:1
	Steam Piping Pressure Drop	
27	Superheater-Turbine, psi	85
41	Steam Drum-Superheater, psi	32
35	Evaporator-Steam Drum, psi	7

Table 5.5-10

SWR DESIGN BASIS

		ASME CODE CATEGORY	
LEAK DESCRIPTION[(1)]	FAILED STEAM GENERATOR	AFFECTED RPST	OTHER STEAM GENERATORS AND IHTS EQUIPMENT IN THE AFFECTED LOOP
44 Small Leak in One Tube	Upset	Normal	Upset
One EDEG* Followed by Two Additional Single EDEG Failures at One Second 61 Intervals (Total 3 EDEG's)	Faulted	Faulted	Emergency

(1) See Section 5.5.3.6 for detailed descriptions and basis

61 * Equivalent double ended guillotine

TABLE 5.5-11

CALCULATED RESULTS FOR LARGE SWR DESIGN BASIS LEAK*

Failure Location	IHX Peak Pressure PSIA @ Sec.	Failed Unit Peak Pressure PSIA @ Sec.	Pump Peak Pressure PSIA @ Sec.	Peak Pressure in Adjacent Steam Generators, PSIA @ Sec.		Time to Clear First Relief Line, Seconds
				Evaporator	Superheater	
Evaporator	331 @ 0.412	395 @ 0.420	373 @ 0.436	320 @ 0.391	337 @ 0.364	4.24
Superheater	304 @ 0.311	333 @ 0.548	311 @ 0.619	254 @ 0.438	-------	3.65

* Water Injection rate = 1.2 lb/sec for $0 \leq t \leq = 0.3$ sec Precursor Leak At t = 0.3 sec, one EDEG occurs. At t = 1.3 sec., one additional EDEG occurs. At t = 2.3 sec., one more EDEG occurs. (total 3 EDEG)

TABLE 5.5.12 MAXIMUM ALLOWABLE STEAM GENERATOR SYSTEM PRESSURE BOUNDARY VALVE LEAK RATES

VALVE	LEAK RATE, LB/HR
Evaporator Inlet Isolation	4.7
Evaporator Inlet Water Dump Isolation	.02
Evaporator Outlet Relief	1.0
Steam Drum Relief	1.0
Superheater Inlet Isolation	4.7
Superheater Relief	1.0
Superheater Outlet Isolation	4.7
Superheater Bypass	1.2
Main Feedwater Isolation	4.7
Steam Drum Drain Isolation	4.7
Nitrogen Supply (per Conn.)	1.0

TABLE 5.5-13 SUMMARY OF U.S. LARGE SODIUM/WATER REACTION TESTS

COUNTRY	TEST DESIGNATION/ OBJECTIVE	TEST VESSEL	TEST BUNDLE	INITIAL PRESS/TEMP. SODIUM PSIG	SODIUM °F	WATER PSIG	WATER °F	WATER INJECTION METHOD	DURATION SEC	WEIGHT LB	SIGNIFICANT RESULTS
U.S.	LLTR Series-1, SWR-1/One Double ended Guillotine (DEG) Failure near lower nozzle, subcooled H_2O	Al-MSG 16- inch ID Vessel Prototypic Height	158 tubes of prototypic material and dimensions, prototypically spaced	122	600	1900	543	Rapid DEG of pre-weakened tube	10	80	No Secondary failures. Maximum wastage on one tube near leaksite = 0.016 inches. Only significant wastage in all 6 Series 1 tests.
	LLTR Series 1, SWR-2/Same as SWR-1 @ mid-span	Same as SWR-1	Same as SWR-1	81	628	1900	543	Same as SWR-1	10	60	No secondary failures
	LLTR Series 1, SWR-3 One DEG @ 1.75 in from upper tube sheet, Two-Phase H_2O	Same as SWR-1	Same as SWR-1	116	800	1900	700	Same as SWR-1	5	40	No secondary failures
	LLTR Series 1, SWR-4 Same as SWR-3 with superheated steam	Same as SWR-1	Same as SWR-1	80	800	1900	700	Same as SWR-1	3	8	No secondary failures
	LLTP Series 1, SWR-5 Same as SWR-4 with 700-F Nitrogen injected	Same as SWR-1	Same as SWR-1	90	800	1900	700	Same as SWR-1	3	zero	Served to calibrate RELAP code
	LLTR Series 1, SWR-5 Same as SWR-4 with Three Equivalent DEG	Same as SWR-1	Same as SWR-1	90	800	1900	700	Same as SWR-1	3	8	No secondary failures. Series 1 served to validate the TRANSWRAP Code.

TABLE 5.5-13 SUMMARY OF U.S. LARGE SODIUM/WATER REACTION TESTS

COUNTRY	TEST DESIGNATION/ OBJECTIVE	TEST VESSEL	TEST BUNDLE	INITIAL PRESS/TEMP. SODIUM PSIG	SODIUM °F	WATER PSIG	WATER °F	WATER INJECTION METHOD	DURATION SEC	WEIGHT LB	SIGNIFICANT RESULTS
U.S.	LLTR Series II, Test A1a, One DEG @ Lower Midspan, Injected Nitrogen, Prototypic Rupture Disk Assembly used on all Series II Tests	Same as A2	Prototypic	125	580	2000	580	Same as SWR-1	30	0	Prototypic rupture disk assembly used on all Series II tests. Served to verify RELAP calibration
	LLTR Series II, Test A1b, Same as A1a except A1b used double disk and minor difference in leak location.	Same as A2	Same as A2	125	580	2000	580	Same as SWR-1	43	0	Served to verify RELAB calibration
	LLTR Series II, Test A2, One DEG @ Lower Midspan, subcooled H_2O	Prototypic Cross-Section 1/2 Length	Prototypic	125	580	1700	580	Same as SWR-1	40	200	No secondary failures. Maximum measured secondary wastage equals 4 mils. Prototypic double disc assembly served to calibrate TRANSWRAP rupture disc model

TABLE 5.5-13 SUMMARY OF U.S. LARGE SODIUM/WATER REACTION TESTS

COUNTRY	TEST DESIGNATION/ OBJECTIVE	TEST VESSEL	TEST BUNDLE	INITIAL PRESS/TEMP. SODIUM PSIG	SODIUM °F	WATER PSIG	WATER °F	WATER INJECTION METHOD	DURATION SEC	WEIGHT LB	SIGNIFICANT RESULTS
U.S.	LLTR Series II, Test A-3, One self-Wastage Leak Simulation @ sub-cooled H_2O @ 0.1 lbm/sec aimed for maximum secondary damage.	Same as A2	Same as A2	145	580	1700	580	Rapid pull-apart of prenotched tube to expose 0.040" dia. hole.	145	144 plus	Secondary failures (less than an EDEG) after long delays (one minute and longer).
	LLTR Series II, Test A6, One DEG @ Lower Midspan Periphery, subcooled H_2O.	Same as A2	Same as A2	125	580	1700	580	Same as SWR-1	36	200	No secondary failures.
											System modified as gas-free Actual test contained large gas space to S.G. TRANSWRAP overpredicted measured pressures where comparable.

TABLE 5.5-13 SUMMARY OF U.S. LARGE SODIUM/WATER REACTION TESTS

COUNTRY	TEST DESIGNATION/ OBJECTIVE	TEST VESSEL	TEST BUNDLE	INITIAL PRESS/TEMP. SODIUM PSIG	INITIAL PRESS/TEMP. SODIUM °F	INITIAL PRESS/TEMP. WATER PSIG	INITIAL PRESS/TEMP. WATER °F	WATER INJECTION METHOD	WATER INJECTION DURATION SEC	WATER INJECTION WEIGHT LB	SIGNIFICANT RESULTS
U.S.	LLTR Series II, Test A7, One DEG @ Lower Midspan, sub-cooled H_2O higher initial sodium pressure.	Same as A2	Same as A2	255	580	2000	580	Same as SWR-1	2	15	Secondary tubes filled with nitrogen @ 400 PSIG.
	LLTR Series II, Test A8, Intermediate-sized superheated steam injection.	Same as A2	Same as A2	180	900	1550	700	Rapid pull-apart of prenotched tube to expose 0.054" dia. hole.	40		No secondary failures deduced from instrumentation and post test helium leak checks. Final confirmation awaits post test destructive examination.
	LLTR Series II, Test A5, Intermediate-sized superheat injection	Same as A2	Same as A2	50	625	1450	625	Rapid pull apart of tube to expose 0.25" dia. hole	58	TBD	Test Report not available. Examination of of test article in progress.

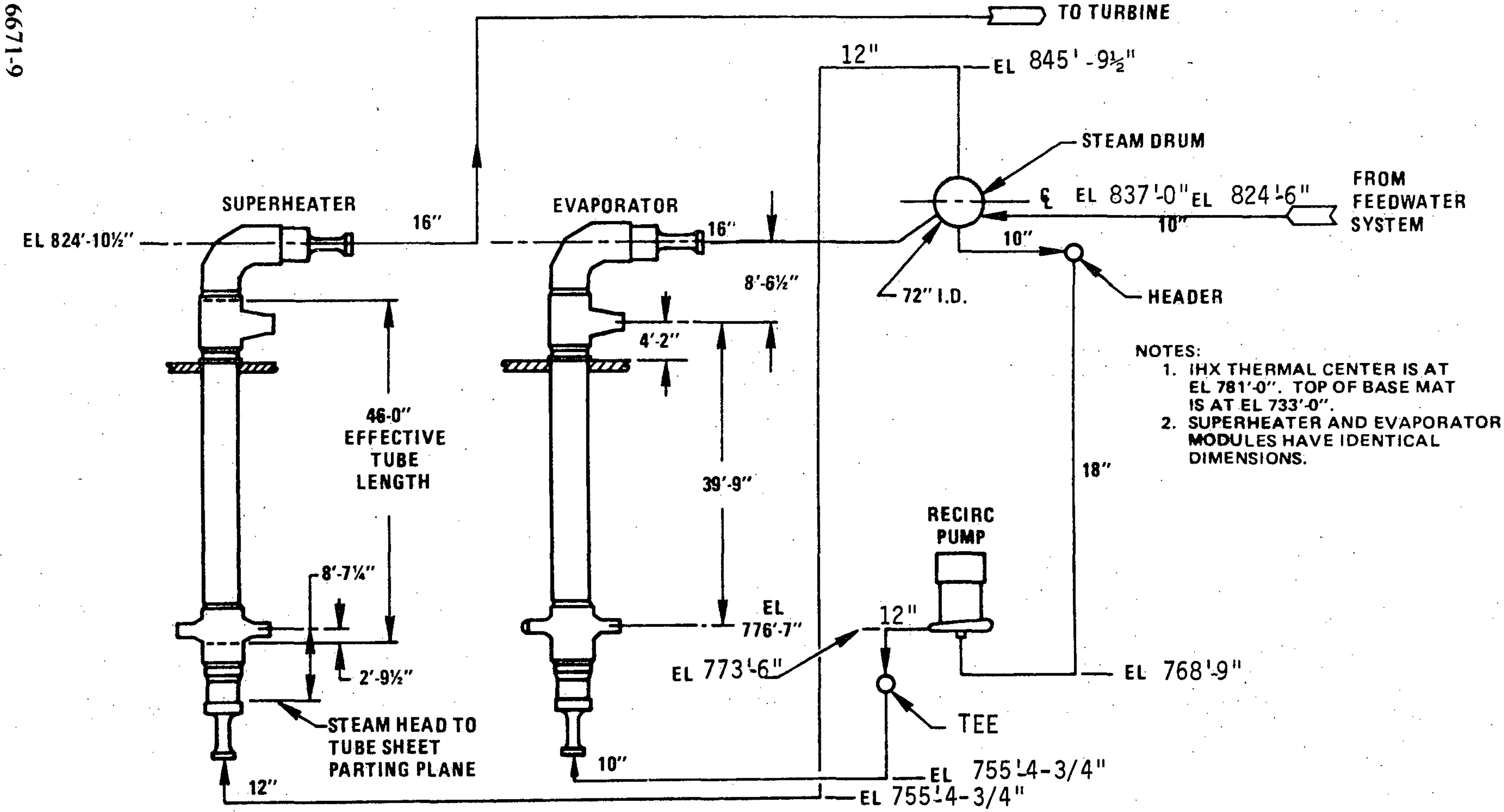


Figure 5.5.1 Steam Generation System Hydraulic Profile

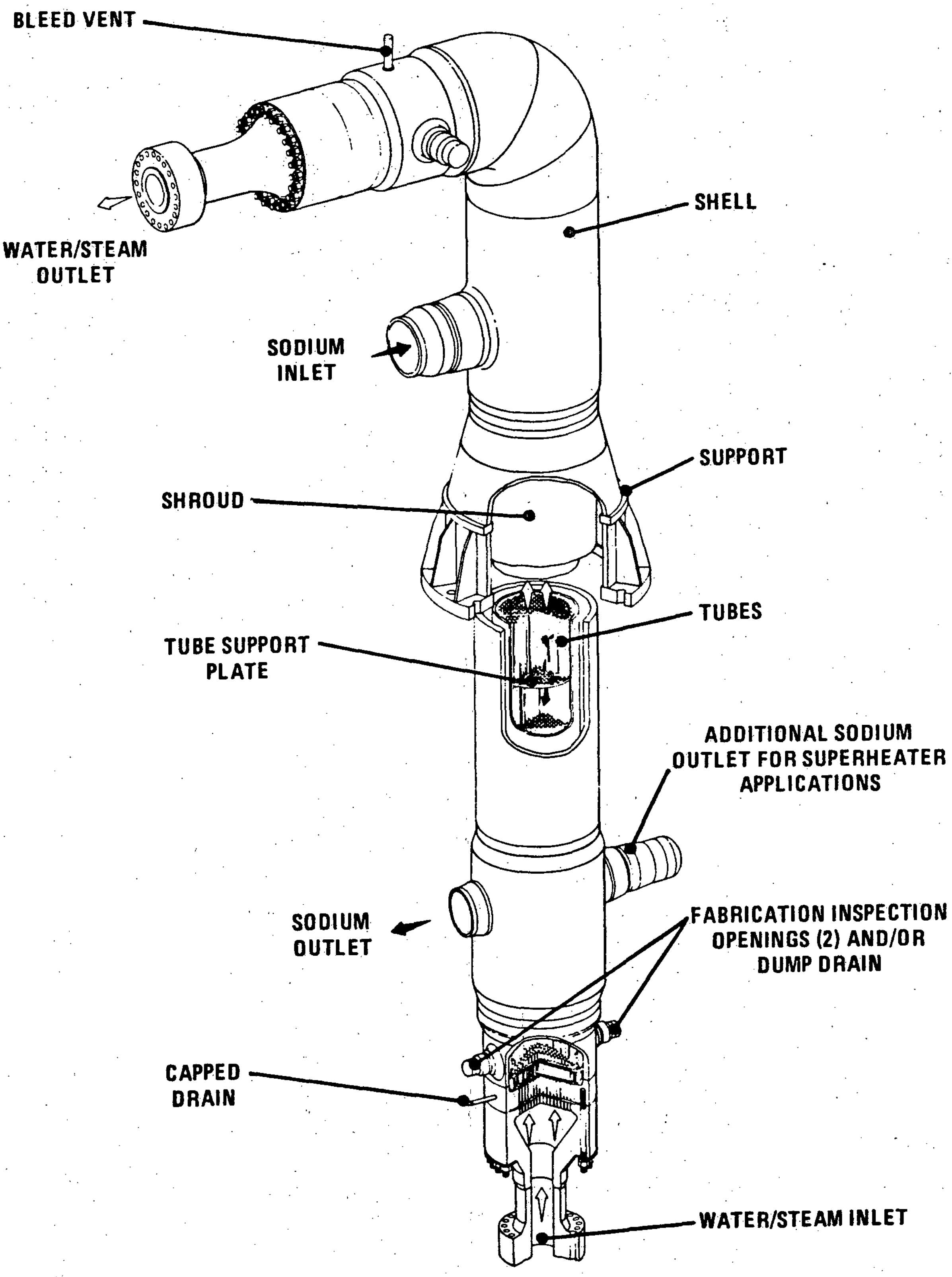
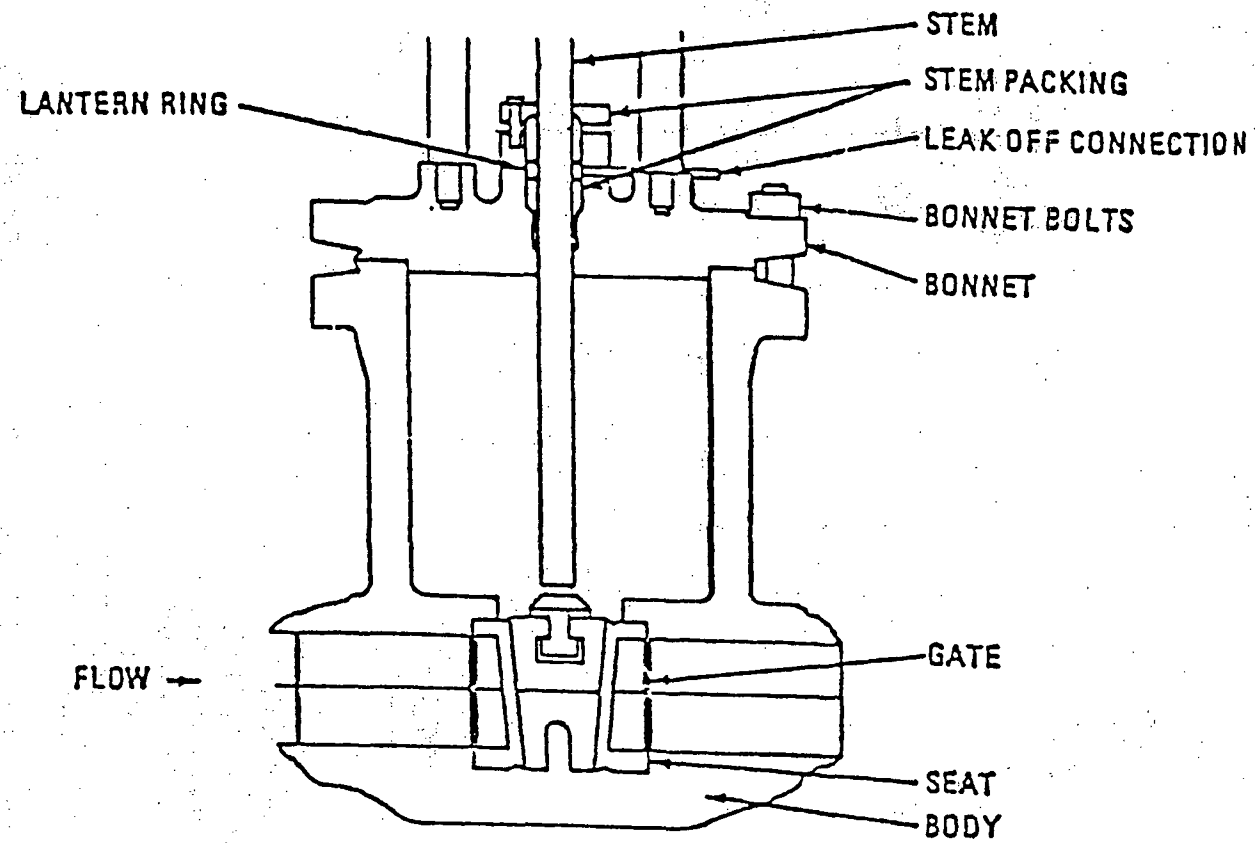
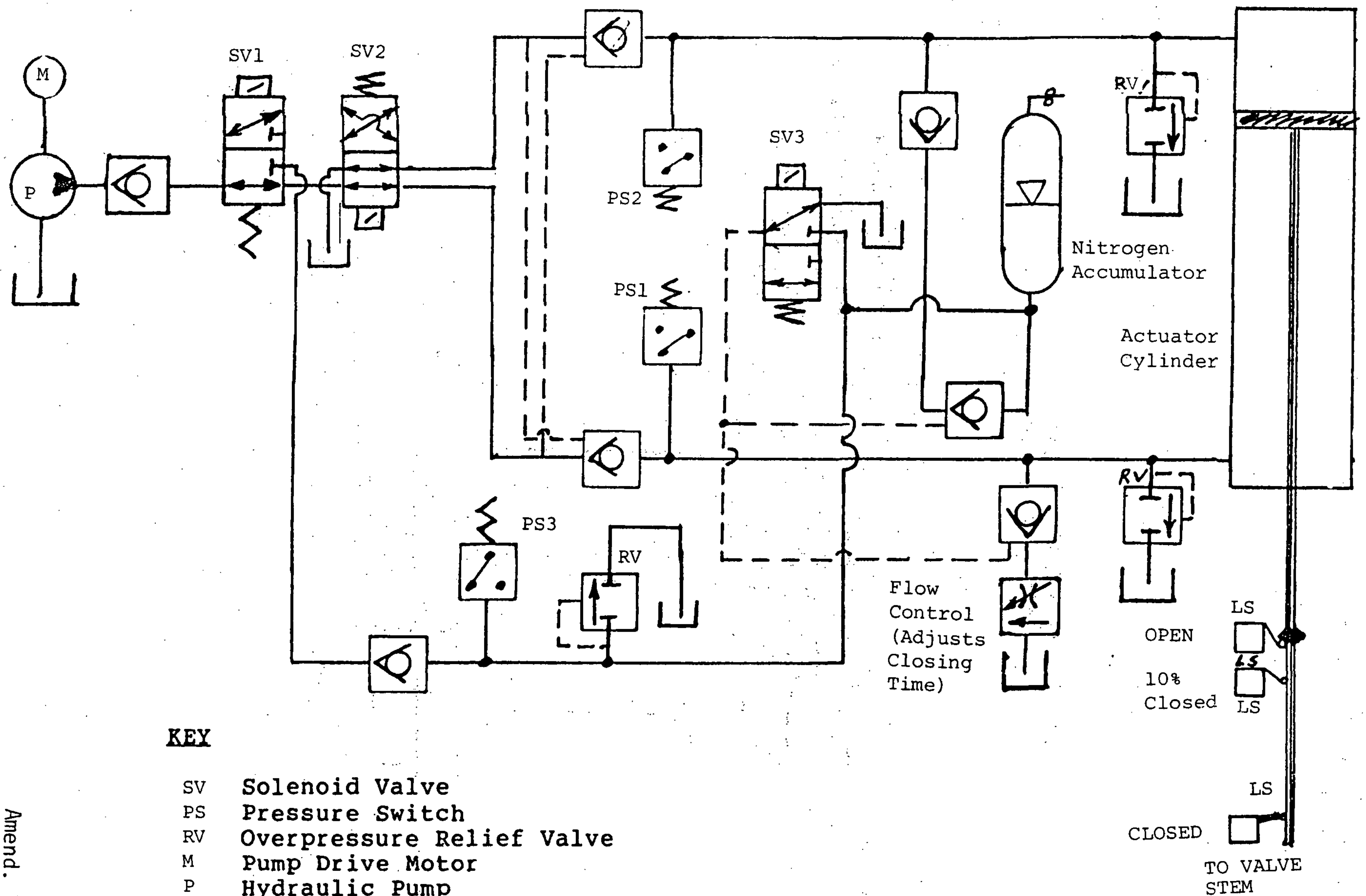


Figure 5.5-2. Steam Generator Module

Figure 5.5-2A Main Steamline Isolation Valve (Superheater Isolation Valve Outlet)

See Figure 5.5-2B for valve operator schematic



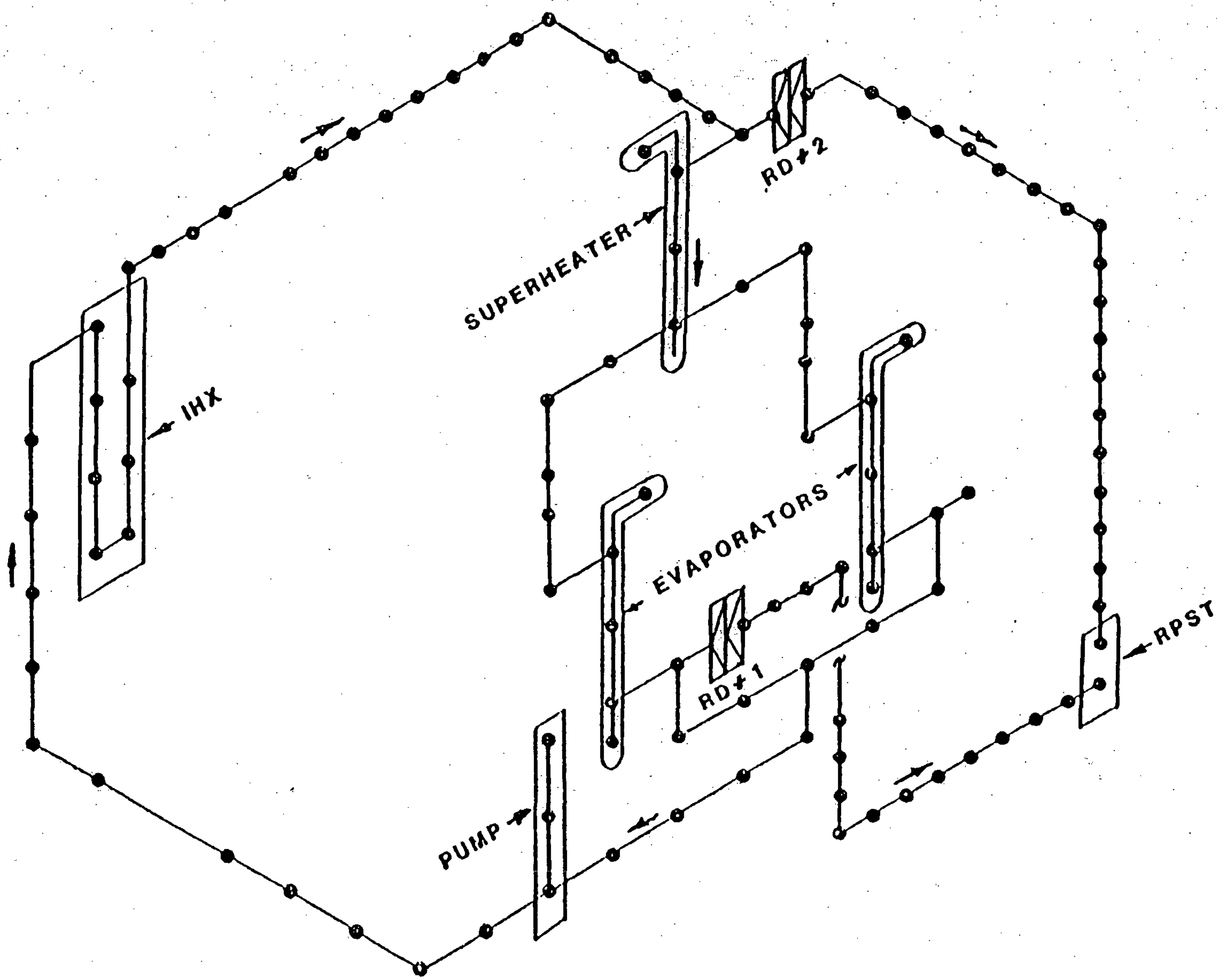


KEY

SV	Solenoid Valve
PS	Pressure Switch
RV	Overpressure Relief Valve
M	Pump Drive Motor
P	Hydraulic Pump
	Reservoir (Low Pressure fluid)
	Pilot-Operated Check Valve

Figure 5.5-2B. ELECTRO/HYDRAULIC OPERATOR FOR TYPICAL QUICK-CLOSING ISOLATION VALVE (Shown in Opening Mode)

FIGURE 5.5-3 TRANSWRAP Model of IHTS and Relief System



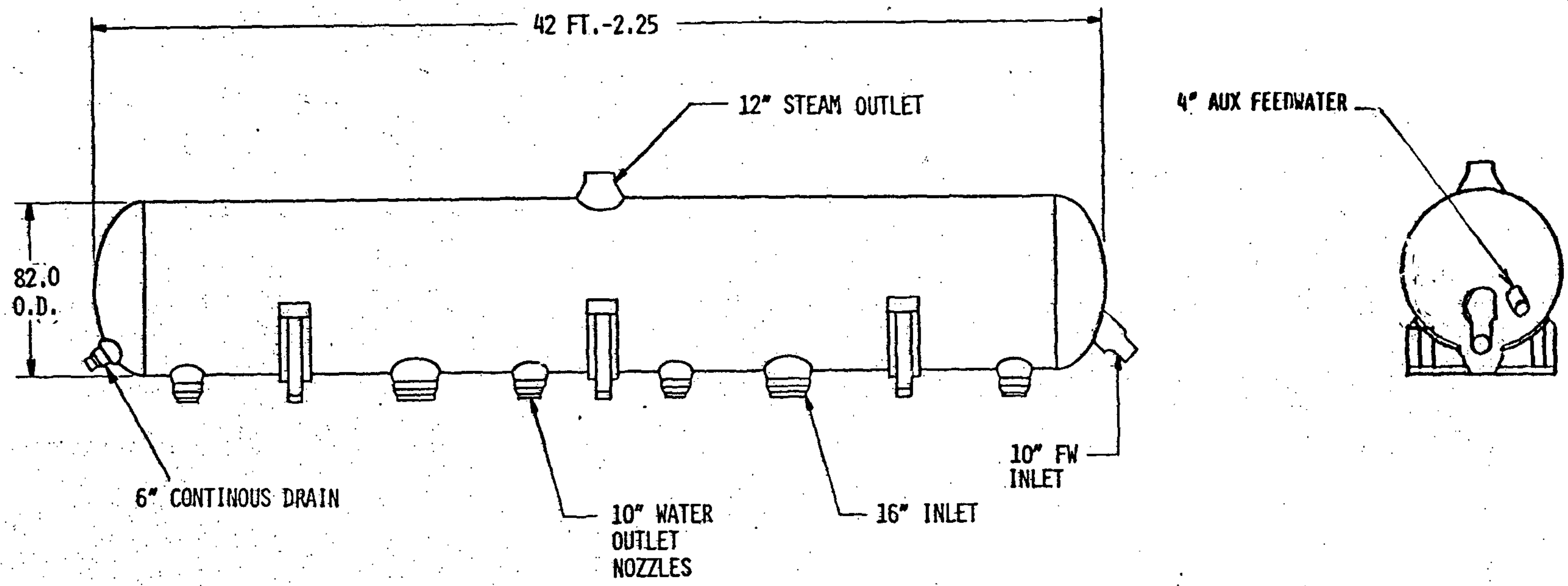


Figure 5.5-4 Steam Drum Outline

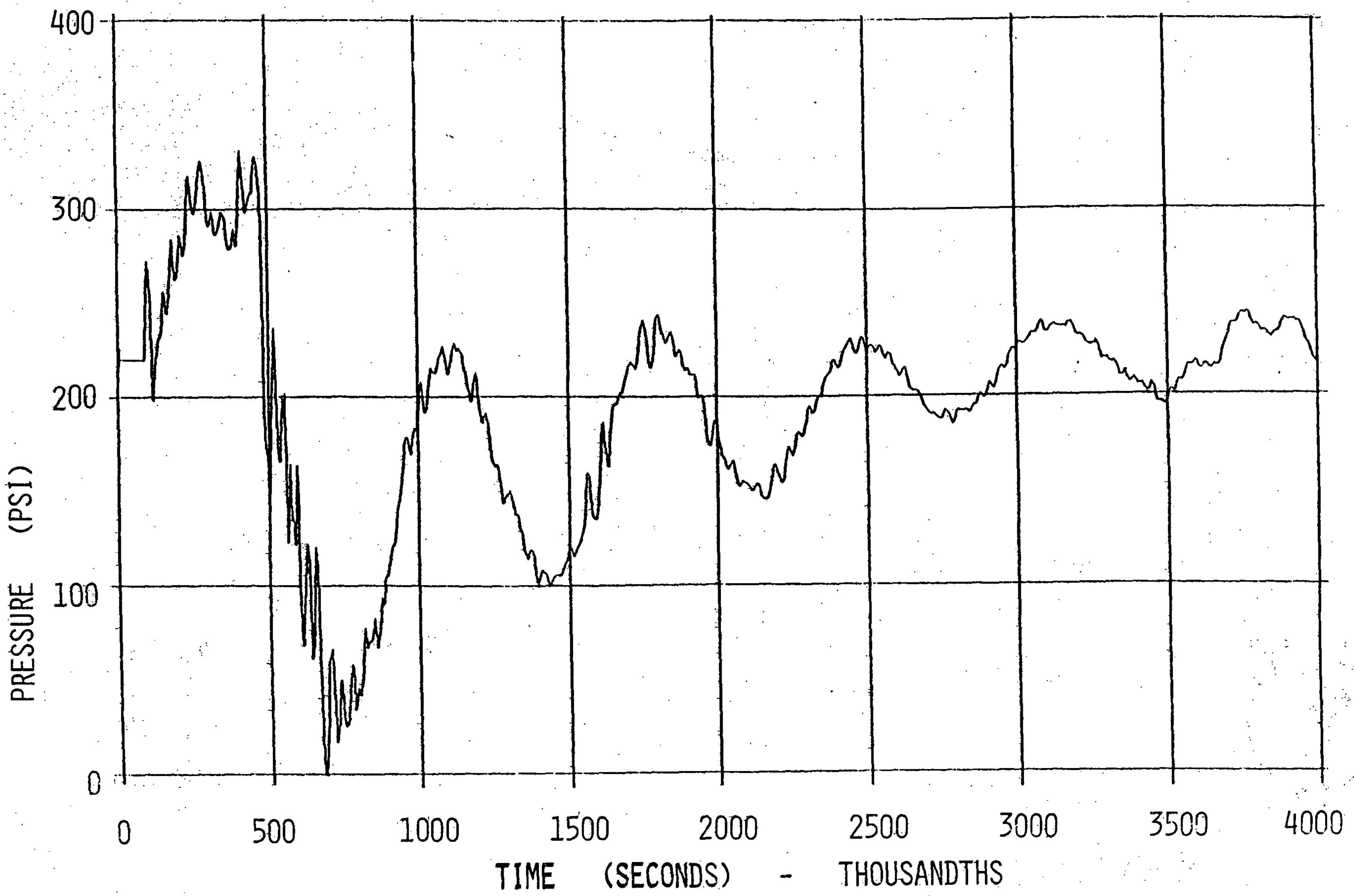


Figure 5.5-4A

Predicted Pressure History in IHX for SWR DBL in Evaporator

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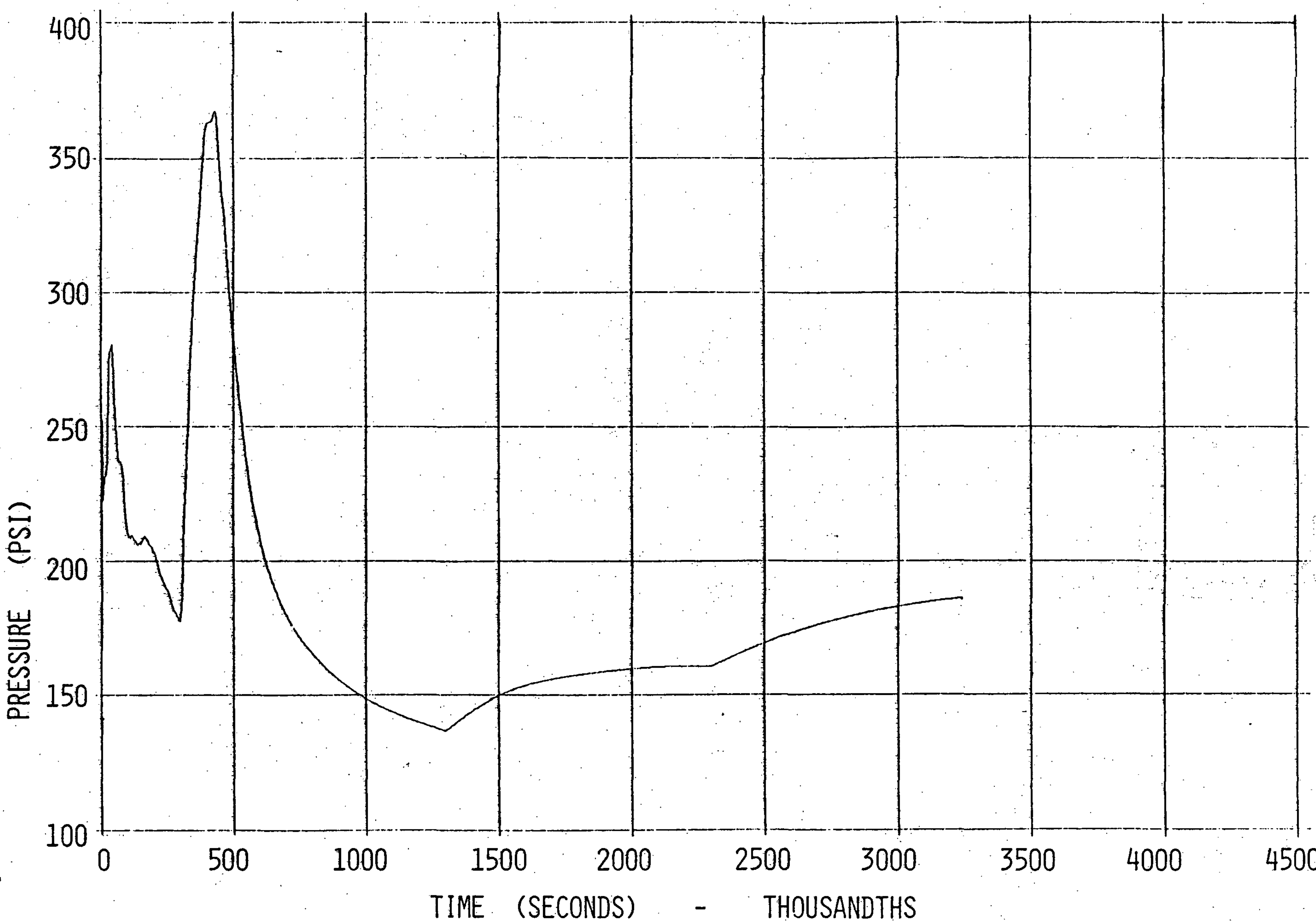

Figure 5.5-4b. Leaksite Pressure History for Sodium Water Reaction DBL in Evaporator

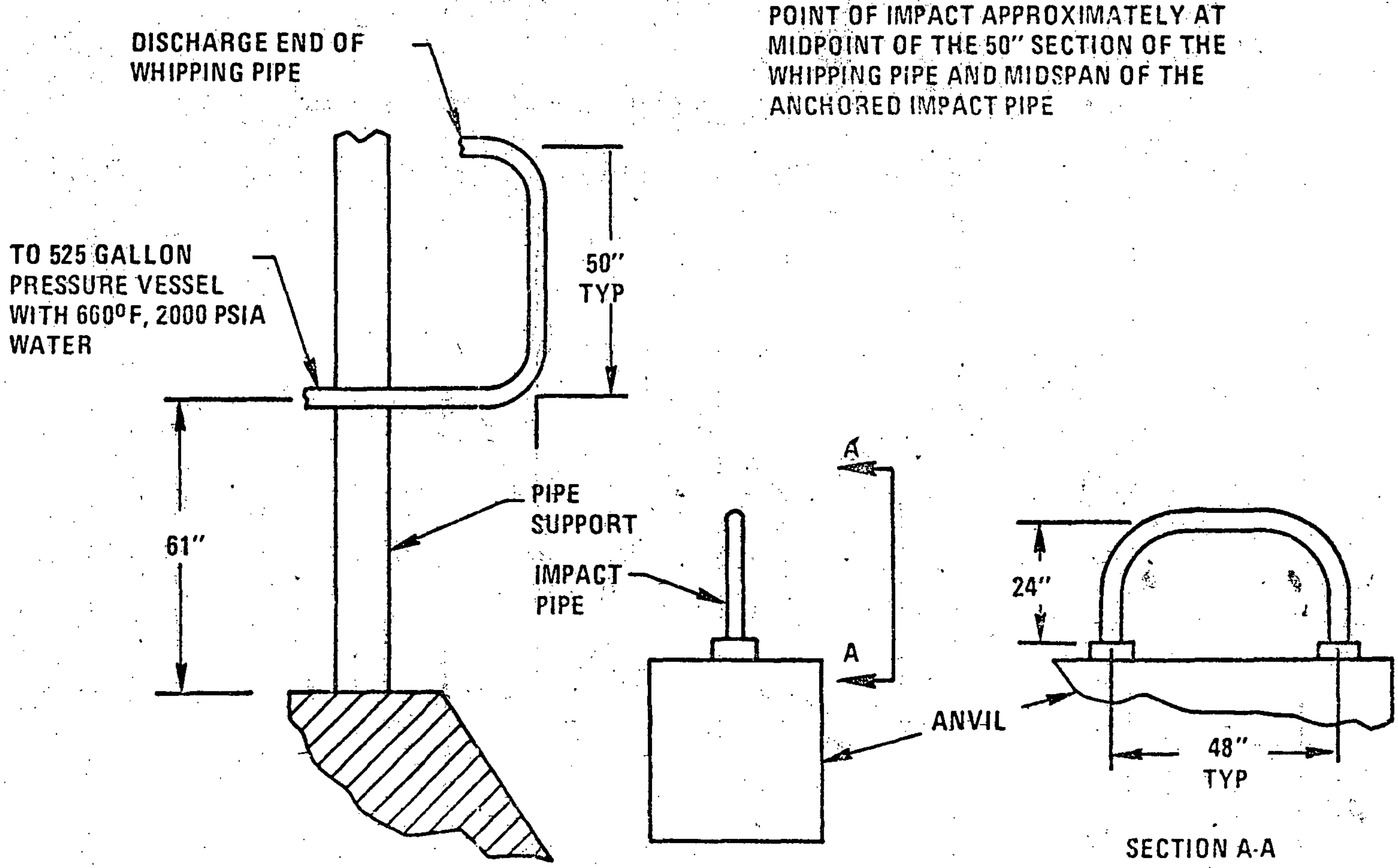

Figure 5.5-5. Pipe Whip Test Configuration, Tests 3 And 4 Of Reference 9

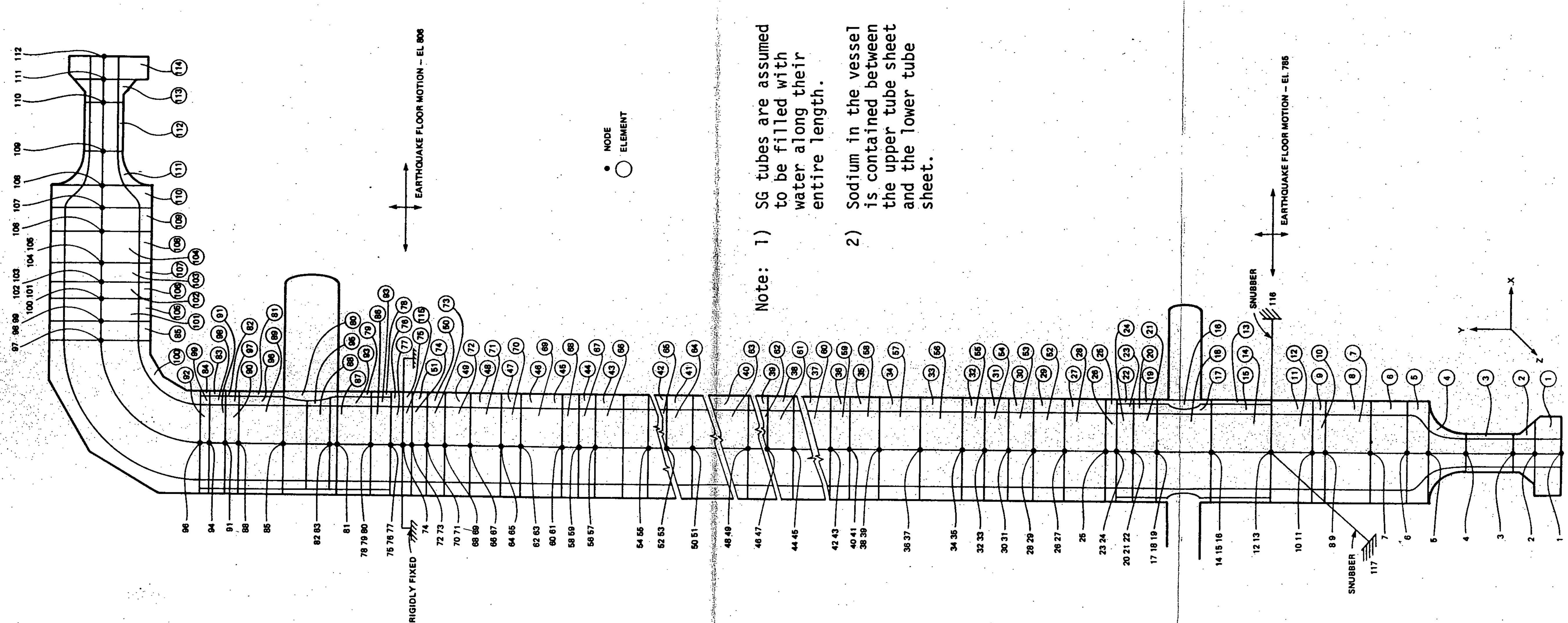

FIGURE 5.5-6
PRELIMINARY SEISMIC MODEL OF THE
STEAM GENERATOR MODULE

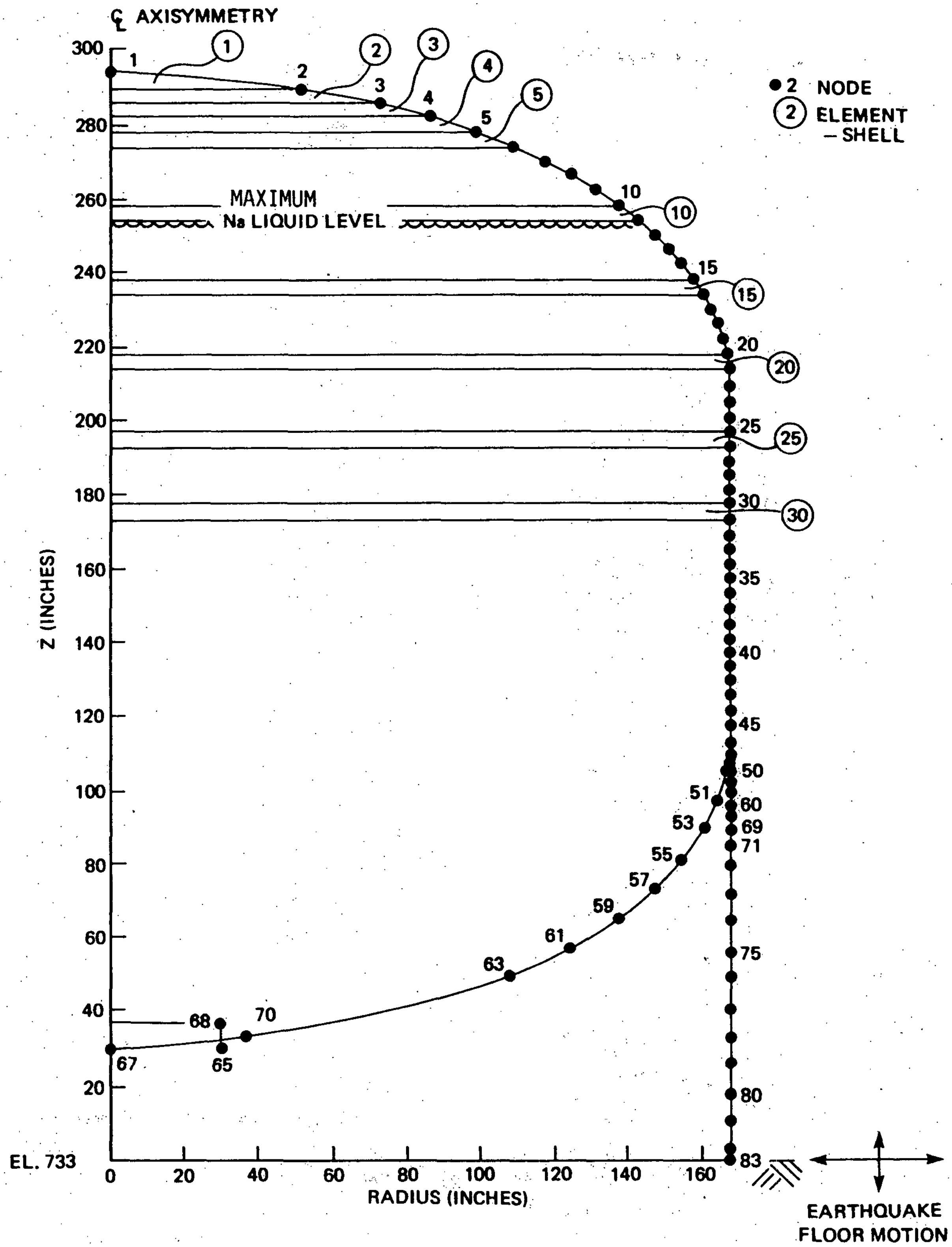

FIGURE 5.5-7 SEISMIC MODEL SODIUM DUMP TANK

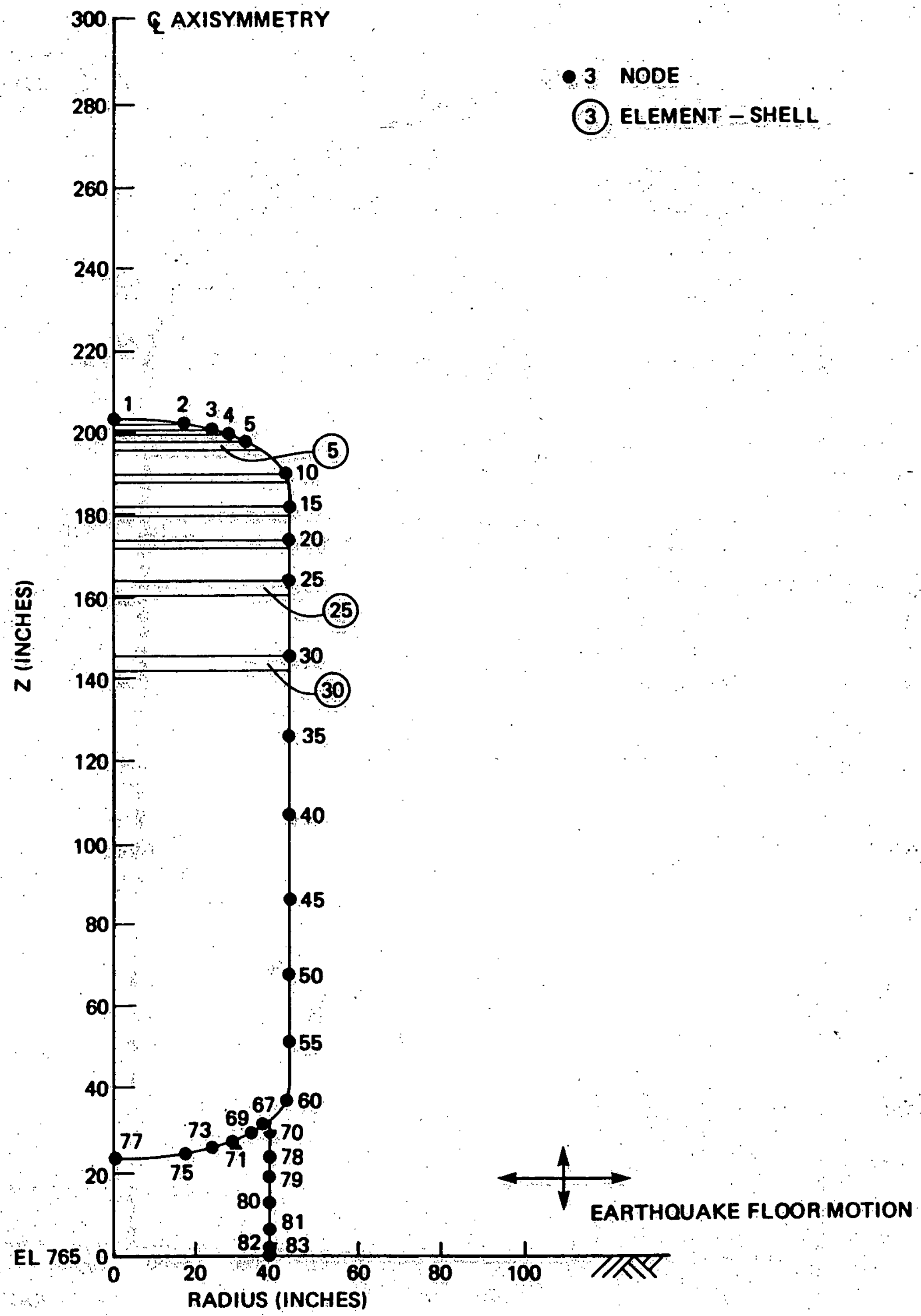

FIGURE 5.5-8 SEISMIC MODEL WATER DUMP TANK

Amend. 17
Apr. 1976

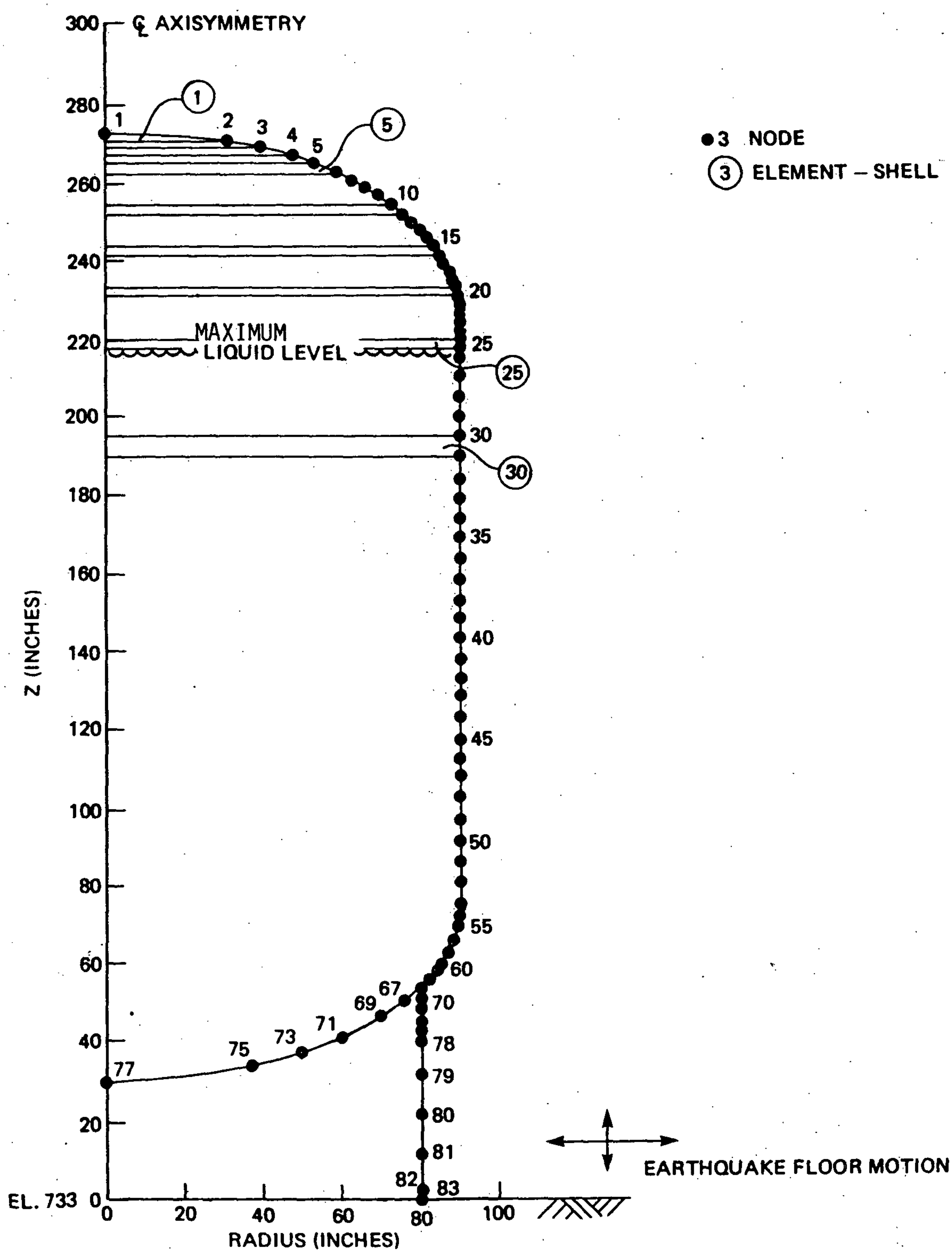

FIGURE 5.5-9 SEISMIC MODEL REACTION PRODUCTS SEPARATOR TANK

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5.6 RESIDUAL HEAT REMOVAL SYSTEMS

The residual heat removal systems include the Steam Generator Auxiliary
26 Heat Removal System (SGAHRS) and the Direct Heat Removal Service (DHRS). These systems are described in Sections 5.6.1 and 5.6.2, respectively.

Sensible and decay heat removal from the CRBRP reactor following rated or two thirds* rated power operation for all normal shutdown conditions and all operational occurrences and postulated accidents is through the primary and intermediate heat transport loops. With pony motor flow each loop by itself is adequate to remove all short term and long term decay heat from the reactor and transport it to the Steam Generator System (SGS). Each Steam Generator loop, regardless of recirculation pump operation, is capable of removing all short term and long term decay heat from the IHTS provided either the Main Condenser/Feedwater train or the SGAHRS is available to that loop.

The Main Condenser/Feedwater train is the normal path for heat removal from the SGS. However, it is comprised of non-safety class equipment and its operation is not required for safe shutdown of the plant.

The SGAHRS is a safety related system designed to provide the ultimate heat sink for all postulated loss of feedwater or loss of normal heat sink incidents. The SGAHRS consists of two subsystems; the Auxiliary Feedwater System (AFWS) and the Protected Air Cooled Condensers (PACC's) which provide short and long term heat sinks, respectively. The AFWS is capable of providing sufficient water to each and every available loop in the SGS to remove all short term decay heat loads via steam venting. The PACC in each SGS loop begins operating at full capability as the AFWS is initiated, and accepts an increasing percentage of the decay heat load as the total load decreases, until the PACC in any loop operating with forced air flow is capable of removing all of the decay heat. The steam venting and hence feedwater flow is reduced automatically as the PACC operation begins to bring the Steam Drum pressure below the vent valve set point. The AFWS continues to provide makeup for losses due to valve leakage, etc.

The three redundant heat removal paths operating as discussed above have sufficient redundant capability, properly qualified equipment and Class IE power supplies to provide adequate short and long term decay heat removal for all design basis events.

The PHTS, IHTS, SGS and SGAHRS are designed to provide decay and sensible heat removal from the reactor via natural circulation in combination with steam venting utilizing a steam turbine driven auxiliary feedwater pump.

---

* The two loop operation power level has not yet been specified. The term "two thirds" is used as a nominal value until a specific value is established. 20

Specifically, the PHTS and IHTS will provide sufficient heat removal to prevent loss of coolable geometry following a normal pump coastdown from full power operation with three loops at natural circulation flow. The SGAHRS turbine driven feedwater pump is sized to provide adequate short term heat removal after full power operation via one loop without recirculation pump or motor driven feedwater pump operation. The SGAHRS Protected Water Storage Tank capacity and PACC heat removal capability are such that the entire long term decay heat load can be carried by a combination of extended steam venting from one SGS loop and operation of one PACC with water side natural circulation and air side forced circulation. These components have been sized to assure a 30 day supply of protected water under the most severe accident conditions.

27 The DHRS is designed to increase the overall post shutdown heat removal reliability by providing long term decay heat removal capability in addition to that provided in the three redundant HTS loops and SGAHRS.

The DHRS consists of one heat removal path capable of removing approxi-
16 27 mately 11 MWt from the primary coolant. That capability is adequate to prevent loss of coolable geometry assuming the SGS heat sink is lost immediately upon shutdown from rated power and active operation of DHRS is initiated one
27 half hour after shutdown.

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Nov. 1980

5.6.1 Steam Generator Auxiliary Heat Removal System (SGAHRS)

In the short term, decay heat is to be removed by the main condenser whenever it is available. The plant operator may initiate heat removal via the Protected Air Cooled Condensers (PACC) for long term heat removal at any time; however, due to the limited heat removal capability of the PACC's (45 MWt total for all three units) either the main turbine generator condenser or the short term steam venting and auxiliary feedwater subsystem must function until the steam generator heat load drops below the capability of the PACCs (∿ 4 1/2% of rated). Both the PACCs and the steam venting and auxiliary feedwater are subsystems of SGAHRS. Under normal three loop shutdown conditions, without using the main condenser, steam venting is expected to cease within one hour
26 **after the plant trip.**

Whenever the normal heat removal path is not available in the short term, activation of SGAHRS will occur automatically with both the auxiliary feedwater and PACC subsystems brought into service. The two subsystems will continue to function concurrently until the heat load is reduced to a level such that venting will cease and the PACC will remove the entire heat load. Operator action is only required to shut off the auxiliary feedwater pumps once the venting and feedwater supply requirements are ended. Failure to do so is not critical, but will result in the gradual heating of the remaining PWST water as this water is recirculated.

Normal decay heat removal, in the short term, will be accomplished using the main condenser via the turbine bypass system. The bypass system setpoint of 1450 psig will be maintained during this time interval while the sodium and water system temperatures are reduced to the ∿600°F level associated with the saturation conditions. During this time interval, the SGAHRS will be involved since a plant trip activates the PACC
3 subsystem. At such a time as the PACCs are functioning and the heat load is equal or less than the PACC capability, the Steam Generator System (SGS) pressure will fall below the bypass system setpoint, and continued decay heat removal will be accomplished in a closed loop manner via the PACC. These conditions would be maintained until a reduction to refueling temperatures is desired at which time the PACC setpoints would be modified and PACC operation would bring the sodium and water system temperatures down to the 400°F level.

17

In the event that the normal heat sink or feedwater supply is unavailable, the SGAHRS short and long term subsystems will be activated by low steam urum level, or high steam-to-feedwater flow ratio trip signals. In the short term, the heat removal is accomplished primarily by venting
43 at 1475 psig from the superheater exit via SGAHRS vent pressure control valves. This is supplemented by steam condensation in the PACC until the PACC can carry the entire heat load. Auxiliary feedwater makeup as required by steam venting will be supplied by SGAHRS auxiliary feedwater pumps automatically. When the steam venting ceases, the PACC control logic will reduce the steam drum pressure to approximately 1400 psig corresponding to approximately 588^{o}F sodium and water system temperatures.

Cooldown to refueling temperatures would follow the procedures used when the main condenser was available.

17

5.6.1.1 <u>Design Bases</u>

5.6.1.1.1 <u>Performance Objectives</u>

<u>System Objectives</u>

a. SGAHRS shall remove sensible heat and reactor decay heat from the reactor and main heat transport loops following those reactor shutdowns in which a heat removal path to the main condenser is not available. SGAHRS will have sufficient capability to remove heat delivered by the HTS to prevent exceeding temperature limitations for the core and the sodium coolant boundary.

b. SGAHRS shall be capable of performing its safety function following a normal reactor shutdown or following any anticipated, unlikely, or extremely unlikely events.

<u>Component Objectives</u>

<u>Auxiliary Feedwater Pumps (AFP)</u>

The objective of the pump(s) is to supply auxiliary feedwater from a protected source or from an alternate source to the steam drum(s) in the event of the loss of the main heat sink and/or the normal feedwater. The pump operation is the same for all anticipated, unlikely, or extremely unlikely plant conditions which require activation of the SGAHRS.

There are three pumps. Two of these pumps are driven by electrical motors, powered by normal or emergency plant AC power. The third pump is
17 driven by a turbine, which in turn is driven by steam from the steam drums. The turbine-driven pump is full size; it can supply the required flow rate to all three loops. Each motor-driven pump is half size, such that the
26 combination can supply the required flow to all three loops.

Protected Water Storage Tank (PWST)

The PWST will provide sufficient water so that the heat removal function can be accomplished by venting of steam until such time as the heat load is reduced to a level which permits using the Protected Air Cooled Condensers (PACC's) as the heat sink.

Sufficient water is provided to accomplish the venting, plus an additional amount to assure that valve leakage and losses due to postulated pipe failures do not result in premature depletion of the protected water supply. Postulated pipe failures, including full ruptures, can be accommodated with operator initiated isolation of the break. Valve leakage makeup for 30 days can be accommodated in addition to recirculation pump seal leakage for a 24-hour
17 period.

Protected Air-Cooled Condenser (PACC)

PACC will provide the long term heat removal capability to remove reactor decay heat when the SGAHRS is activated following a reactor shutdown. The capacity of the PACC on a single loop will be sufficient to assume the total decay heat load prior to the time that the PWST water is depleted below the level required to provide the 30-day emergency makeup water supply with heat removal being
17 accomplished via the PACC.

The PACC shall provide capability for removing reactor decay heat during a long term planned outage.

Design Parameters

The design pressure, temperature and other significant parameters for each of the SGAHRS components are listed in Table 5.6-1. The SGAHRS will be routinely tested during plant life and prior to installation as required to meet the intent of Section XI of the ASME Code. The SGAHRS is Seismic Category I and
17 will be operative following any anticipated, unlikely, or extremely unlikely plant condition. Seismic loading conditions are specified in Section 3.7.

5.6.1.1.2 Applicable Code Criteria and Cases

The SGAHRS pressure vessels and appurtenances, piping, pumps, valves and other components shall be constructed in accordance with Section III of the ASME Code
43 as indicated in Table 5.6-2. Code Cases 1606 and 1607 will be applied to Class 2 and 3 piping and vessels subjected to the SGAHRS operating conditions. Code Cases 1729 and 1739 will be applied to AFW pumps. Code Case 1797 will be applied to the finned tubing in the PACC. Effective dates of Codes and Code
43 17 Cases will be those that are in effect at the time the contract is let.

5.6.1.1.3 Surveillance Requirements

The only toughness degradation phenomenon which is a consideration is strain aging. Since the degradation is anticipated to be small in magnitude and localized in nature, and since post fabrication stress relief will be employed if necessary (see Section 5.6.1.3.10.2) fracture toughness surveillance specimens are not considered to be required for the SGAHRS.

5.6.1.1.4 Material Considerations

The following information pertains to materials which will be used in the SGAHRS.

High Temperature Design Criteria

Code Case 1481 will be applied to the superheater vent valves and the superheater vent valve piping since they have a design temperature above 800°F. The remainder of the SGAHRS components and piping have a design temperature below 800°F so no high temperature design criteria are necessary.

26

Material Specifications

A list of material specifications for the SGAHRS vessels, piping, pumps, and valves is given in Table 5.6-3. Corresponding weld materials specifications are listed in Table 5.6-4.

5.6.1.1.5 Leak Detection Requirements

Leak detection requirements for the SGAHRS are as follows:

a. Excessive leakage of high pressure and temperature water from the steam generator system into the SGAHRS will be detectable

b. Excessive leakage of low pressure water will be detectable

The methods used for leak detection are described in Section 5.6.1.2.5.

5.6.1.1.6 Instrumentation Requirements

Functional requirements of the SGAHRS instrumentation are to monitor the following parameters and to warn the plant operator of any abnormal or dangerous conditions in the following parameters:

1. Protected water storage tank level, pressure, and temperature
2. Auxiliary feedwater pump inlet pressure and temperature
3. Auxiliary feedwater pump discharge temperature and pressure
4. Auxiliary feedwater flow and temperature
5. Position of all isolation and control valves
6. Drive turbine steam supply and discharge pressure
7. Operating status of protected air cooled condenser
8. Operating status of all motors
9. Startup of air-cooled condenser
10. Startup of auxiliary feedwater pumps.
11. Open position or leakage of a SGAHRS vent control valve or a PACC non-condensible vent, using acoustic detectors in vent line.

The plant protection system instrumentation and control equipment associated with the active components which must operate to insure that SGAHRS performs its safety function are described in Section 7.4. The SGAHRS control set points are included in Table 7.4-2 "SGAHRS Nominal Set Points."

5.6.1.2 Design Description

5.6.1.2.1 Design Methods and Procedures

5.6.1.2.1.1 Identification of Active and Passive Components which Inhibit Leaks

The equipment of the SGAHRS is shown schematically in Figure 5.1-5. Valves and pumps within the SGAHRS are classified as active or inactive, and their operating mode is given in Tables 5.6-5 and 6.

In the event of a pipe break in the auxiliary feedwater portion of SGAHRS, continued heat removal capability will be assured by the multiple loop feature of the SGAHRS and heat transport system.

If a large pipe break occurs in any portion of a steam generator loop, this will result in a reactor shutdown and an AFW initiation. Automatic isolation of the AFW supply to the affected loop will occur within approximately 2 minutes when the steam drum pressure falls below 200 psig. Operator action as a backup is available.

If after an AFW initiating event, a pipe break were to occur in the auxiliary feedwater piping between the steam drum and the isolation valves immediately downstream of the control valves, the flow in the effective loop will increase until limited by the control valve (at approximately 110% of rated flow). A flow limit alarm in the control room will alert the operator to the fact that corrective action is necessary. Following the control valve flow limit alarm, the operator verifies a leak from information provided by the following instrumentation:

a) Safety-related steam drum level and pressure indication are provided on each loop to assist in making a break determination. An inability to recover level or maintain pressure on any steam drum with a corresponding flow limiting alarm on AFW provides a break indication.

b) The Steam Generator Building (SGB) Flooding Protection Subsystem annunciates abnormal SGB temperature, humidity, and sump level in the control room to alert the operator to pipe breaks that could compromise SGAHRS operation (see Section 7.6.5).

c) The plant trip signal: high or low steam to main feedwater flow ratio or low steam drum level. A trip of this type will direct the operator's attention to the steam/water-side of the plant.

Operator action in the control room will close two AFW supply isolation valves to isolate the defective loop. In addition to the above, automatic isolation will occur when the AFW flow remains above 150% for 5 sec. (indicating a flow limiter failure). Due to the flow limiting capability of the control valves, the leakage flow will be minimized and proper flow to the two remaining steam drums will continue even though one loop has suffered a pipe break.

If a leak were to occur in the AFW system upstream of the AFW isolation valves during normal plant operation (AFW in standby), it would be detected by a change in indicated water level in the PWST.

The PWST is sized to retain sufficient water for at least 30 days of SGAHRS operation while allowing for 10 minutes to isolate the AFWS from a large break. A large break creates restricted flow in both AFW control valves to the affected loop (154 lb/sec combined flow). Analysis of SGAHRS water use, following a pipe break, determines the required usable water volume to complete a 30-day mission (see Table 5.6-9). This evaluation uses the specified operator action time of 10 minutes which shows that water remains in the PWST after 30 days. The water volume remaining in the initially full PWST and steam drums after normal 30-day SGAHRS operation is actually sufficient to allow 22 minutes prior to loop isolation by operator action following a large break. Smaller break sizes with less than control valve limited flow would allow longer operator action times.

PWST water supply for at least 30 days is assured with an operator design response time of 10 minutes to isolate the loop with a pipe break. Operator action within 10 minutes is facilitated by control room information noted above and the control room operation to isolate the affected loop. Exceeding 22 minutes for operator initiated isolation may reduce the PWST water supply to the SGAHRS to less than 30 days. Prior to depletion of the PWST supply, an alternate water supply is likely to be available.

The piping runs from the AFWP to the AFW isolation valves and from the turbine drive steam supply isolation valve to the turbine drive are low-pressure and low-temperature lines during normal plant operation. Both lines are subjected to high-pressure during AFW operating periods and the turbine supply line is also subjected to high-temperature conditions during the time the turbine is operating. However, the SGAHRS operating time is anticipated to be less than 2% of the plant operating time since the auxiliary feedwater portion of SGAHRS will not be utilized unless either the normal heat rejection system (main condenser) or feedwater supply system has been lost. Therefore, no piping breaks will be postulated for these piping runs.

A pipe break between the pump suction isolation valve and the protected water storage tank is not considered credible because of the low temperature (200^{o}F) and the low pressure (15 psia) operating conditions. If such an event were to occur, the alternate water supply could be brought into service by the operator. The alternate supply is provided by a separate header connecting the feedwater pumps to the 250,000 gal. condensate storage tank.

5.6.1.2.1.2 <u>Design of Active Pumps and Valves</u>

In order to assure the functional performance of active components of the SGAHRS, the active ASME Section III Class 2 or 3 pumps and valves will be designed and tested in accordance with Reference 12, PSAR Section 1.6.

5.6.1.2.1.3 Surveillance and In-service Inspection

The SGAHRS system will be inspected in accordance with the intent of
17 58 Section XI Division 1 of the ASME Code.

5.6.1.2.1.4 Protection Against Accelerated Corrosion and Material Degradation

In water and/or steam, carbon steels are susceptible to pitting in the presence of chloride and oxygen. Furthermore, below 550°F, these materials are susceptible to caustic gouging and, perhaps, caustic stress corrosion cracking. Maintaining the water purity consistent with the requirements for chlorides, caustics and oxygen for short term operation will prevent these forms of localized attack.

Carbon steel is also susceptible to hydrogen embrittlement under SGAHRS operating conditions. However, maintaining the specified water purity will prevent this occurrence. Administrative procedures will be established to assure that water purity will be maintained.

5.6.1.2.1.5 Material Inspection Program

The SGAHRS material inspection program will be based on the require-
58 **ments of the ASME Code, Section III, for carbon steel and 2¼ Cr 1mo. steel.**

5.6.1.2.2 Material Properties

The materials used in the SGAHRS are described and discussed in
58 Section 5.6.1.1.4.

5.6.1.2.3 Component Descriptions

The major SGAHRS components have been designed with sufficient margin to assure that they will provide adequate cooling after a plant shutdown from power operation up to 115% of rated power. The decay heat levels shown in Figure
5.6-6 were used for component sizing and system response calculations for SGAHRS. 25

5.6.1.2.3.1 Protected Air Cooled Condensers (PACC)

Component Description

The PACC is a tube-type steam condenser constructed of carbon steel. Heat is rejected to the atmosphere by condensing the saturated steam from the steam drums by forced circulation of air over the tube bundles.

Each unit is sized to reject 15 MWt under conditions of forced convection on the air side and natural circulation flow on the steam/water side. Each PACC has two half-size tube bundles, two variable blade pitch fans and two sets of variable position louvers to control airflow and, therefore, heat rejection. The electrical power supplies and instrument and control circuits for the PACCs are Class 1E. Refer to PSAR Section 7.4 for information on the power sources and I&C.

The arrangement of PACC is illustrated in Figures 5.6-8 and 5.6-9. Air is delivered from axial fans (one for each tube bundle) into the insulated plenum surrounding each tube bundle. Air flows circumferentially around the tube bundle, then radially inward through the fin tube bundle into a central core. Air then flows upward through the central core and exhausts through louvers to an exhaust stack.

Each tube bundle consists of 50 finned tubes connected in parallel between vertical pipe headers. Each tube is approximately 100 ft. long and, of the 100 ft. length, 95 ft. is finned. The individual finned tubes are formed in a conical spiral of approximately four concentric turns with a slope toward the center. The tubes are connected in parallel between vertical pipe headers. The inlet header is on the outside and outlet header is in the center of spiraled coils. The finned tubes are made of 2 inch O.D. tubes with 0.156 inch minimum wall as shown on Figure 5.6-10. The O.D. of the fin is 3.28 inches. The fins are serrated into 0.156 inch segments from continuous strip 0.050 inch thick x 0.75 inch wide. The strip is first formed into the shape of an "L". The strip is then wound around the tube O.D. to complete the footed fin attachment to the tube. There are two separate tube bundles in each PACC.

Design Data

Design Conditions:

Pressure	2200 psig
Temperature	650oF

Thermal Hydraulic Performance:

Heat Removal	15 MWt (7.5 MWt per tube bundle)
Steam Pressure	1450 psig
Steam Temperature	592oF
Moisture	0%
Condensate Temperature	592oF
Air Temperature	100oF
Air Pressure	14.3 psia

Design Criteria

The power supplies to the PACC fans, instrumentation and controls are Class IE. The Instrumentation and Control System is a safety related system and as such will meet the requirements of the regulatory guides and standards as listed in Tables 7.1-2 and 7.1-3 of the PSAR. The means of compliance are described in Section 7.1.2.

Three PACC units are provided, one for each heat transport loop, each capable of removing the total decay heat approximately 1 hour after shutdown. Each unit is single active failure proof in that no single active failure will result in the loss of more than 50% of heat removal capability. This is provided by utilizing two tube bundles, two fans, etc., such that at least half capacity is retained following the failure. The PACC unit is a Seismic Category I design, hardened against tornado missiles and designed to withstand the pressure loads from tornados. The PACC tube bundle design is based upon standard techniques for steam-to-air heat exchangers.

Operation and Control

The airflow is regulated by the use of variable position inlet louvers and fans with variable blade pitch. There are separate controls for the air side of each PACC for each of the two fans and for each of the two sets of louvers. The inlet louvers and fan blade pitch are positioned by controllers which compare steam drum pressure to the setpoint and generate position demand signals to the louvers and fan blade pitch drives as required to maintain pressure at the setpoint value. In order for the PACC to effect heat rejection control over the range of operation there are two modes of air side operation:

(1) Forced convection with the louvers open and airflow varied by changing the fan blade pitch.

(2) Natural circulation with airflow varied by changing the position of the inlet louvers.

The range of automatic operation is from 15% to 100% heat rejection. From 100% down to approximately 30% (4.5 MWt) the unit is operating in the first mode, and from 30% to 15% in the second mode. Control is accomplished by sensing and maintaining the steam pressure at the desired set point.

5.6.1.2.3.2 Auxiliary Feedwater Pumps (AFWP)

The AFWP will be a multi-stage, centrifugal pump selected from a commercial vendor's equipment line. No special requirements should be necessary since these pumps have been proven to be reliable in commercial applications. The turbine driven pump will be sized to deliver a 1432 GPM flow rate at 3927 feet developed head, and the two motor driven pumps will be sized to deliver one-half of this flow rate each at the same head. The predicted constant speed head/flow curves for the turbine driven and motor driven AFW pumps are shown on figures 5.6-11 and 5.6-12 respectively.

AFWP Motor Drives

These motor drives will be synchronous speed squirrel cage induction motors of 980 horsepower. These motors will be selected from a vendor's standard line and no special requirements are anticipated.

AFWP Turbine Drive

This component will be obtained from an experienced vendor and will be sized to produce 1960 horsepower. The turbine will be constructed with sufficient quality assurance coverage to assure its reliability during service.

The auxiliary feedpump turbine is not kept hot for quick start operation. The drive turbine concept selected for the Auxiliary Feed Pump is based on the capability of this turbine to withstand severe service conditions. This is accomplished by constructing the turbine wheel from a single forging with buckets milled into the forging. The start-up procedure is similar to that for the RCIC turbine in a BWR in that it will occur without pre-warming.

Pump Integrity

The auxiliary feed pumps will be designed to the requirements of ASME B&PV Code, Section III, Class 3. In addition, the pumps and their supports will be designed to Seismic Category 1 requirements. Allowable stress limits are specified in Table 3.9-3 and pressure limits are specified in Table 3.9-4.

5.6.1.2.3.3 Protected Water Storage Tank (PWST)

The PWST holds the protected water to be supplied to the steam drums in the event of loss of normal feedwater or normal heat sink. The size is determined by detailed analysis of the heat removal conditions during the first several hours after shutdown and by anticipated component leakage rates. The tank will be constructed to the requirements for an ASME Section III/Class 2 vessel and it will operate at low temperature (<200°F) and low pressure (<15 psig).

5.6.1.2.3.4 SGAHRS Piping and Support

The SGAHRS piping is described below and is shown in Figure 5.1-5. The SGAHRS piping will be designed in accordance with the ASME Code Section III as specified in Section 5.6.1.1.2. The material
49 specifications are discussed in Section 5.6.1.1.4.

The SGAHRS piping runs can be categorized as follows:

a. <u>PWST Fill Line</u>

This 3 inch low pressure, low temperature, Class 3 carbon steel line runs from the 10 inch alternate water supply line through the motor-driven, normally closed PWST fill valve to the PWST inlet.

b. <u>Protected Water Storage Tank (PWST) to Auxiliary Feedwater Pump (AFP) Inlet</u>

There are three low pressure, low temperature, uninsulated carbon steel lines from the PWST to the three auxiliary feedwater pump inlets. Two of the lines, each of which leads to a half size, motor-driven pump are 6 inches in diameter and the third line to the full size turbine-driven pump is 8 inches. All three lines contain a manually
58 operated, locked open valve and an electrically operated, normally open isolation valve. These lines are Class 2
49 from the PWST to the electrically-operated isolation valve and then Class 3 to the pump inlet.

c. <u>Alternate Supply Line to AFWP Inlet</u>

The alternate supply line provides the capability for the AFW pumps to take suction from the condensate storage tank. A 10 inch carbon steel line runs from the feedwater and condensate
58 system junction to the first branch line. An 8 inch branch line
49 passes through an electrically-operated, normally closed isolation valve and tees into the 8 inch turbine pump inlet piping. Two 6-inch branch lines each pass through electrically-
49 operated, normally closed isolation valves and then tee into the 6 inch motor-driven pump inlet piping. The total run of piping is Class 3.

d. <u>Auxiliary Feedwater Pump Discharge to Discharge Header (Inclusive)</u>

The 6 inch carbon steel turbine pump discharge line leads to a 6 inch discharge header. This header in turn has three discharge points, one to each steam drum feedwater supply loop. a 6 inch carbon steel line from each motor driven pump feeds into a 6 inch header which also has three discharge points, one
43 17 to each drum.

Amend. 58
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The pump discharge lines contain check valves to prevent back flow through inoperable pumps. The motor driven pump discharge lines also contain a manually operated, locked open isolation valve downstream of the check valve. All three Class 3 discharge lines also have a 2 inch pump recirculation line containing an electrically-operated, normally closed isolation valve, branching off and running back to the PWST.

e. Auxiliary Feedwater Supply Lines

The six auxiliary feedwater supply lines from both the turbine and motor driven pump discharge headers are 4 inch diameter and contain (in order and in direction of flow) a manually operated, locked open isolation valve; a normally open electro-hydraulic control valve; a normally closed, electric operated isolation valve; and a manually operated, locked open isolation valve. After the final isolation valve, the turbine and motor driven pump supply lines are joined. The resulting 4 inch carbon steel line, which contains two check valves and a manual isolation valve, is then routed to the steam drum.

Routing of the auxiliary feedwater supply lines is such that high pressure lines (high pressuring during normal plant operation) are not located in cells containing the PWST, auxiliary feedwater pumps or other SGAHRS equipment whose failure could cause a loss of SGAHRS safety function.

f. AFW Pump Test Loop

Downstream of the tee where the motor-driven and turbine-driven pump supply lines join at the loop #1 valve station, an AFW pump test line returns flow to the protected water storage tank during periodic testing. This line contains redundant automatic valves for isolating the AFW supply from the PWST should SGAHRS be initiated during testing.

g. Steam Supply Line From Steam Drum to AFWP Drive Turbine

There are three 4 inch steam supply lines, one from each steam drum. Each of these lines contains a locked open, manual isolation valve, an electrohydraulicly operated, normally closed isolation valve, a check valve, and another locked open, manual isolation valve. Downstream of the final isolation valves, the three lines are headered together. The resulting 4 inch line then passes through a normally closed, electrohydraulicly operated pressure control valve before entering the drive turbine.

Routing of the turbine steam supply lines is such that they do not pass through the PWST cell. When the turbine lines pass through adjacent cells, protection is provided from missiles and jet impingement.

h. Steam Drum to Protected Air Cooled Condenser (PACC)

This is a high temperature, high pressure insulated 8 inch diameter carbon steel line. There are three parallel lines, one to each of the PACCs, which are separated by the Steam Generator Building Containment walls. Each line, which supplies steam from the steam drum to the PACC, has two locked open, manually operated isolation valves. Before entering the PACC, each 8 inch line tees into the 6 inch lines, each of which leads to one of the PACC's two half size tube bundles. During normal plant operation, these lines remain hot due to the PACC heat losses and natural circulation flow.

i. Protected Air Cooled Condenser to Steam Drum Recirculation Lines

Condensate from each of the half size PACC tube bundles will be piped in a separate 8-inch insulated line down to an elevation 3 feet below normal water level in the steam drum (See Figure 5.6-7). These separate lines assure that each half size PACC bundle is isolated from the other by a water seal. The isolation allows one half-size PACC bundle to be started and operated independently of the other. At an elevation 3 feet below the normal water level the 8-inch half PACC returns join to a single 6-inch line which continues down to the recirculation header 19 feet below the normal water level. This common condensate return line contains two locked open manual isolation valves and a venturi flowmeter. Above the water seal elevation, condensate flow will be a vertical annular or stratified two phase gravity flow pattern. A large line size (8-inches) is used to assure the two phase gravity flow remains stable and does not result in entrainment over the PACC operating range. (See Section 5.6.1.3.2.3) The lines from each PACC to its steam drum are separated from the lines for other PACCs by the Steam Generator Building walls.

j. Steam Drum and Superheater Steam Vent Lines

These two lines, one branching from the steam drum to superheater piping and the other branching from the superheater to main turbine line, contain a locked open, manual isolation valve and a normally closed electro-hydraulic operated pressure control valve. Both lines are used to vent steam from the system to release heat from the plant and maintain the steam drum at a pressure below the design head of the auxiliary feedwater pumps. The superheater vent valve and vent line are made of 1 1/4 CR - 1/2 Mo steel; the steam drum vent valve and vent line are carbon steel. Following the plant trip and the initial pressure reducing transient, these valves will normally be used as the only means for venting steam during SGAHRS operation. Power relief valves located at the superheater outlet will serve as a backup should both the SGAHRS superheater and steam drum vent valves be unavailable. These steam generator system valves will be set to open at a higher pressure. The advantage of separate SGAHRS vent valves is a controlled steam drum pressure by venting through valves designed for low erosion rather than the on/off operation of the safety valves.

The design of the piping must be such that a pipe failure does not result in failure of parallel and redundant systems. This requires use of pipe restraints to limit pipe whip, shields to protect critical piping from missile damage, and design of the system so that in the event of a pipe break, the redundant systems are not damaged.

Major SGAHRS components such as the pumps and pump drives will be anchored and the piping designed to provide the required flexibility. The piping stress analysis will include stresses from thermal expansion, seismic loads, dead weight loads and thermal transients.

The SGAHRS piping will be supported from the building structure with support hangers and restrained with seismic snubbers.

All piping within the Steam Generator Auxiliary Heat Removal
17 System smaller than 1 inch will be field run.

5.6.1.2.3.5 <u>Valves</u>

Special or unusual requirements will not be necessary for any of the SGAHRS valves. Valves that can meet and exceed all design specifications for the SGAHRS are commercially available. Therefore, it is not anticipated that any special data on valve performance will be
58 required. All check valves are testable with external position indicators.

5.6.1.2.4 Overpressurization Protection

The high pressure portions of the SGAHRS such as the PACC, and the
piping and equipment downstream of the auxiliary feedwater pumps have design
pressures exceeding the steam generator safety relief valve set point and the
17 shut-off head of the auxiliary feed pumps, respectively. The auxiliary feed
pump drive turbine is designed to operate with a throttle pressure of 1000
psig and the pressure on its casing down stream of the admission valves must
not exceed its structural design pressure of 1250 psig. To prevent overpres-
sures during startup, shutdown, and standby periods, the drive turbine inlet
including the admission valves are designed to withstand the highest possible
17 steam drum pressure consistent with the steam drum safety valve set points.

5.6.1.2.5 SGAHRS Leakage Detection Methods

Leakage of steam generator water into the low pressure portion of the
SGAHRS will be unlikely due to the fact that leakage will have to backflow
through two check valves, a closed isolation valve, and a third check valve.
However, if this does occur, the leakage will be detected by temperature sensors
located in the auxiliary feedwater supply lines to the steam drums and a high tem-
58 perature alarm will be initiated. Leak detection from the low pressure portion of the
SGAHRS is accomplished by monitoring the water level in the PWST. A low water
17 level alarm is sounded when the level reaches a minimum acceptable level
corresponding to the maximum water volume required for shutdown heat removal.

5.6.1.3 Design Evaluation

5.6.1.3.1 Analytical Methods and Data

17 The thermal and hydraulic characteristics of the SGAHRS will be
analyzed using standard engineering data and references. Since the SGAHRS
fluid is water, this presents no uncommon analysis problems. However, the
performance analysis of the protected air cooled condenser, and the auxiliary
feedwater pump turbine drive will require data input from the vendors.

This data will then be used in the system analysis after suitable review and verification.

5.6.1.3.1.1 Compliance with Code Requirements

The classification for the various SGAHRS components that are to be constructed in accordance with Section III of the ASME Code are listed in Table 5.6-2. The design of these components, as described in Section 3.9, will be in conformance with the intent of Regulatory Guide 1.48.

5.6.1.3.1.2 Category I Seismic Design

The entire SGAHRS will be designed to Seismic Category I requirements since it is required for plant heat removal following seismic events. Figure 5.1-5 shows the limits of Category I design. The steam generator building provides the interface between Category I and non Category I piping for the line from the condensate storage tank to the Protected Water Storage Tank. All components of SGAHRS shown on Figure 5.1-5 are Category I. 14

17 5.6.1.3.1.3 Analytical Methods for Pumps, Turbine, and Heat Exchangers

Time independent, elastic analysis will be used, where required, on the SGAHRS pumps, turbine and heat exchangers (PACCs).

58 5.6.1.3.1.4 Operation of Active Valves Under Transient Loadings

The qualification test program used to verify that active valves within the SGAHRS will operate under the transient loadings experienced during service life is discussed in Reference 12, PSAR Section 1.6.

17

5.6.1.3.1.5 Analytical Method for Component Supports (Vessels, Piping, Pumps, and Valves)

In accordance with the ASME Code, component supports will have the same code classification as the components they support. Design of each component support will comply with the ASME Section III design rules corresponding to the component support classification. In order to provide assurance that the component support stresses comply with limits specified in Section 5.6.1.1, analysis of each component support will be performed. The applicable analytical techniques and applicable computer codes discussed in Section 5.3.3.1.5 will also apply to detailed analysis of support components. The classification of components within the SGAHRS is included in Section 5.6.1.1.1.4. Allowable stress limits and pressure limits are specified in Tables 3.9-3 and 3.9-4.

5.6.1.3.2 Thermal Hydraulic Design Analysis

5.6.1.3.2.1 Natural Circulation

The Protected Air Cooled Condensers (PACC) operate on natural circulation on the steam and water side. In the event of loss of all AC power, a two hour blackout, the PACC units will operate at reduced power levels with natural circulation on the air side. The relative elevations are shown in Figure 5.1-6.

Since the relative densities of water and steam are 10:1, there will be no difficulty in ensuring steam supply to the PACC. The condenser design will permit adequate circulation within the condenser tubing. The PACC design will be verified by analyses and by proof testing after installation.

The PACC closed loop schematic is shown on Figure 5.6-7. The steam/water side natural circulation is comprised of two parts as follows:

(1) Steam flow from the steam drum superheater supply piping, through the steam inlet piping, into the tube bundle.

(2) Condensate flow from the tube bundle through the condensate return piping, to the recirculation pump header located below the steam drum.

The tube bundles during normal plant operation are filled with saturated steam at steam drum conditions and kept on hot standby (i.e., isolation from ambient by air side isolation louvers). Assuming 3% heat loss through the insulated isolation louvers (design goal) during standby, condensate is formed at the rate of 2974 lbm/hr. The condensate outflow from the tube bundle during its period is due to gravity.

Upon SGAHRS initiation signal, the isolation louvers are opened, the fan is turned on, and steam condensation increases. Condensation causes a volume collapse inside the finned tube bundle. This volume collapse causes the bundle pressure to drop below the steam drum pressure as makeup flow from the drum is established. The return piping connected to the recirculation pump header is supplied with water from the steam drum. Because this line contains relatively high density water (43.2 lbm/ft^3 for water as compared to 3.36 lbm/ft^3 for steam) the low pressure in the bundle causes the liquid level in the return piping to rise above the steam drum liquid level while steam flows into the tube bundle through the supply line. The units are designed to condense 89,000 lbm/hr of saturated steam from the steam drum. The combined pressure drops associated with flow of steam through the inlet piping, steam/water mixture through the tube bundle, and water through the return piping is calculated to be 4 psi. This causes the liquid level in the condensate return pipe to rise 16 ft. above the steam drum liquid level. This height is 11 ft. below the low point of the tube bundle (i.e., the tube bundle exit header nozzle). This 11 ft. margin is enough that the tube bundle pressure drop could be as high as 4.6 psi without drawing water into the tube bundle. The tube bundle pressure drop is not expected to be more than the 2 psi allowed by the PACC Equipment Specification.

The condensate outflow from the tube bundle is caused by two factors as follows:

(1) Shear forces resulting from flow of steam over the condensate formed in the tubes. These forces are directly proportional to the velocity differential between the steam and the condensate as predicted by the relation:

$$\tau = \mu\left(\frac{\partial u}{\partial y}\right) y = \delta$$

where:

τ = The shearing stress at steam/condensate interface
μ = Steam viscosity
u = Steam velocity
δ = Location of the steam/condensate interface

(2) Gravitational forces causing the condensate to flow to the low point of the tube bundle.

The tube bundle length may be divided in three parts. The condensate flow through the first region is primarily due to shear forces as described above. In the second region the steam velocity is greatly reduced and both gravitational and shear forces cause condensate to flow towards the tube bundle exit header. The governing forces in the third region are gravitational, shear, and pressure gradient induced. These forces cause the condensate to flow into the tube bundle exit header where it is returned to the recirculation header. The steam inlet nozzle location (high point of the tube bundle) with respect to the condensate return nozzle (low point of the tube bundle) also serves to insure flow of all condensate steam towards the condensate return pipe.

Each PACC air side will have forced circulation capability to remove 15 MWt of heat. The PACC units operated with natural circulation on the air side will remove an estimated 30% of the PACC rated heat load. The cooler inlet air (max. 100°F) will be drawn in at the cell base and rise through the 600°F condenser and exit out the cell roof through a stack.

5.6.1.3.2.2 Thermal Analysis of PACC

A thermal hydraulic analysis of the PACC has been performed. The steam side calculations represent a coiled, finned condenser tube from steam inlet to condensate outlet. The appropriate water/steam side heat transfer coefficient for each segment is determined for the flow regime for each segment based on water/steam properties and film temperature drop.

For a ratio of steam-to-water density $\geq$ 0.125, the Boyko-Kruzhilin correlation (Ref. 5.6-2) is used. For a ratio of steam-to-water density < 0.125, the heat transfer coefficient is selected based on Baker's flow regime transition data. (Ref. 5.6-3) When the flow regime in a tube segment is stratified, the modified Soliman, Schuster and Berenson correlation (Ref. 5.6-4) is used. When the flow regime is dispersed or slug, the Chato correlation (Ref. 5.6-5) is used.

The pressure drop and void fraction correlations used in determining the flow rate in the heat exchanger tubes are:

Reynold's Number	Void Fraction	Frictional Pressure Drop
Re < 2000	Lockhart-Martinelli (ref-5.6-6)	Lockhart-Martinelli (ref-5.6-6)
Re > 2000	Boroczy (ref 5.6-7)	Boroczy (ref 5.6-8)

The analysis considers a fouling resistance of 0.0005 (Hr-Ft^2-°F)/Btu on the steam side after 30 years of service. This value is consistent with thermal standards of the Tubular Exchanger Manufacturers Association (TEMA).

For the air-side, the convective heat transfer coefficient is computed using the method outlined in the ESCOA Fintube Engineering Manual (Ref. 5.6-1) as a function of the gas-side mass flow and the tube, fin and coil geometry. Several air-side heat transfer coefficients were examined. The correlation in the ESCOA manual most resembles the conditions in the PACC design.

The ESCOA air-side heat transfer coefficient is based on the total overall heat transfer area between the steam and air. However, for PACC an individual air-side heat transfer coefficient is used for each increment of the model. Local air-side thermal and physical properties are used based on the local air and steam temperatures. Furthermore, row dependent air side heat transfer coefficients are calculated separately for each of the four turns by employing an averaging method since each turn of the coil has a different air flow area. All these are taken into account to derive the appropriate local air-side heat transfer coefficient for each increment.

An air-side fouling resistance of 0.001 (Hr-ft^2-°F)/Btu is applied in the PACC thermal analysis. This value is consistent with recommendations of the ESCOA Fin Tube Engineering Manual (Ref. 5.6-1).

Using the above described correlations and assuming 10% plugged tubes, the nominal PACC capacity has been determined to be 29.07 x 10^6 Btu/hr/tube bundle or 8.53 MW (17.06 MW/PACC).

The approach to conservatively determine PACC thermal performance included a sensitivity study where reasonable uncertainties were assumed in turn for each critical parameter. The parameters and uncertainty ranges evaluated and their effect on heat transfer are given in Table 5.6-14.

The most critical source of uncertainty, the air side heat transfer coefficient, was conservatively assumed to have a 25% uncertainty. This was based on ESCOA data (Ref. 5.6-1) which indicates a maximum uncertainty of 20%. This uncertainty in air side heat transfer coefficient could result in a 12.7% reduction in PACC capacity. Combining all the uncertainties given in Table 5.6-14 using the method described by Kline and McClintock (Ref. 5.6-9) results in a minimum PACC heat rejection rate of 15 MW.

5.6.1.3.2.3 Thermal Hydraulic Stability

Thermal hydraulic stability in the PACC condensate return line and in the PACC heat exchanger bundle will be investigated analytically and experimentally (during component test) to assure there will be no significant impact on the PACC performance. The large condensate return lines will be designed to assure stability of the two phase gravity flow.

Flow instability is sometimes associated with parallel condensing tube bundles. Features to ensure steam/water flow stability have been successfully applied to eliminate flow instabilities in gravity drains of moisture separator reheater tube bundles in steam turbine cycles of LWR plants. This will be addressed in the analysis.

In the PACC unit, shell-side air or tube-side steam maldistribution would cause similar effects. Analytical and experimental development programs are currently underway to develop circumferential and axial flow distribution features (e.g., bundle air-side inlet and outlet perforated plate distribution screens) to provide sufficiently uniform air flow distribution to assure that the outlet conditions of all tubes remain uniformly subcooled. A three-dimensional flow analysis will be used to define the design of the required air flow distributing features. The design of these air flow distribution features will be refined and verified by tests using a one-fifth scale isothermal air flow model. Final confirmation of acceptable air flow distribution and resulting uniformity of condensate subcooling will be obtained during the PACC lead unit test. This will preclude unstable flow or cyclical cooling modes which have been observed under variable heat load conditions in similar heat exchangers. Inlet orifices with four size variations to overcome the effect of manifold pressure variation are provided to preclude tube side flow maldistribution.

Individual tube oscillations may occur, even in the absence of system oscillations, since this is a function of the nature of individual tube flow. For PACC, oscillatory flow, corresponding to slug or plug flow, will be predicted over part of the condensation path. These flow regimes are expected to impart some individual tube oscillations. The design bases of the PACC tube to outlet header connection include a conservatively postulated cyclical subcooling for which ASME code fatigue requirements will be met.

It is expected that the PACC air flow design and tube orificing capability will obviate system instability across the heat exchanger bundle. Experience with reheater tube bundles suggests the individual tube oscillations will be acceptable. Specifically, this will be less severe than conservatively assumed cycling to be used in PACC fatigue design analysis.

A full-scale PACC test conducted prior to FSAR submittal with low pressure (165-250 psia) steam provides a means for simulating PACC operation at CRBRP refueling conditions. This condition is included as part of the PACC duty cycle which occurs during plant startup from and plant shutdown to refueling temperature of 400°F. This full scale PACC test, to be performed on the lead unit, will address thermal performance and stability.

5.6.1.3.3 <u>Pump Characteristics</u>

The feed pump head and torque as a function of feed pump flow rate will be determined as the design progresses.

5.6.1.3.4 <u>Valve Characteristics</u>

The flow coefficients of all valves and the closing times of all isolation valves will be determined as the design progresses.

5.6.1.3.5 <u>Pipe Leaks</u>

Pipe leaks can be categorized under identified leakage and unidentified leakage. Identified leakage is leakage into closed systems, such as from pump seals or valve packing leaks where the leakage is captured and directed to a sump or collection tank. These leaks occur in system components where it is not practical to make the components 100% leaktight. The existence of identified leakage is known in advance and is provided for in the system design.

17 Unidentified leakage is all other leakage. The existence of unidentified leaks is not anticipated; however, the occurrence of such leaks could reduce the SGAHRS efficiency or jeopardize the safety of reactor plant operations. As a result, pipe leaks will be postulated in each piping run of the SGAHRS and investigated to identify those areas where unexpected loading and possible damage may result. Pipe leaks will be postulated in a way similar to that used for postulating pipe breaks (see Section 3.6), except that leaks will be postulated for all pipe runs or branch runs. In pipe leak investigations, the leak size and the poten-
17 tial for propagation will be considered.

Section 3.6.5 contains a discussion of methods to be used to postulate the effects of leaks on nearby equipment and the measures to be used to prevent propagation of damage.

Leak detection capabilities for the SGAHRS are discussed in Section 5.6.1.2.5.

5.6.1.3.6 Evaluation of Steam Generator Leaks

In the event of a steam generator leak, there will be no effects on the SGAHRS components. The SGAHRS is connected to the steam drum which is part of the water side of the steam generators, and there are no pressure increases in the water side in the event of a leak.

5.6.1.3.7 Inadvertent Operation of Valves

All isolation valves in the main water flow path of the SGAHRS, except one in each steam drum auxiliary feedwater supply line, are intended to be in the open position during normal plant operation so that the system functions with a minimum of valve operation.

Inadvertent closing of any one valve downstream of the feedwater pumps will not affect the operation of the SGAHRS feedwater supply capabilities due to the redundant system arrangement (i.e., redundant feedwater supply lines to each drum). Furthermore, if the pump suction isolation valve closes, the water supply will not be affected because of the availability of redundant auxiliary pumps. The alternate water supply is also available following manual switching. If a manual isolation valve in the PACC lines is inadvertently closed, the initiation of the PACC in that loop will be delayed until the valve is opened. However, the two remaining loops will continue to reject the necessary plant heat load via both steam venting and PACC operation. The affected loop will be limited to
17 steam venting.

Maintenance Valves

All maintenance valves will be locked open during normal plant operation. The effect of inadvertently closing a maintenance valve will depend on its location in the system, as described above. Administrative procedures will be provided to preclude this occurrence.

Auxiliary Feedwater Supply Isolation Valve

The electrically-operated valve in each of the six steam drum auxiliary feedwater supply lines is the only valve in the main flow paths of the SGAHRS that is normally closed during plant operation. Inadvertent opening of this valve will not affect the operability of the SGAHRS or of the plant because redundant check valves will prevent reverse flow.

Control Valve

The electro-hydraulic operated control valve in each of the six steam drum auxiliary feedwater supply lines will fail "as is" upon loss of electrical power to the valve, thereby allowing feedwater supply to continue. If a failure occurs in the control signal to the valve, and the valve opens to greater flow than required, the steam drum level in the affected loop could possibly rise to the trip level. Closure of the motor driven and turbine driven auxiliary feedwater supply system isolation valves for the affected loop will be initiated by independent logic trains when the steam drum level rises to 8 and 12 inches above normal, respectively. Auxiliary feedwater supply to that steam drum would be terminated until the level fell below the drum level setpoint. The isolation valve would then open and flow would be supplied until reaching the trip point. The sequence would then be repeated. Heat removal capabilities would be available through all loops.

Drive Turbine Steam Supply Isolation Valve

The electrohydraulicly operated normally closed isolation valve in each of the three steam lines to the feedwater pump drive turbine is in series with the normally closed pressure control valve. Inadvertent opening of any of these isolation valves will not have an adverse affect on the SGAHRS system or the rest of the plant. Inadvertent closure of any of these isolation valves during SGAHRS operation will have no effect due to the redundant steam supplies from each drum.

Steam Pressure Control Valve

58 This electrically operated, normally closed valve is located between the steam line manifold and the drive turbine inlet. Inadvertent opening of the valve will not affect the SGAHRS long term operation or normal plant operation since the isolation valves upstream are normally closed. Inadvertent opening during SGAHRS short term operation will require the turbine throttling valve to operate over a wider range with a resultant loss in speed setting accuracy. AFW flow will not be affected. If, during SGAHRS operation, the control valve closes, the drive turbine, and therefore, part of the system redundancy will be lost, but plant heat removal can continue with the motor driven auxiliary feedwater pumps
17 supplying makeup feedwater.

Pressure Controlled Bypass Valves

The bypass valve on each auxiliary feedwater pump discharge is
normally open. If one of these valves inadvertently remained opened during
SGAHRS operation, the flow capability of the pump would be degraded by
58 approximately 25%. However, the SGAHRS would still be able to function
43 due to the pumping redundancy provided and the fact that full feedwater
flow is not needed except for the first few minutes after a trip from
43 stretched conditions (115% rated power) and the fact that the auxiliary
feedwater supply system is designed to provide sufficient flow with margin
58 even after a pipe break. A failure to remain open when needed could result
in overheating the pump. Pump redundancy protects the decay heat removal
17 function.

Alternate Water Supply Valves

These valves are normally closed. Inadvertent opening will not
affect SGAHRS unless the alternate water (unprotected) supply line has been
17 broken below the PWST level so that reverse flow can occur in this line. In-
advertent opening while SGAHRS is not operating will result in water from the
feedwater and condensate systems flowing out the PWST overflow. The system
58 operation will not be impacted. In the event that this valve is inadvertently
opened when the alternate water supply is not available, the feedpump should not
be affected unless the normal inlet valve from the PWST is closed in which case
the pump would lose suction. Protected water cannot normally be lost by this
path since the alternate supply lines rise well above the protected storage
17 tank level before leaving the steam generator building.

Pump Suction Isolation Valve

The isolation valves in the pump suction lines are normally open.
Inadvertent closing of one of these valves would isolate the affected feed
17 pump from the protected water supply causing the pump to lose suction.
Redundant pumps prevent loss of the heat removal function.

5.6.1.3.8 Performance of Pressure Relief Devices

No pressure relief devices are utilized on SGAHRS as discussed in
17 Section 5.6.1.2.4.

5.6.1.3.9 Operational Characteristics

The SGAHRS is required primarily for safety and is designed to remove
17 plant stored heat and reactor decay heat for off-normal plant conditions by
(1) supply of auxiliary feedwater (AFW) for steam venting through vent control
valves, and (2) closed-loop heat removal by the Protected Air Cooled Condenser
58 (PACC). In addition, the PACC will also be used for decay heat removal during
long-term outages following normal or off-normal plant shutdowns. For testing
as well as normal and off-normal plant shutdowns, the SGAHRS will be activated
a substantial number of times during the plant lifetime.

58

The SGAHRS will respond essentially in the same manner for all
off-normal events requiring SGAHRS. That is, whether SGAHRS is activated
because of feedwater loss or condenser loss, AFW will be supplied to the
drums and heat will be removed by venting steam and through operation of
58 the PACCs. The PACCs are, however, activated following all plant trips.

Analysis Methods

In order to size critical components of SGAHRS, such as AFW pumps, PACC's and the protected water storage tank (PWST), the transient heat load on SGAHRS must be known. Plant stored heat and reactor decay heat must be removed through SGAHRS initially over a period of several hours, and then decay heat must be removed through PACC's for an indefinite period. The DAHRS (Demo Auxiliary Heat Removal Simulation) computer code (Appendix A) is used to compute the heat load on SGAHRS after shutdown, while accounting for both sources of heat.

To provide an example of normal SGAHRS response to abnormal plant operation, DAHRS was used to evaluate the heat load following a loss of off-site power. Reactor decay heating and flows (sodium, main feedwater, and recirculation water) are inputs; system temperatures and AFW flow are computed values.

Analysis Input Assumptions

Initial conditions are assumed to be at 15% above rated power (the "stretch" condition), with system temperatures adjusted to the maximum possible within estimated instrument error. The higher initial temperatures are conservative for heat loading on SGAHRS. Sodium flows are assumed to be at their maximum reasonable values above rated, in order to assure conservative heat loads in
17 SGAHRS sizing calculations. Specific values are as follows: 8

Power: 1121 Mwt

Flows: Primary--4030 lb/sec/loop
Intermediate--3910 lb/sec/loop
Recirc H_2O--617 lb/sec/loop
Main FW--355 lb/sec/loop 8

Temperature: Pri Hot Leg--1015°F
Pri Cold Leg--725°F
Int. Hot Leg--960°F
Int Cold Leg--662°F
Main FW--465°F 8

Response to the initiating events was computed with the following assumptions:

- All three heat transport loops behave identically (i.e., no additional failures).
- Sodium flows coastdown to pony motor values (384 lb/sec/loop and 426 lb/sec/loop for primary and intermediate, respectively).
- Main FW goes to zero; recirculating water flow coasts down to 20% of initial.
- Venting at 1450 psia occurs at the superheater steam outlet
- AFW at 125°F is supplied to match vented steam mass.
- PACC heat removal begins 4 minutes after shutdown and is 15 MWt/loop after 1 additional minute. 8

Upon plant shutdown, steam is vented until the heat load on SGAHRS can be removed by the PACC alone. Thereafter, pressure is assumed to remain constant and PACC heat removal is assumed to be equal to the decreasing heat load to the steam generator. As the transient proceeds, the plant cool down will be controlled by adjusting the PACC airflow which will control the heat removal rate.

Reactor decay heat is taken as 125% of nominal. Stored heat in the metal throughout the plant includes an appropriate margin:

- 110% of nominal for piping and components,
- 125% of nominal for reactor vessel region.

Analysis Results

Results of DAHRS analysis of a loss-of-offsite-power event are presented in Figures 5.6-1, 5.6-2 and 5.6-3. These results will serve to explain SGAHRS operation in response to the event. Transient heat load on SGAHRS is plotted in Figure 5.6-1. Several features of this figure are noteworthy:

Heat load on SGAHRS peaks at about 0.07 hours after shutdown, when the PACC's are started. The peak at 0.15 hours is due to the positive temperature transient in the IHTS hot leg reaching the steam generators (see Figure 5.6-3).

Heat removed by the superheater is seen to be the difference between the two curves, peaking at the same time as heat load on the evaporators and then falling to zero as steam venting subsides. 8

17 Startup of the PACC 5 minutes after shutdown produces a dramatic reduction in heat load. This effect is due to reduced water needs for venting. The reduced water flow (at 125°F) to the drum (see Figure 5.6-2) allows the drum water temperature to rise toward saturation,resulting in rising temperatures at the water inlet and sodium outlet of the evaporators (see Figure 5.6-3).

17 Venting continues until heat load decreases to the level at which the PACC's can reject all incoming heat, at about 1 hour after shutdown.

During the venting period, the great majority of heat removed is plant stored heat (decay heat at 0.1 hour is about 3%, decreasing to below 2% at 1 hour).

Figure 5.6-2 displays cumulative heat removed and AFW performance. Note that AFW usage occurs early in the transient and total water vented amounts to about 3130 ft^3.

17

Figure 5.6-3 shows sample sodium temperature responses at several plant locations. The key feature to note is the substantial decrease in the cold leg temperature due to the addition of the 125°F AFW. This temperature decrease accentuates the heat load on SGAHRS during the venting process.

The temperatures presented in Figure 5.6-3 result from the conservative evaluation of feedwater requirements during SGAHRS operation. The analyses include piping and component stored heat to maximize the SGAHRS heat load. These temperature plots are not used for evaluation of thermal stresses on the systems and components. Thermal transient analyses are generally performed without inclusion of piping stored heat to provide conservatism. 8

Component Sizing Analyses

The sizing of the SGAHRS components was performed for a heat rating 15% above the rated operating condition. This increased heat rating results in a more conservative design for rated operation and does not reduce the safety of the system in any way.

The various components in the SGAHRS were sized using the results of the DAHRS code as input data. Each component is discussed separately in the following sections.

Protected Water Storage Tank (PWST)

In evaluating the required capability of the PWST, the water makeup requirements under the most severe "Safety Design Basis" events must be con- 31
sidered. The worst initiating events, with respect to the PWST, are the Safe Shutdown Earthquake or the Design Basis Tornado since they can destroy the normal heat sink, the alternate water supply (condensate storage tank) and 31
the off-site power sources. In addition, active failures such as the partial loss of a PACC will further increase the water venting requirement. As an added conservatism, the event selected includes a pipe rupture (on one loop)
58 17 in the normally pressurized region of Auxiliary Feedwater Subsystem. In order 31

to show sufficient conservatism to accommodate an extended period without AC power, a two hour loss of off-site and on-site AC power is assumed as a PWST size evaluation event.

The DAHRS model (See Appendix A) was used to determine the steam venting requirements, in reducing the residual heat loads to a level within the capability of the PACC units, for the various events. The inputs to the DAHRS model included those discussed above, except PACC startup was arbitrarily delayed for twenty minutes.

Note that the calculations performed by DAHRS will be conservative due to the uncertainty margins applied to plant sensible and decay heat loads as well as the conservatively high initial sodium and PWST temperatures.

Table 5.6-7 shows the results of the DAHRS computations. As can be seen from this table, the worst event, with respect to venting required, is the two hour loss of AC power sources. The loss of AC power event is based upon natural circulation in three heat transport loops during the two hour period. The PACC units are also assumed to be operating with natural circulation on the air side for this period. Following this interval, power is assumed to be restored and cooling can be resumed via forced circulation in the HTS loops and on the air side of the PACCs. As can be seen in Table 5.6-9, the total water requirements are slightly more for this event than for the three loop operation pipe break event. Under these conditions, 5100 ft.^{3} of water is vented before power is restored and the PACCs accept the total steam generator heat load.

In addition to the venting volume calculated by DAHRS, it is necessary to consider the sensible heat released as the Auxiliary Liquid Metal System cools down. For the primary and intermediate heat transport service systems combined, the sensible heat release will require an additional 250 ft^3 of make up water if the complete sodium volume is circulated during the venting period. Since it is the conservative approach, the full 250 ft^3 of additional vent flow will be needed for Auxiliary Liquid Metal System cool down.

Allowance must also be made for leakage from the various valves and pumps. As Tables 5.6-7 and 8 show, the PWST is sized to assure adequate feedwater makeup for a 30 day period. If the normal heat sink cannot be restarted during this period, the operator must draw water from the condensate storage tank or have water brought in from an off-site source. In computing the 30 day leakage, it has been assumed that the operator will stop the leakage from the recirculating pump seals, by shutting off the seal cooling water, within 24 hours after the initiating event. These leakage values and their 30 day sum are shown on Table 5.6-8.

Table 5.6-9 summarizes the total PWST volume requirement by summing the venting, leakage, and the heel requirements. As shown by Table 5.6-9, the resulting design basis PWST tank volume is 9591 ft^3. This includes a 22.9% margin on the volume requirement. Margin for the venting volume is built into the DAHRS model in the form of increasing the heat from 10 to 25%.

Auxiliary Feedwater Pumps (AFP)

These pumps are sized to deliver enough flow to the steam drums to prevent drum dryout under the conditions of a pipe break on a loop, venting from the superheater on a second loop, and venting from the steam drum on the third. Preliminary calculations indicate that the turbine driven AFP should be sized to deliver 1432 gpm. The two motor-driven AFPs are each sized to deliver one-half this flow rate.

Following a plant trip with SGAHRS initiation, the steam drum pressure is controlled to 1475 psig by the superheater vent control valve. If this normal means of venting is not available, the steam drum pressure is controlled to 1550 psig by the steam drum vent control valve. The AFP sizing is such that it can provide adequate auxiliary feedwater during venting from the steam drum vent control valve (1550 psig). This requirement has resulted in pumps capable of supplying 1432 GPM (199 lbm/sec) at 1700 psi discharge pressure (1700 psi is necessary to overcome friction and elevation losses while supplying auxiliary feedwater to the steam drum at 1550 psig).

Auxiliary Feedwater Pump Drives

The drives for the AFP's are sized for the AFP requirements. For the 1432 gpm capacity of the turbine driven AFP, a 2000 hp drive turbine is
17 58 required. The half size motor drives are rated at 1050 hp.

Protected Air Cooled Condensers (PACC)

These components are sized to be capable of rejecting the shutdown heat load within one hour and to be consistent with the size of the PWST. At these times, the total heat load to SGAHRS is less than the 15 MWt/loop capacity of the PACCs. The PACCs are designed for forced circulation on the air side when rejecting 15 MWt.

HTS Transients

The SGAHRS is designed to handle the maximum shutdown heat removal duties expected to occur following a trip from 115% of rated power. Conservative sodium temperatures and flows in addition to a conservative decay power are included in these sizing calculations to assure adequate heat removal capability from a 115% power plant trip. This conservatism is applied to sizing of the auxiliary feedwater pumps and piping. The PACC heat removal capability of 15 MWt is not influenced by these conservative factors but is a fixed value. While the auxiliary feedwater system contains margin in its flow capability it will not lead to overcooling of the Heat Transport System (HTS).

Some transient events which require SGAHRS to initiate the injection of the relatively cold auxiliary feedwater into the Steam Generator System (SGS) will result in a down-side transient on the Heat Transport System (HTS). Excess flow will not occur since FW addition is limited by the FW flow control system. However, the cold water addition will cause a slight down-side transient. This transient has been accounted for in the evaluation of the plant duty cycle events since those thermal transient events are included in the design duty cycle events for the HTS components. Therefore the SGAHRS startup down transients do not constitute overcooling.

In the time period following the initial thermal transients the SGAHRS control system will maintain system temperature within their acceptable ranges such that no excessive cooling will result.

If the main condenser and normal feedwater are available, short term heat removal will be accomplished via the turbine bypass system. SGAHRS will be involved since a plant trip activates the Protected
58 Air Cooled Condensers (PACC). The PACC is designed to adjust air flow and thus its heat removal rate in a manner which will hold the entire plant at the ~600°F temperature level until the setpoint is adjusted by the operator.

If the main condenser or normal feedwater systems are not available, the SGS pressure after the initial pressure increase transient will
43 be controlled to 1475 psig by the superheater outlet SGAHRS vent valve. Steam venting through the vent valve will be initiated and makeup feedwater will be supplied by SGAHRS. The flow from SGAHRS will be controlled by the steam drum levels. The plant will be cooled down to the saturation
43 temperature at the vent pressure (1475 psig) of $\sim 595^{\circ}$F. After the SG heat load drops below the capability of the PACC's, venting will cease as the
43 steam drum pressure begins to drop below the 1475 psig set point. Heat removal will continue as described for the normal shutdown.

The HTS has been designed to provide natural circulation flow capability during SGAHRS operation. The potential for natural circulation in the sodium systems will be increased as a result of a reduction in the evaporator sodium outlet temperature during the thermal transients discussed above. The potential excess flow capability within SGAHRS will tend to enhance natural circulation during this initial period. The potential excess cooling and its effects will be included in the natural circulation evaluation. The natural circulation contribution is included in the
17 analysis of the thermal transient.

5.6.1.3.10 Material Considerations

5.6.1.3.10.1 Structural Materials for Elevated Temperature Service

The design temperature for the majority of SGAHRS components and piping is less than 800°F, for which elevated temperature material requirements will not be required. Code Case 1481 will be applied to the superheater vent control valve and inlet line since they have a design temperature
3 of 935°F.

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5.6.1.3.10.2 Fracture Toughness

To demonstrate adequate margins of safety against fracture for the components of the SGAHRS, the following information is provided.

Fracture Control Procedures

Pertinent design procedures and operating requirements (relating to fracture prevention) such as permissible rates of temperature change are covered in ASME Section III, Appendix G. The basis of the fracture control procedure will be fracture mechanics methodology, wherein design stresses and flaw sizes are related to material toughness requirements.

Fracture Analysis

The planned fracture analysis includes the following items:

a. The SGAHRS will be designed in accordance with the current ASME Section III rules which account for expected normal, upset, emergency, and faulted conditions. The general criteria for establishing that components will be subjected to a fracture control system are the following: a) integrity of the component is critical to system performance or safety, b) the component is subjected to forces that
58 could produce fracture during tests or operational service, and c) the possibility exists that cracks or other flaws may be present in the component of a size and character which could result in failure or unsatisfactory performance.

b. The stress and strain analysis to be used in design will be in accordance with ASME Section III and applicable code case requirements.

c. The minimum size of flaws to be assumed to be present initially will correspond to the most severe flaws which might escape detection during inspection. The quality control procedures in the ASME Code will serve as a guideline.

d. The fatigue analysis to be used will incorporate the combined effects of fatigue and corrosion. If component criticality warrants, fracture mechanics crack growth data will also be generated in order to have test data which is generally applicable to all component geometries, stress states, and flow types. The analysis and test data application would be simi-
58 lar to that discussed in Section 5.5.3.11.2.

e. Current fracture mechanics methodology will be employed to ensure an adequate margin of safety against fracture possibilities. Required toughness values will reflect any degradation resulting from exposure to pertinent plant operating and environmental conditions. This work will provide further guidelines for any surveillance requirements.

Time-Dependent Operational Limitations

Time-dependent phenomena to consider for carbon steel are methana-
26 tion, corrosion, and strain aging. As cited in 5.6.1.3.10.4, maintaining
specified water purity will prevent problems from methanation and corrosion
17 in the SGAHRS. Strain aging, resulting from elevated temperature service
will be considered in design. Post fabrication heat treatment (stress relief)
would alleviate any potential for strain aging as a result of fabrication.
Strain aging at the plastic zones formed at any flaw tips during hydrostatic
testing (according to ASME Section III) will remain a consideration. Adequate design allowances will be made, although such regions of potential embrittlement would be quite small and localized.

Various embrittling phenomena may occur in 1 1/4 Cr - 1/2 Mo steel
as it ages at certain temperatures. The most notable are temper embrittlement
and creep embrittlement. Embrittlement of 1 1/4 Cr - 1/2 Mo steel can be mini-
mized by heat treatment. Specifically, heating the steel to a temperature
greater than $1300^{o}F$ and holding for at least one hour followed by atmospheric
26 cooling.

5.6.1.3.10.3 Austenitic Stainless Steel

As set forth in Section 5.6.1.1.4, the materials presently expected to be used in the SGAHRS are carbon steel and 1 1/4 Cr - 1/2 Mo steel. There are currently no austenitic stainless steels specified for this system.

5.6.1.3.10.4 Compatibility with Coolant

All components of the SGAHRS, as currently specified, will be of
26 carbon steel and 1 1/4 Cr - 1/2 Mo steel (Sec. 5.6.1.1.4). These components
will be exposed to water and/or steam.

In water and/or steam, carbon steels and 1 1/4 Cr - 1/2 Mo steel are susceptible to pitting in the presence of chloride and oxygen. Furthermore, below $550^{o}F$, these materials are susceptible to caustic gouging and perhaps caustic stress corrosion cracking. Maintaining the water consistent with requirements for chlorides, caustics and oxygen concentration will prevent these forms of localized attack.

Carbon steel may also be susceptible to methanation (hydrogen embrittlement). However, maintaining the specified water purity will also prevent this occurrence. Administrative procedures will be established to assure that proper water chemistry and purity levels are maintained.

5.6.1.3.10.5 Compatibility with External Insulation and Environment Atmosphere

Carbon steels and 1 1/4 Cr - 1/2 Mo steel are compatible with external
26 insulation during normal operation in the absence of excessive moisture or
chloride. Excessive moisture is prevented by quality controlled installation and
operating procedures.

5.6.1.3.11 Protection Against Environmental Factors

Protection against floods, fires, missiles, earthquakes and accidents is provided in the Steam Generator Building as discussed in Sections 3.4, 9.13, 3.5, 3.7 and 3.8, 3.6 and 15.6, respectively.

5.6.1.3.12 Compliance with BTP APCSB 10-1

BTP APCSB No. 10-1 requires that the following items be incorporated into the auxiliary feedwater system. A comment explaining how each requirement is met follows each item.

1. "The auxiliary feedwater system should consist of at least two full capacity independent systems including diverse power sources".

   The Steam Generator Auxiliary Heat Removal System (SGAHRS) meets this requirement by employing two 50% capacity motor-driven pumps, each connected to a separate diesel generator, and one 100% steam turbine-driven pump. Thus, there is 200% total capacity with half of it electrically powered and half steam powered. There are also two parallel auxiliary feedwater paths to each drum, one from the turbine-driven pump through a control valve and one from the headered motor-driven pumps through another control valve. The drum level setpoints are set such that normally flow is supplied from the motor-driven pumps only with the turbine pump recirculating. If the drum level falls below the turbine pump setpoint, the control valve downstream of that pump opens and begins supplying flow.

2. "Other powered components of the auxiliary feedwater system should likewise share the concept of separate and multiple source of motive energy".

   Power can be supplied through transformers and rectifiers from either the off-site supply or the emergency diesels. An additional source of power is a 125 V DC for each instrumentation and control division. The levels of redundancy are as follows: off-site power, diesel-generator power, battery power. The batteries in conjunction with the steam-driven auxiliary feedwater pump will permit auxiliary feedwater subsystem operation even with the loss of both off-site and emergency diesel supplied AC power.

3. "The piping arrangement, both intake and discharge, for each system should be designed to permit the pumps to supply feedwater to any combination of steam generators. This should take into account pipe failure, active component failure, power supply failure, and control system failure that would defeat the diversity requirement. One method that would be acceptable is crossover piping containing valves that can be operated remote manually from the control room using the power diversity principle for the valve operators and actuation systems".

This is one basis of the SGAHRS design. First, each pump can draw water from either the protected water storage tank or the condensate storage tank. The appropriate valving is included in the system to make this switch from the control room. Second, two complete auxiliary feedwater supply subsystems are used for each drum exclusive of the final piping into the drum. The discharge of the turbine-driven pump supplies water to each drum through a separate control valve and isolation valve. The two motor-driven pumps are headered together and also supply flow through a separate control valve and isolation valve. The isolation valves can be manually operated from the control room to stop flow from breaks. The design allows flow to any combination of steam drums and minimizes the effect of pipe failure, active component failure, power supply failure, and control system failure.

4. "The auxiliary feedwater system should be designed with suitable redundancy to off-set the consequences of any single active component failure; however, each subsystem need not contain redundant active components".

   The required redundancy has been provided by SGAHRS. Water can be drawn from either the protected water storage tank or the condensate storage tank. Redundant pumps are provided. Redundant control valves are provided for each loop. Three loops, any one of which can remove the maximum decay heat load, can be utilized in any required combination.

5. "When considering a high energy line break, the system should be so
58 arranged as to permit the capability of supplying necessary emergency feedwater to the steam generators, despite the postulated rupture of any high energy section of the system, assuming a concurrent single active failure".

   The only part of the Auxiliary Feedwater Subsystem which is pressurized during plant operation, startup, hot standby, or shutdown is the section of line between the normally closed SGAHRS isolation valve and the main feewater system. Should a rupture occur in that area, the two auxiliary feedwater isolation valves to that loop can be remote manually closed by the operator from the control room. If one valve fails to close, the control valve on the supply line which failed to isolate can be shut providing the isolation required. Water supply to the other drums would be uninterrupted. Prior to the time the break is isolated, the flow limiting capability of the control valves will limit break flow to approximately 110% of rated to insure a flow supply to the other operating loops. Should the flow limiting function fail, the AFW isolation valve
3 is automatically closed. Lines are properly separated and protected such that that the effects of a rupture in one loop cannot effect operation of
17 another loop.

5.6.2 Direct Heat Removal Service (DHRS)

5.6.2.1 Design Bases

5.6.2.1.1 Performance Objectives

The Reliability Program, discussed in Appendix C, will provide verification that SGAHRS removes residual heat following a reactor shutdown with a high level of reliability. Hence it is judged that only steam and feedwater trains backed by SGAHRS are necessary to safely and reliably remove residual heat following shutdown from three loop, full power operation. To enhance the reliability of decay heat removal, the DHRS provides a fourth redundant heat removal path and heat sink. The impact on overall shutdown heat removal reliability by inclusion of the DHRS is being determined by the Reliability Program described in Appendix C. The DHRS provides this supplementary capability by satisfying the following objectives:

a. Function to remove reactor decay heat following reactor shutdown from three loop rated power operation, assuming loss of all heat transfer through the IHXs at the time of reactor trip. Operation of three primary pump pony motors and maximum reactor decay power is assumed. Under these conditions, the DHRS is designed to provide sufficient cooling to ensure primary coolant boundary integrity and prevent loss of in-place coolable geometry of the core.

   To meet this objective, DHRS components will be sized such that, under these conditions, the average bulk primary sodium temperature will be limited to approximately 1140°F. Capability will be provided to permit remote manual initiation of DHRS from the Control Room. The overflow and makeup circuit and the spent fuel cooling system will be able to be cross connected in a manner which permits both EVST NaK cooling trains to be used when DHRS is removing full capacity heat load.

b. Accommodate the thermal transients resulting from normal, upset, emergency and faulted plant events in which continued performance of its function is not impaired.

c. Accommodate floods (Section 3.4), tornadoes (Section 3.3), missiles (Section 3.5), and earthquakes (Section 3.7), in which continued performance of its function is not impaired.

d. Function in a manner which will not significantly reduce the reliability and availability of the EVST heat removal chain. This objective requires the EVST NaK cooling circuits to be designed to remove concurrently the heat generated by the spent fuel and the reactor decay heat. If a single failure in DHRS occurs, procedures will require the EVST heat load to be transferred to the natural circulation cooled EVST loop.

e. Accommodate the failure of a single active component. To meet this objective, the equipment performing the DHRS function will be redundant to the extent that failure of a single active component will not result in complete loss of DHRS heat removal capability (i.e. adequate heat removal capability to maintain coolable core geometry will remain following failure of a single active component based on analyses where conservatisms are selectively applied.

An FMEA to demonstrate the single failure tolerance of the DHRS will be prepared for the FSAR.

5.6.2.1.2 Applicable Code Criteria and Cases

The components of the DHRS shall be designed, fabricated, erected, constructed, tested and inspected to the standards of Section III of the ASME Code, 1974 edition through the summer 1974 Addenda, Class 1 or 2.

Applicable code cases will be used to supplement the design analysis required by the ASME Code.

5.6.2.1.3 Surveillance Requirements

The need for surveillance of the DHRS piping and components will be determined as the system design progresses and as the need to monitor austenitic stainless steel is determined by ongoing programs. If a requirement is identified, a surveillance program will be designed in accordance with the philosophy of 10 CFR 50, Appendix H.

5.6.2.1.4 Material Considerations

High Temperature Design Criteria

High temperature components in the DHRS will be analyzed in accordance with the requirements specified in ASME Boiler and Pressure Vessel Code, Section III, as supplemented by the applicable code cases and RDT standards.

Material Specifications

Stainless steel materials which satisfy the requirements of the ASME Code will be specified for use in the DHRS system, as noted in Table 5.6-13.

5.6.2.1.5 Leak Detection Requirements

The DHRS will be monitored for sodium and NaK leaks and leak indication will be provided in the Control Room by the leak detection system described in Section 7.5.5.

5.6.2.1.6 Instrumentation Requirements

DHRS is remote manually activated and controlled from the Control Room. Instrumentation required to monitor the condition of the DHRS consists of thermocouples on the EVST sodium outlet lines (3 loops) level indicators in the EVST and the Reactor Vessel (RV), and makeup pump low flow indication. These instruments confirm that the sodium levels in the RV remain above the loop outlet nozzles, that temperatures remain below design limits, and that adequate flow is being discharged from the makeup pumps. Other DHRS diagnostic instrumentation is not essential for DHRS operation as the pumps and air blast heat exchanger are being operated at maximum design rates. When the reactor decay heat load has dropped sufficiently, the cooling capacity of the system may manually be reduced by lowering flow-rates or fan speed, or by shutting down one of the EVST cooling trains.

Instrumentation in the primary Na storage and processing system that is a part of DHRS is described in 9.3.2.5 (Overflow and Makeup Circuit). Instrumentation in the EVS sodium processing system
26 that is a part of DHRS is described in 9.3.3.5 and 9.1.3.1.5.

5.6.2.2 <u>Design Description (See Figure 5.1-7)</u>

5.6.2.2.1 <u>Design Methods and Procedures</u>

5.6.2.2.1.1 <u>Identification of Active and Passive Components Which Inhibit Leaks</u>

The equipment from the overflow and makeup circuit and the spent fuel cooling and cleanup system which is cross connected to provide the direct heat removal service is shwon schematically in Figure 5.1-7. The EVST airblast heat exchanger fans and louvers, pumps and certain valves are classified as active components in DHRS. The airblast fans and louvers are normally operated (for EVST cooling) at partial speed and partially open. During changeover to DHRS duty, the fans and louvers are manually set at full speed and full open position. The classification and operating mode of DHRS valves are given in Table 5.6-10. All active valves are remote manually operable from the Control Room. The classification and operating mode of DHRS pumps are given in Table 5.6-11.

In the remote case of a leak in one of the systems which provide DHRS, system leak detection or tank level instrumentation will
26 warn the operator to manually isolate the leak. An undetected leak in any of the piping or components providing DHRS cannot significantly lower the reactor vessel sodium due to the high elevation of the overflow nozzle.

5.6.2.2.1.2 <u>Design of Active Components</u>

The active valves associated with DHRS operation are listed in Table 5.6-10. Valve locations are shown in Figure 5.1-7. Active pumps associated with DHRS operation are listed in Table 5.6-11.

In order to assure the functional performance of active components in the DHRS, the components will be designed and tested in
44 accord with Reference 12, PSAR Section 1.6.

As described in Section 9.3, the components used for DHRS are designed for two plant events: DHRS initiation one-half hour after
26 reactor shutdown (design-governing) and DHRS initiation twenty-four hours after reactor shutdown. The design stress limits imposed for active DHRS pumps for the one-half hour event are those for emergency
46 conditions as defined in Section III of the ASME Code. The design stress limits imposed for active valves for the 1/2 hour event are those for faulted conditions as defined in Section III of the ASME Code,

46 with the stipulation that operability following the event must be assured. The design stress limits imposed for active DHRS pumps and valves for the twenty-four hour event are those for upset conditions as defined in Section III of the ASME Code. These limits will insure that no deformation resulting from DHRS conditions will prevent operability of these
26 components.

The configuration of the active DHRS valves for CRBRP is basically the same as the valves used in the Fast Flux Test Facility (FFTF). FFTF valves have been tested and the results of this test program have been factored in the design of the CRBRP valves. A test program for the CRBRP valve is planned to demonstrate valve operability following the DHRS events.

The active DHRS pumps are all electromagnetic pumps similar to the design used for the FFTF. Some changes have been incorporated into the CRBRP design as a result of the tests of the FFTF pumps. Water flow tests have been conducted on the revised throat section of the CRBRP pump. These test results have been factored into the design of the pumps. A prototype pump will be built and sodium tested with the results being factored into the final design of the CRBRP plant units. The first plant unit will also be sodium tested with its associated control consoles to
46 verify performance and capacity. Verification of operational capability 19
at temperatures associated with DHRS events will be done by analysis.

5.6.2.2.1.3 Surveillance and Inspection Program

The DHRS equipment will be inspected on a regular basis. The inspection program will cover critical welds, valves, and components. As the detailed design of DHRS equipment progresses, details of the
26 inspection program will be identified. Building arrangements will provide adequate access for inspection.

5.6.2.2.1.4 Protection Against Accelerated Corrosion and Material Degradation

Precautions similar to those discussed in Section 5.3.2.1.4 will be incorporated into the fabrication, transportation, installation, and test procedures for DHRS components.
26

5.6.2.2.1.5 Materials Inspection Program

The materials inspection program for DHRS will be specified as
26 the design progresses.

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5.6.2.2.2 Material Properties

The materials which form the sodium and NaK boundary for the DHRS will be 300 series stainless steels with properties consistent with the requirements of ASME B&PV Code, Section III. Other material requirements will be specified as the design of DHRS progresses.

26

5.6.2.2.3 Component Descriptions

All components of the DHRS are part of the Auxiliary Liquid
26 Metal System and are described in Section 9.3.

5.6.2.2.4 Overpressure Protection

Inert cover gas pressure will be maintained in the primary sodium overflow tank and in the NaK expansion tanks. Overpressure protection
26 will be provided by Inert Gas Receiving and Processing System (Section 9.5) relief valves.

5.6.2.2.5 Leak Detection System

The description of methods to be used for DHRS (as part of the Auxiliary Liquid Metal System) leak detection are in Section 7.5.
26

5.6.2.3 Design Evaluation

5.6.2.3.1 Analytical Methods and Data

Thermal and hydraulic characteristics of the DHRS will be evaluated to insure that the performance objectives are met. Reactor vessel upper plenum mixing assumptions will be verified by the program discussed in Section 1.5.1.4. Heat sources to be considered for the analysis are conservative estimates of decay heat and sensible heat. Additionally, the spent fuel
26 heat load will be conservatively considered as a heat source. The heat sink will be the two EVST NaK airblast heat exchangers cooled by air whose temperatures will be conservatively predicted.

The analysis will be a transient calculation which reflects the characteristics of the DHRS. As primary sodium temperatures increase, heat
26 exchanger ΔT's and the heat transfer rate to the air will increase. The analysis will demonstrate that the primary sodium temperature peaks at an acceptable value and then decreases.

5.6.2.3.1.1 Stress Evaluation Plan (SEP)

26 The procedure for developing SEP's for the DHRS will be consistent with the procedure described for the PHTS in Section 5.3.3.1.1.

5.6.2.3.1.2 Stress Analysis Verification

The analytical requirements in the ASME B&PV code, Section III for components under normal, upset, emergency, and faulted conditions will be used as design limits for DHRS. Code case supplements will be used when required.
26

5.6.2.3.1.3 Compliance With Code Requirements

The components of DHRS will be designed as Seismic Category I com-
26 ponents and in accordance with Section III of the ASME Code as listed in Table 9.3-1. The design of such components, as described in Section 3.9, will conform with the intent of Regulatory Guide 1.48.

5.6.2.3.1.4 Category 1 Seismic Design

All DHRS components will be designed to Seismic Category 1 require-
26 ments since they are required for heat removal following the seismic events described in Section 3.9. Lists of components and supporting structures are contained in Table 3.2-2. The limits of Category 1 for connecting components is shown on Figure 5.1-7. Component design is currently in conceptual or preliminary stages and detailed seismic models have not been developed. Where preliminary loadings are necessary, an equivalent static method is used to generate the loads. As the design and arrangement of system components become finalized, a dynamic analysis based on detailed seismic models will be performed. 14

5.6.2.3.1.5 Analytical Methods for Pumps, Valves, and Heat Exchangers

Elastic and dynamic analytical methods and, when required, inelastic methods will be used to evaluate stresses for pumps, valves, heat exchangers, and mixing tees. Computer programs used for these evaluations will be verified to assure their accuracy and applicability.

5.6.2.3.1.6 Operation of Valves under Transient Loadings

To insure proper valve operation during their design life, an
26 in-service testing program will be implemented for the DHRS. After installation and prior to initial service each valve will be given tests required by the program. These tests will be conducted under conditions similar to those to be experienced during subsequent in-service tests. Once the valves are placed in service they will be full-stroke exercised on a regular basis.

After a valve and/or its control system has been replaced, repaired, or undergone maintenance that could effect its performance, it will be tested as necessary to demonstrate that the performance characteristics are within acceptable limits.

Remote indicators will be visually checked on a regular basis to confirm that remote valve indications accurately reflect valve operations.

Additional assurance of proper valve operation is provided by requiring the valves be designed, fabricated, installed, tested and operated in accordance with Reference 12, PSAR Section 1.6.

44

5.6.2.3.1.7 Analytical Methods for Component Supports

Components supports will have the same ASME code classifications as the components that they support. Design and analysis of each support will comply with the appropriate ASME B&PV code, Section III, paragraph NF rules.

5.6.2.3.1.7 Analytical Methods for Component Supports

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5.6.2.3.2 Natural Circulation

The DHRS will not be designed to provide heat removal by natural circulation. Electrical power must be available to supply the primary sodium makeup and EVST NaK pumps in addition to at least one PHTS pony motor when the DHRS is removing decay heat. DHRS has no natural circulation heat removal capability since the overflow concept requires pumping and is not designed to operate when the PHTS loops are naturally circulating.

5.6.2.3.3 Pump Characteristics

Pump characteristics will be determined as the designs of the overflow and makeup circuit, EVST cooling chain, and DHRS progress.

5.6.2.3.4 Valve Characteristics

Valve characteristics will be determined as the designs of the overflow and makeup circuit, EVST cooling chain and DHRS progress.

5.6.2.3.5 Overflow Heat Exchanger Characteristics

The Overflow Heat Exchanger characteristics will be determined as design of the Auxiliary Liquid Metal System progresses.

5.6.2.3.6 Coolant Boundary Integrity

Pipe leaks could reduce the DHRS capability and could cause an operational and radiation hazard. As a result, pipe loadings resulting from off-normal events will be predicted and the regions of highest stress will be identified. Pipe leak investigations will be made for the DHRS pipe runs to determine the possible damage the postulated leaks will have on the DHRS components and their surroundings. These investigations will be similar to those discussed for the PHTS in Section 5.3.3.6.

5.6.2.3.7 Inadvertent Operation of Valves

Initiation of the DHRS is a remote-manual operation accomplished from the Control Room, consisting of opening or closing the active valves listed in Table 5.6-10 and remote manual operation of the makeup pump, EVST NaK pump, and EVST airblast exchanger (refer to Section 5.6.2.2.1.1, "Active and Passive Components"). The switchover to DHRS is done in accordance with the clearly defined procedures implemented over a period of less than one-half hour. During that time, board-mounted temperature indications provide verification of proper operation of the system. A sequence is provided to automatically position valves and control the pumps and ABHXs to minimize required operator actions during DHRS startup. Remote-manual initiation of DHRS is acceptable, because DHRS is not required to mitigate the effects of any Design Basis Event. Additionally, the operator has sufficient time to assess the plant condition and to take appropriate action to switch over to DHRS utilizing either the DHRS startup panel sequence or manually energizing a series of component controls from a single panel in the Control Room. Once the system is put into operation, no further valve positioning changes are made, except in the event that a makeup pump fails, it would be necessary to close the outlet valve from the failed pump. Consequently, inadvertent operation of valves is extremely remote.

In the extremely unlikely event, however, that a valve is mispositioned, some of the DHRS heat removal capability may be initially lost. For instance, closure of a valve on the makeup pump suction will initially reduce the heat removal capability by about 50%; however, the capacity of the remaining pump can be increased to 600 gpm, which results in no long term reduction in the DHRS heat removal capability. Closure of one of the redundant DHRS OHX inlet isolation valves will not result in any loss of cooling capability. If one makeup pump becomes inoperative, most of the flow from the operating pump may recycle (back-flow through the inoperative pump) until the discharge valve of the inoperative pump is closed. In any event, sufficient time exists to correct the situation and restore complete cooling capability with no deleterious effects. Total loss of cooling will result in about a 50°F/Hr rise in the temperature of the primary system (maximum decay heat at time of DHRS initiation - 24 hrs.). It is estimated that restoration of service can be accomplished in less than 15 min. with the resultant temperature rise of only about 12°F.

Other misoperations, such as inadvertent shutdown of an airblast heat exchanger, will reduce DHRS capability by 1/2. Restart can also be accomplished in less than 15 minutes with no serious impact on the system. In this case, the maximum temperature rise is estimated at less than 6°F.

5.6.2.3.8 Performance of Pressure Relief Devices

DHRS pressure relief is achieved through the inert cover gas system. The pressure relief devices are in contact only with cover gas which is described in Section 9.5.

5.6.2.3.9 Operational Characteristics

DHRS components function to remove decay heat only when the three steam generator heat removal paths are not available. However, the DHRS components routinely function as part of the overflow and makeup circuit and the spent fuel cooling and cleanup system. As a result, the majority of transients to be considered for component design will result from their normal service application. Additional transients will be used which represent operation of the components when they are cross connected to provide the direct heat removal service.

The component structural design conditions, when acting in the DHRS capacity, are described in the appropriate sections of the plant duty cycle given in Appendix B. Initiation of DHRS for full capacity load at 24 hours after shut down is considered an emergency event and, at one half hour, a faulted event.

The Direct Heat Removal System provides a separate, diverse, fourth heat removal path which can adequately cool the reactor core after a shutdown. The DHRS equipment will have adequate heat removal capacity to maintain the maximum average Primary Heat Transport System coolant temperatures at or below 1140°F for the following design basis transient.

1. Reactor trip from 3 loop thermal hydraulic design power level. Complete loss of heat transfer through the IHXs is assumed at the time of reactor trip.

2. DHRS is remote-manually initiated from the Control Room and is fully operational 30 minutes after the reactor trip. It is assumed that all PHTS loops are intact with sodium circulated by all three pony motors.

Plant equipment is being designed to accommodate the event under criteria which assures that (1) the coolant and DHRS boundaries remain intact, (2) the primary pumps retain their ability to function at pony motor speed, and (3) active DHRS components retain their ability to perform as intended. The requirement for the reactor is that coolable core geometry be maintained. Table 5.6-12 shows various DHRS operating cases and sensitivities.

5.6.2.3.9.1 Preliminary Assessment

The following discussion is qualitative and based on design requirements and conservative scoping calculations.

During the 1/2 hour between shutdown and DHRS initiation, the flowing sodium reduces the difference between the maximum and minimum primary system temperatures. At the same time, the reactor decay power slowly heats the overall primary system. At 1/2 hour the average primary temperature is estimated to be 1050°F. The peak hot channel temperature occurs shortly after shutdown, and is similar to the peak temperature following a normal scram.

This estimate is based on an available DEMO calculation in which the reactor was tripped and heat flux from the IHX tube to intermediate side was set to zero to initiate the transient. Heat transport via the overflow system was conservatively set to zero.

The simulation conservatively neglected piping, check valve and reactor vessel heat capacities as well as heat transfer from these components. The upper plenum was assumed to stratify without mixing, effectively neglecting a portion of the plenum and associated structures heat capacity. The piping transport was represented as a pure delay of one dimensional slug flow. Thus the known smoothing of temperature profiles in the primary piping due to thermal diffusion (mixing and conduction) was neglected. The flow from the three loops was assumed to mix perfectly in the lower plenum and transfer heat to or from a single node representation of the lower plenum, assembly inlet and support structures.

As shown in Figure 5.6-4, the reactor vessel outlet temperatures, shortly after the event is initiated, are similar to those following a normal reactor trip. The vessel outlet temperature is determined by the collapsing reactor ΔT in the first 5 seconds, stratification and settling of the cooler fluid to the lower parts of the upper plenum in the next 15 seconds, and finally, a direct circuit, caused by the assumed complete stratification between core outlet and reactor vessel nozzle outlet during the remainder of the transient. As shown, the front of decreasing temperatures is transported to the IHX inlet in about 100 seconds for the first part of the transient during which flow is coasting down. Later, at pony motor flow rates, this transport time is about 175 seconds.

The adiabatic condition applied to the IHX tube surface precipitates a rise of the IHX outlet temperature to near 1015^{o}F, the initial inlet temperature. The outlet temperature maximizes near 400 seconds and then decreases because of lower IHX inlet temperatures. These lower inlet temperatures are a result of the reactor trip as previously discussed. The IHX outlet temperature rise is transported back to the reactor vessel inlet in
25 about 20 seconds while flows are coasting down. Later, at pony motor flow rates, the transport time is about 75 seconds. The hotter fluid entering the lower plenum is assumed to mix, transfer heat to the structure and enter the core. In the core, thermal energy from decay heat is added, effectively determining the fluid exit temperature and completing the simulation loop.

As these processes are repeated by the loop recirculation, the thermal capacitance of the system and decreasing decay power act to further reduce the system-wide temperature differences. The actual reduction would probably be greater than shown in Figure 5.6-4, since piping heat capacity and heat loss, as well as piping mixing, were neglected in the simulation.

The one lump average system temperature at 1/2 hour used to scope the DHRS size requirements was extrapolated from these curves. The extrapolation was based on the fact that at 1/2 hour the system ΔTs were small, and that the reactor outlet temperature projects to near 1050^{o}F. 22

The reactor temperatures for both the radial blanket and fuel assembly hot channels are shown in Figure 5.6-5. Immediately following the initiation of the event, the transient in the hot channel is similar to that following a normal reactor trip. Later the hot channel temperatures follow the increasing core inlet temperatures. The results shown are based on a calculated 8.7% core flow after the pump coastdown. With the highly improbable combination of worst case pressure drops and pump characteristics the minimum flow could be as low as 7.5%. Therefore, the higher hot channel temperatures for this lower flow were estimated.

Based on results calculated for a reactor trip followed by minimum flow (7.5%) from 3 pony motors, the peak temperature immediately following initiation of this event would be no higher than 1428°F. At longer times, the peak blanket temperatures were estimated by multiplying the calculated inlet to blanket ΔT in Figure 5.6-5 by the ratio of calculated to minimum flow (8.7/7.5). These results show the maximum temperature at 650 seconds is 1190°F, at 1400 seconds it is 1200°F, and at 1/2 hour the blanket maximum temperature projects to 1260°F, compared to a sodium boiling temperature of over 1700°F.

Following DHRS initiation at 1/2 hour, the transient system temperatures are governed by the steadily decreasing decay heat, the DHRS heat rejection rate, and the system heat capacity. Scoping calculations show the maximum system temperature, 1140°F for the design basis transient, occurs about 10 hours after shutdown.

The temperature of the reactor cladding hot spot over that of the system was estimated with the maximum decay power and minimum reactor flow from 3 pony motors. At DHRS initiation, 1/2 hour after the shutdown, the maximum cladding hot spot temperature is 1260°F. At the time of the maximum system temperature the maximum cladding hot spot temperature is 1245°F.

5.6.2.3.9.2 Updated Assessment of the Base Case F-2 Event

A standard F-2 event was analyzed with updated decay heats and heat capacities. The results indicate that, with adiabatic IHXs at time zero, the peak hot leg temperature is 1052°F, as shown in Figure 5.6-13. The difference between the revised and previously estimated PHTS and reactor heat capacities is small. For example, the (MC_p) for the reactor and one PHTS loop is 549,000 Btu/F. The old value was 557,000 Btu/F. Therefore, it can be concluded that the lower peak temperature of the present analysis is a result of the new and revised core decay power.

5.6.2.3.9.3 Single Failure Assessment

The requirement to accommodate the failure of a single active component, as described in paragraph 5.6.2.1.1.e, resulted in the necessity to perform a single failure analysis. The inputs for this updated DHRS single failure analysis include:

- Two PHTS loops operating
- No IHTS heat capacity
- No SGS heat capacity

- One Na makeup pump at 600 gpm
- One NaK pump at 600 gpm
- One air blast heat exchanger
- Conservative decay heat used
- No EVST heat load
- Heat losses through insulation included

The results of the single failure analysis indicate a peak hot leg temperature of approximately 1137°F, as shown in Figure 5.6-14.

The peak temperatures for the single failure assessment slightly exceed (by 7°F) the structural peak temperature analyzed for the F-2 transient. The detailed transient curves for the preliminary assessment peaked at 1130°F and were included in specific equipment specifications. The detailed structural analysis is not planned to be redone since scoping analysis indicates the capability of the primary boundary to maintain structural integrity, based on creep rupture considerations of Code Case 1592, at temperatures of approximately 1200°F is approximately 100 hrs.

Specific features which incorporate the single failure capability include 1) a redundant valve in parallel with the existing valve at the OHX inlet to ensure OHX flow if either valve fails to open, and 2) the addition of remote manual operators on the valves at the discharge from the makeup pumps to permit operation from the Main Control Room. These features are shown in Figure 5.1-7 and in Table 5.6-10.

5.6.2.3.10 Material Considerations

5.6.2.3.10.1 Structural Materials for Elevated Temperature Service

DHRS materials will satisfy the requirements of ASME Code Section III and the applicable code cases. Methods for predicting short time strength degradation will be developed as the design progresses. The methods for extrapolating creep-rupture and creep fatigue-data will be those which have been proposed and reported by the ASME materials committees.

5.6.2.3.10.2 Fracture Toughness

A coordinated program, similar to that discussed in Section 5.3.3.10.2, will be designed to prevent the non-ductile failure of components which contain radioactive material or which are essential to the protection of the health and safety of the public.

5.6.2.3.10.3 Austenitic Stainless Steel

Procedures will be used to insure that DHRS materials are not degraded by cleaning solutions or other contaminants. Heat treatment requirements are detailed in Section 5.3.3.10.3.2. Delta ferrite will be controlled to optimum levels in accordance with Section III of the ASME B&PV Code.

5.6.2.3.10.4 Compatability with Coolant

The major DHRS construction material in contact with sodium is 300 series stainless steel and its compatability is discussed in Section 5.3.3.10.4.1.

5.6.2.3.10.5 Compatability with External Insulation

All piping in the system will be thermally insulated with materials which will cause minimum corrosive element contamination of the piping.

5.6.2.3.11 Protection Against Environmental Factors

Protection against fires, floods, missiles, earthquakes and accidents is provided in Seismic Category I buildings which house components used for DHRS as discussed in Sections 9.13, 3.4, 3.5, 3.7 and 3.8, and 15.6, respectively.

5.6.2.3.12 Pump Testing

5.6.2.3.12.1 EM and Primary Pump Testing at DHRS Temperatures

The Na/NaK Electromagnetic pump testing was performed on the prototype pump in a sodium loop in the Sodium Component Test Installation at the Energy Technology Engineering Center (ETEC) operated by Atomics International. One test consisted of head-flow tests performed at a sodium temperature of 1130°F. Cavitation, variable voltage and head flow mapping tests were performed at 1130°F. Figure 5.6-13 shows the 1130°F Head-Flow Mapping Efficiency versus Sodium Flow Rate and summarizes the EM pump capability at 1130°F.

The primary sodium pump tests are to be performed on the prototype pump in a sodium loop in the Sodium Pump Test Facility at ETEC. One of the tests to be conducted involves a heat-up of the test loop sodium by loop heaters and pump power (with the pump on main motor flow) to 1100°F over a priod of twelve hours. Temperature will then be decreased to 1005°F over the next twelve hours.

These tests will demonstrate the ability of the EM pumps and the primary sodium pumps to operate at the temperatures associated with DHRS operation.

5.6.2.3.12.2 EM Pump Capability at DHRS Temperatures

The EM pump capacity is 600 gpm, based on test results, even though it is specified as a 400 gpm flat, linear induction, electro-magnetic pump. During the EM pump sodium test program, head-flow mapping tests were performed at different sodium temperatures (1130°F, 800°F, 600°F, and 450°F) and varying voltages. Each of the test series at the four temperatures included flow rates of 600 gpm. These tests demonstrate the capability of the EM pump at 600 gpm under various DHRS operating conditions. Additionally, a design limit test was performed to determine the amount of excess capacity available for emergency operation. As demonstrated, a flow rate of 800 gpm is possible at the design limit.

REFERENCES TO SECTION 5.6

5.6-1 ESCOA Fintube Engineering Manual, 1979.

5.6-2 Boyko, L. D. and Kruzhilin, G. N., "Heat transfer and hydraulic resistance during condensation of steam in a horizontal tube and in a bundle of tubes", Int. J. Heat Mass Transfer, 10, pp. 361-373 (1967).

5.6-3 Baker, O., "Simultaneous flow of oil and gas", Oil and Gas Journal 53, p. 185 (1954).

5.6-4 Soliman, M., Schuster, J. R. and Berenson, P. J., "A general heat transfer correlation for annular flow condensation", J. Hear Transfer 90, p. 267 (1968).

5.6-5 Chato, J. C., "Laminar condensation inside horizontal and inclined tubes", ASHREA Journal, Vol. 4, No. 2, p. 52, February 1962.

5.6-6 Lockhard, R. W. and Martinelli, R. C., "Proposed correlation of data for isothermal two-phase, two-component flow in pipes", Chem. Eng. Progress 45, p. 39 (1949).

5.6-7 Baroczy, C. J., "Correlation of liquid fraction in two-phase flow with application to liquid metals", NAA-SR-8171 (April 1963), Atomics International, Canoga Park, California.

5.6-8 Baroczy, C. J., "A systematic correlation for two-phase pressure drop", Chem. Eng. Prog. Symp. Ser. 62, No. 64, p. 232 (1966).

5.6-9 Kline, S. J. & McClintock, F. A., "Describing Uncertainties in Single-Sample Experiments," Mechanical Engineering, January 1953.

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5.6-1 ESCOA Fintube Engineering Manual, 1979.

5.6-2 Boyko, L. D. and Kruzhilin, G. N., "Heat transfer and hydraulic resistance during condensation of steam in a horizontal tube and in a bundle of tubes", Int. J. Heat Mass Transfer, 10, pp. 361-373 (1967).

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5.6-4 Soliman, M., Schuster, J. R. and Berenson, P. J., "A general heat transfer correlation for annular flow condensation", J. Heat Transfer 90, p. 267 (1968).

5.6-5 Chato, J. C., "Laminar condensation inside horizontal and inclined tubes", ASHRAE Journal, Vol. 4, No. 2, p. 52, February 1962.

5.6-6 Lockhart, R. W. and Martinelli, R. C., "Proposed correlation of data for isothermal two-phase, two-component flow in pipes", Chem. Eng. Progress 45, p. 39 (1949).

5.6-7 Baroczy, C. J., "Correlation of liquid fraction in two-phase flow with application to liquid metals", NAA-SR-8171 (April 1963), Atomics International, Canoga Park, California.

5.6-8 Baroczy, C. J., "A systematic correlation for two-phase pressure drop", Chem. Eng. Prog. Symp. Ser. 62, No. 64, p. 232 (1966).

5.6-9 Kline, S. J. & McClintock, F. A., "Describing Uncertainties in Single-Sample Experiments," Mechanical Engineering, January 1953.

TABLE 5.6-1

SGAHRS MAJOR COMPONENTS DESIGN SUMMARY

Protected Air Cooled Condenser (PACC)

Number: 1 Per Loop, 3 Per Plant
Design Pressure = 2200 psig (1)
Operating Pressure = 1400 psig
Design Temperature = 650°F
17 Operating Temperature = 588 °F
26 Heat Rejection Fans Operating = 15 MWt (2)
Cooling Air Temperature = 100°F

Turbine Driven Auxiliary Feedwater Pump (AFP)

Number: 1 Per Plant
Design Flow Rate = 1432 gpm
17 Design Developed Head = 3927 ft
Design Temperature = 200°F (water); 120°F (ambient)
43 58 Minimum NPSH 25 ft
17 26 Design Pressure = 2200 psig (1)

Motor Driven Auxiliary Feedwater Pumps (AFP)

Number: 2 Per Plant
Design Flow Rate = 716 gpm
43 17 Design Head = 3927 ft
Design Temperature = 200°F (water); 120°F (ambient)
Design Pressure = 2200 psig (1)
43 17 58 Minimum NPSH = 25 ft

AFP Motor Drives

Number: 2 Per Plant
26 Maximum Horsepower = 1050
17 Design Voltage = 4160 AC
RPM = 3600

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(1) Maximum test pressure will be determined as design progresses. 9

(2) Each PACC consists of two one half size tube bundles, fans and louvers so that no single active failure can reduce the heat rejection capacity of a given PACC below 50% of rated (7.5 MWt).

26 (3) Admission valves are designed to 2200 psig and 650°F. Remaining turbine parts are designed as indicated.

TABLE 5.6-1 (Cont'd)

AFP Turbine Drive

Number: 1 Per Plant
Design Pressure = 1250 psia [1] [3]
Design Temperature = $600^{o}F$ [3]
Design Steam Inlet Quality = 96.0%
45 Design Steam Flow Rate = 100,000 lb/hr
43 Maximum Horsepower = 2000
26 RPM = 4000

Protected Water Storage Tank (PWST)

Number: 1 Per Plant
Tank Volume = 9591 ft^3
Usable Water Volume = 9111 ft^3
Design Temperature = $200^{o}F$
Design Pressure = 15 psig

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(1) Maximum test pressure will be determined as design progresses.

(2) Each PACC consists of two one half size tube bundles, fans and louvers so that no single active failure can reduce the heat rejection capacity of a given PACC below 50% of rated (7.5 MWt).

(3) Admission valves are designed to 2200 psig and $650^{o}F$. Remaining turbine parts are designed as indicated.
26

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TABLE 5.6-2

CLASSIFICATION OF SGAHRS COMPONENTS

COMPONENT	SAFETY CLASS	NATIONAL CODES	QUALITY STANDARDS*	QUALITY ASSURANCE ASME
Protected Water Storage Tank (PWST)	SC-2	ASME III/2	Group B	NA-4000
PWST Piping	SC-2	ASME III/2	Group B	NA-4000
PWST Valves	SC-2	ASME III/2	Group B	NA-4000
Protected Air Cooled Condenser (PACC)	SC-3	ASME III/3	Group C	NA-4000
PACC Piping	SC-3	ASME III/3	Group C	NA-4000
Auxiliary Feedwater System (AFS) Piping	SC-3	ASME III/3	Group C	NA-4000
AFS Pumps	SC-3	ASME III/3	Group C	NA-4000
AFS Valves	SC-3	ASME III/3	Group C	NA-4000

17 * NRC Regulatory Guide 1.26 "Quality Group Classifications and Standards," March 23, 1973.

TABLE 5.6-3

SGAHRS EQUIPMENT LIST AND MATERIAL SPECIFICATIONS

SGAHRS COMPONENT	ASME SECTION III CODE CLASS	MATERIAL*	DESIGN TEMP (^{o}F)	DESIGN PRESSURE (PSIG)
Air Cooled Condenser Bundle	3	CS	650	2200
Air Cooled Condenser Fan, Motor, Louvers	-	--	100	----
Auxiliary Feedwater Pump	3	CS	200	2200
Pump Motor Drive	-	--	104	----
Pump Turbine Drive				
Downstream of Admission Values	-	CS	600	1250
Upstream of Admission Values	-	CS	650	2200
Water Storage Tank	2	CS	200	15
SGAHRS Piping:				
FWST to First Isolation Valve	2	CS	200	15
First Isolation Valve to AFW Pumps	3	CS	200	100
AFW Pumps to AFW Headers	3	CS	200	2200
AFW Headers	3	CS	200	2200
AFW Headers to Electrically Operated Isolation Valve	3	CS	200	2200
AFW Pump Test Loop to and between Isolation Valves	3	CS	650	2200
AFW Pump Test Loop From Isolation Valves to FWST Fill Line	3	CS	200	100
Isolation Valve to Main FW Line	3	CS	650	2200
Superheater Inlet Line to PACC	3	CS	650	2200
PACC to Evaporator Recirc Line	3	CS	650	2200
AFW Pump Recirc to Orifice	3	CS	200	2200
Orifice to FWST-Recirc	3	CS	200	250
Superheater Vent Line (Upstream of Valve)	3	1 1/4 Cr-1/2 Mo	935	1900
Steam Drum Vent Line (Upstream of Valve)	3	CS	650	2200
Superheater Vent Line (Downstream of Valve)	3	1 1/4 Cr-1/2 Mo	850	250
Steam Drum Vent Line (Downstream of Valve)	3	CS	400	250
Steam Supply Line to Drive Turbine	3	CS	650	2200
PACC Vent Line Upstream of Vent Orifices)	3	CS	650	2200

*CS - Carbon Steel

TABLE 5.6-3 (Cont'd)

SGAHRS COMPONENT	ASME SECTION III CODE CLASS	MATERIAL*	DESIGN TEMP (^{o}F)	DESIGN PRESSURE (PSIG)
PACC Vent Line (Downstream of Vent Orifices)	3	CS	400	250
PWST Fill Line	3	CS	200	100
AFW Pump Alternate Supply Line	3	CS	200	100
Drive Turbine Exhaust	3	CS	340	100
SGAHRS Valves:				
Alternate AFW Supply	3	CS	200	100
PWST Fill	3	CS	200	100
PWST Drain	2	CS	200	15
PWST Level Indicator	2	CS	200	15
AFW Pump Inlet (Manual)	2	CS	200	15
AFW Pump Inlet (Electrical)	2	CS	200	100
Alternate AFW Pump Inlet	3	CS	200	100
Pump Recirculation	3	CS	200	2200
Pump Recirculation c/v	3	CS	200	2200
Pump Discharge C/V	3	CS	200	2200
Pump Discharge Isolation	3	CS	200	2200
AFW Supply Isolation (Manual)	3	CS	200	2200
AFW Supply Control	3	CS	200	2200
AFW Supply Isolation (Electrical)	3	CS	650	2200
AFW Supply C/V	3	CS	650	2200
AFW Supply Isolation (Manual)	3	CS	650	2200
AFW Supply C/V	3	CS	650	2200
AFW Pump Test Loop Isolation	3	CS	650	2200
Superheater Vent Control	3	2 1/4CR-1 Mo	935	1900
Steam Drum Vent Control	3	CS	650	2200
Drive Turbine Steam Supply Isolation (Elect.)	3	CS	650	2200
Drive Turbine Steam Supply C/V	3	CS	650	2200
Drive Turbine Steam Supply Isolation (Manual)	3	CS	650	2200
Drive Turbine Steam Supply Pressure Control	3	CS	650	2200
PACC Steam Supply	3	CS	650	2200
PACC Steam Supply Bypass	3	CS	650	2200
PACC Condensate Return	3	CS	650	2200
PACC Noncondensible Vent	3	CS	650	2200
PACC Noncondensible Vent Isolation	3	CS	650	2200
Pressure Instrument (Pump Inlet)	3	CS	200	100
Pressure Instrument (Pump Discharge)	3	CS	200	2200
Pressure Instrument (Turbine Inlet)	3	CS	650	2200
Chilled Water Isolation	3	CS	200	100

*CS - Carbon Steel

TABLE 5.6-4

SGAHRS WELD FILLER METAL SPECIFICATIONS

	BASE MATERIAL	ASME SECTION II	SPECIFICATION
	Carbon Steel	SFA-5.1	Specification for Mild Steel Covered ARC-- Welding Electrodes
26	1 1/2 Cr-1/2 Mo	SFA-5.5	Specification for low-alloy steel covered arc-welding electrodes.

Amend. 26
Aug. 1976

TABLE 5.6-5

SGAHRS VALVE CLASSIFICATION

VALVE	ACTIVE/ INACTIVE	NORMAL POSITION	OPERATING MODE
Alternate AFW Supply	Inactive	Open	Isolation
PWST Fill	Inactive	Closed	Isolation
PWST Drain	Inactive	Closed	Isolation
PWST Level Indicator	Inactive	Open	Isolation
AFW Pump Inlet (Manual)	Inactive	Open	Isolation
AFW Pump Inlet (Electrical)	Inactive	Open	Isolation
Alternate AFW Pump Inlet	Inactive	Closed	Isolation
Pump Recirculation	Active	Closed	Isolation
Pump Discharge C/V	Active		Check
Pump Discharge Isolation (Manual)	Inactive	Open	Isolation
AFW Supply Isolation (Manual)	Inactive	Open	Isolation
AFW Supply Control	Active	Open	Flow Control
AFW Supply Isolation (Electrical)	Active	Closed	Isolation
AFW Supply Isolation (Manual)	Inactive	Open	Isolation
AFW Supply C/V	Active		Check
AFW Supply C/V	Active		Check
AFW Pump Recirculation Check Valve	Active		Check
AFW Pump Test Loop Isolation Valve	Active	Closed*	Isolation
Steam Drum Vent Control	Active	Closed	Vent, Pressure Control
Superheater Vent Control	Active	Closed	Vent, Pressure Control
PACC Steam Supply	Inactive	Open	Isolation
PACC Steam Supply Bypass	Inactive	Open	Isolation
PACC Condensate Return	Inactive	Open	Isolation
PACC Noncondensible Vent	Active	Closed	Vent
PACC Noncondensible Vent Isolation	Inactive	Open	Isolation
Drive Turbine Steam Supply Isolation (Electrical)	Active	Closed	Isolation
Drive Turbine Steam Supply C/V	Active		Check

*Open During Test, Valve Closes if SGAHRS is Actuated.

TABLE 5.6-5 (Cont'd)

	VALVE	ACTIVE/INACTIVE	NORMAL POSITION	OPERATING MODE
	Drive Turbine Steam Supply Pressure Control	Active	Closed	Pressure Control
	Pressure Instrument (Pump Inlet)	Inactive	Open	Isolation
	Pressure Instrument (Pump Discharge)	Inactive	Open	Isolation
	Pressure Instrument (Turbine Inlet)	Inactive	Open	Isolation
50 26 54	Pressure Instrument (Turbine Exhaust)	Inactive	Open	Isolation

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May 1980

TABLE 5.6-7

DAHRS RESULTS STEAM VENTING (1)

Incident, Plant Response	Failure Discussion	Vent Time (hr)	Total S.G. Heat Load (10^6 Btu)	Total PACC Heat Load (10^6 Btu)	Total Heat Vented (10^6 Btu)	H_2O Vented (ft^3)
A.(3) 3-Loop Operation at "stretch" (1121 Mwt)						
1. Loss of off-site power, diesels start	U-18 event, SGAHRS works properly	0.9	383	127	256	3116
2. Water-side pipe break, half PACC fails	Design basis: All loops on pony motors with one steam-side loop dried out; one PACC @15 Mwt, one @ 7.5 Mwt(2)	L1:0 L2:4.4 L3:1.5	 320 186	 111 74	 321	 4025
3. Loss of all bulk AC power	All loops and PACCs on natural circulation (4)	L1,2,3:1.6	480	90	390	5100
4. Sodium fire, delayed PACC start-up	Design basis: One loop lost due to sodium spill; second loop lost due to evaporator safety relief valve failure; PACC @ 15 Mwt after 5000 second startup delay in third loop.	L1,2:0 L3:5.7	 555	 219	 336	 4207

NOTES:

(1) Assumes off-site power and balance of plant equipment (i.e., condensate storage tank and main FW pumps) are not available.

(2) Each PACC unit contains two half size fans and two sets of inlet and outlet louvers. Thus, single active failures (fan or control louver) cannot reduce the capacity of a given PACC to less than one-half its rated capacity. For two loop operation, each fan is connected to a separate power source.

(3) Unless otherwise stated, PACC starts at 4 minutes and ramps to full operation over 60 seconds. Auxiliary feedwater is at 100°F (all cases).

(4) Assumes power restored after two hours at which time the turbine-driven pump is turned off and the PACCs assume complete shutdown heat removal based upon an estimated PACC heat rejection of 30% of rated capacity.

TABLE 5.6-8

SUMMARY OF SYSTEM WATER/STEAM LEAKAGE

Component	Leakage/Plant ft^3
Safety/Relief and Vent Valves	1655(1)
Isolation Valves	295(1)
Recirculation Water Sampling Line	15(2)
Recirculation Pump Seals	93(2)
Thirty Day Sum (three loops)	2058

(1) Leakage value for 30 days.

(2) Assumes operator isolates and bypasses the recirculation pump, including the recirculation water sampling line, within 30 minutes.

TABLE 5.6-9

TOTAL PWST VOLUME REQUIREMENTS

CASE(3)

	PIPE BREAK (THREE LOOP OPERATION)	PIPE BREAK (TWO LOOP OPERATION)	SODIUM FIRE DELAYED PACC START UP	LOSS OF BULK AC POWER
Water Vented (per DAHRS)	4025 ft^3	4164	4207	5100
Pipe Break Loss(1)	1481	1481	1481(4)	---
Leakage(2)	1372	686	686	2058
Auxiliary Liquid Metal System	250	250	250	250
Required Usable Volume	7128 (3)	6581	6624	7408
Margin (22.9%)				1703
Usable Volume				9111
Heel (5%)				480
Total Tank Volume				9591 ft^3

(1) Assumes operator isolates break within 10 minutes. Break flow sum based on 110% flow (77 lb/sec) from each of two auxiliary feedwater supply lines for a period of 600 seconds.

(2) Leakage is a function of the number of loops remaining following the initiating event.

(3) Cases correspond to those of Table 5.6-7.

(4) In the sodium fire event, this quantity of water is lost due to the assumed failure of the safety relief valve, in the open position, at the superheater exit. A ten minute operator action time is assumed to isolate the loop.

TABLE 5.6-10

DHRS VALVE CLASSIFICATION

LOCATION	VALVE NO.**	ACTIVE/INACTIVE	NORMALLY* OPEN/CLOSED	OPERATING MODE
Makeup Pump Suction	116	Inactive	Open	Isolation
	132	Inactive	Open	Isolation
	114, 133	Inactive	Closed	Isolation
Overflow VSL Drain/Sample Return	153	Active	Open	Isolation
Makeup Pump Dicharges	104	Active	Open	Isolation
	118	Active	Open	Isolation
	102	Active	Open+	Isolation
	109	Active	Open+	Flow Control
	103, 107	Active	Closed+	Isolation
Overflow Ht. Exch. Sodium Out.	150	Inactive	Open	Isolation
	151	Inactive	Open	Check
	149	Inactive	Open	Isolation
Overflow Ht. Exch. to Airblast (hot leg)	416, 357	Active	Closed+	Isolation
	359, 420	Active	Open+	Isolation
Airblast to Overflow Ht. Exch. (cold leg)	358, 415	Active	Closed+	Isolation

* The position indicated is the valve position during normal plant operation. Valves with + are those which are manually opened, or closed, prior to initiation of DHRS operating mode. All active valves are remote-manual operable from the Control Room.

** Valve numbers are those shown on Figure 5.1-7.

TABLE 5.6-10

(Continued)
Page 2

DHRS VALVE CLASSIFICATION

LOCATION	VALVE NO.**	ACTIVE/INACTIVE	NORMALLY* OPEN/CLOSED	OPERATING MODE
NaK Drain Lines	364, 424, 446 455	Inactive	Closed	Isolation
EVST NaK Exp. Tank	300P, 300Q	Inactive	Open	Isolation
Cover Gas	301A	Inactive	Closed+	Isolation
Sodium Vent (Typ.)		Inactive	Closed	Isolation
Sodium Sample	100A	Inactive	Open	Isolation

* The position indicated is the valve position during normal plan operation. Valves with + are those which are manually opened or closed prior to initiation of DHRS operating mode. All active valves are remote-manual operable from the Control Room.

** Value numbers are those shown on Figure 5.1-7.

TABLE 5.6-11

DHRS PUMP CLASSIFICATION

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PUMP	ACTIVE/INACTIVE	OPERATING MODE NORM. PLANT OPER.	 DHRS DUTY
Makeup Pump (s) (Operating)	Active	Reduced Power Flow	100% Power Flow
46 26 45			
EVST NaK Pump (Operating)	Active	Reduced Power	100% Power
EVST NaK Pump (Standby)	Active	Reduced Power	100% Power

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Aug. 1978

Table 5.6-12

DHRS OPERATING AND SENSITIVITY EVALUATIONS

DHRS OPERATING CASE (1)	PEAK TEMPERATURE	NUMBER OF PHTS LOOPS	NUMBER OF IHTS LOOPS	NUMBER OF DHRS TRAINS	CONSERVATIVE DECAY HEAT	EVST HEAT LOAD	NA FLOW GPM	NAK FLOW GPM
Design Case	∿1120°F	3	0	2	Yes	Yes	560	800
Updated Design Case	∿1052°F	3	0	2	Yes [2]	Yes	560	800
Single Failure Evaluation #1	∿1055°F	2[3]	2	1	Nominal Values Used	No	600	600
Single Failure Evaluation #2	∿1137	2	0	1	Yes [2]	No	600	600
Delayed Start Design Case (∿ 45 minutes to DHRS operation)	∿1078	3	0	2	Yes [2]	Yes	560	800

(1) Conservatisms included in all cases: Shutdown at end of cycle at full power; no consideration of insulation, no consideration of SGS heat capacity.

(2) Decay heat values have been revised to reflect current conservative design values.

(3) The two PHTS loops assumption conservatively envelopes the assumption that a PHTS pump or pony motor fails to operate.

Table 5.6-13

DHRS EQUIPMENT LIST AND MATERIAL SPECIFICATIONS

	Component	ASME Section III Class	Material	Design Temperature (°F)	Design Pressure (psig)
	Overflow Vessel	1	SS	900	15
	Makeup Pump	1	SS	900	100
48	Overflow Heat Exchanger	1	SS	650	100
46	EVST Airblast Heat Exchanger	2	SS	650	100
	EVST Nak Exp Tanks	2	SS	650	100
	DHRS Nak Exp. Tank	2	SS	650	100
	EVST Nak Pumps	2	SS	650	100
	EVST Nak Diff. Cold Traps	2	SS	650	100
	Sodium Piping:				
	Overflow Line	1	SS	900	15
	Makeup Pump Suction	1	SS	900	15
46	Makeup Pump Discharge to Reactor	1	SS	900	100
26	Nak Piping	2	SS	650	100

TABLE 5.6-14

PACC HEAT TRANSFER UNCERTAINTIES

Parameter	Uncertainty	Heat Transfer Reduction, %
Air-side heat transfer coefficient	± 25%	12.68
Water-side heat transfer coefficient	± 20%	.95
Air-side fouling factor	+ .0015 / - .0005 hr-ft^2-F/Btu	.33
Water-side fouling factor	± .0005 hr-ft^2-F/Btu	1.45
Contact thermal resistance between the tubes and fins	± .001 hr-ft^2-F/Btu	2.26
Staggered/in-line flow ratio	+ 0.66 / - 0.34	2.72
Circumferential flow variation	± 30%	.67
Axial flow maldistribution	± 10%	1.05
Air-side system pressure drops		
- transition	± 10%	.18
- inlet louvres	+ 200% / - 50%	.90
- diffuser	+ 62% / - 50%	.36
- casing	± 10%	.04
- screen (perforated plate)	± 20%	.27
- coil assembly	± 30%	.72
- coil center	± 15%	.76
- expansion	± 15%	.09
- outlet louvres	+ 200% / - 50%	1.26

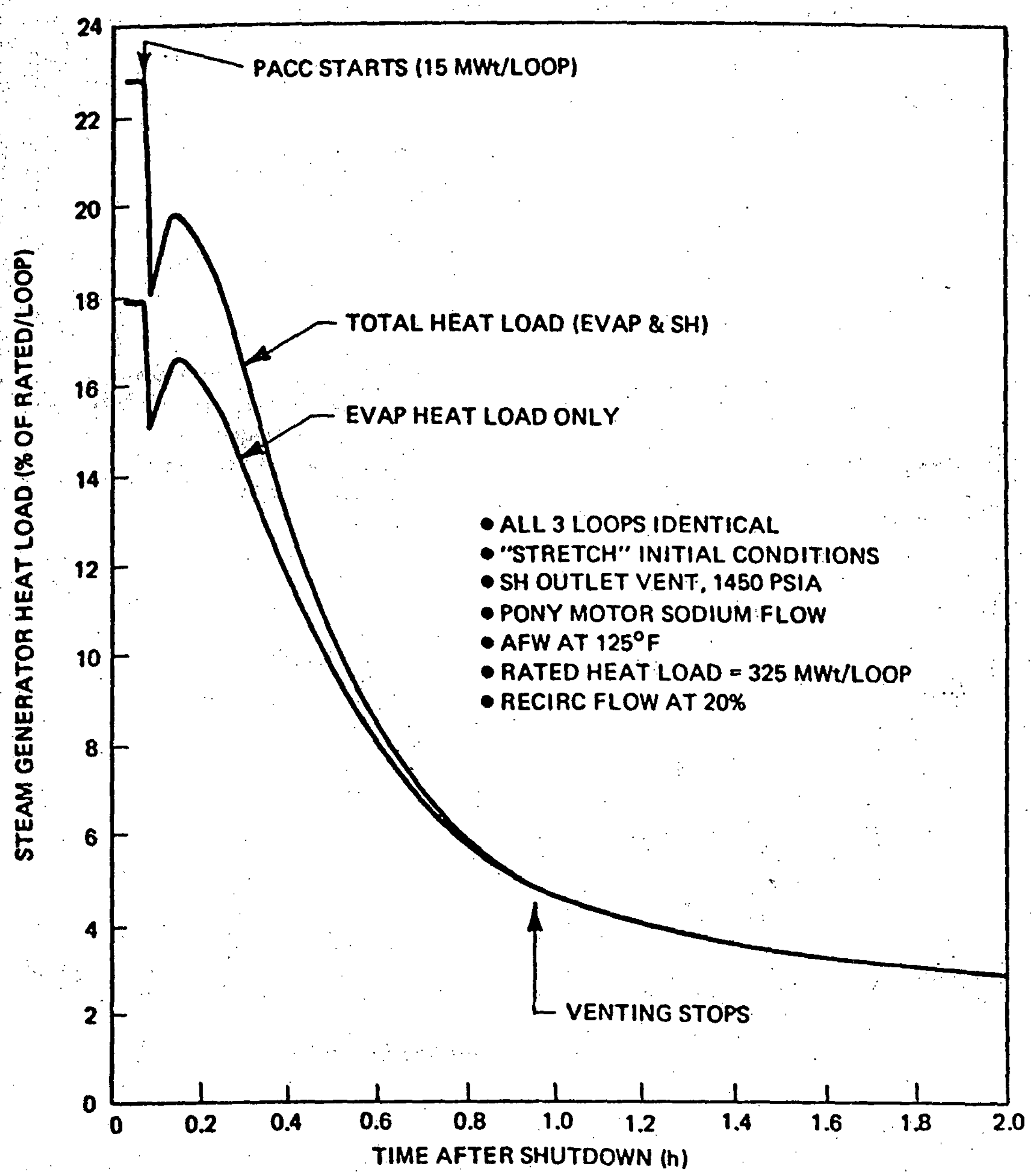

FIGURE 5.6-1 HEAT LOAD ON SGAHRS (U-18 EVENT)

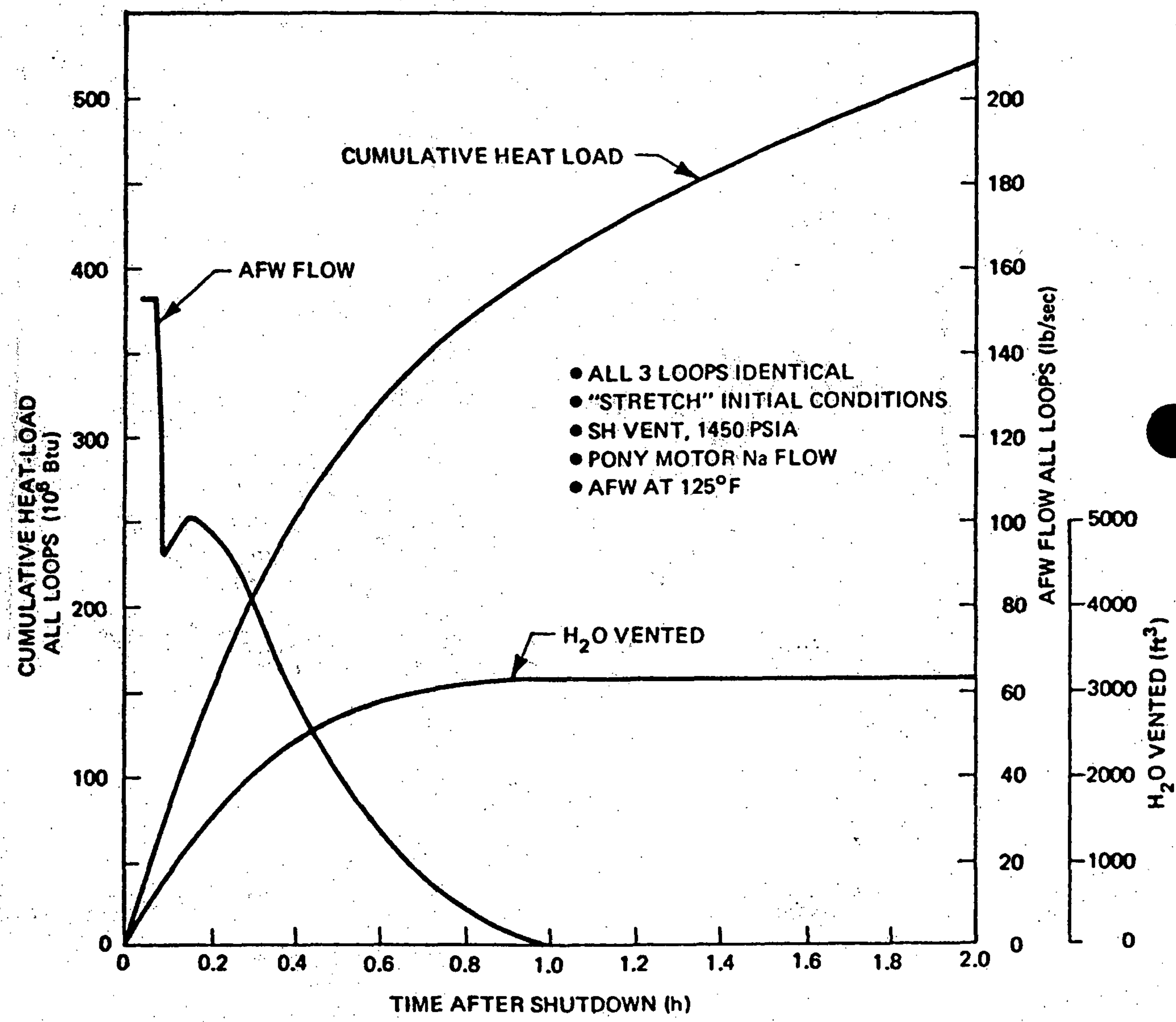

FIGURE 5.6-2 SGAHRS RESPONSE (U-18 EVENT)

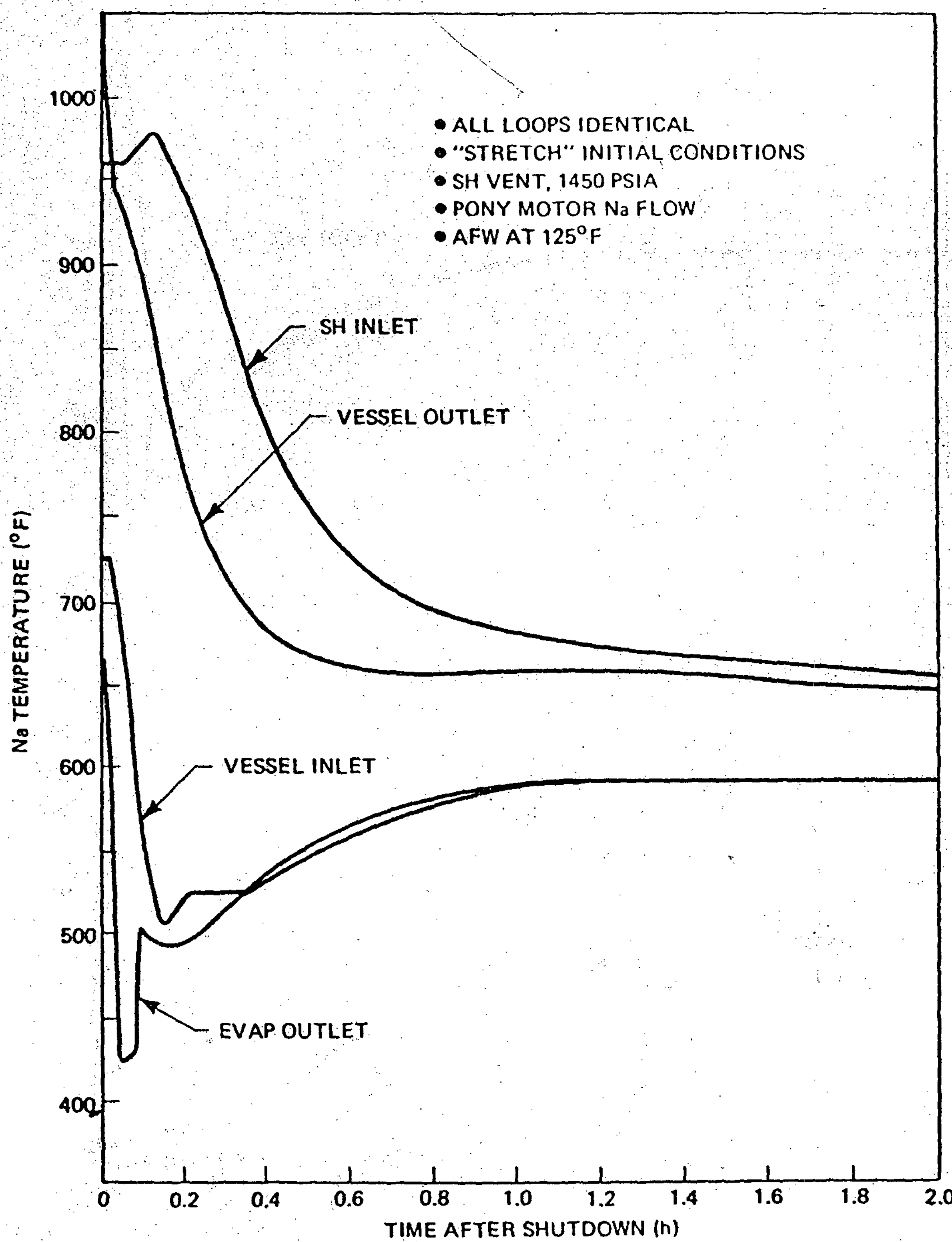

FIGURE 5.6-3 Na TEMPERATURE RESPONSE (U-18 EVENT)

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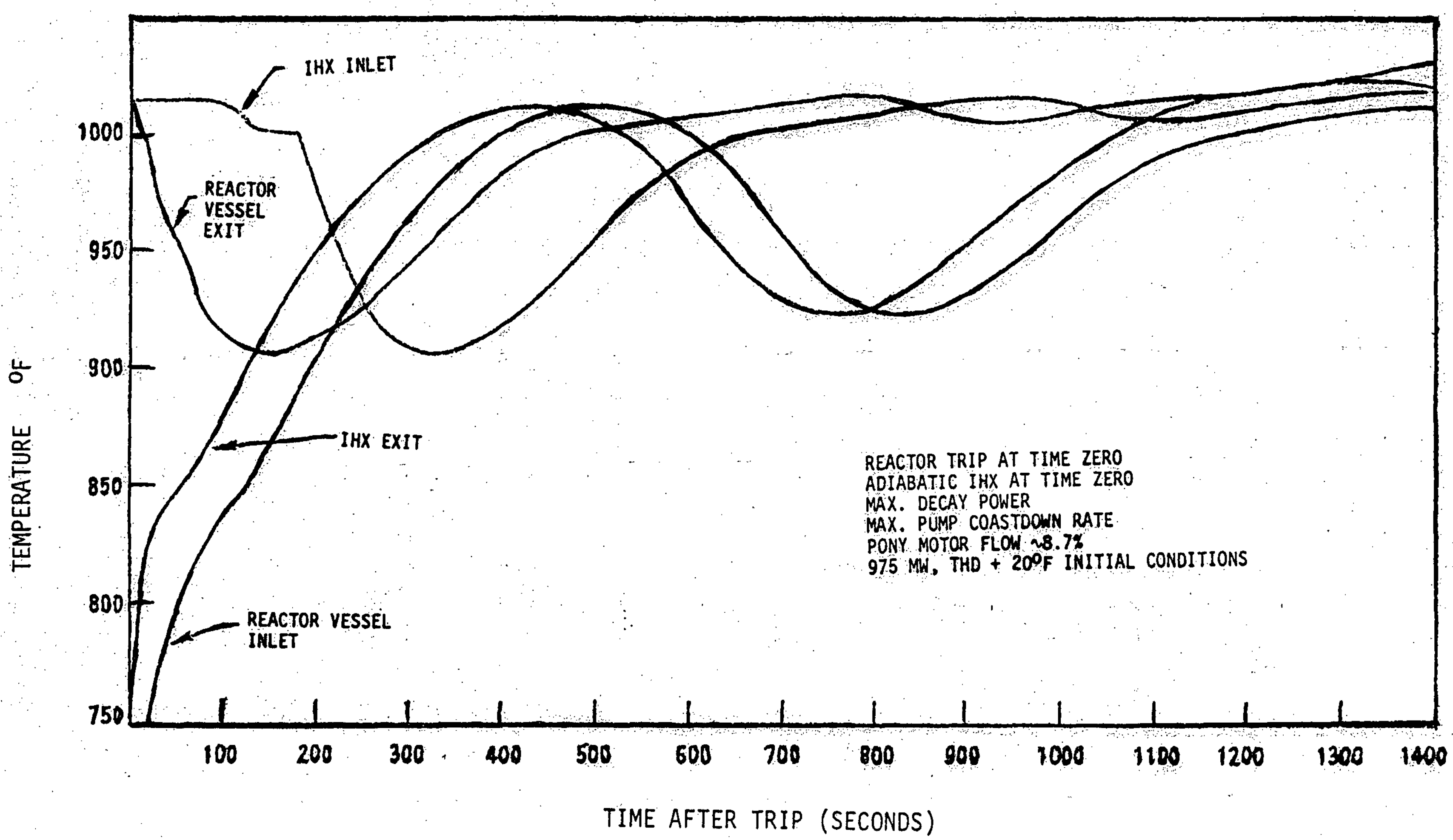

Figure 5.6-4. Temperature Estimates For Early Stages Of F-2 Transient (Primary Loop Temperatures)

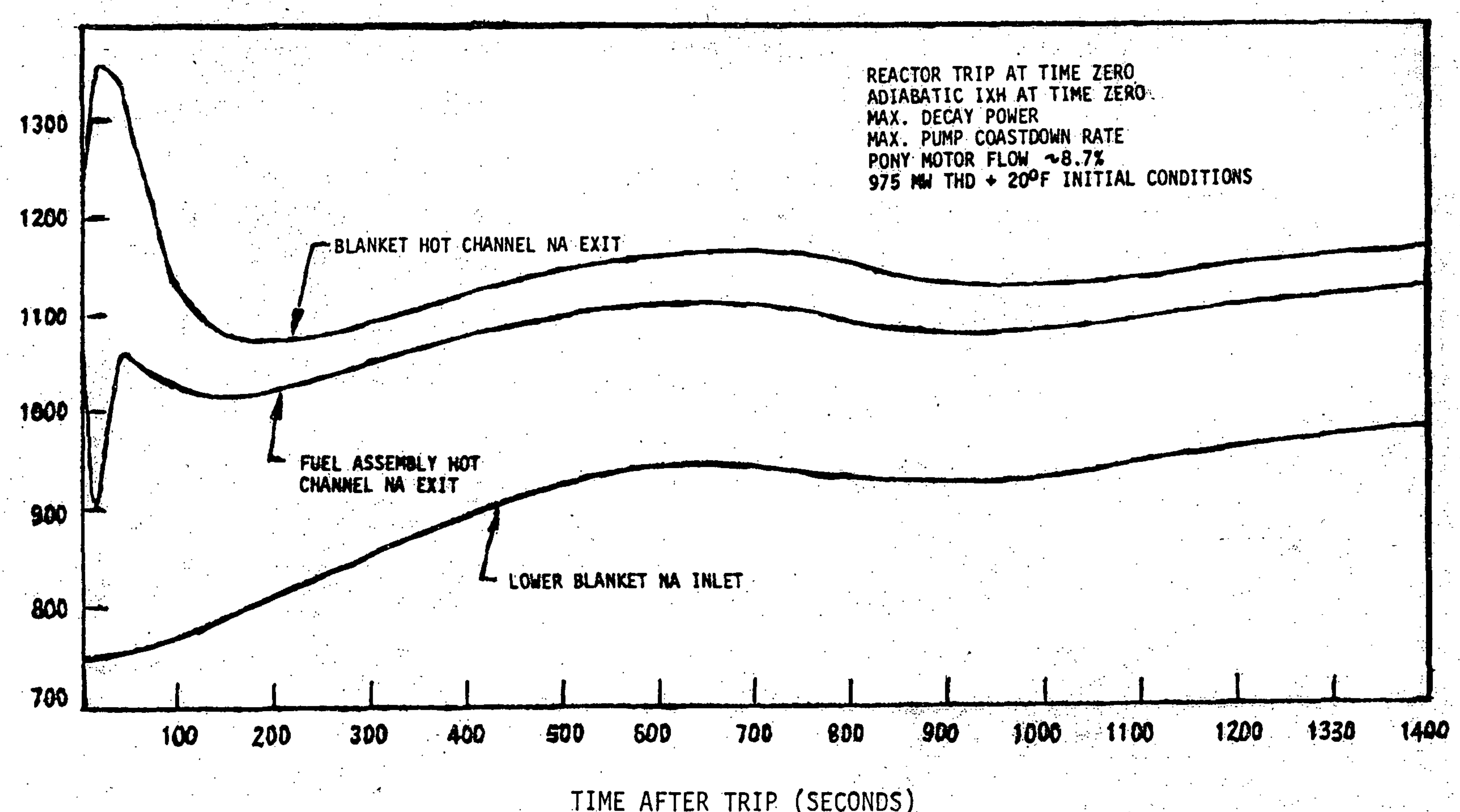

Figure 5.6-5. Temperature Estimates For Early Stages Of F-2 Transient (Reactor Temperatures)

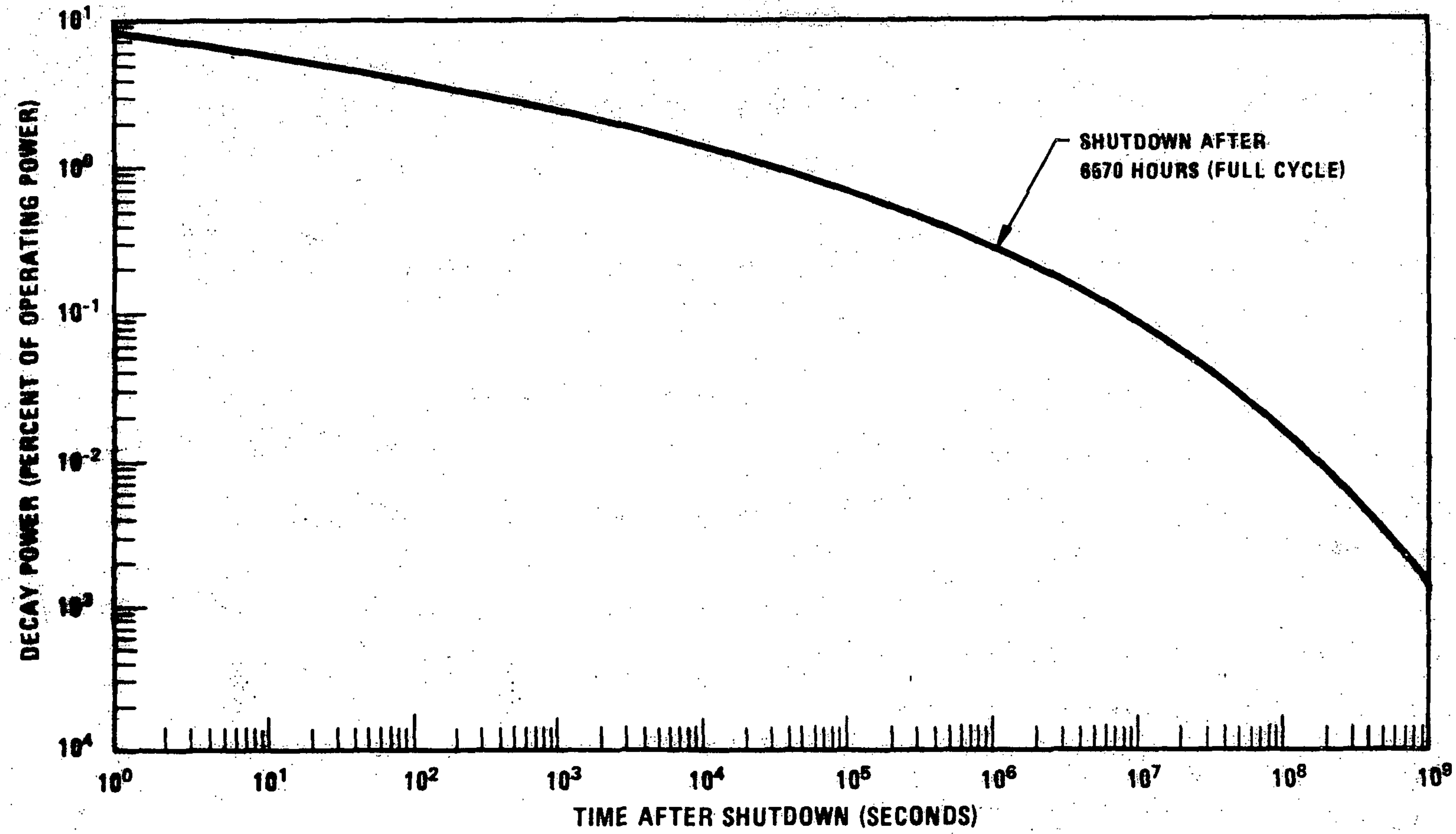

FIGURE 5.6-6 Decay Power after Shutdown Equilibrium Cycle Conditions Maximum Values.

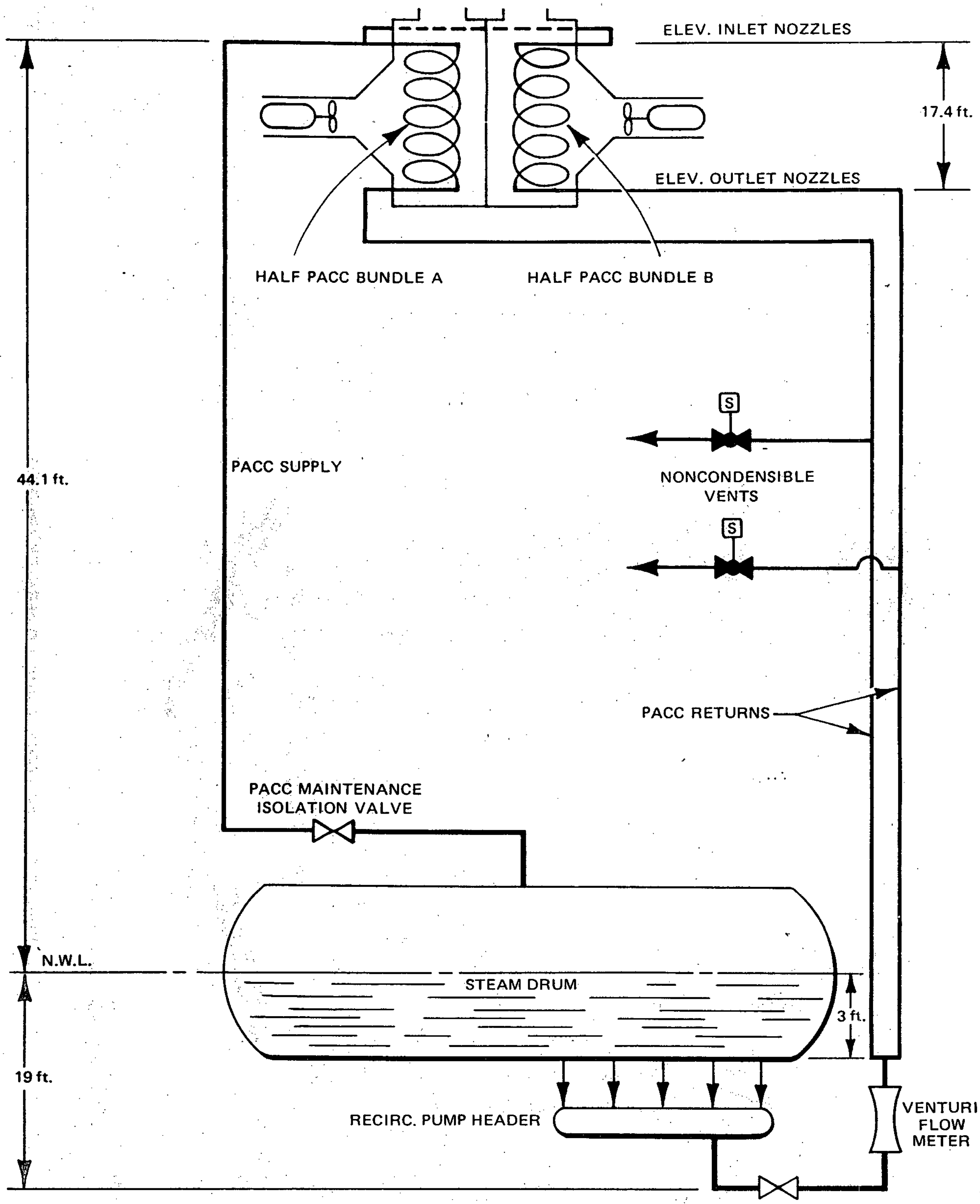

FIGURE 5.6-7 PACC CLOSED LOOP SCHEMATIC

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Feb. 1983

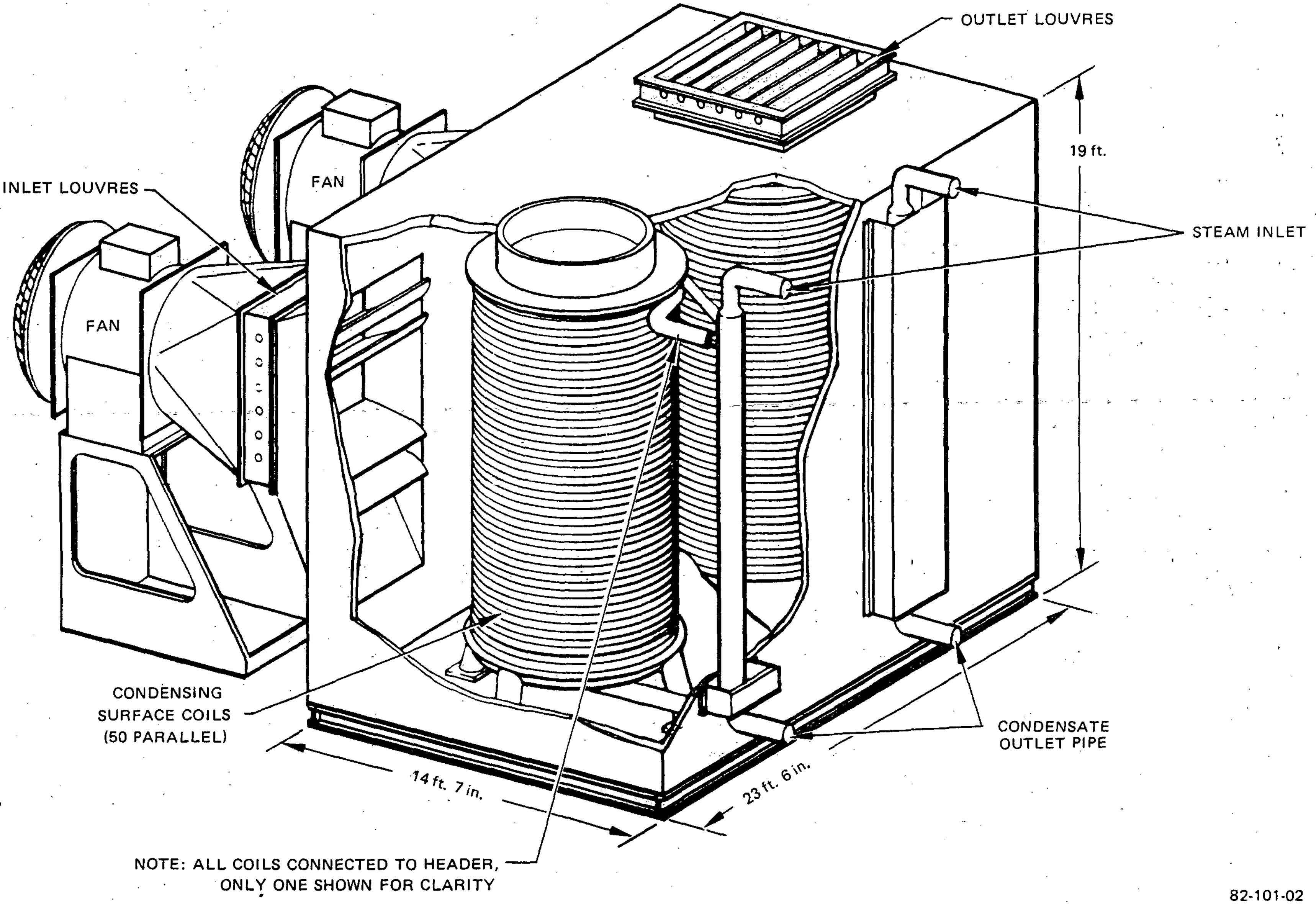

Figure 5.6-8 PROTECTED AIR-COOLED CONDENSER

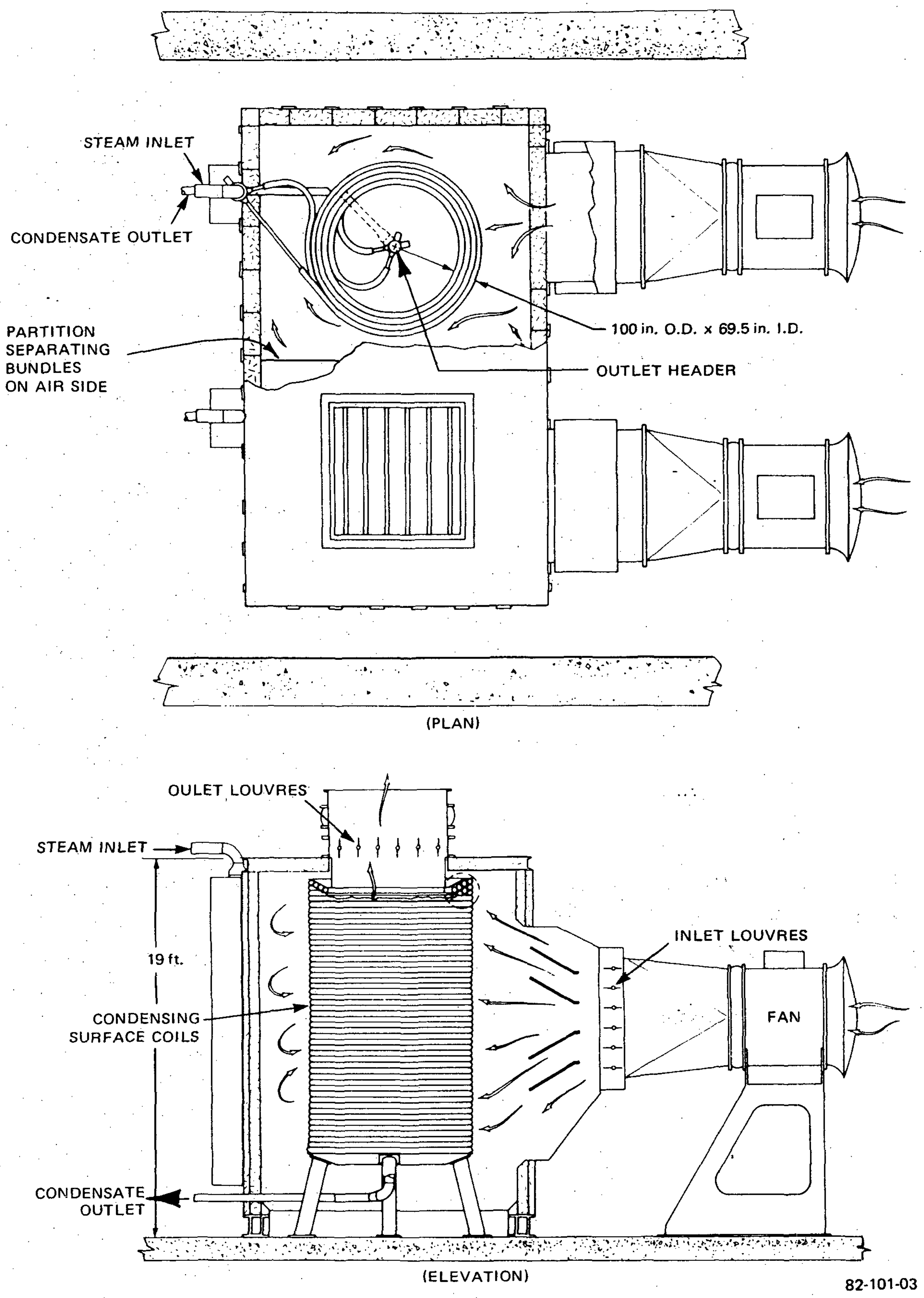

Figure 5.6-9 PROTECTED AIR COOLED CONDENSER

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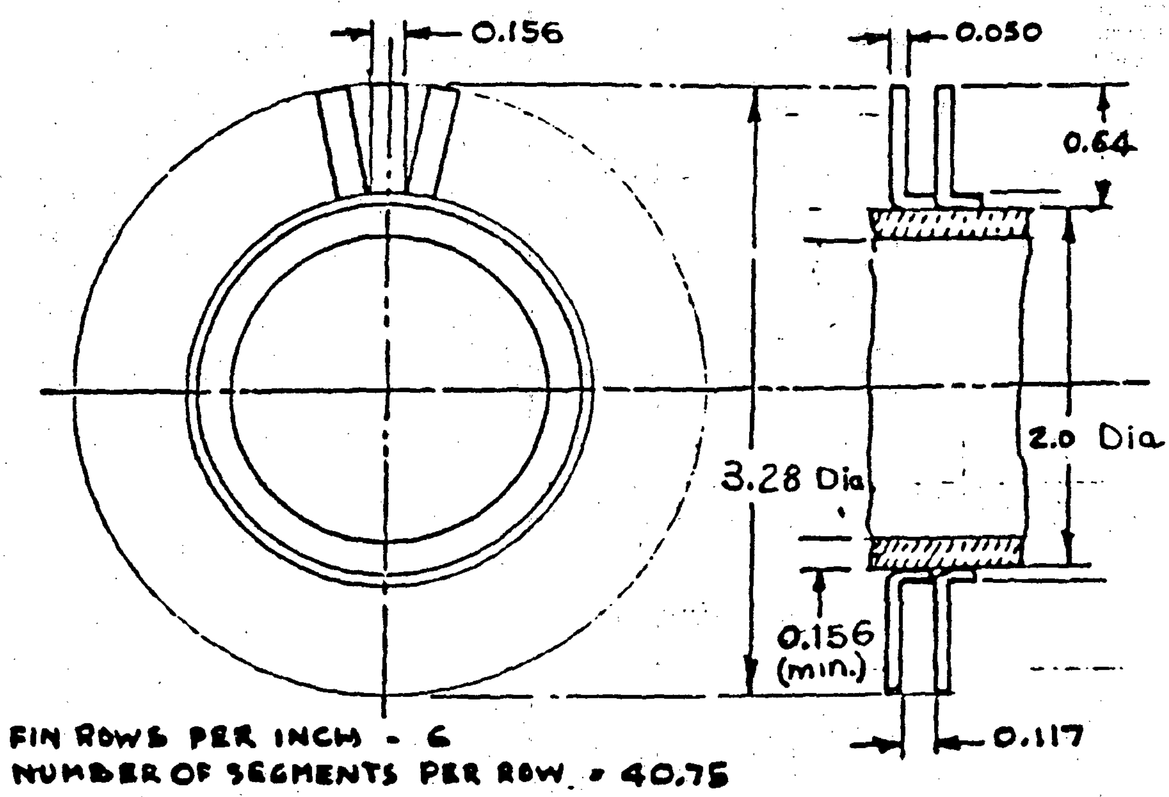

Figure 5.6-10 Nominal PACC Tube and Fin Geometry

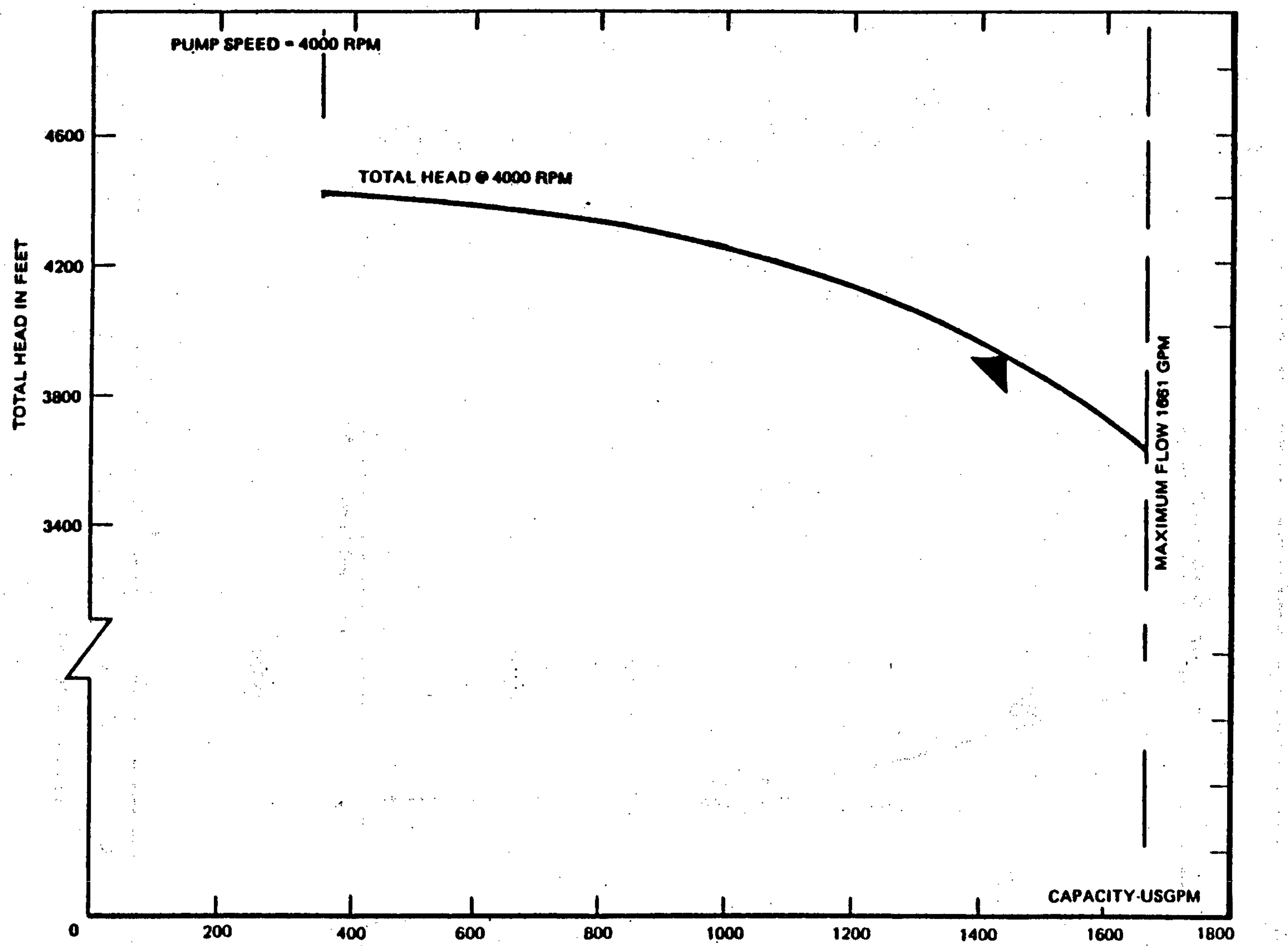

Figure 5.6-11 Turbine-Driven AFW Pump Characteristics

80-632-02

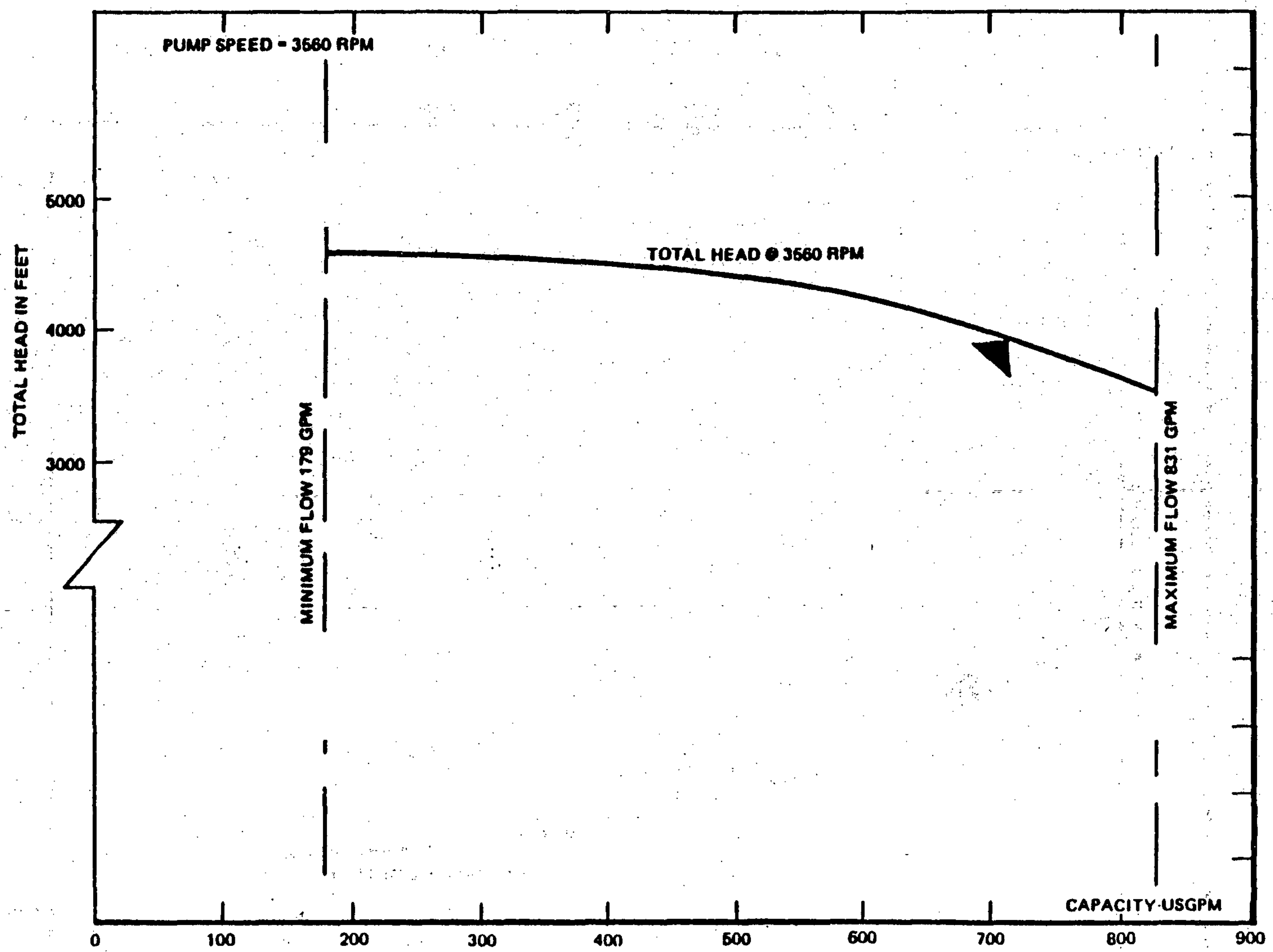

Figure 5.6-12 Motor-Driven AFW Pump Characteristics

80-632-03

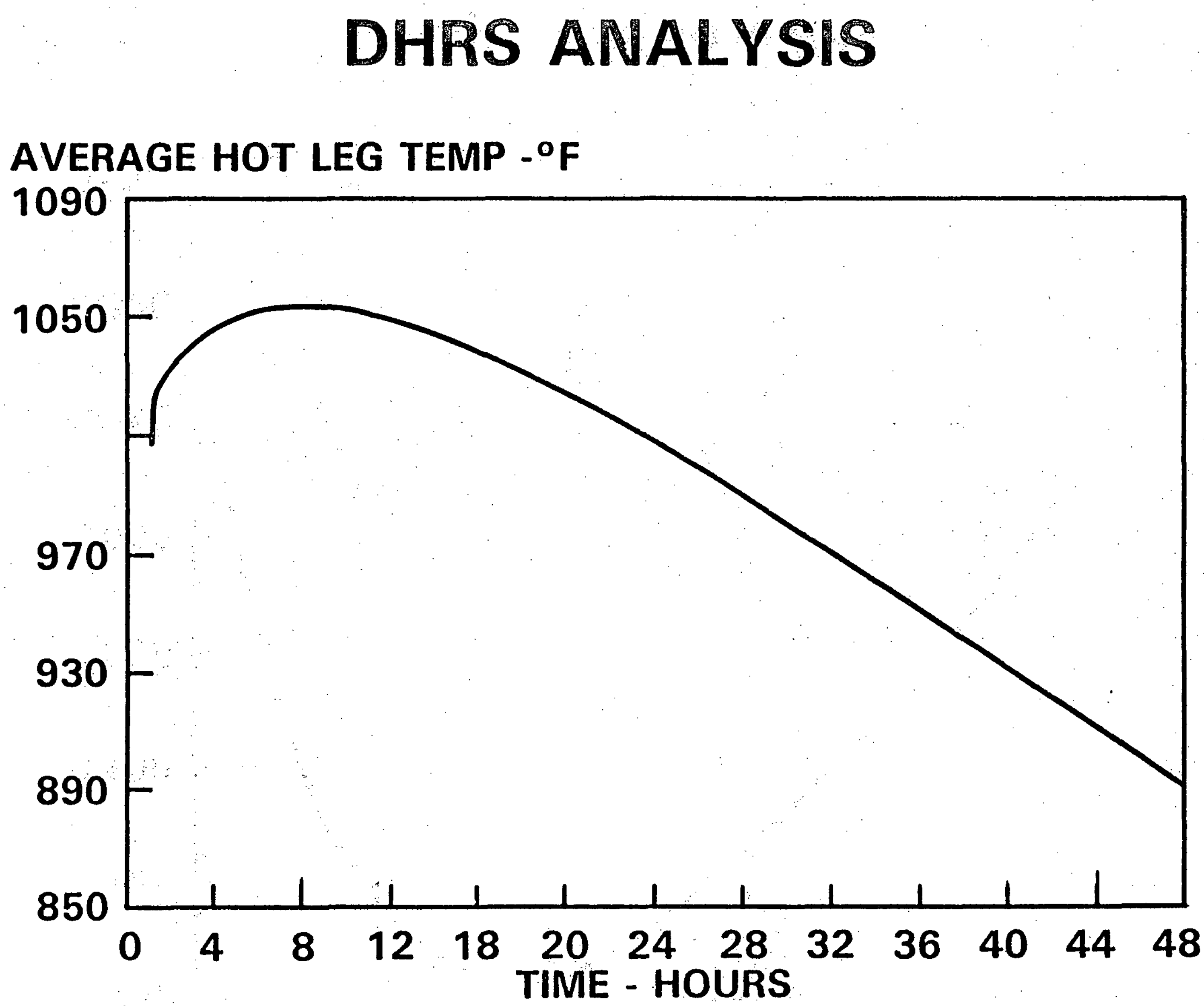

Figure 5.6-13 Results of Updated Base Case F-2 Analysis

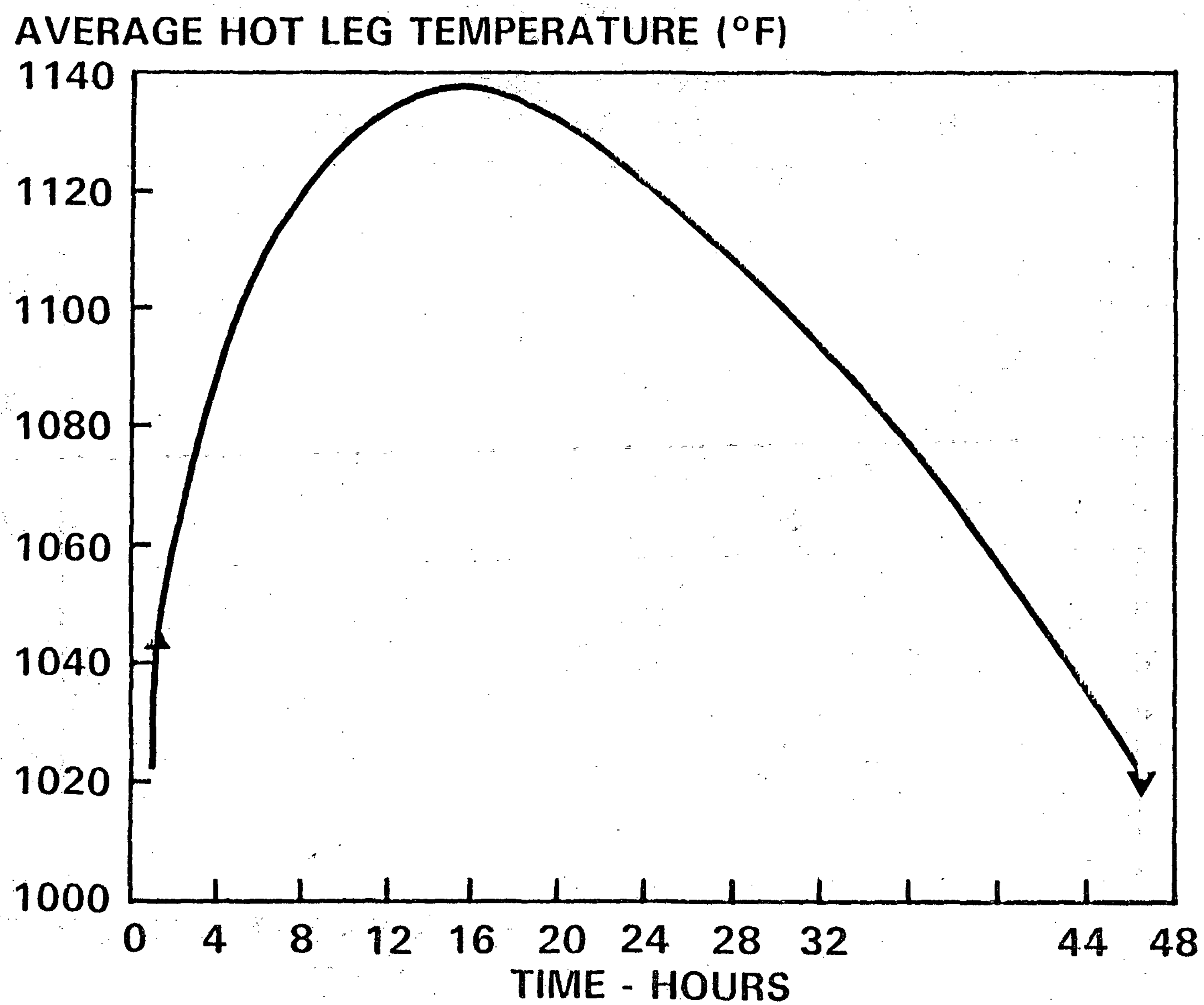

Figure 5.6-14 Results of DHRS Single Failure Analysis

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5.7 OVERALL HEAT TRANSPORT SYSTEM EVALUATION

5.7.1 Startup and Shutdown

5.7.1.1 Plant Startup

Plant startup basically encompasses the operations that take the reactor plant from either hot standby or refueling conditions to the operating conditions associated with 40% reactor thermal power. However, the initial plant startup also includes Reactor Heat Transport System (HTS) preheat and sodium fill as well as Steam Generator System (SGS) fill and heatup. Figures 5.7-A and 5.7-B present the typical startup characteristics of the HTS and SGS.

During initial startup, the HTS preheat operation consists of heating the HTS, along with the reactor systems, from a dry, argon filled, cold
35 iron condition (~70°F) to 400°F at a nominal rate of 3°F/hr.
This operation is accomplished utilizing the electrical resistance heaters which are part of the Piping and Equipment Electrical Heating and Control System (see Section 9.4). Prior and subsequent to the preheating, the reactor systems and HTS are evacuated and backfilled with argon to remove impurities; this function is performed utilizing the Inert Gas Receiving and Processing System (see Section 9.5). For the initial startup, the SGS is filled with water at approximately ambient temperature from the Condensate System and pressurized with nitrogen to meet system requirements. The SGS is then heated to 400°F by water/steam utilizing the Auxiliary Steam Boiler in addition to recirculation pump heat input.

With the reactor systems, HTS and SGS being maintained at 400°F, the reactor systems and HTS are filled with sodium at 400°F from the Auxiliary Liquid Metal System. During the sodium fill operation, the reactor systems and HTS are maintained at 400°F by the electrical resistance heaters and associated temperature controls; the SGS is maintained at 400°F utilizing heat from the Auxiliary Steam Boiler and recirculation pumps. Subsequent to the sodium fill, the appropriate cover gas pressures are established for the reactor systems and HTS, and the sodium pumps (primary and intermediate) are started on their pony motors and ~10% sodium flow is established; at this point, the reactor plant is at refueling conditions.

After completion of all pre-operational tests, the reactor systems, HTS and SGS are heated from refueling to hot standby conditions at an average rate not exceeding 50°F/hr. Heat for this operation is provided primarily by the sodium pumps. However, reactor heat will also be available subsequent to reactor criticality and the completion of the appropriate reactor physics tests. Due to the small amount of reactor power (~2%) that may be used during this operation, the heatup to 600°F will be essentially isothermal (spatially). When the reactor systems and HTS reach 600°F, and the steam drum pressure is established at 1450 psig, the reactor plant is considered to be at hot standby conditions. At this point the sodium pump speeds are reduced to 40% of full flow in preparation to increasing the reactor power.

The final phase of startup consists of taking the reactor plant from hot standby conditions to 40% reactor thermal power. Initially, the reactor is operated at a low power level (<5%) and the SGS blowdown and chemistry are

1

established. Also, at this time, the Main Steam and Steam Dump System (see Chapter 10.0) is warmed and the turbine is pre-warmed. The reactor power is then increased and the HTS primary and intermediate hot leg to cold leg ΔT's are formed by releasing steam from the superheater at a rate which will limit the primary hot leg temperature average heatup rate to 150°F/hr. This change in temperature will be accomplished by making discrete steps in reactor power level which will result in temperature changes at a rate limit of 1°F per second for 25 seconds every 10 minutes. As steam flow increases, the turbine is warmed up and loaded while dumping excess steam. Reactor power is increased to ∿15% by manual rod control and the turbine is brought up to synchronous speed. The generator is placed in parallel with the grid and the initial electrical load picked up; bypass steam is then secured. Reactor power is further increased and the HTS ΔT's are fully formed when the reactor thermal power, steam flowrate and primary and intermediate sodium flowrates are at 40% of their full power values. At this point, plant control is shifted from manual to automatic load follow control.

It should be noted that all startup as well as shutdown operations will be performed to preclude the possibility of sodium freezing in the HTS. 1

5.7.1.2 Plant Shutdown

Plant shutdown encompasses the operations that take the reactor plant from 40% reactor thermal power to either hot standby or refueling conditions. Figures 5.7-C and 5.7-D present the typical shutdown characteristics of the HTS and SGS.

At the 40% thermal power position, the plant control is shifted from automatic load follow to manual control. At this point, steam flow to the condenser via the bypass system is established in order to maintain proper steam pressure. Reactor power is then decreased at a rate which limits the primary hot leg cooldown to 150°F/hr, while the primary and intermediate sodium flows are maintained at flows corresponding to 40% power. The turbine is tripped at approximately 10% power and the generator output breaker opened. The reactor power is then further decreased to the decay heat level at a rate which limits the primary hot leg cooldown rate to 150°F/hr. During this phase of the shutdown, the superheater steam flow is also reduced and the Protected Air Cooled Condenser (PACC) of the Steam Generator Auxiliary Heat Removal System is placed into operation. When the PACC is removing all the decay heat, the steam bypass flow is secured.

The reactor and sodium pumps are not tripped until the HTS hot and cold leg temperatures are approximately equal (at the hot standby condition of 600°F) and the control rods are properly positioned; this minimizes the system temperature transients as the pumps shift from main to pony motors. If the plant is to be maintained at hot standby conditions, the system temperatures will be controlled by operation of the PACC and electrical resistance heaters. For those cases where the reactor plant temperatures are to be reduced to refueling conditions, the cooldown is accomplished by heat removal from the SGS steam drum by the PACC. This cooldown operation is controlled so as not to exceed the 50°F/hr limit. As in the case of hot standby conditions, the reactor plant is maintained at refueling conditions by operation of the PACC and the electrical resistance heaters.

5.7.2 Load Following Characteristics

The plant is designed to follow changes in load at a maximum rate of 3% of rated thermal power per minute over the range of 40% to 100% of rated thermal power. In addition, the plant is designed to respond to step load changes of ± 10% of rated thermal power. The plant follows these load changes in a smooth method and avoids tripping of the reactor and dumping of steam Load follow changes are implemented using the control configuration described in Chapter 7. The capacity to follow load changes is accomplished by automatic adjustment of rod positions in concert with changes to both primary and intermediate sodium flow. Nominal expected steady state temperatures over the range of 40% to 100% rated thermal power (975 MWt) for the primary sodium, intermediate sodium, steam, feed water, and evaporator water side inlet are presented in Figure 5.7-1. Nominal expected steady state flows over the range of 40% to 100% of rated thermal power for the primary sodium, intermediate sodium, feed water, and evaporator are presented in Figure 5.7-2. Note in Figure 5.7-2 that evaporator flow is essentially constant over the range of 40% to 100% of rated thermal power and that primary sodium flow varies linearly with power over the same range. These temperature and flow profiles, while not unique (due to possible variation in actual heat transport system component performance), are typical of those expected for the 40% to 100% power operation of CRBRP.

5.7.3 Transient Effects

To provide the necessary high degree of integrity for the equipment in the Heat Transport System, the transient conditions selected for equipment structural evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. The transients selected are representative of operating conditions which presently should be considered to occur during plant operation and are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of transients which, used as a basis for component structural evaluation, provide confidence that the component is appropriate for its application over the design life of the plant. Appendix B describes the events which result in transients on heat transport system components. Table 5.7-1 presents a summary of a preliminary selection of those transients.

Several events and examples of their affects on the components of the heat transport system are discussed and provided below to illustrate the transient behavior of the Heat Transport System. (More detailed discussions, including plots of temperature, flow, and pressure as a function of time, are included in Chapter 15):

a. Reactor Trip from Full Power

A reactor trip from full power results in the release of safety and/or control rods. Sodium pumps coast to pony motor speed. The continued transfer of heat results in rapid temperature reductions at the reactor vessel outlet, primary pump, IHX primary inlet, superheater sodium inlet and outlet, and evaporator sodium inlet. The primary hot leg temperatures drop about 300°F in 200 seconds while the superheater inlet sodium temperature drops about 200°F in the same time. Superheater outlet and evaporator inlet sodium temperatures fall about 170°F and then increase the same amount in a total of 100 seconds. The latter affect results from controlled dumping of steam to the condenser through the turbine bypass to maintain pressure at the turbine admission valve at 1450 psig to avoid lifting of safety or power relief valves. The transient is most severe when it occurs with minimum plant decay heat conditions since decay heat tends to slow the rate of temperature reduction. Figure 5.7-3 depicts the transient at the reactor vessel outlet where the rate of temperature change is the highest.

Substantial flow oscillations do not occur following reactor scram as discussed below.

The free surfaces in the reactor coolant system are 1) the free surface in the reactor vessel, 2) the free surfaces in each of the three primary pump tanks and 3) possibly a free surface at the high point of the primary side of the IHX in the annulus between the outer shell and the tube bundle support cylinder.

The only gas which would be under any significant pressure would be that which may accumulate in the IHX. The volume of this gas will deliberately be kept as small as possible by locating the vent line between the IHX and the pump tank as high as possible. The position of the vent line from the IHX is shown in Fig. 5.3-15 and shows the possible trapped gas volume to be extremely small. When the pump is tripped and the pressure in this gas space drops off rapidly from about 165 psia to approximately 15 psia, there will be an expansion of this gas and a lowering of the free surface. The volume of this gas when expanded will be small compared to the gas volume in the reactor vessel and pump tanks and as such, will not significantly affect sodium levels in either the pumps or reactor vessel.

The pump tank cover gas pressure during full flow conditions will be equal to or only slightly higher than the reactor vessel cover gas pressure (which is equalized with the overflow tank gas pressure through an equalization line). When the pumps are tripped, the level in the pump tanks will rise and submerge the stand pipe bubbler nozzle thereby cutting off communication of the pump cover gas with the rest of the cover gas in the primary system. The level rise in the tank is limited by the compression of the trapped gas. The increase in pump tank level is at the expense of the level in the reactor vessel but any oscillation in free surfaces in the pump and reactor vessel is precluded by providing a flow restriction between the pump hydraulics region and the pump tank which will critically damp any potential oscillation.

b. Uncontrolled Rod Movement

Control system malfunctions may cause uncontrolled control rod movement resulting in undesired insertion or withdrawal of one or more control rods. Uncontrolled insertion of a control rod, which could occur without a compensating reduction in sodium flows, results in rapid plant temperature reductions similar to those which occur from a reactor trip from full power.

Uncontrolled withdrawal of a control rod may occur under various initial conditions. If uncontrolled rod withdrawal occurs from 100% power, reactor vessel outlets, IHX primary inlets, and primary sodium pump temperatures will increase to values higher than normal and higher than from any other event. When power reaches 115%, a reactor trip occurs. Since temperatures just prior to reactor trip are higher than just prior to reactor trip from full power, a more severe transient will occur. Although the rate of temperature change is about the same as that for a reactor trip from full power, the extent of the transient is greater since it starts from a temperature about 60°F higher than that observed at 100% power. Figure 5.7-4 illustrates the nature of this transient.

Uncontrolled rod withdrawal during startup also results in an up temperature transient at the reactor vessel outlet although the transient occurs at a lower temperature than when the rod withdrawal starts from 100% power. Figure 5.7-5 depicts the transient initiated during startup.

c. <u>Operating Basis Earthquake (OBE)</u>

The operating basis earthquake results in reactive forces acting on the plant components as described in the Seismic Criteria Document. Five OBEs, each with 10 maximum peak response cycles, are assumed to occur over the design life of the plant. Four of these OBE's are assumed to occur during the most adverse Normal Operating Conditions determined on a component and design limit basis. The other one OBE is assumed to occur during the most adverse upset event determined on a component and design limit basis, and at the most adverse time in the upset event. Thus, the plant components are simultaneously exposed to the thermal effects of the thermal transients as well as the stresses of the OBE.

d. <u>Loss of Steam Generator Load</u>

Isolation and dumping of the water/steam sides of both evaporators and the superheater removes the load from that loop. This results in up temperature transients on the steam generator modules, the intermediate cold leg, the IHX intermediate inlet, the IHX primary outlet, and the reactor vessel inlet. The ensuing reactor trip then causes down temperature transients on these components. The intermediate cold leg temperature increases approximately 350°F in 400 seconds; then decreases approximately 220°F in 300 seconds. This transient is then transported to the IHX primary outlet and reactor vessel inlet. Figures 5.7-6 a-k presents the resulting transient at the intermediate sodium pump, core & steam generators.

e. <u>Inadvertent Opening of Superheater Outlet Power or Safety Relief Valve</u>

This event results in a large increase in load without an accompanying increase in reactor power or sodium flows. It occurs when a super-heater relief valve inadvertently opens to increase steam flow from 40% to 100%. The event results in a reactor trip but overcooling occurs due to the open relief valve. The steam generators, inter-mediate cold leg, IHX intermediate inlet, primary cold leg and reactor vessel inlet drop in temperature about 150°F in 100 seconds. The reactor vessel outlet, primary hot leg, and IHX primary inlet drop in temperature about 200°F in 75 seconds. Figure 5.7-7 depicts the transient at the intermediate pump.

f. Primary Pump Mechanical Failure

Primary pump mechanical failure involves the instantaneous stoppage of the impeller of one primary pump due to such reasons as seizure or breakage of the shaft or impeller. Flow in the affected loop rapidly goes to zero and a reactor trip occurs almost immediately after seizure based on primary to intermediate flow ratio. The event is characterized by a down transient in the intermediate hot leg and a check valve slam in the primary cold leg of the affected loop. The down-temperature transient in the intermediate hot leg results from the sudden loss of primary sodium flow while intermediate sodium flow continues. The intermediate hot leg temperature drops 300°F in about 200 seconds. The check valve slam, which results from the check valve being forced shut by reverse flow from the reactor vessel, results in significant pressure fluctuations at the reactor vessel inlet, the check valve, and the IHX primary outlet. Figure 5.7-8 presents the temperature transient at the superheater inlet while Figure 5.7-9 depicts the pressure effects of the check valve slam at the check valve inlet and outlet.

g. Saturated Steam Line Rupture

A rupture of the saturated steam line between the steam drum and the superheater inlet isolation valve results in immediate cessation of superheater steam flow in that loop and initiation of a reactor trip. The superheater rapidly becomes isothermal at the sodium inlet temperature due to the loss of cooling. Sodium leaving the evaporators of the affected loop initially drops in temperature due to over cooling as the water flow increases and flashes to atmospheric pressure through the steam drum. Then, as the loop blows dry through the rupture, evaporator sodium temperature rapidly increases to the superheater inlet temperature. This transient is the most severe that the evaporator and intermediate pump experience. The transient is propagated through the intermediate cold leg and results in similar severe transients on the intermediate pump, the IHX intermediate inlet, the IHX primary outlet, the primary cold leg and check valve, and the reactor vessel inlet nozzle. Subsequently, these components experience down temperature transients as a result of the reactor trip. Intermediate cold leg temperature drops 200°F in about 60 seconds and then increases 500°F in about 100 seconds. Figure 5.7-10 illustrates the transient at the intermediate pump.

h. Loss of One Primary Pump Pony Motor with Failure of the Check Valve in that Loop to Shut

This event occurs subsequent to a shut down or reactor trip and results in reverse flow in the affected primary loop as a result of the head developed by the two operating pumps. The reverse flow of primary sodium at reactor vessel inlet temperatures results in rapid down temperature transients at the IHX primary inlet, the primary pump, and the reactor vessel outlet nozzle of the affected loop. A core temperature increase occurs as a result of the bypassed flow. Primary hot leg temperature drops 425°F in about 150 seconds. Figure 5.7-11 depicts a typical transient.

5.7.4 Evaluation of Thermal Hydraulic Characteristics and Plant Design

Heat Transport System Design Transient Summary

The heat transport system design transients for the individual heat transport system components are described in Appendix B. Table 5.7-1 presents a preliminary summary listing of design transient events as well as the frequency of each event assigned to the reactor vessel, IHX, primary pump, intermediate pump, primary check valve, evaporator and super-heater.

It should be noted that the assigned frequency for a particular event varies among the components in some cases. This is the result of the method used in establishing the design transients. The events listed in Appendix B are the result of grouping less severe events under more severe events and applying the total frequency of all events in the group to the most severe event in the group. This approach was applied separately to each component so that the most transients discussed in Appendix B (where a particular transient applies to more than one component) do not have the same frequency applied to each component. This approach was required because each event does not result in the same transient effect on each component.

5.7.5 Natural Circulation

Although the HTS normally transfers reactor decay heat to the steam generator system with forced sodium circulation in at least one of the three main heat transport loops using pony motor drives on the main sodium pumps, the PHTS, IHTS and SGS and SGAHRS have been designed to assure adequate decay heat removal by natural circulation.

The objective to design the plant with natural circulation capability has influenced component and piping arrangements and component hydraulic requirements (See Sections 5.3, 5.4, 5.5 and 5.6.1.). The relative locations of major heat transfer components were selected to provide the necessary thermal driving head for adequate loop flows. Requirements on pump coastdown and locked rotor characteristics and on the PHTS cold leg check valve pressure drop versus flow characteristic were established. The elevations of PHTS and IHTS components are shown in the HTS Hydraulic Profile on Figure 5.1-3. The elevations of the steam drum and the SGAHRS components are shown in the SGAHRS Hydraulic Profile on Figure 5.1-6.

5.7.5.1 Preliminary Assessment

A preliminary assessment of the CRBRP natural circulation decay heat removal capability was made in 1976 for several cases including: heat removal through three loops following three loop rated power operation, two loops following three loop rated power operation, and two and one loops following 2/3 rated power operation on two loops. (See Reference 1) The principal case is the first of these cases (3 loop natural circulation following 3 loop rated power). It has been analyzed on the basis of the following event description.

> From initial conditions of three loop full power operation, complete loss of all electrical power to the main motors and pony motors in the HTS is assumed, initiating a PPS trip in 0.5 seconds. Immediately upon loss of electrical power, the sodium pumps coast down and stop (in approximately 55 seconds) when the normalized speed has decreased to approximately 1% and inertia cannot overcome internal friction. Flow in all sodium loops is then maintained by the natural circulation thermal driving heads. Auxiliary feedwater flow at 70°F is available from the auxiliary feedwater portion of the SGAHRS in approximately 30 seconds. The turbine driven auxiliary feed pump takes suction from the protected water storage tank to maintain drum levels during the transient. Final decay heat rejection is through SGAHRS via the superheater outlet and steam drum vent valves. Heat removal via the exhaust from the auxiliary feed pump turbine was conservatively neglected.

The results of this conservative analysis demonstrate that the transient follows a well-defined pattern and that sufficient flow and therefore decay heat removal exists to keep the core temperatures within design limits.

Analysis of the other cases indicated that the temperature transients in the core would not be significantly different from the three-to-three case for the first 600 seconds. Beyond this time, the magnitude of the core temperatures for any of the cases is not high enough to be of concern.

The analyses were limited in scope to five specific cases (four cases initiated from full power and one from 40% power) analyzed for a maximum of about 17 minutes. The trends shown at 17 minutes (steady flow, decreasing cladding temperature, etc.) taken with the various areas of conservatism indicate that the extrapolation of the short term conclusions on their behavior to the long term are justified.

5.7.5.2 Current Assessment

Changes in plant design and improvements in the analysis methodology and input data, necessitated an update of the preliminary assessment of the CRBRP natural circulation capability. Because the preliminary assessment indicated that there is very little difference in the four cases analyzed, only the principal case of three loop natural circulation from initial three loop full power operation was analyzed.

The two significant changes in plant design from the preliminary assessment are: 1) a change from a core design having a relatively "homogeneous" central core region of fuel assemblies surrounded by blanket assemblies, to a relatively "heterogeneous" design, which includes blanket assemblies interspersed in the central fuel region, in addition to the outer or radial blanket region (PSAR Chapter 4.0, Amendment 54), and 2) a change to the auxiliary feedwater sparger location in the steam drum from a submerged location to one in which the auxiliary feedwater is sprayed into the steam space of the drum. The most significant change in input data as a result of component testing was to the pump characteristics. These changes are best seen in a comparison of coastdown times. The time to stop from design speed for the primary pump was increased from 55 seconds in the preliminary assessment to 131 seconds in the current assessment.

The results of the current assessment, as well as a description of the model and input changes from the preliminary assessment, are presented in Reference 2. This current assessment confirmed the conclusions of the preliminary evaluation that the CRBRP design provides adequate decay heat removal capability by natural circulation. A margin to boiling exceeding 150°F for the peak core temperatures is predicted. Selected parameters for this analysis are presented in Figures 5.7-12 and 5.7-13. Figure 5.7-12 presents the HTS hot leg temperatures and Figure 5.7-13 presents the HTS flows.

5.7.5.3 Verification of Natural Circulation

The emphasis for verification of natural circulation decay heat removal capability of CRBRP is on testing at the system and component levels during the transition to and operation in the natural circulation mode. This testing has demonstrated the validity of the methodology for predicting CRBRP plant-wide natural circulation capability. The plant implemented to achieve this verification is documented in Reference 3. The goal of the plan is to verify the conclusion of the preliminary assessment and current assessment that adequate decay heat removal capability by natural circulation exists for CRBRP. The basis of this plan is threefold.

- Prior to plant startup testing, conclusions concerning plant wide natural circulation can be based on analysis.

- Models verified on an individual component basis (with the correct integration of these individual models confirmed through prototypic system data) provide a valid basis for these analyses.

- The use of CRBRP methodology to evaluate natural circulation testing in FFTF and EBR-II verifies the integrated methodology and provides a significant contribution to the verification of the models used in natural circulation analysis.

A sufficient portion of the verification program has been completed to verify that the CRBRP natural circulation prediction methodology is conservative. Additional verification activities are planned to provide support for further refinement of the model.

Comparisons of data from EBR-II natural circulation tests and comparisons of predictions (both pre-test and post-test) and data from FFTF natural circulation tests were made to evaluate and to demonstrate the systems modeling conservatism in the system analysis codes. These tests are a key element in the natural circulation verification program.

The results of EBR-II tests provide a basis for verification of parts of the DEMO, COBRA-WC, and FORE-2M models. This analysis of EBR-II and comparisons with measured data serve to verify the whole-plant modeling capability of DEMO. The conclusion of this study was that the models accurately predict natural circulation transients (Reference 6). Verification of the COBRA-WC and FORE-2M models from the EBR-II tests centered on the results of detailed flow and temperature data collected from two instrumented core assemblies (Reference 7 and 8). The conclusion of this study was that the models conservatively predict natural circulation transients.

A description of the DEMO/FFTF modeling methodology and comparisons of the natural circulation test results with both the pre-test and post-test predictions for the primary and secondary sodium systems are presented in Reference 4. A description of the verification of the CRBRP natural circulation core analyses methodology with data from FFTF natural circulation tests is provided in Reference 5. The FFTF tests and predictions demonstrated that the DEMO, COBRA-WC and FORE-2M codes used for CRBRP analysis provide adequate (conservative) predictions of the dynamic response of a similar, loop-type, sodium cooled reactor.

Based on successful completion of the Natural Circulation Verification Program, it is concluded that the methodology, the analysis codes, and the modeling techniques have been verified, and with their use, the response of CRBRP to natural circulation will be adequately characterized. In order to further demonstrate the natural circulation capability, a natural circulation acceptance test will be performed to confirm the as-built system's thermal and hydraulic analyses.

5.7.5.4 Natural Circulation Testing in CRBRP

The startup test program for CRBRP will include tests designed to quantify the natural circulation decay heat removal capability in the plant. The objective of the tests is to confirm the thermal-hydraulic computer codes which have been used to predict the natural circulation behavior. The natural circulation test program will provide information to:

1. Ensure adequate prediction of natural circulation capability (transition as well as steady-state flow and temperatures).

2. Ensure adequate prediction of the effect on temperatures and flow caused by variations in the heat sinks available (e.g., venting, PACC's, two heat transport system loops).

References:

1. R. R. Lowrie and W. J. Severson, "A Preliminary Evaluation of the CRBRP Natural Circulation Decay Heat Removal Capability," WARD-D-0132, Rev. 0, March 1976.

2. W. J. Severson, et al., "Summary Report on the Current Assessment of the Natural Circulation Capability with the Hetergeneous Core," WARD-D-0308, February 1982.

3. "Verification of Natural Circulation in Clinch River Breeder Reactor Plant - An Update," May 1982 (Provided to NRC in letter HQ:S:82-042, dated June 10, 1982).

4. A. C. Cheung, "DEMO Post-Test Analysis of the FFTF Transient Natural Circulation Tests," WARD-NC-94000-6 (Provided to NRC in letter HQ:S:82-068, dated July 15, 1982, Docket No. 50-537).

5. A. C. Cheung, R. D. Coffield, K. D. Daschke, and Y. S. Tang, "Verification of the CRBRP Natural Circulation Core Analyses Methodology with Data from FFTF Natural Circulation Tests, "CRBRP-ARD-0310, June 1982 (Provided to NRC in letter HQ:S:82-068, dated July 15, 1982.

6. R. Schurko et.al., "Comparisons of DEMO Calculations to EBR-II Secondary Loop Natural Circulation Measurements," WARD-NC-94000-2, April 1980.

7. E. U. Khan, W. A. Prather, T. L. George and J. N. Bates, "A Validation Study of the COBRA-WC Generalized Parameters for Steady State Analysis," PNL-4128, December 1981 (Available from Technical Information Center).

8. Y. S. Tang, R. D. Coffield, and F. C. Luffy, "Verification of FORE-2M Computer Code, Part II, Comparison with EBR-II Natural Circulation Experiments," WARD-D-0315, July 1982.

TABLE 5.7-1

PRELIMINARY SUMMARY OF HEAT TRANSPORT SYSTEM DESIGN TRANSIENTS

	DUTY CYCLE EVENT NUMBER[1]	Event Title	Frequency (Lifetime): Reactor Vessel	IHX	Primary Pump	Inter. Pump	Primary Check Valve	Evap.	Super-heater
	N-1	Dry system heatup and cooldown, sodium drain and fill	5	13	13	30	13	30	30
		Startup							
	N-2a	1. From refueling	140	140	140	140	140	214	214
	N-2b	2. From hot standby	886	728	728	728	728	668	668
		Normal shutdown							
	N-3a	1. To refueling	140	60	60	60	60	60	60
	N-3b	2. To hot standby	329	250	260	260	260	210	210
		Load follow							
42	N-4a, N-4b[2]	1. Loading (total of 40-100% range and 80-100% range)	57,426	57,114	57,124	57,128	57,122	57,139	57,139
	N-4a, N-4b[2]	2. Unloading (total of 40-100% range and 80-100% range)	56,480	56,610	56,610	56,610	56,610	56,550	56,550
41	N-6	Steady state temperature fluctuations	30×10^6	30×10^6	30×10^6	--	--	30×10^6	30×10^6
	N-7	**Steady state pressure fluctuations**	10^{10}	10^{10}	10^{10}	10^{10}	10^{10}	10^{10}	10^{10}
		Reactor trip from full power							
	U-1a	1. Normal decay heat	180	268	258	--	--	--	--
41	U-1b	2. Minimum decay heat	97	144	193	454	400	315	315
		Uncontrolled control rod movement							
	U-2a	1. Rod insertion from 100% power	--	10	10	--	--	10	10
	U-2b	2. Rod withdrawal from 100% power	22	18	14	--	32	10	10
41	U-2d	3. Rod withdrawal from startup	20	20	20	--	--	20	20
	U-2e	4. Plant loading at maximum rod withdrawal rate	10	10	10	10	10	10	10
	U-4a	Intermediate pump control failure	--	7	--	--	--	--	--
41	U-11a	Water side isolation and dump of both evaporators and superheater of a loop	153	15	--	6	41	15	15
	U-20b	Turbine bypass valve fails open following reactor trip	--	5	5	5	--	--	--
41	U-18	Loss of offsite power supplies	35	9	11	19	21	84	84
	U-21a	Evaporator outlet relief valves open	--	--	--	--	--	3	3
	U-11a[3]	Unaffected loops for water side isolation and blowdown-Superheater and (2) Evaporators	--	--	--	--	--	116	116

Notes:
1. Event number and description are as found in Appendix B.
2. Event N-4a describes loadings and unloadings over the range of 40-100% power. Event N-4b describes loadings and unloadings (referred to in the duty cycle as load fluctuations) over the range of 80-100% power.
3. Refers to transient eff s on the two unaffected loops when this event occurs in one loop.

TABLE 5.7-1 (continued)

PRELIMINARY SUMMARY OF HEAT TRANSPORT SYSTEM DESIGN TRANSIENTS

DUTY CYCLE EVENT NUMBER[1]	Event Title	Frequency (Lifetime) Reactor Vessel	IHX	Primary Pump	Inter. Pump	Primary Check Valve	Evap.	Superheater
U-11b	Water side isolation & blowdown of evaporator module	--	--	--	--	--	7	7
U-11b	Adjacent evaporator during water side isolation and blowdown of evaporator	--	--	--	--	--	9	9
U-21a	Adjacent evaporator outlet relief valves open	--	--	--	--	--	3	3
E-9a	Superheater isolation & blowdown-outlet valve open	--	--	--	--	--	Note 4	Note 4
E-14	Inadvertent dump of intermediate sodium	--	--	--	--	--	Note 4	Note 4
OBE[5]	Operating basis earthquake	5	5	5	5	5	5	5
E-16	Three loop natural circulation	Note 4						
U-21b	Inadvertent opening of superheater outlet power or safety relief valve	42	19	24	14	26	13	13
U-23	Inadvertent opening of evaporator inlet dump valve	--	33	--	37	--	--	--
U-8	Primary pump pony motor failure	*15	5	--	--	--	5	5
E-1	Primary pump mechanical failure	Note 4	Note 4	--		Note 4	--	--
E-5	Loss of one primary pump pony motor with failure of check valve in that loop to shut	Note 4	Note 4	Note 4	--	--	Note 4	Note 4
E-6	Design basis steam generator sodium/water reaction	--	Note 4	--	Note 4	--	Note 4	Note 4
E-7	One loop natural circulation (from initial two loop operation)	Note 4	Note 4	--	--	--	Note 4	Note 4
E-15	DHRS Activation 24 Hours After Scram	2	2	--	2	2	2	2
E-16	Three loop natural circulation	Note 4	Note 4	Note 4	--	--	Note 4	Note 4

Notes: 4. Each component, or part of a component, must accommodate 5 occurrences of the most severe emergency transient for that component or part of a component (one every 6 years) and two consecutive occurrences of the most severe event (or of unlike events if consecutive occurrences of unlike events provide a more severe effect than two occurrences of the most severe event).

5. See Paragraph 5.7.3(c)

TABLE 5.7-1 (continued)

PRELIMINARY SUMMARY OF HEAT TRANSPORT SYSTEM DESIGN TRANSIENTS

	DUTY CYCLE EVENT NUMBER[1]	Event Title	Frequency (Lifetime) Reactor Vessel	IHX	Primary Pump	Inter. Pump	Primary Check Valve	Evap.	Super-heater	
	F-1	Safe shutdown earthquake	1	1	1	1	1	1	1	
35	F-2	DHRS Activation without SGS cooldown	1	1	1	1	1	1	1	1
41	F-3	Feedwater line ruptures	1	1	1	1	1	1	1	
	F-3a	Feedwater line rupture between steam drum and inlet isolation valve	1	1	1	1	1	1	1	
	F-3b	Feedwater line rupture in main incoming header	1	1	1	1	1	1	1	
	F-4	Steam line ruptures	1	1	1	1	1	1	1	
	F-4a	Saturated steam line rupture	1	1	1	1	1	1	1	
	F-4b	Main steam line rupture	1	1	1	1	1	1	1	
	F-4c	Rupture between superheater module outlet and superheater outlet isolation valve	1	1	1	1	1	1	1	
	F-4d	Rupture between superheater outlet isolation valve and main steam line	1	1	1	1	1	1	1	
	F-5	Recirculation line breaks	1	1	1	1	1	1	1	
	F-5a	Recirculation line break between drum and recirculation pump inlet	1	1	1	1	1	1	1	
43	F-5b	Recirculation line break between evaporator outlet and drum inlet	1	1	1	1	1	1	1	

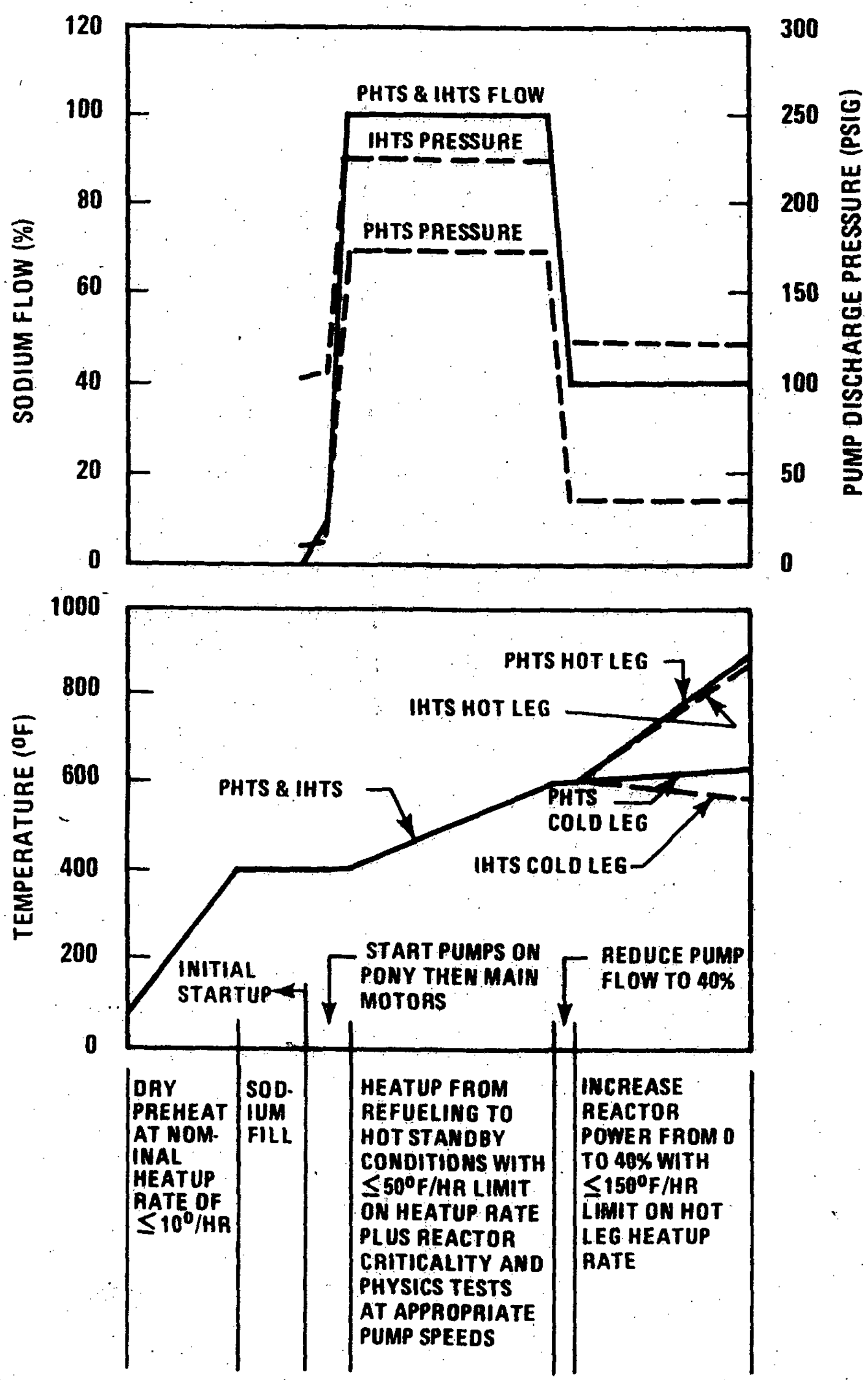
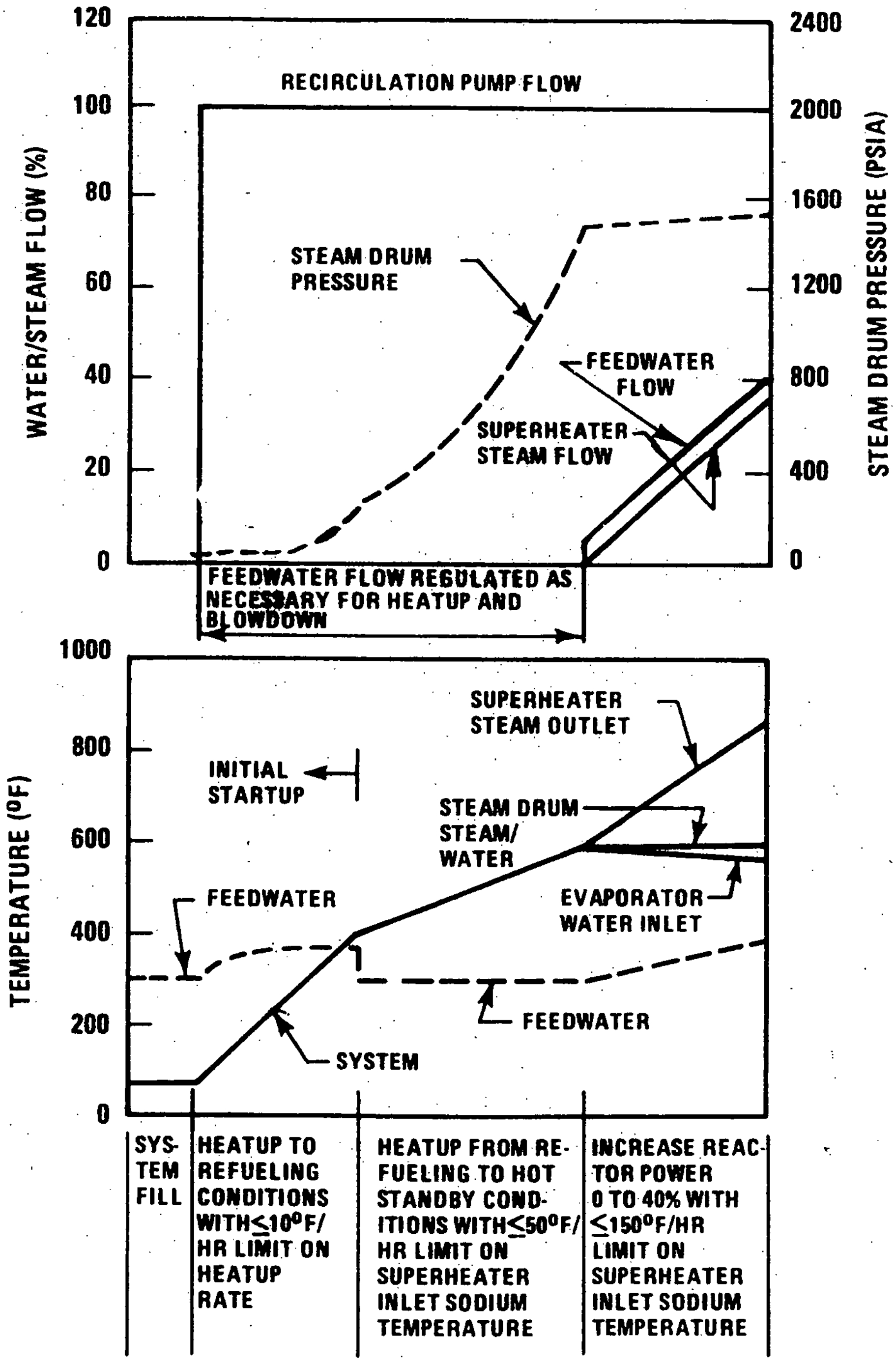


Figure 5.7-A Typical Startup Characteristics of Reactor Heat Transport System



Fiqure Figure 5.7-B Typical Startup Characteristics of Steam Generator System

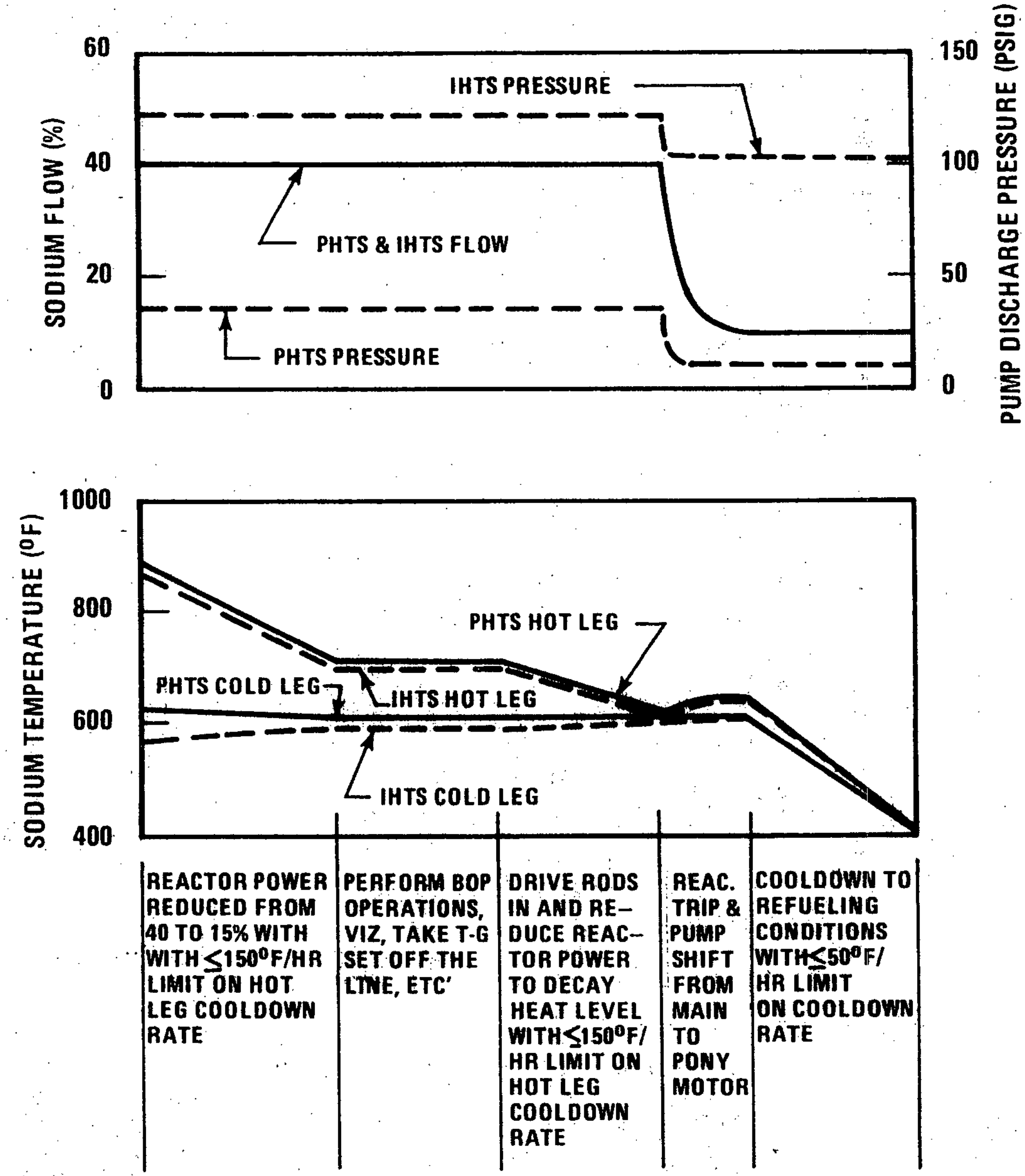


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Figure 5.7-C. Typical Shutdown Characteristics of Reactor Heat Transport System
Amendment 1

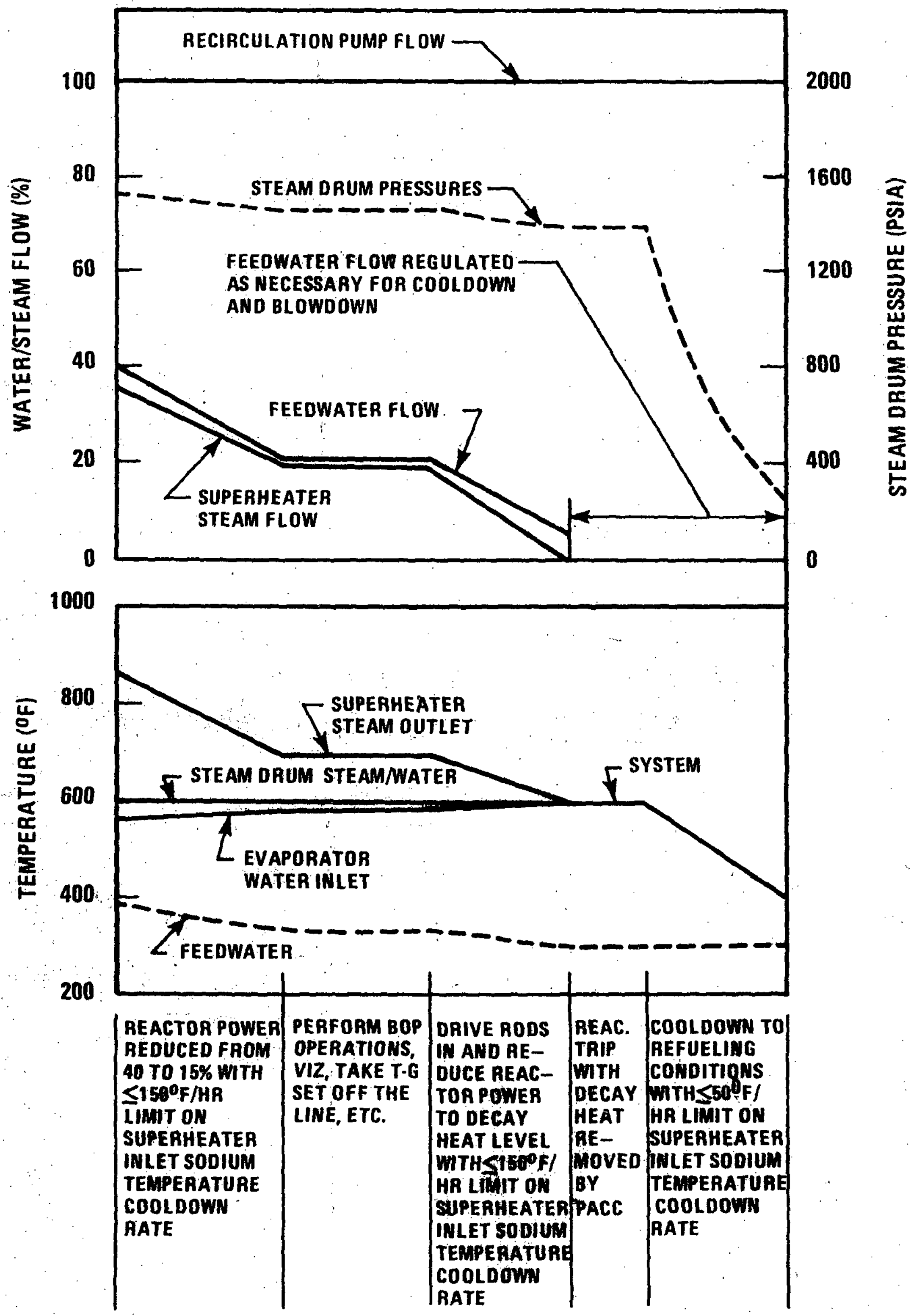

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Figure 5.7-D. Typical Shutdown Characteristics of Steam Generator System Amendment 1

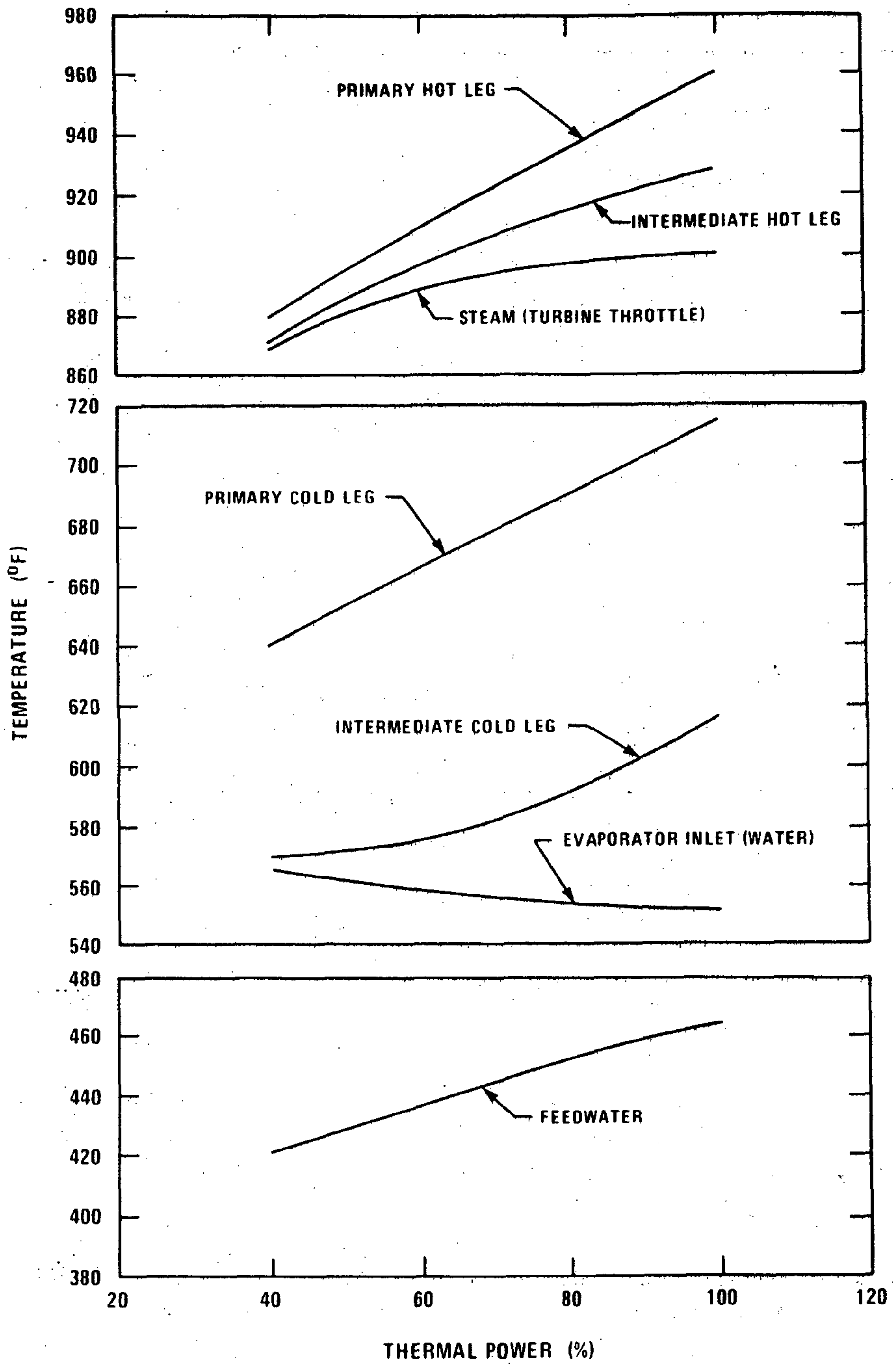

Figure 5.7-1. Expected Loop Thermal Parameters Vs. Power Level
Constant Primary Flow to Power Level Ratio
Constant Steam Pressure at 1450 psig
Linear Primary Hot Leg Temperature

5185-2

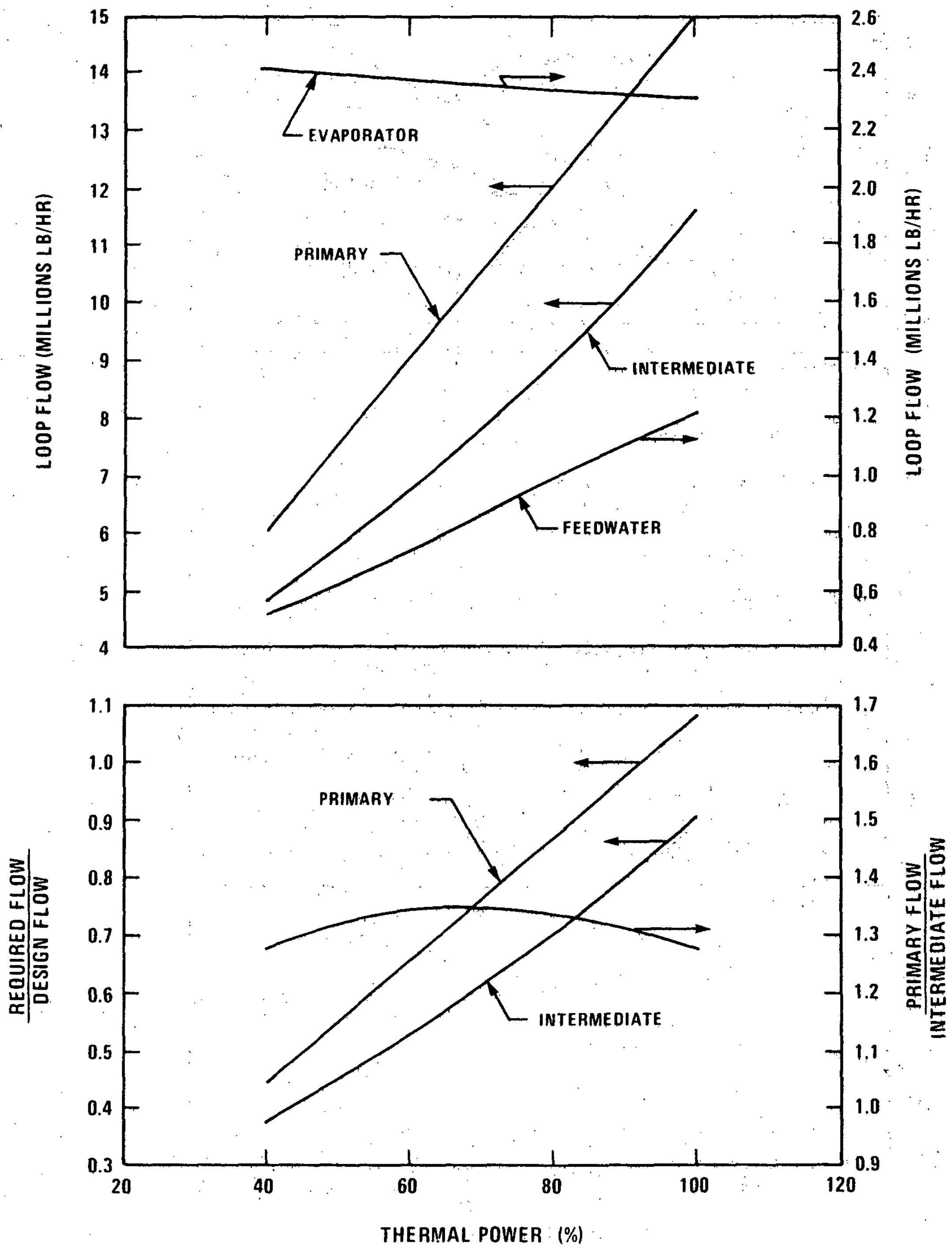

Figure 5.7-2. Expected Loop Flow Parameters Vs. Power Level
Constant Primary Flow to Power Level Ratio
Constant Steam Pressure at 1450 psig
Linear Primary Hot Leg Temperature

5185-1

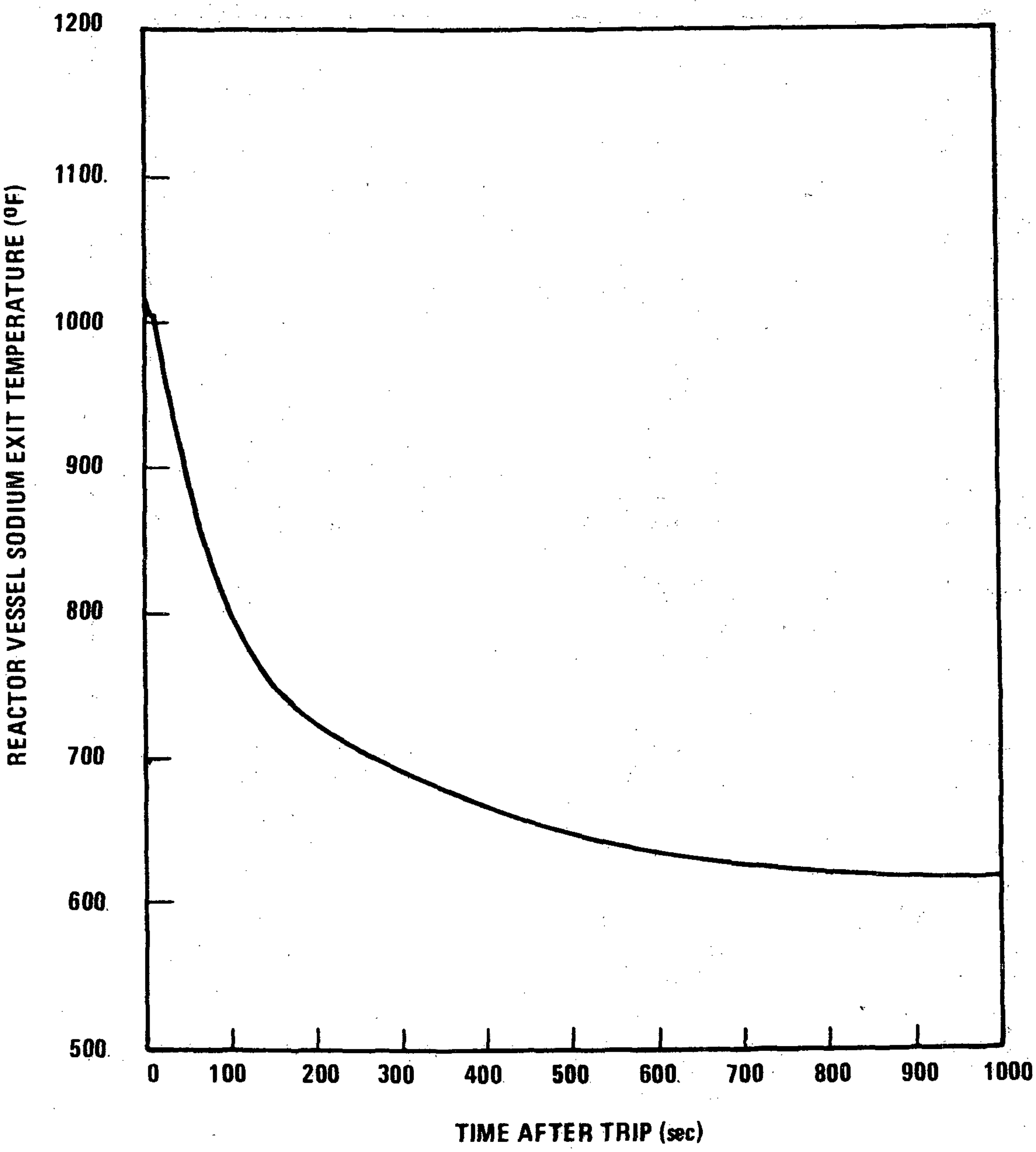

Figure 5.7–3 Reactor Vessel Sodium Exit Temperature Vs. Time for Reactor Trip From Full Power

6673-3

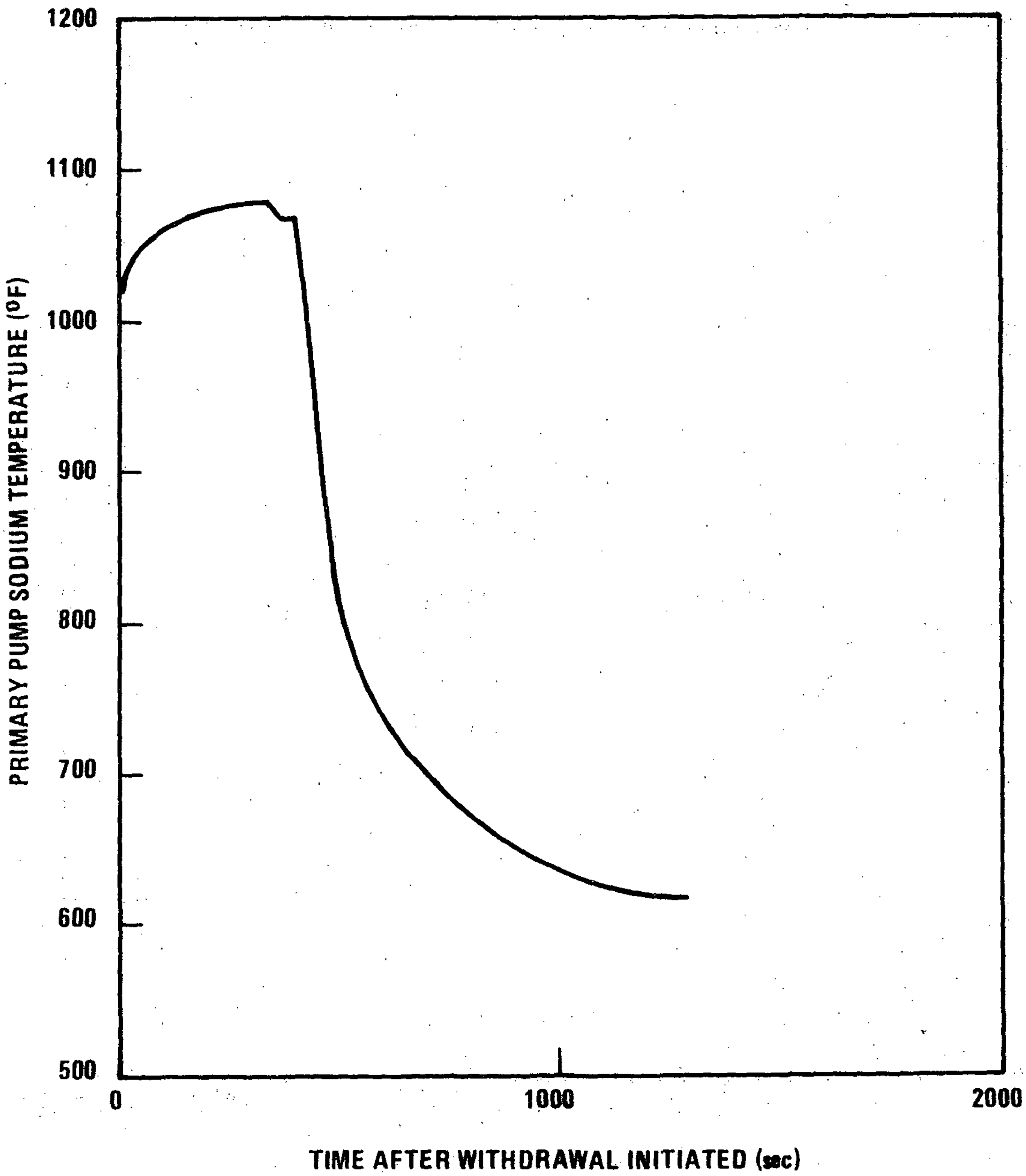

Figure 5.7–4 Primary Pump Sodium Temperature Vs. Time for Uncontrolled Rod Withdrawal With Trip

6673-4

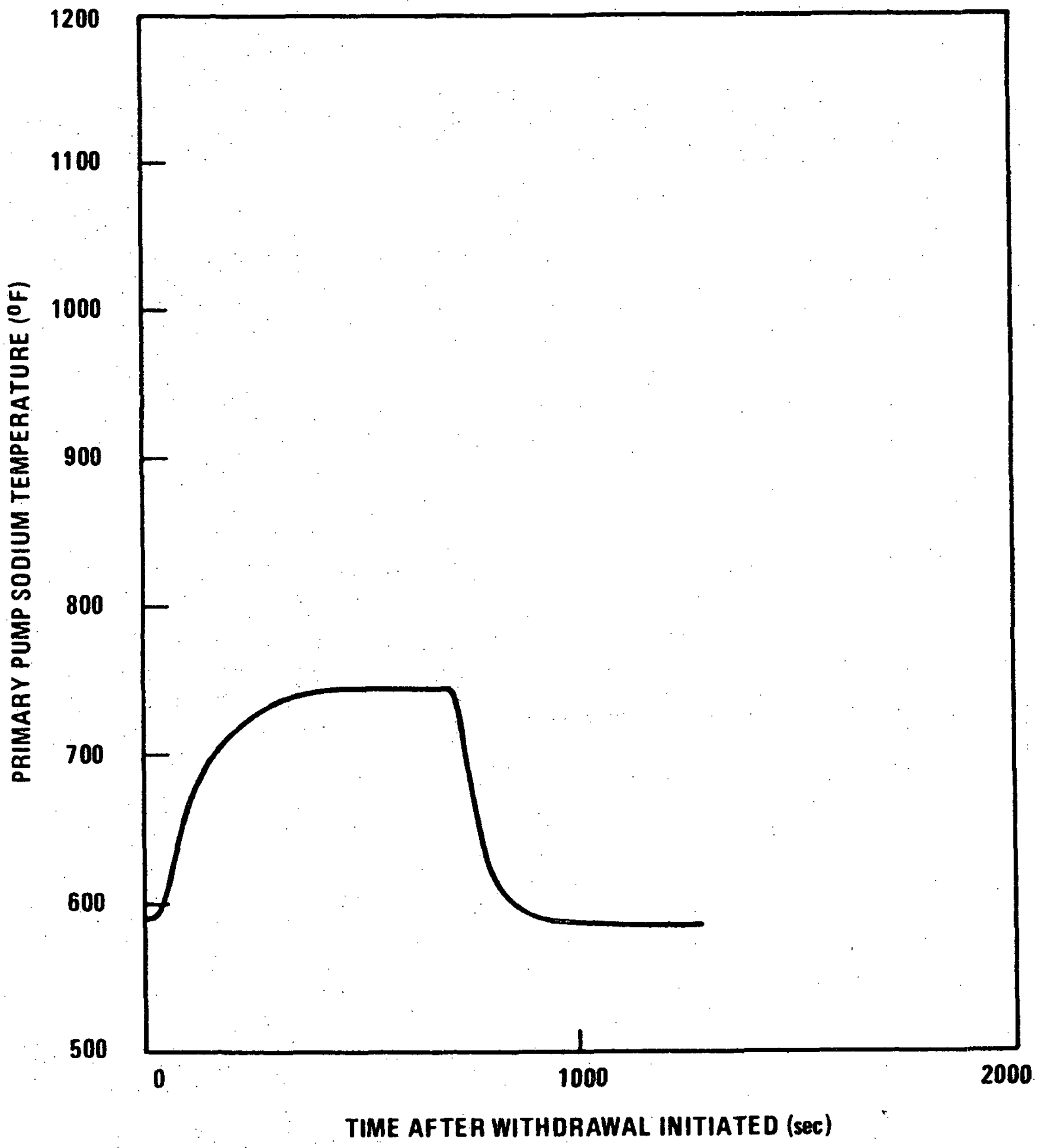

Figure 5.7–5 Primary Pump Sodium Temperature Vs. Time for Uncontrolled Rod Withdrawal From Startup

6673-5

FIGURE 5.7-6

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Figure 5.7-6a Average Channel Sodium Exit Temperature Top of Active Core vs. Time for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater)

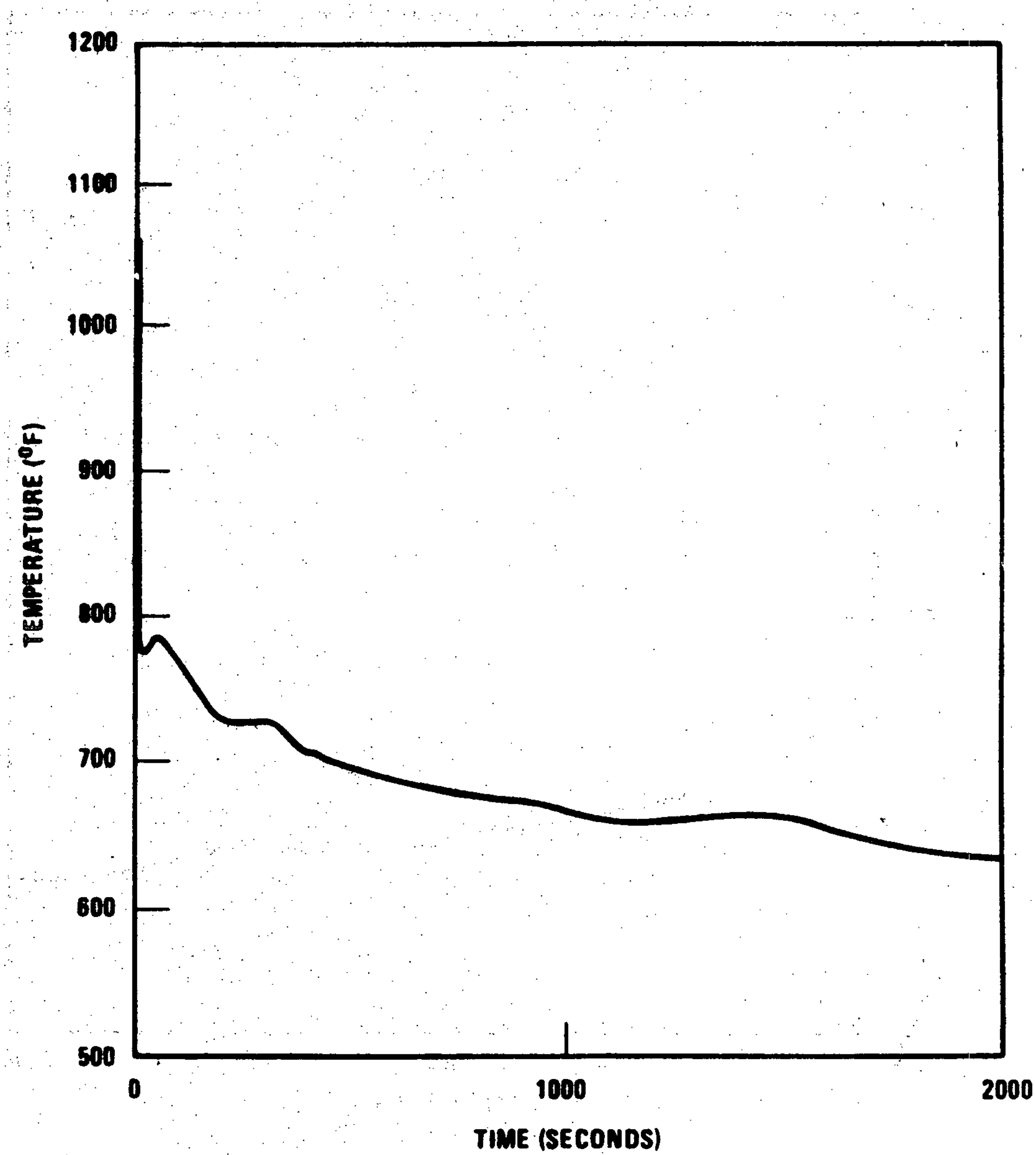

Figure 5.7-6b Maximum Channel Sodium Exit Temperature, Top of Active Core for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater).

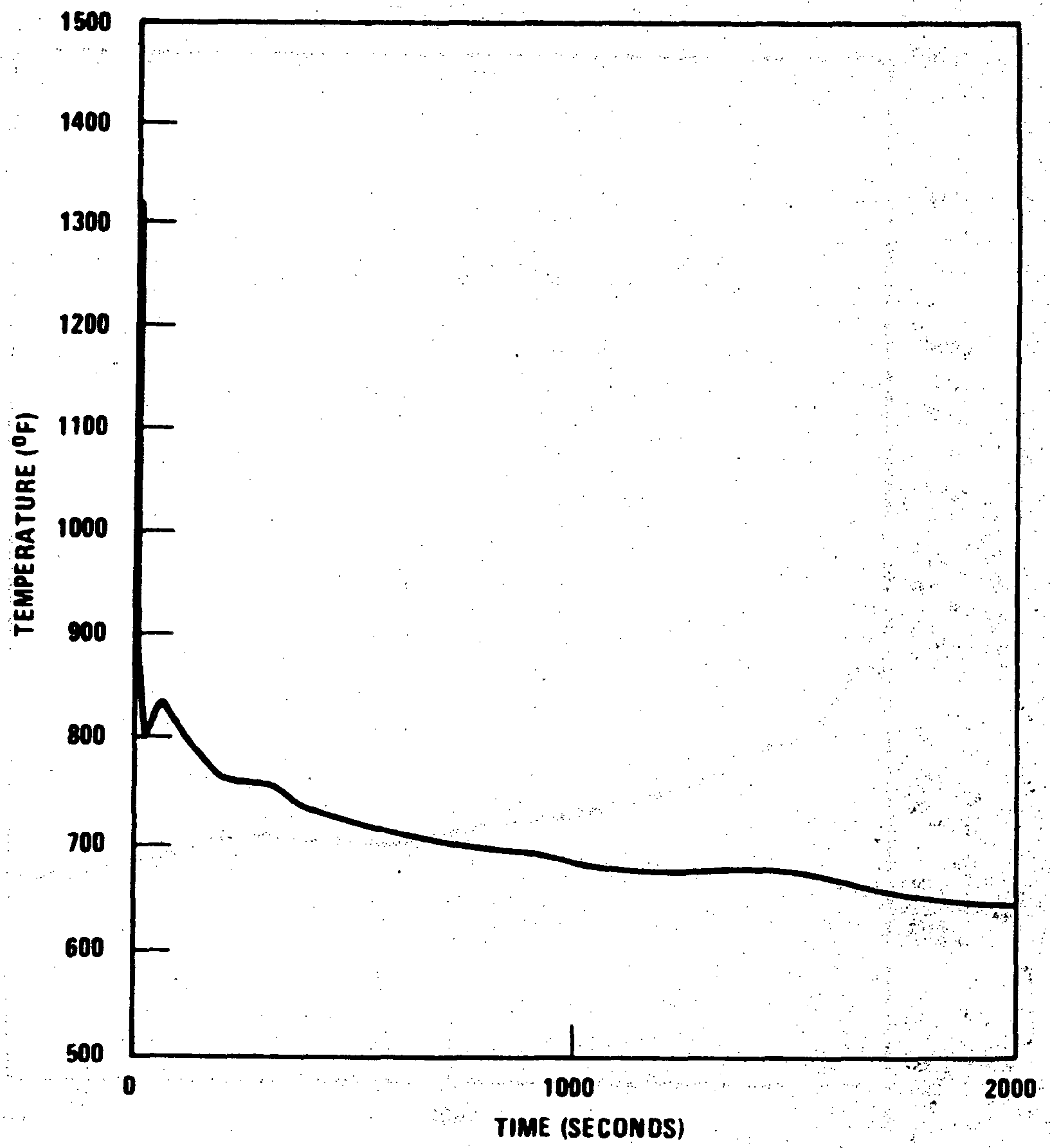

Figure 5.7-6C Blanket Hot Channel Sodium Outlet Temperature for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater).

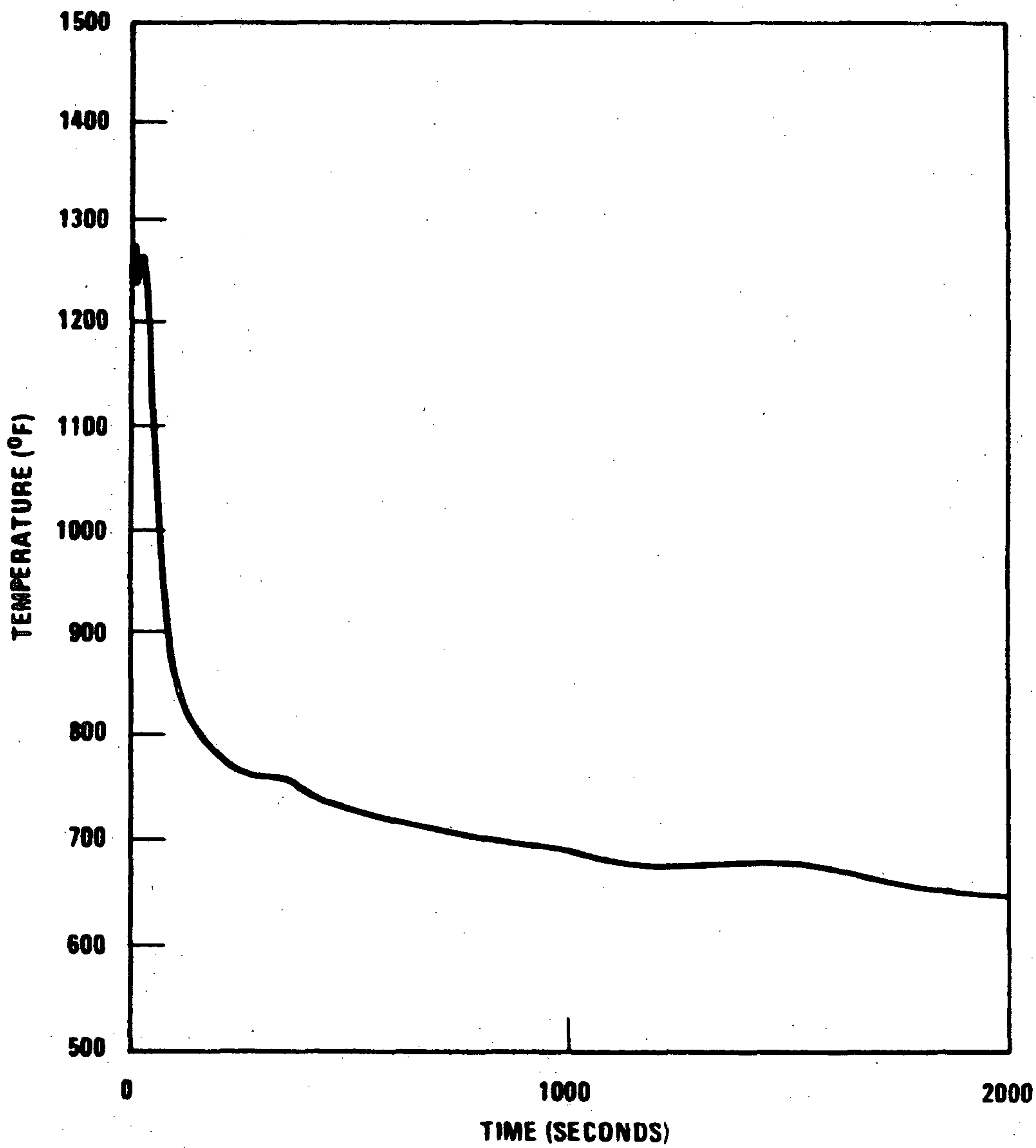

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Figure 5.7-6D Reactor Vessel Exit Temperature for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater).

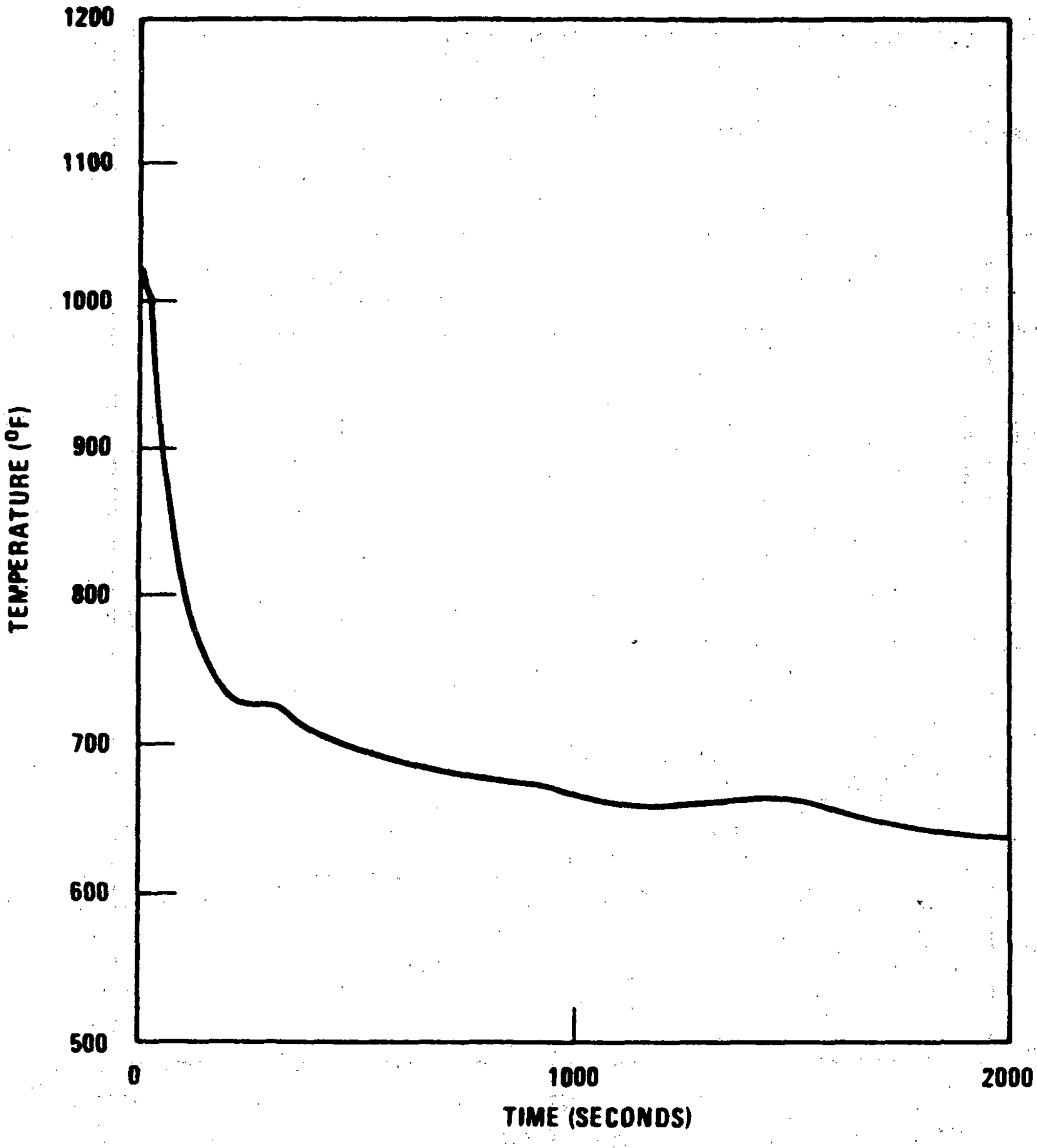

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Figure 5.7-6E Affected Loop Superheater Sodium Inlet Temperature for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater).

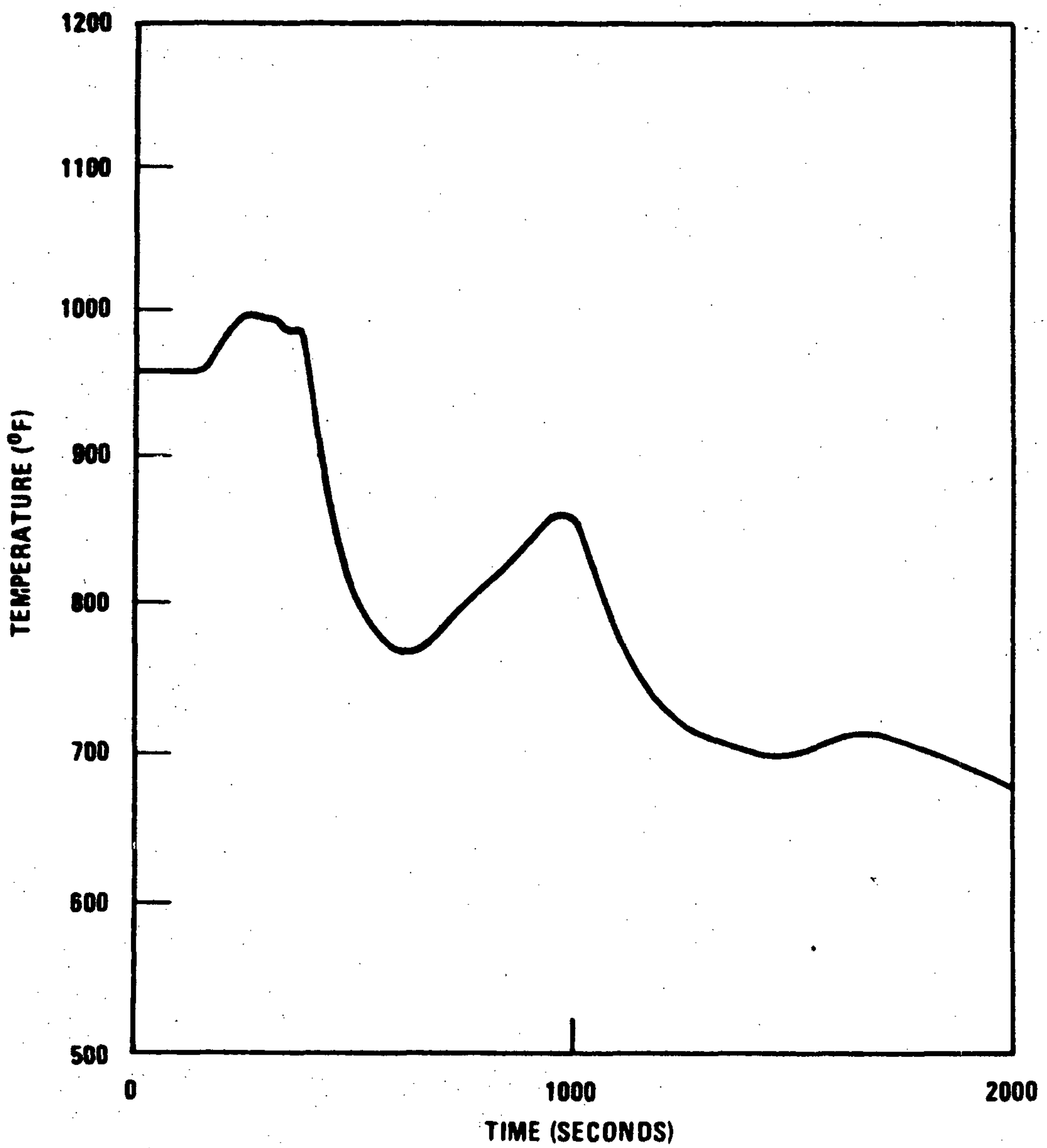

Figure 5.7-6F Affected Loop Evaporator Sodium Inlet Temperature for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater).

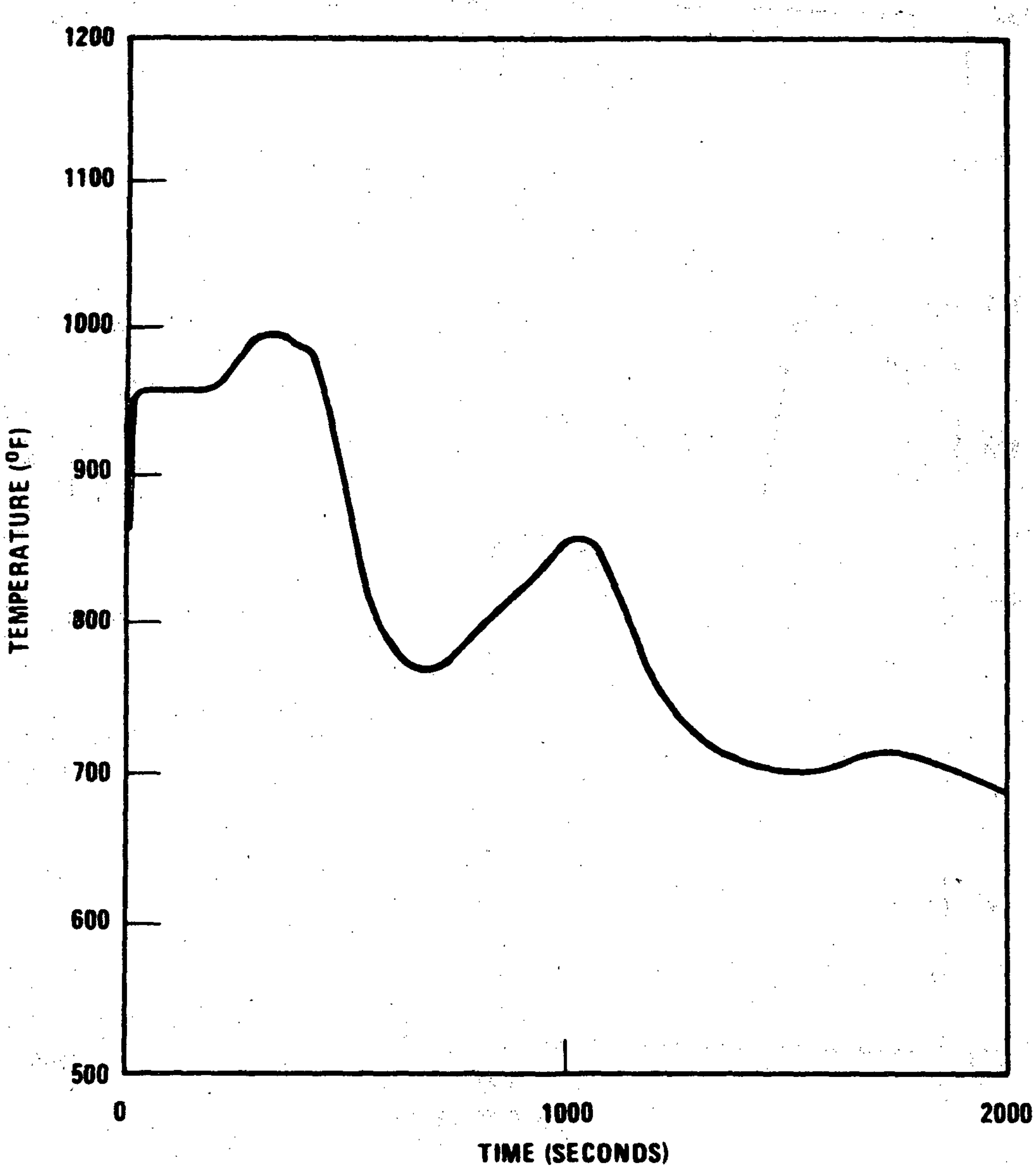

Figure 5.7-6G Affected Loop Evaporator Sodium Exit Temperature for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater).

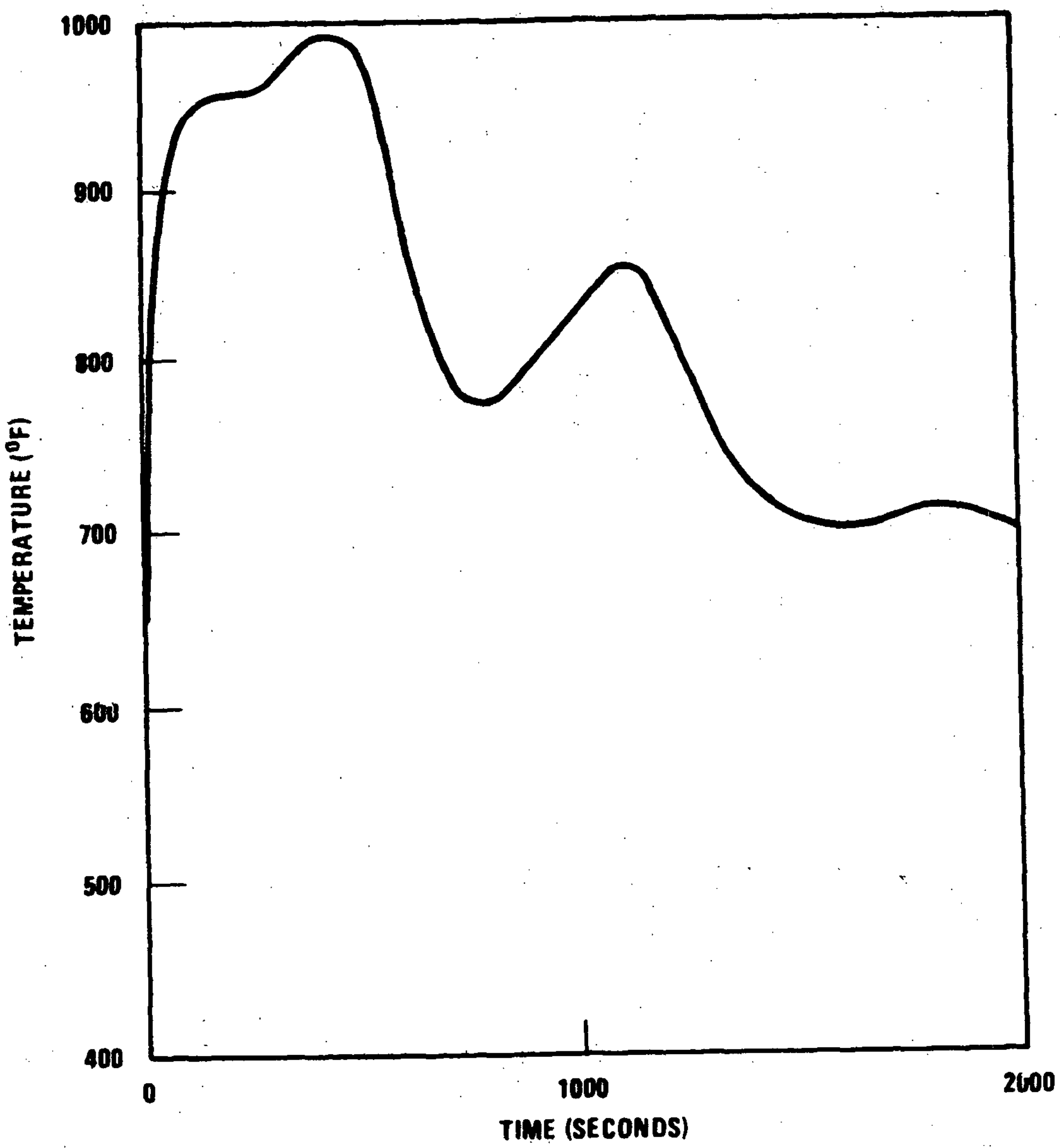

Figure 5.7-6H Intermediate Pump Sodium Temperature Vs. Time for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater).

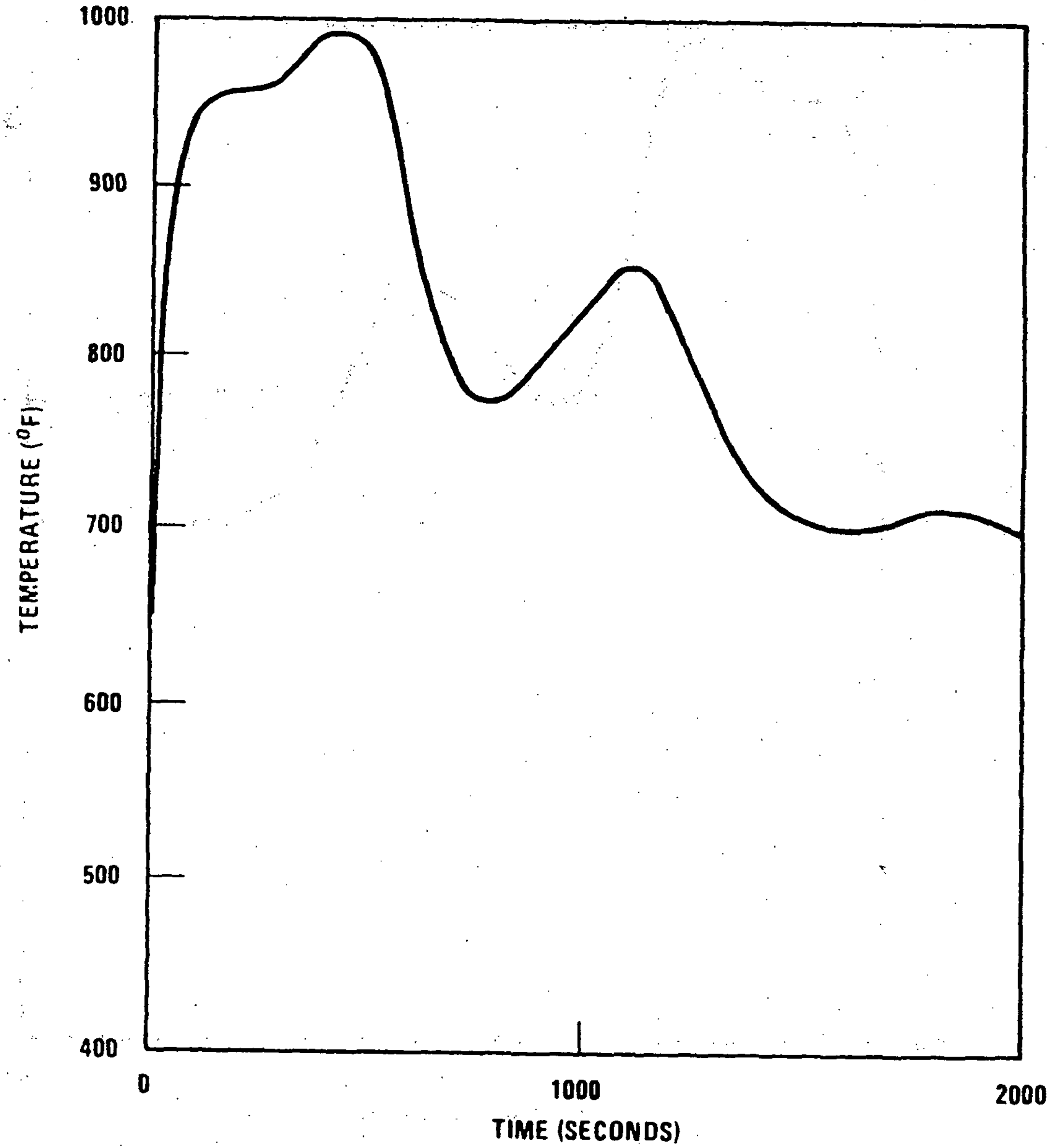

Figure 5.7-61 Affected Loop Drum Steam Temperature for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater).

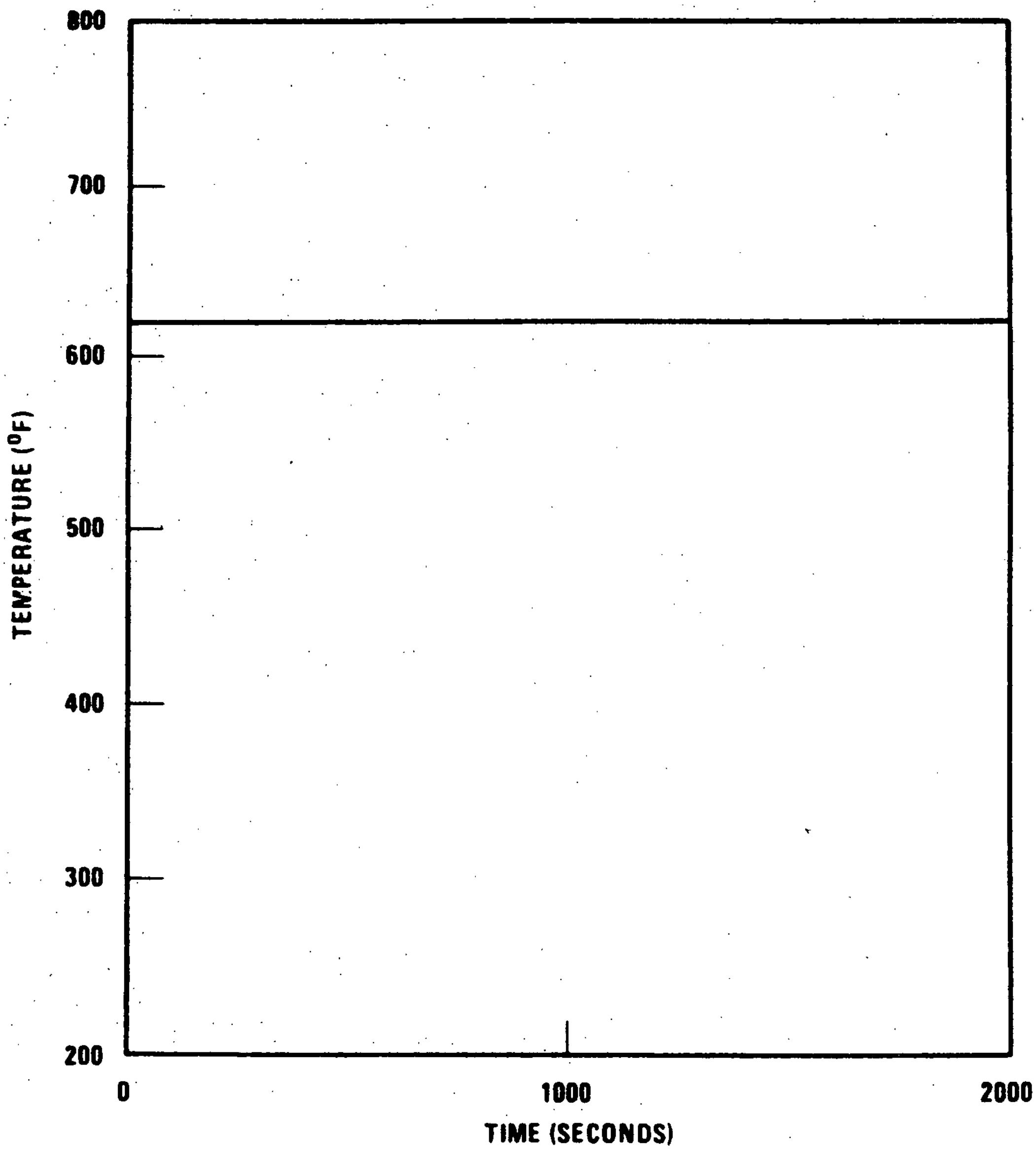

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Figure 5.7-6J Affected Loop Evaporator Inlet Water Temperature for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater).

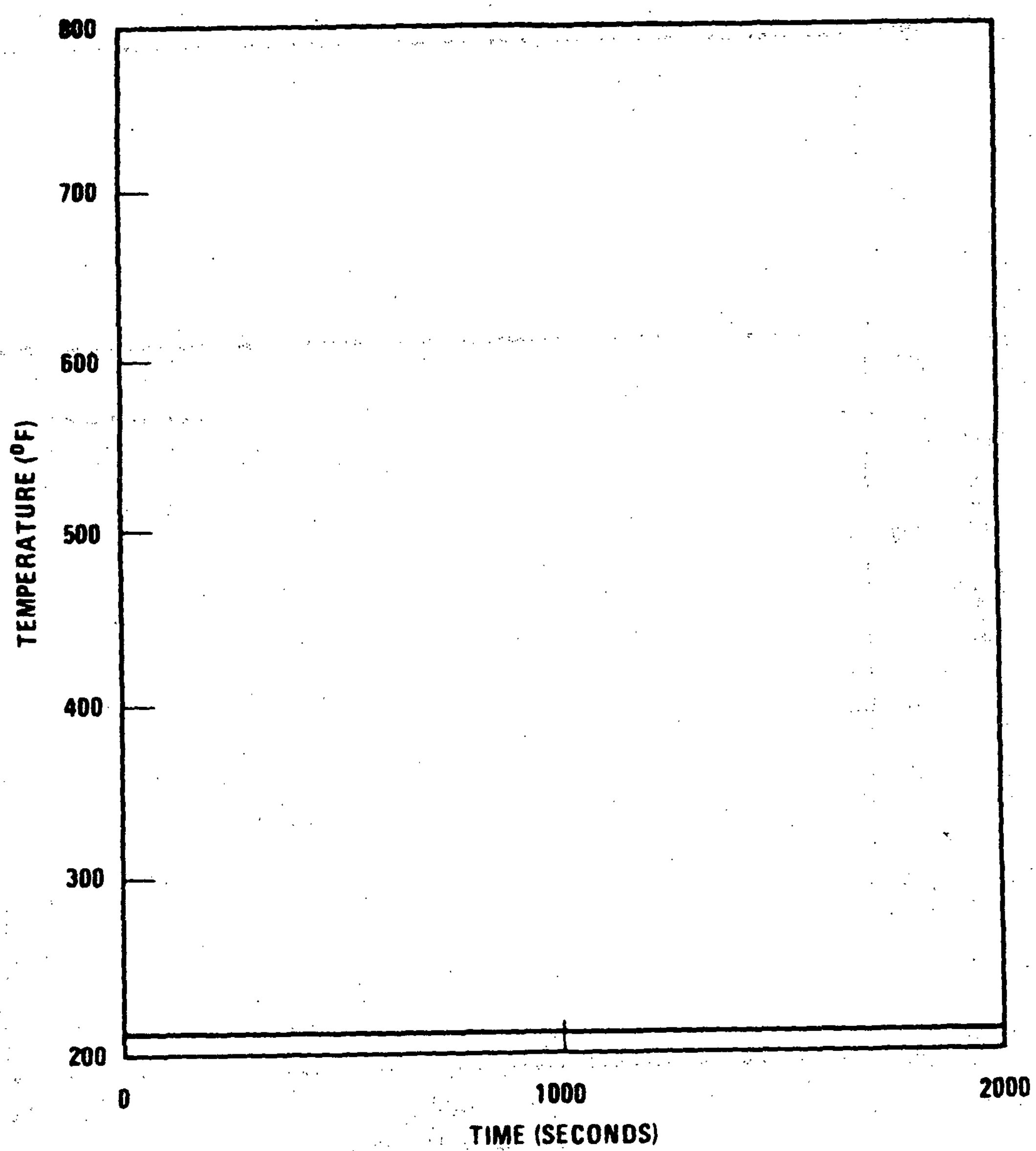
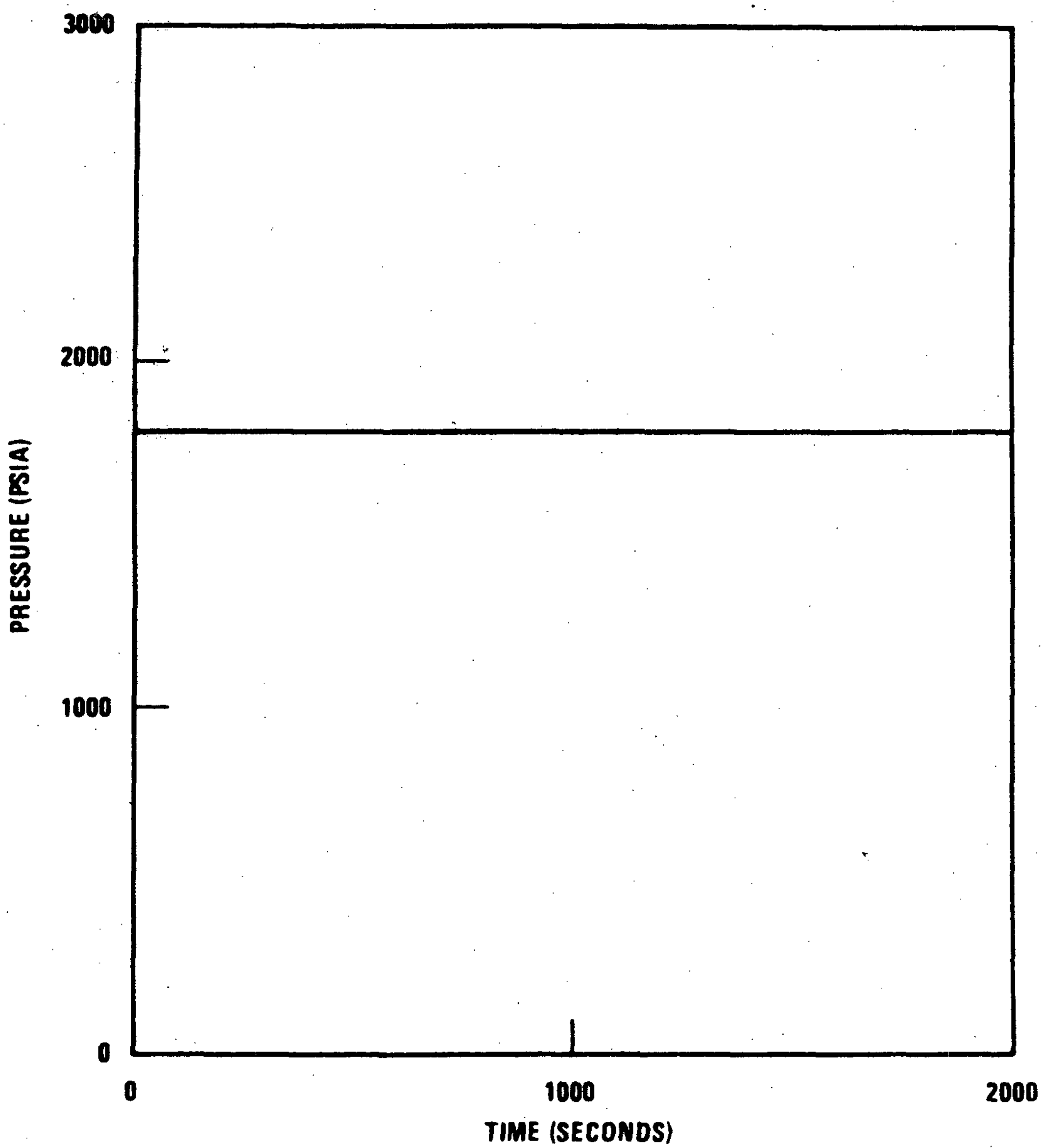


Figure 5.7-6K Affected Loop Drum Pressure for Loss of Steam Generator Load (Dumping of Water/Steam Sides of Both Evaporators and the Superheater).



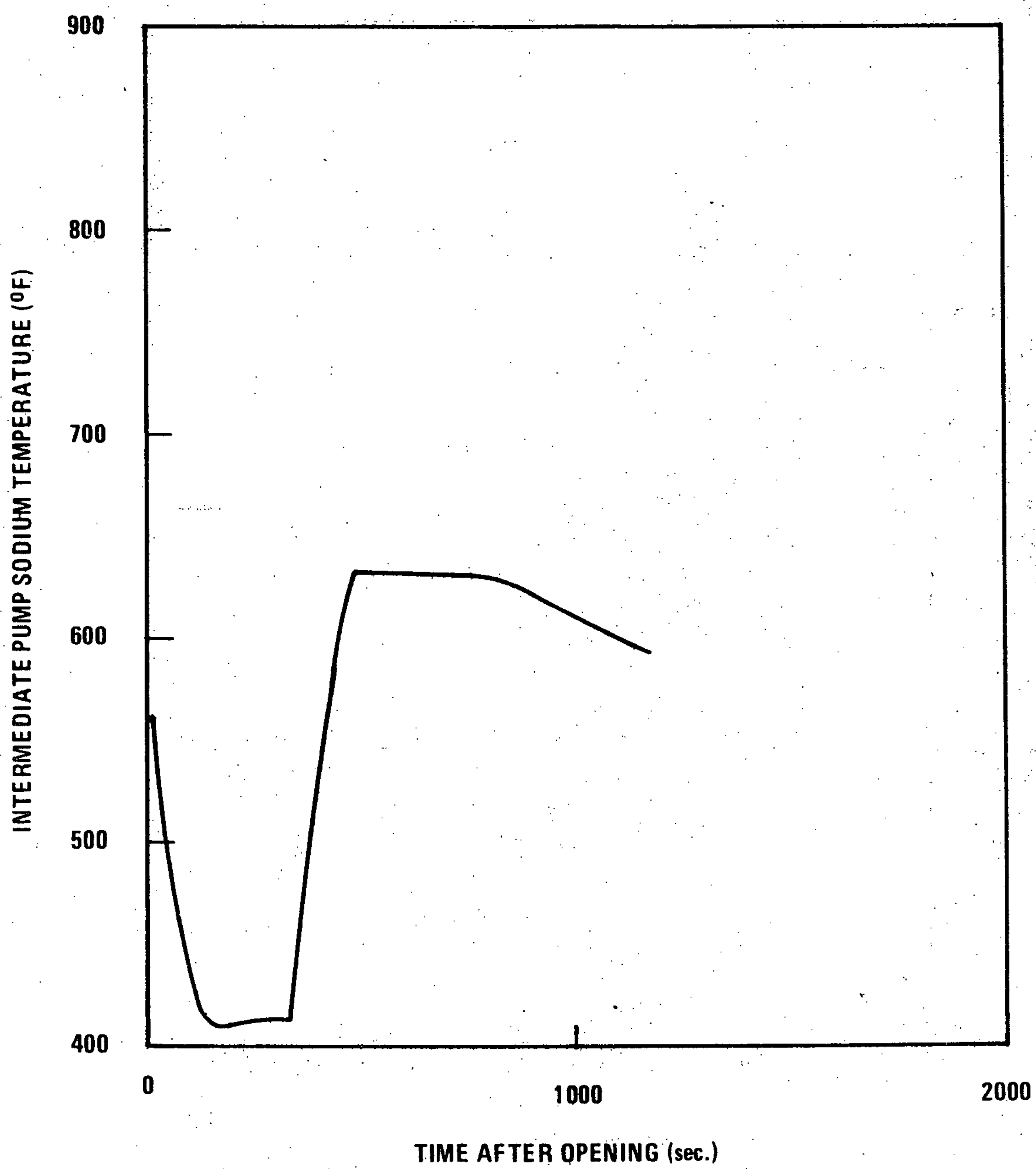


Figure 5.7–7 Intermediate Pump Sodium Temperature Vs. Time for Inadvertent Opening of Superheater Outlet Relief Valves

6673-7

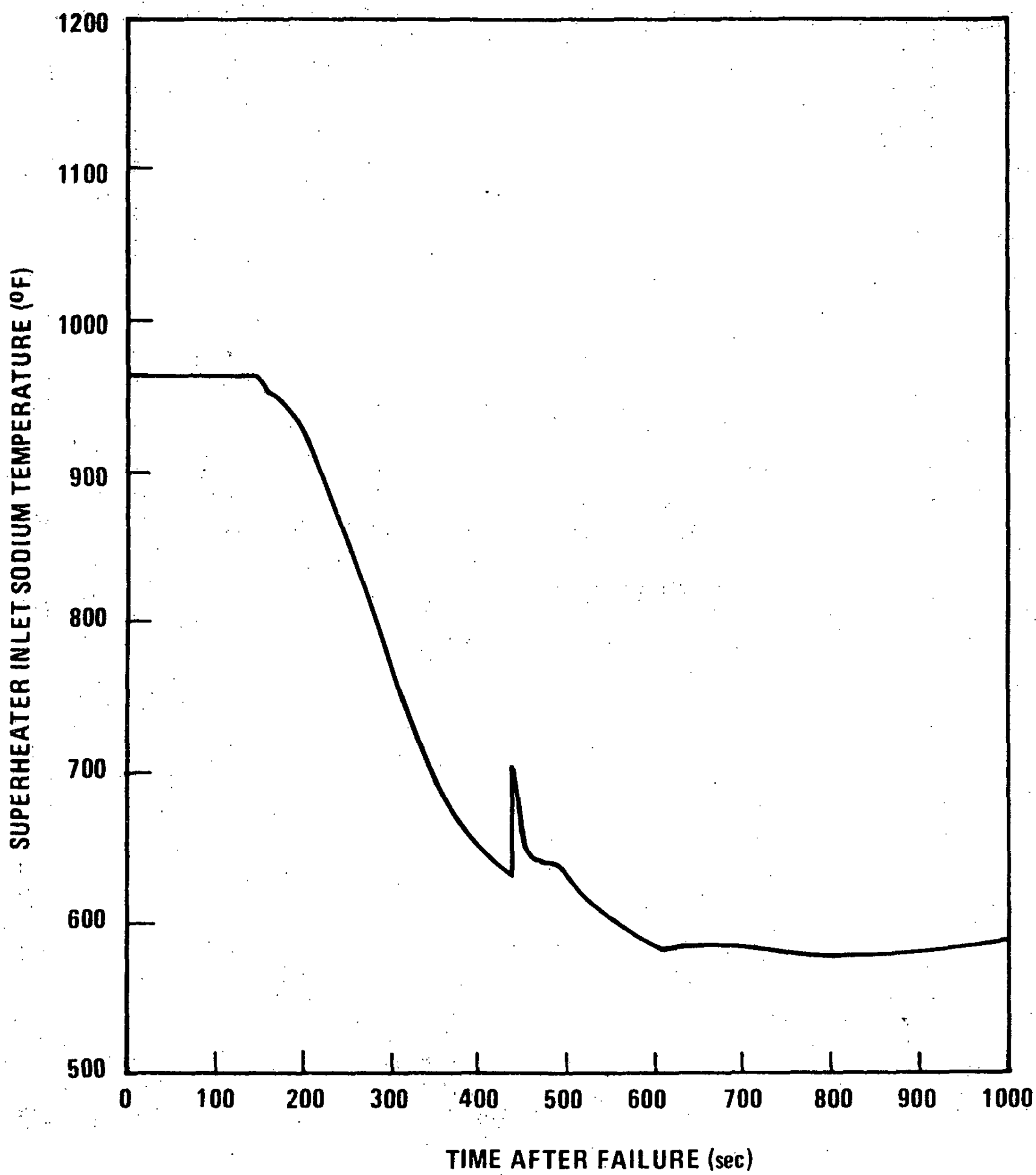


Figure 5.7–8 Superheater Inlet Sodium Temperature Vs. Time for Primary Pump Mechanical Failure

6673-8

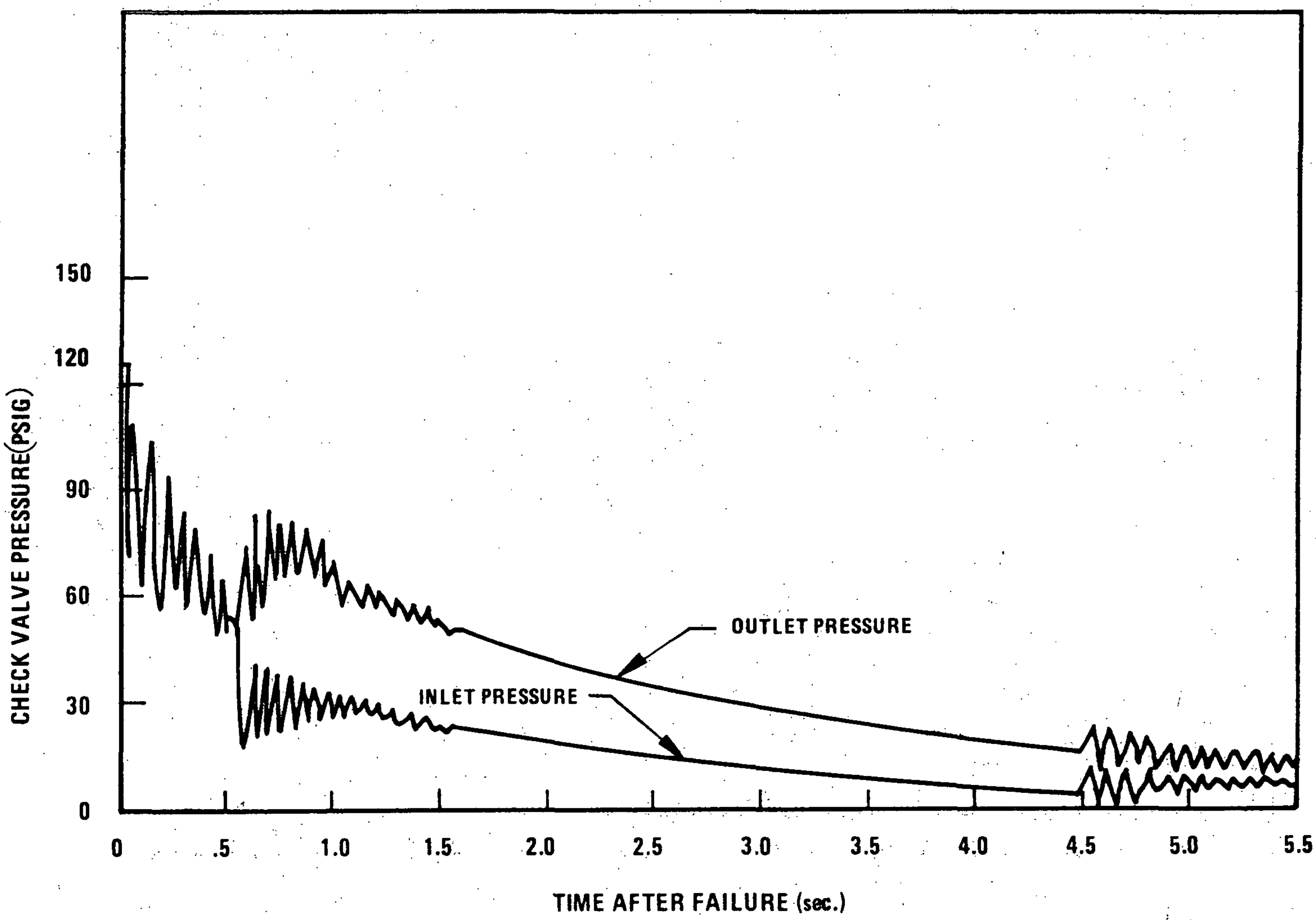


Figure 5.7–9 Check Valve Pressure Vs. Time for Primary Pump Mechanical Failure

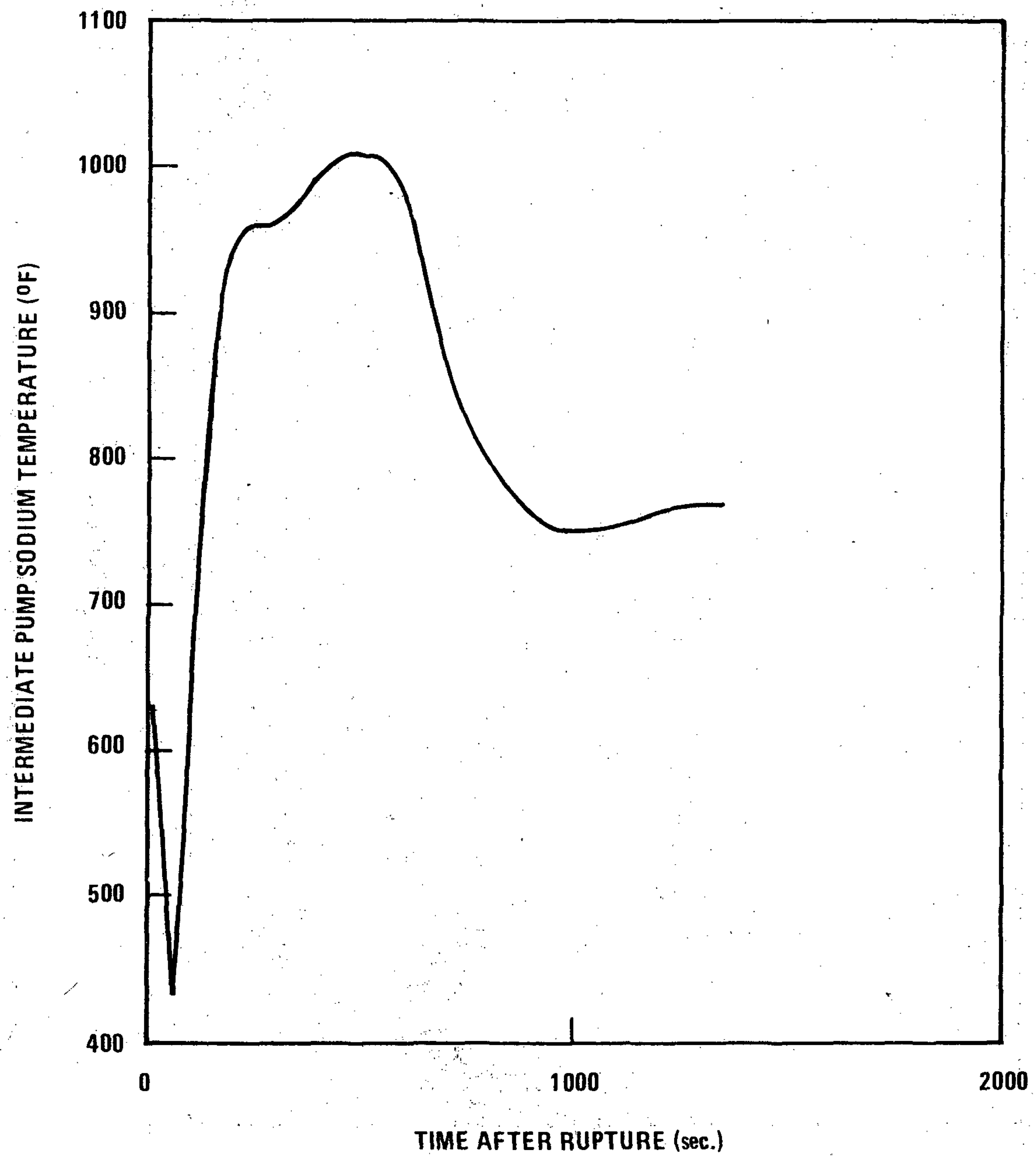


Figure 5.7–10 Intermediate Pump Sodium Temperature Vs. Time for Saturated Steam Line Rupture

6673-10

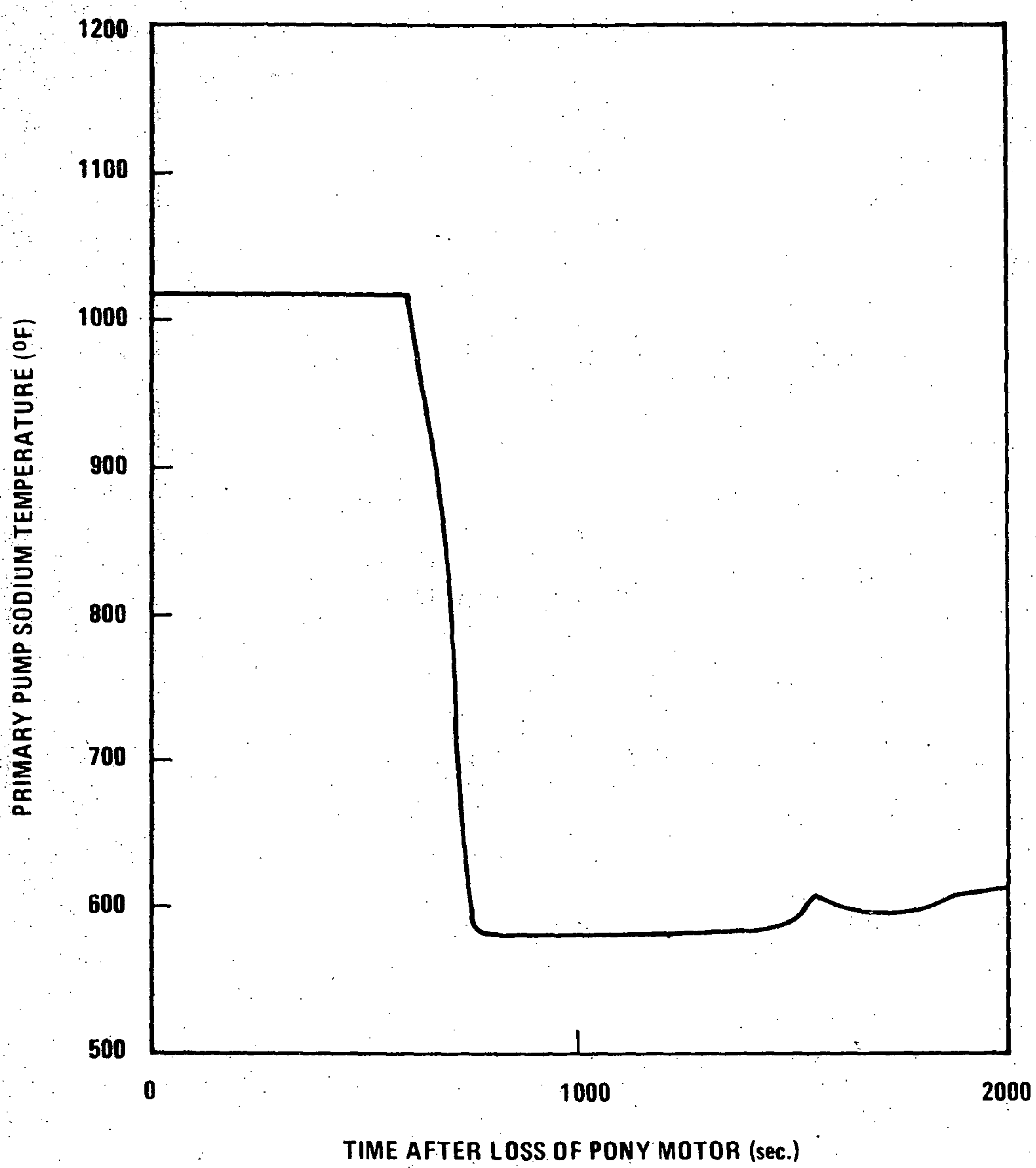


Figure 5.7–11 Primary Pump Sodium Temperature Vs. Time for Loss of Primary Pony Motor With Failure of Check Valve to Shut

6673-11

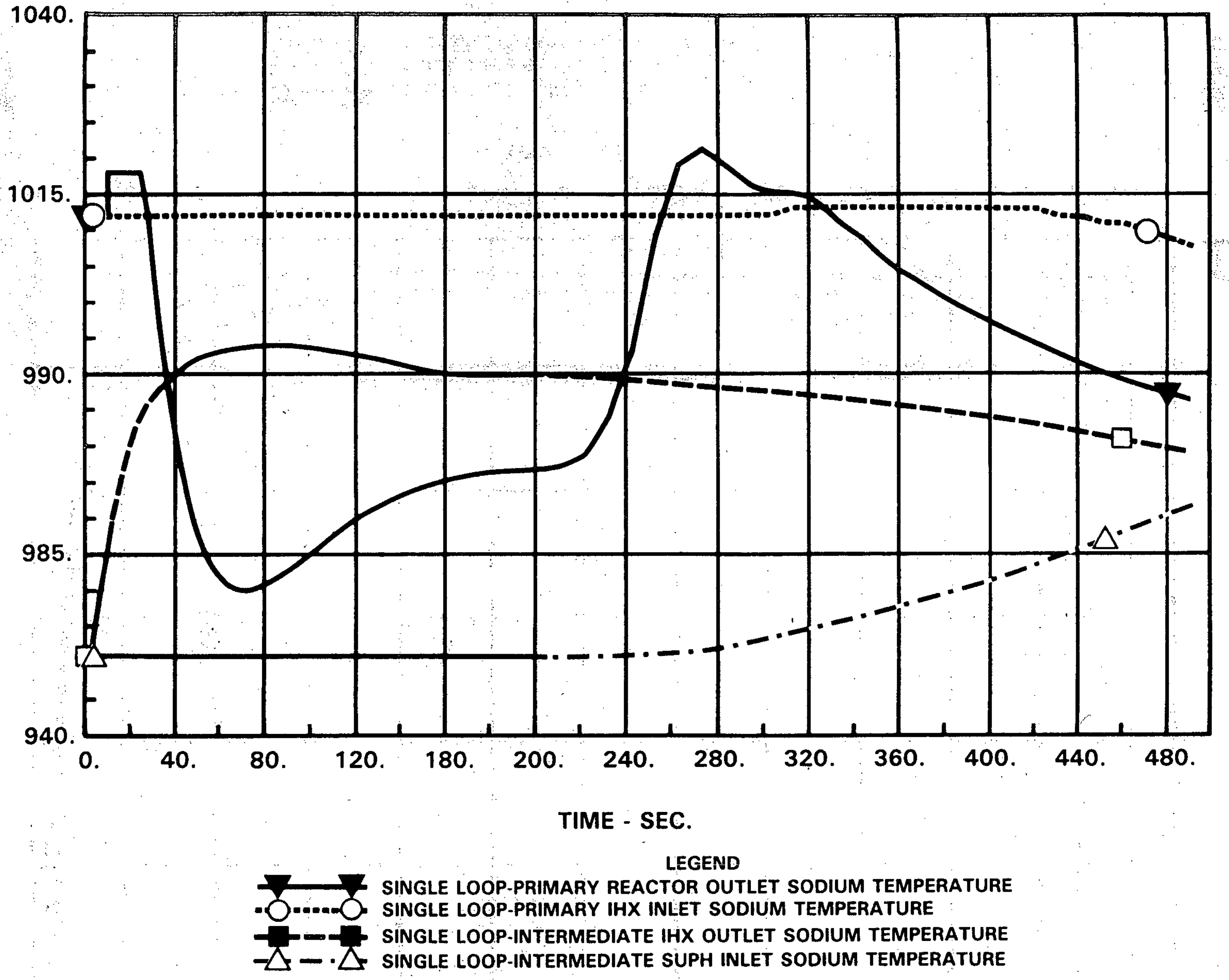


Figure 5.7-12
Natural Circulation, 3 to 3 Loop Case, Temperature vs. Time

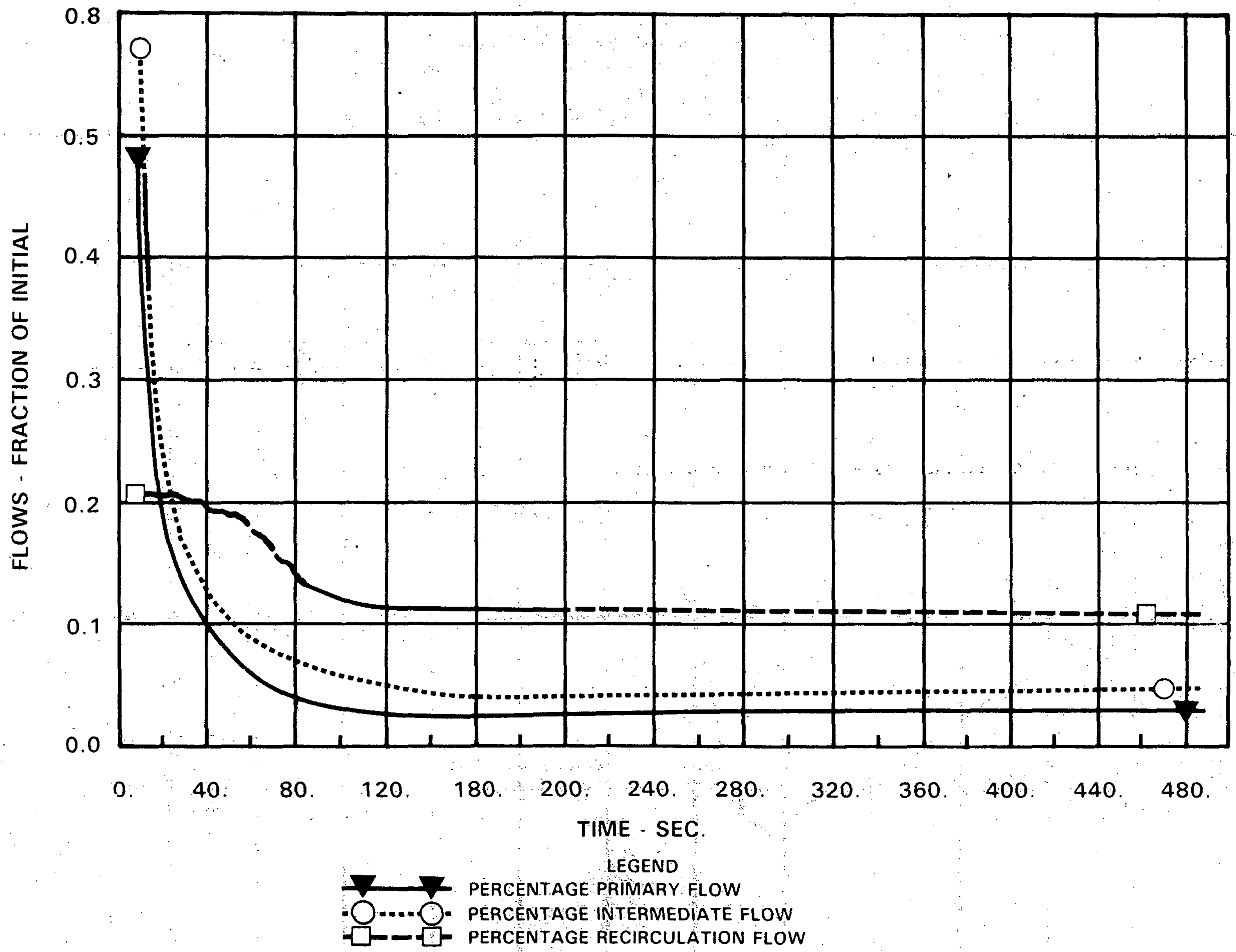


Figure 5.7-13
Natural Circulation, 3 to 3 Loop Natural Circulation