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PRISM™

Preliminary Safety Information Document

Prepared for U.S. Department of Energy
Under Contract No. DE-AC03-85NE37937

Volume II

Chapters 5-8

APPLIED TECHNOLOGY

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Letter dated 5/26/93

GENERAL ELECTRIC
ADVANCE NUCLEAR TECHNOLOGY
SAN JOSE, CALIFORNIA

AMENDMENT 9

87-568-02

DISCLAIMER

Department of Energy
Washington, DC 20585

MAY 2 6 1993

Mr. Dennis M. Crutchfield
Associate Director for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Crutchfield:

As you indicated in your letter, dated April 29, 1993, you are completing the final Preapplication Safety Evaluation Report (PSER) for the "Power Reactor Innovative Small Module" (PRISM) Advanced Liquid Metal Reactor design. You expressed concern about meeting one of the Commission's objectives of public disclosure since the PSER will be based on documents on which the Department of Energy (DOE), Office of Nuclear Energy, placed a restrictive distribution labeled "Applied Technology." We hereby approve your request for public disclosure and you are authorized to remove the "Applied Technology" (AT) distribution limitation from all of the DOE documents titled Preliminary Safety Information Document. The documents are:

"PRISM - Preliminary Safety Information Document" (PSID) - GEFR-00795

Volume I - December 1987, Chapters 1-4
Volume II - December 1987, Chapters 5-8
Volume III - December 1987, Chapters 9-14
Volume IV - December 1987, Chapters 15-17
and Appendices A-E
Volume V - February 1988, Amendment to PSID
Volume VI - March 1990, Appendix G

With regard to the Modular High Temperature Gas-Cooled Reactor (MHTGR), we would like to request that public disclosure of its AT information be delayed until publication of the MHTGR PSER becomes more imminent. We would appreciate your understanding of this situation and assure you that we will release MHTGR AT for public disclosure when needed to support the PSER issuance. We will be happy to meet with you and your staff to discuss this further at your convenience.

Sincerely,

Jerry D Griffith

Jerry D. Griffith
Director
Office of Advanced Reactor Programs
Office of Nuclear Energy

cc:
Salma El-Safwany, DOE/SF
James Quinn, GE
Richard Hardy, GE
Robert Pierson, NRC
Ray Mills, PDCO

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ABSTRACT

This document is a Preliminary Safety Information Document (PSID) for a PRISM (Power Reactor Inherently Safe Module) electric power plant. The PSID is the document in the PRISM licensing plan that provides the description and evaluation of the conceptual design using nine reactor modules. Each module is a compact liquid metal reactor of the pool type design. The reactor module has unique passive safety characteristics that enhance the safety of the design. These include passive shutdown heat removal and passive reactivity shutdown. The document presents design criteria, design description and analyses that demonstrate these favorable safety characteristics. The format is similar to the standard format for safety analysis reports, however, the design description and evaluations are consistent with the conceptual design level. Design basis accidents are described in Chapter 15 and a preliminary PRISM probabilitic risk assessment is included in Appendix A.

CHAPTER 5
REACTOR COOLANT SYSTEM

CHAPTER 5

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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CHAPTER 5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

5.1.1. Reactor Module

PRISM is a compact pool-type LMR. A summary description of the entire reactor module is given here as an introduction to more detailed description and evaluation of the primary heat transport system (PHTS) that is contained within the module. The reactor module, shown in Figure 5.1-1, consists of the containment vessel, reactor vessel, reactor closure with rotatable plug, intermediate heat exchangers (IHX), electromagnetic (EM) pumps, control rod drives, reactor internal structures, fuel transfer machine, fuel storage racks, core, shielding, and the module support structure.

The reactor vessel and containment vessel are attached to the reactor closure and are the major components of the reactor enclosure which provides the container and support structure for the reactor core, primary sodium, and structures within. The reactor vessel performs its support and container functions during all temperature, pressure, and load variations which occur during the operating lifetime. The reactor and containment vessels and the closure are designed and constructed to the requirements of the ASME B&PV Code, Section III. The reactor vessel has no penetration and no attachments other than those for connecting the core support structure to the hemiellipsoidal bottom head and those used as temporary shipping supports.

The reactor vessel is suspended from the closure. The closure is supported on the containment vessel flange, which in turn is supported as described in the previous paragraphs.

The dimensions of the reactor and containment vessel are sized not only to hold the core and perform as a reactor system but also to have sufficient surface area to transfer decay heat by direct convection and by radiation to the collector cylinder for removal by natural circulation of air. The chrome-moly containment vessel is about 20 feet in diameter and about 60 feet long. The slightly smaller reactor vessel is constructed of Type 316 stainless steel.

Reactor Module Support - The reactor module is top supported. It has twelve radial support brackets resting on Lubrite plates to accommodate radial, horizontal thermal displacement. The Lubrite plates and reactor support brackets are anchored to a circular recessed steel ledge embedded in the HAA floor adjacent to the reactor module. The seismic isolators, located in a separate gallery below the RVACS horizontal plena, support the module, the operating floor at elevation -15 feet, the RVACS shield walls below the operating floor, a shield slab at elevation -17 feet, the concrete shielded RVACS vertical ducting and the RVACS horizontal plena and collector cylinder as depicted in Figures 1.2-2 and 1.2-4. Seismic isolation is used to mitigate the horizontal seismic motion transmitted to the reactor module. The special flexible bearings (isolators) shift the horizontal fundamental frequency of the isolated structure to a range below that of the earthquake frequencies which are harmful to the reactor module. A typical rubber and steel isolator is shown on Figure 1.2-4b. Each rubber layer is bonded on both sides to steel shim plates. Each isolator is fitted with upper and lower steel base plates which are connected to the isolated structure and basemat (foundation) by dowel pins. See Table 1.5-1 for isolator qualification program. Twenty isolators are used to support each reactor module, Figure 1.2-4a. Support columns are used to provide a 5 ft. high space for inspection and maintenance of the isolators. A $1\text{-}^1/2$ ft. thick circular shield wall located adjacent to the reactor provides radiation shielding for the isolator gallery.

The principal elements of the support are 12 stiffened L-shaped brackets. These are designed to be installed in the field prior to setting the reactor module. Each bracket is mechanically attached to the containment vessel and has a shear lug that fits into a keyway of the containment vessel flange. There is one keyway, machined as a chord to the flange's outside diameter, for each lug. The keyways, brackets and their attachment hardware are designed so as not to interfere with the ISI ports and the reactor closure holddown bolts. The vertical weight of the module is supported on the 12 shear lugs. Lateral seismic restraint is provided by the close fit of the bracket anchor bolts within the radially slotted holes of the brackets.

The radially slotted holes allow the support brackets to move radially, relative to the stationary anchor bolts, as the top end of the module cycles through its temperature range. The attachment is also designed to resist module motions and accelerations relative to the support ledge occurring during seismic events. Shims are provided as required to assure the proper alignment and leveling of the reactor module relative to the building.

EM Pumps - There are four EM pumps in the reactor module, each pump delivers 10,500 gpm at a discharge pressure of 120 psi. The pump's compact size, low net positive suction head (NPSH) requirements, low maintenance, absence of seals, moving parts, and dynamic auxiliary systems makes it ideal for this application. The lack of any substantial superstructure on the reactor closure where space is at premium is also very advantageous. The pumps have no moving parts and are cooled by the sodium in which they are submerged. This requires an electrical insulation of higher temperature rating than has been previously used for EM pumps. Development work is underway at ANL to select an insulation and the method of application.

Intermediate Heat Exchanger - There are two intermediate heat exchangers (IHX) in each reactor module. Each IHX is rated at 215 MWt to accommodate the core heat load of 425 MWt and approximately 5 MWt from the four EM pumps. The two IHX's are located above the reactor core in the annular region between a reactor shield barrel and the reactor vessel wall. The IHX cross section is kidney shaped to most effectively use the reactor space available and to minimize the overall diameter of the reactor vessel. All components of the IHX are constructed of austenitic stainless steel and the IHX is a safety-related component.

The IHX design consists of an upper and lower tubesheet separated by straight tubes, with a central downcomer and riser for incoming and outgoing intermediate sodium, respectively. Primary sodium from the hot sodium pool enters the IHX at an elevation below the upper tubesheet. The primary sodium flows downward between the tubes and shell to above the lower tubesheet, and exits into the IHX lower plenum. This lower plenum has two outlet nozzles where the primary sodium exits into the cold sodium pool of the reactor.

The cold leg intermediate sodium flows down the central downcomer, and splits into two streams below the lower tubesheet. Each stream then flows up through the straight tubes. The intermediate sodium exits the bundle just above the upper tubesheet into the annular space between the concentric external riser cylinder and internal downcomer cylinder. This sodium leaves the *IHX through the intermediate outlet nozzle for use in the* intermediate heat transport loop.

Control Rod Drives - There are six control rod drive (CRD) assemblies to regulate power, compensate for burnup, scram the reactor, and hold it subcritical during refueling. Each CRD consists of a drive mechanism, driveline, absorber bundle, and the absorber channel. The drive is mounted on top of the rotatable plug of the reactor vessel closure and controls axial motion of the absorber bundle in the core. The six control assemblies are scram actuated by a magnetic coil which, when de-energized, releases a latch mechanism allowing the absorber assembly to fall by gravity into the core. A drive-in mechanism is also incorporated into each control drive assembly to ensure rod insertion following scram release. The driveline is disconnected from the control rod and raised above the tops of the core channels to allow plug rotation during refueling.

The detailed description and evaluation of the drives is contained in Chapter 4.0.

In-Vessel Transfer Machine - The in-vessel transfer machine (IVTM) is mounted on the reactor rotatable plug which is centered within the reactor closure. The IVTM consists of two subassemblies, the above-head drive assembly and the in-vessel pantograph machine. The latter extends to reach all removable core assembly locations when used in conjunction with the rotatable plug.

The IVTM is used to move fuel and other core components between the core, storage racks, and the transfer station from which they are removed from the reactor by a shielded transfer cask. The in-vessel portion of the IVTM remains in the reactor vessel during operation, stored within the confines of the upper internal structure, which also provides support for the CRD drivelines. The above head portion is moved from module to module.

Reactor Core - The PRISM core has a heterogeneous configuration with an active core height of 46 inches. The core is comprised of 42 fuel, 25 internal blanket, 36 radial blanket, 60 radial shield, and 6 control assemblies. The metal core operates at low temperatures due to its high thermal conductivity with peak specific power of <14 kW/ft and due to the core outlet temperature of 875°F used in conjunction with a simple saturated steam cycle. The burnup reactivity swing is also relatively low such that the maximum reactivity insertion associated with a single rod withdrawal is about 0.22$.

The detailed description and evaluation of the core is contained in Chapter 4.0.

Core Support Structure - The core support structure provides the restraint of the reactor core assemblies necessary to maintain them in their prescribed geometry during all modes of reactor operation. This integrally welded structure is attached to the reactor vessel shell, also by welding, to form a rigid radial beam structure. This approach has large design margins and the added advantage that failure in a single member is of negligible consequence. The core support is located in the bottom end of the reactor vessel where the operating temperatures will be the lowest of the entire system and thermal transients, because of the distance from their source, will be benign.

The core restraint hardware assures the positioning of the core assemblies. It constrains any thermal, neutronic, seismic, or other condition that causes distortions and displacements of the core assemblies at the elevation of their top load pads. Load transfer is through the core assembly load pads to the plate segments and then to the former ring and the core barrel.

The inlet plenum, located in the central region of the core support structure and below the core, receives primary sodium from the eight primary pipes and distributes it to the core via the nosepiece receptacles. There are 169 receptacles (one for each core assembly). These are located in a triangular pitch to match the core array map. The receptacles participate in the core orificing. The depth of the inlet plenum is established by the space required for the inlet piping nozzle forging welds and the volume necessary to assure uniform flow distribution to all the core assemblies. This flow distribution is further enhanced by the design of the receptacles which are necked down on their lower end to increase the available flow volume.

Support Cylinder - The support cylinder is the structural member extending vertically upward from its attachment at the core support structure to the upper regions of the reactor hot plenum. It has a 10-foot outside diameter and is 2.0 inches thick at its lower end and 1.0 inch thick over its top four feet of length. Radially it is located outboard of the core barrel and inboard of most of the fixed shielding. The function of the support cylinder is to provide a base of support for all the reactor internal components except for the core. It also serves as the boundary between the reactor hot plenum and cold plenum regions and thus provides both thermal insulation and low pressure separation between these regions.

Fixed Shielding - Radiation shielding is provided within the reactor to limit the activation of secondary sodium flowing through the IHX, to limit the activation of impurities in the air flowing through the RVACS, to provide a radiation environment that accommodates the various neutron flux monitors, and to minimize the secondary fissioning in the stored fuel caused by thermal neutrons in the hot plenum. A small portion of this shielding also provides required irradiation protection for the core barrel structural welds. Other portions of the core support are sufficiently remote to not require dedicated shielding to prevent excessive material property degradation.

The fixed shielding is provided in three regions within the vessel: (1) core barrel shielding inside the core barrel, (2) IHX effluent shielding at the core elevation and outside the core barrel, and (3) IHX secondary sodium shields attached to the support cylinder at the IHX elevation level.

Primary Pump Inlet Manifold - Primary sodium is directed to the pump intakes through a manifold structure that surrounds the near core fixed shielding. Cold pool sodium enters this manifold after passing upward from the bottom of the reactor vessel through the four annuli surrounding the fixed shielding cylinders. Numerous flow holes in the brackets supporting the tops of the fixed shielding provide the communicating path between the annuli and the manifold. Sodium is collected here by a conical extension of the flow guide. This extension is attached to the support cylinder to close the manifold boundary. The pressure drop through the shield region is small because of the large flow area available.

The conical extension of the manifold has local appurtenances to interface with the primary pumps. These appurtenances are attached through sleeves to the seal plate and the pumps are provided with piston ring type seals at their bottom end to minimize bypass leakage.

<u>Reactor Vessel Liner and Seal Plate</u> - The reactor vessel liner provides steady state and transient thermal protection for the reactor vessel and forms a portion of the pressure boundary between the hot (outlet) plenum and the cold (inlet) plenum regions within the reactor. This boundary is completed by the seal plate which spans the gap between the liner and the support cylinder. The thermal liner provides support for the horizontal baffles which force thermal stratification in the upper volumes of the cold pool and thus minimize heat transfer between the hot and cold pools. Additionally, the fixed portions of the reactor closure shield and insulation plates are supported by the top of the vessel liner.

The vessel liner is a cylindrical member located 1.5 inches inside the reactor vessel between module reference elevations -1 ft-6 inches and -37 ft-9 inches. It is provided with slots at its top end that under normal operating conditions are always above the sodium level. During reactor operation these slots are approximately 12.0 inches above the hot pool sodium level and the cold pool communicates freely with the annulus between the liner and the reactor vessel at the support plate elevation. The free surface of the sodium within the inner annulus formed by the vessel and the liner will be approximately 10 feet below the hot pool free surface due to the IHX primary pressure drop. This causes the portion of the reactor vessel adjacent to the hot pool to be exposed to cover gas over its entire length, thus helping to insulate the vessel and minimize steady state heat losses to the RVACS. The liner also isolates the reactor vessel from rapid temperature changes in the hot pool sodium that result from duty cycle events, thus minimizing the thermal loading on the vessel, its attachment to the reactor closure, and to the containment vessel.

<u>Pump Discharge Manifold and Seals</u> - There are four EM pumps in the PRISM reactor module. These pumps interface with the reactor internals through two manifold assemblies that collect the high pressure pump discharge and distribute it, through pipes, to the core inlet plenum. Each manifold consists of a closed annular (90° arc length) shaped chamber and accommodates two of the EM pumps. Four 12-inch pipes exit the bottom of each discharge manifold and are routed downward to the reactor inlet plenum.

These pipes (total of eight) are designed to accommodate the pump discharge pressure and the differential thermal and seismic movements between their anchor points that arise during prescribed duty cycle events. The pipes penetrate the seal plate at reactor elevation -37 feet-9 inches where an Inconel 718 piston ring interfaces with the pipe periphery to form the seal to the seal plate.

<u>In-Vessel Fuel Storage</u> - Provisions are made within the reactor vessel for storage of spent fuel assemblies during reactor operation to allow them to decay to power levels sufficiently low as to permit "dry" handling and ex-vessel storage. This allows use of simpler equipment for fuel removal from the reactor, for fuel transfer between on-site buildings, and for spent fuel storage. Fuel is stored within the vessel for a period of one refueling outage. There are positions for storage of 22 core assemblies within the reactor. The decay power level of blanket assemblies is sufficiently low so as not to require in-vessel storage before removal.

The bottom end of the fuel assemblies in the storage positions are supported in a manner similar to that used in the core matrix. That is, a nosepiece receptacle is provided with internal features identical to the inlet plenum receptacle. The top end of the assembly is supported in a cup utilizing interfacing features on the outlet nozzle.

Core Assembly Transfer Station - Transfer of core assemblies into and out of the reactor vessel is accomplished with a straight push-pull type device operating through a fixed port in the reactor deck that is immediately outboard of the rotatable plug. A station is located directly below this port used for the temporary parking of the fuel transfer bucket while core assemblies are being transferred into and out of the bucket. Within the reactor vessel, assemblies are moved into and out of the bucket using the in-vessel transfer machine (IVTM). Detailed description and evaluation of the fuel handling equipment is provided in Chapter 9.1.

Upper Internals Structure - The upper internals structure (UIS) is attached to the rotatable plug of the reactor closure and cantilevered downward into the reactor hot pool. Its bottom end is located 2.0 inches above the top of the core assemblies during power operation. The principal functions satisfied by the UIS are: (1) lateral support of the control rod drivelines, (2) protection of the drivelines from sodium flow induced vibration, and (3) support of the above core instrumentation drywells.

The UIS overall length is 38 feet-8 inches and its outside diameter is 52 inches. Principal features of the UIS are the control rod driveline shroud tubes, the instrumentation drywells including the conduit and ducting used for their support, several horizontal structural plates, the structural cylinder, and insulation and shielding in the region below the reactor closure. Its major interface within the reactor system is with the in-vessel fuel transfer machine.

5.1.2 Primary Heat Transport System

The PHTS sodium flow path is contained within the reactor vessel. Sodium is routed through the reactor core, the hot pool, the shell side of the IHX, the cold pool, the pumps, the pump discharge piping and the core inlet plenum.

Flow paths in the PHTS are identified in Figure 5.1-2. Sodium from the hot pool enters and flows through the two IHX's where it is cooled. The sodium exists the IHX at its base and enters the cold pool. From there, cold pool sodium is drawn through the fixed shield assemblies into the pump inlet manifold. The four EM pumps intake the cold pool sodium from the manifold and discharge it into the high pressure core inlet plenum through the piping connecting each manifold to the plenum. The sodium is then heated as it flows upward through the core and back into the hot pool.

Each pump draws cold pool sodium from an inlet plenum beneath the pump. This is cold pool sodium from the IHX which has passed through the fixed radial shield region. Within the pump, the sodium enters the tapered inlet section of the pump duct where velocity is increased from ∿30 fps to the design velocity of ∿50 fps through the remaining two-thirds of the pump duct. The sodium discharge at the top of the pump passes radially outward into a plenum from which it is piped to the core inlet plenum.

5.1.3 Intermediate Heat Transport System

The intermediate heat transport system (IHTS) transfers reactor-generated heat from the PHTS to the steam generator system. The IHTS performs this function during normal power operation, during shutdown (decay heat removal), and during transient conditions. There is an IHTS loop for each reactor module. Each IHTS loop is thermally coupled to the reactor PHTS by two intermediate heat exchangers and to the steam generator system by a steam generator. IHTS non-radioactive sodium is circulated by a centrifugal pump located in the loop's cold leg to transport heat from the IHX to the steam generator. The IHTS piping extends from the IHX's tube side (IHTS) outlet nozzles to the steam generator shell side inlet nozzle and from the steam generator shell side outlet nozzle to the IHX tube side (IHTS) inlet nozzle. Each loop also contains a sodium expansion tank. The expansion tank allows the IHTS to function as a closed loop system without need for sodium makeup or removal to accommodate its thermal expansion.

The IHTS also includes the sodium drain piping, the sodium-water reaction pressure relief subsystem, and vent piping from the steam generator. The sodium-water reaction pressure relief subsystem provides pressure relief and gas venting capability to mitigate the effects of a tube failure in the steam generator. The IHTS piping is covered by guard piping within the head access area to mitigate the effects of a sodium leak. The IHTS piping and components in the steam generator building are located in separate, unshielded accessible cells.

5.1.4 Steam Generator System

The steam generator system provides independent steam generation capability for each of the IHTS loops. Each independent steam generator system is comprised of the following subsystems:

Steam Generator and Water/Steam Subsystem
Leak Detection Subsystem
Water Dump Subsystem
Auxiliary Heat Removal Subsystem

The steam generator subsystem obtains feedwater from the feedwater system. Feedwater enters the steam drum where it is mixed to subcool the saturated water from the steam generator. The subcooled water is then circulated by the recirculation pump from the drum back to the steam generator inlet nozzle. In the steam generator tubes, the subcooled water is heated and partially vaporized by the sodium flowing counter-current on the shell side. The saturated water and steam exiting from the steam generator tubes then flows to the drum where separators, internal to the drum, separate the water and steam. A small percentage of the saturated water (before the saturated water is mixed with the incoming feedwater) is then drained from the drum for water chemistry control and returned to the feedwater and condensate system. The saturated steam then flows through dryers, internal to the drum, to the turbine.

In order to reduce the amount of water which may be admitted to the IHTS in the event of large sodium/water reaction in steam generator, the system is isolated and emptied by the water dump subsystem. Water dump valves are located at the inlet to the steam generator. Water dump piping directs the water/ steam from the water dump valve to a dump tank where the flashed steam is vented to the atmosphere.

The steam generator leak detection subsystem monitors the sodium exiting the steam generator. Monitoring is by three hydrogen meters which provide a signal relative to the hydrogen concentration level in the sodium. In the event of a water-to-sodium leak, a change in the hydrogen concentration levels is detected. The resulting signals are conditioned, transmitted, and displayed. Off-normal conditions are annunciated.

5.1.5 Shutdown Heat Removal Systems

For normal plant operation when a reactor module is brought from full power down to standby or refueling, shutdown heat removal is accomplished by condenser cooling. Steam from the affected steam generator/drum is bypassed directly to the steam condenser. Here the steam is condensed and water is fed back to the steam drum, using one of the feedwater pumps. Feedwater flow to the steam drum is controlled by the drum local control system. Sodium in the intermediate loop is kept circulating by pony motor operation of the IHTS pumps. Flow of primary sodium through the core is maintained by natural circulation.

In the event condenser cooling is not available, shutdown heat removal for each reactor module is accomplished by the auxiliary cooling system (ACS) and limited venting of steam from the steam generators initially when water is still available. The ACS removes heat from the surface of the steam generator by natural circulation air flow. Steam venting can continue only until steam generator dryout conditions are reached. Before that condition is approached, reactor heat loads will have diminished to lower levels which can be handled easily by the combined cooling capability of RVACS and the ACS. This method would be used as a backup to the normal condenser cooling mode.

The RVACS removes heat directly from the reactor and acts as a backup to the normal condenser cooling and ACS. The RVACS transports heat to the atmosphere by natural circulating air flow. It functions continuously with its heat transport rate dependent on the primary sodium temperature. During a shutdown transient and simultaneous loss of the ACS and condenser cooling, the resultant higher primary sodium, reactor vessel and containment vessel temperatures will automatically increase the RVACS heat removal rate.

The inherent mechanisms employed in RVACS are natural convection of sodium within the reactor vessel for core cooling and thermal radiation combined with a natural convecting air stream to remove the decay heat from the reactor and containment vessels. Natural convection within the reactor vessel is promoted by heating of the sodium in the core and cooling along the vessel wall. Heat removal from the reactor vessel is accomplished primarily by radiation with a small contribution by natural convection of argon in the gap between reactor and containment vessels. The containment vessel in turn is cooled by natural convecting air and by radiation to the outside air duct wall referred to as the collector cylinder.

The RVACS operates continuously, but functions with its intended high heat removal rate only when all other reactor heat removal systems are inoperative. Should such a highly unlikely event occur, RVACS performs its function without any human or mechanical action.

5.1.6 Heat Transport Flow Diagram

The heat transport and power generation systems for a power block are shown in Figure 5.1-3. Heat is removed from the reactor core by the primary heat transport system, transferred to the intermediate heat transport system (IHTS) via the intermediate heat exchanger, and then transferred to the turbine generator system via the steam generator system. The PRISM 1245 MWe plant is made up of three identical power blocks each containing three reactors and one turbine-generator system. The complete standard plant employs nine reactors and three turbine-generators. One primary heat transport system, one intermediate heat transport system and one steam generator system is associated with each reactor module. The steam from the three steam generators (one per module) is combined and piped to a single turbine generator. Within each reactor module, four EM pumps circulate primary sodium and two intermediate heat exchangers transport reactor thermal power to a single intermediate heat transport system loop. A sodium expansion tank is provided to accommodate the thermal expansion/contraction of the sodium in the loop. A sodium water reaction pressure relief subsystem is provided to protect the IHTS in the event of a sodium leak in the steam generator.

Within the reactor module, the core generates 425 MWt of thermal power. Primary sodium core flow is 17.5×10^6 lbm/hr. The temperature delta between the hot and cold plena is 265°F.

The intermediate heat transport system circulates sodium from the tube side of the two IHX's, located within the reactor module, to the shell side of the steam generator. At full power, the intermediate sodium flow rate is 18×10^6 lbm/hr. Hot and cold leg temperatures are 800°F and 540°F, respectively. The two IHX's have a combined thermal rating of 430 MWt which accommodates the core power plus the heat inputs by the EM pumps.

The steam generator is designed to transfer 432 MWt from the intermediate system to the steam system. The steam generator obtains feedwater at a temperature of 420°F and flow rate of 1.88×10^6 lbm/hr from the feedwater system and produces 1.85×10^6 lbm/hr of saturated steam at 1000 psia and 546°F. The recirculation ratio is 1.2:1. The recirculation water temperature and pressure are 440°F and 1030 psi, respectively.

The turbine for each power block is a single 1800 rpm tandem compound four-flow unit with a rated output of 453 MWt at inlet steam conditions of 940 psig, 540°F. The turbine exhausts to a condenser operating at an average back pressure of 2.0 inches Hga. The condenser accommodates steam flows of approximately 3.1×10^6 lbm/hr and a heat load equal to 3.0×10^9 Btu/hr.

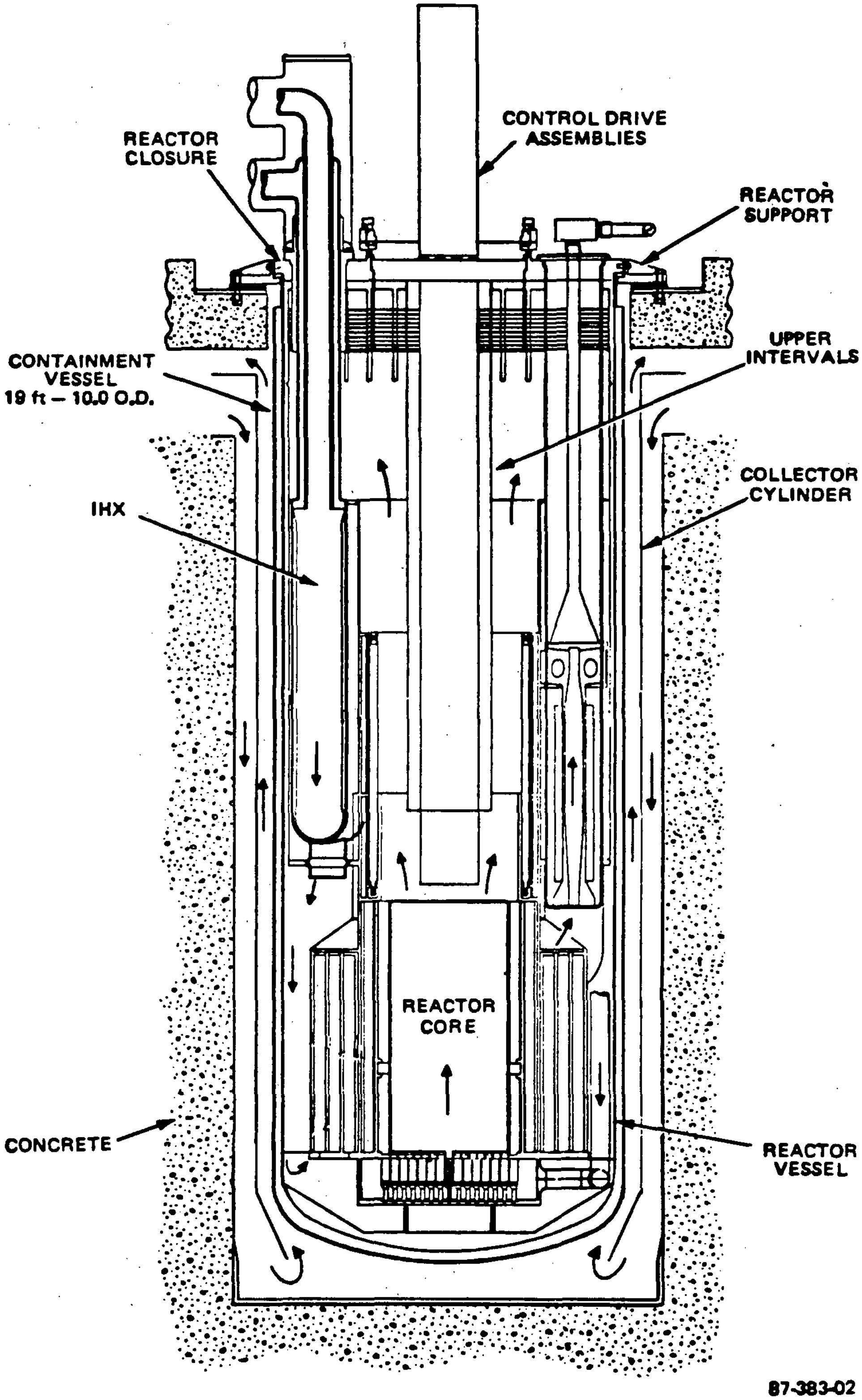

Figure 6.2–1 PRISM REACTOR MODULE

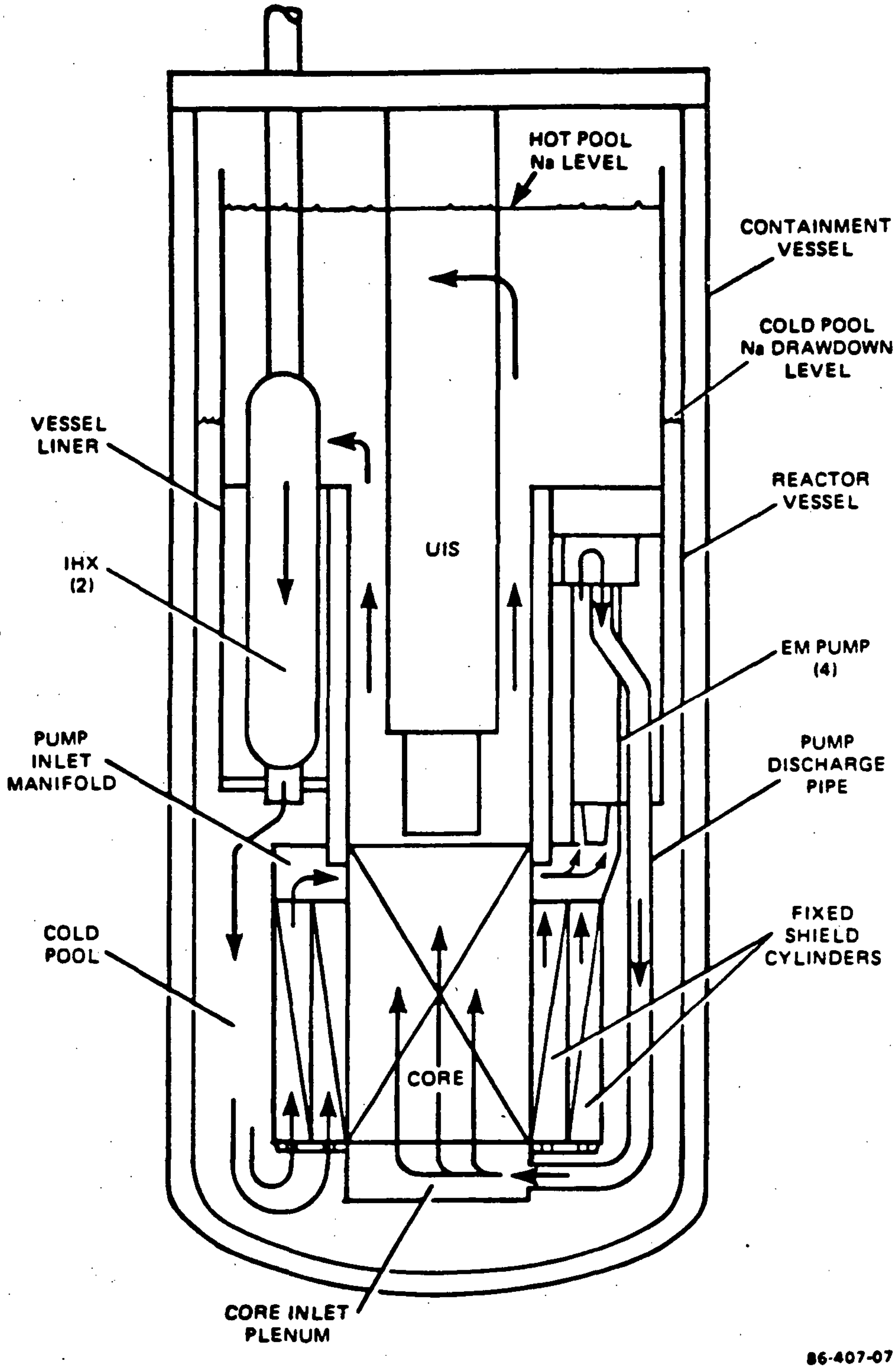

Figure 5.1–2 PRISM REACTOR PRIMARY SODIUM FLOW CIRCUIT – NORMAL OPERATION

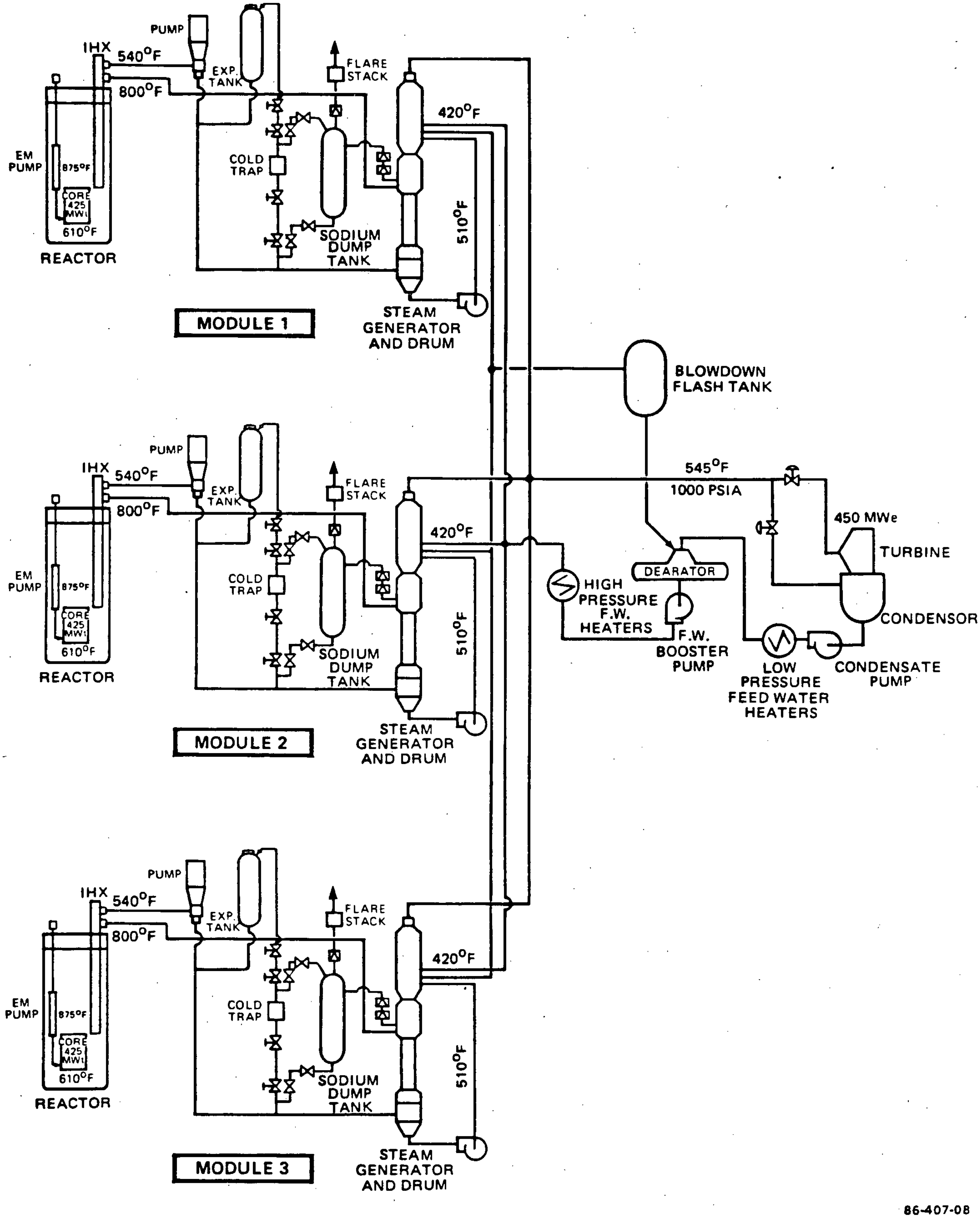

Figure 5.1–3 PRISM POWER BLOCK HEAT TRANSPORT FLOW DIAGRAM

5.2 Reactor Vessel, Containment Vessel and Closure Head

5.2.1 Design Basis

5.2.1.1 Functional Requirements

1. The reactor vessel shall support the reactor, reactor internal structures, and provide the primary boundary for the primary sodium coolant and its cover gas.

2. The reactor vessel and closure head shall provide mechanical support for various components, including those of primary heat transport and reactivity control subsystems.

3. The containment vessel shall provide a barrier to the release of radioactive material in the event of a leak in the reactor vessel. Its net volume shall be sufficiently small so that subsequent to a reactor vessel leak it shall maintain the primary sodium level in the reactor vessel at an elevation that will allow natural circulation of the sodium for cooling the reactor core.

4. A service life of 60 years shall be used as a basis for the reactor vessel, containment vessel, and closure head. The design shall be based on the operating hours from the design duty cycle specified in Appendix D.

5. The containment vessel, at its top flange, shall be the support point for the entire reactor module through its structure interfaces with the reactor closure and with the reactor module support subsystem components.

6. The reactor vessel and closure head shall provide a hermetically sealed containment boundary around the primary system fluids.

7. The reactor vessel and closure head (including the rotatable plug) and the reactor module supports shall limit vertical displacement of the center of the rotatable plug to less than 0.25" when the static loads of normal operation are applied.

5.2.1.2 Structural Requirements

The reactor vessel, containment vessel and closure head shall be designed to withstand all of the pressures, temperatures, and forces which are likely to be imposed in them. The design component conditions shall umbrella all of the respective service conditions. Evaluation for structural adequacy shall include duty cycle events and the operating basis and safe shutdown earthquakes. The design of the vessels and closure shall accommodate all fabrication, handling, transportation, and installation loads.

Design Conditions

The loading conditions to be taken into account in designing the vessels and closure head shall include but not be limited to the following: internal and external pressure, weight of the component and its contents, superimposed loads from other components, vibration and seismic loads, reactions at supports, temperature effects, irradiation effects, and the effects of the sodium environment. Design basis pressures and temperatures are shown in Tables 5.2-1 and 5.2-2. These conditions shall be used in conjunction with the plant duty cycle (Appendix D) to establish the thermal and mechanical loading conditions for the vessels and closure.

Seismic Criteria

The reactor vessel, containment vessel and closure head shall be capable of withstanding the effects of the Operating Basis Earthquake (OBE) without loss of capability to remain functional and to withstand the effects of the Safe Shutdown Earthquake (SSE) without loss of capability to perform their safety functions.

Appropriate analyses are required, using the ground motion inputs below, to define the specific design loads and accelerations for the containment vessel, reactor vessel, and closure.

(a) OBE Conditions

- The OBE horizontal and vertical maximum ground accelerations are 0.15g. The OBE response spectra shall be 0.5 times the SSE values. SSE values are given in Paragraph (b) below.

- Five OBE's, each with 10 maximum peak response cycles, shall be assumed to occur over the design life of the plant.

- Four of these OBE's shall be assumed to occur during the most adverse normal operating conditions (Level A Service Limit) determined on a component limiting basis. The fifth OBE shall be assumed to occur during the most adverse Service Level B event determined on a component limiting basis, and at the most adverse time in the event. The low probability for the OBE makes this event a Service Level B loading.

(b) SSE Conditions

- The SSE horizontal and vertical maximum ground accelerations are 0.3g. The SSE response spectra are given in Section 6.8.1 of Reference 5.2-1.

- One SSE, with 10 maximum peak response cycles, shall be assumed to occur over the design life of the plant.

- The SSE shall be assumed to occur during the most adverse Service Level A or Service Level B events. The probability of the SSE make it a Level D service condition.

Design Criteria

Design of the reactor vessel, containment vessel and closure head (including the rotatable plug) shall conform to the ASME B&PV Code, Section III, Subsections NCA and NB and for the vessels, also Code Cases N-47, N-48, N-49, N-50, and N-51.

Establishment of the limiting values for design stress intensity shall include allowances for any known or predictable degradation of mechanical properties that may occur as a result of irradiation, stress at service temperatures and changes in material properties over the design life.

The vessels and closure shall be designed as Seismic Category I structures.

5.2.1.3 Material Requirements

The materials of construction of the reactor vessel, containment vessel and closure head shall be selected on the basis of performance in fast reactor and liquid sodium environments.

1. The effects of environmental conditions such as neutron radiation exposure, temperature, and sodium shall be included in determining the allowable value of material properties.

2. Appropriate heat treatments and processes shall be utilized during component fabrication to minimize sensitization.

5.2.1.4 Instrumentation

1. The reactor closure head shall provide instrumentation that indicates when the rotatable plug is locked in the "zero" position for reactor operation. The plug shall have a direct reading angular scale for visual verification of plug position.

2. The reactor enclosure head shall provide temperature sensing devices to allow verification that elements of the closure are operating within prescribed temperature limits.

3. The reactor closure head shall provide access for instrumentation discussed in Chapter 7.0.

5.2.2 Design Description

5.2.2.1 Reactor Vessel

The reactor vessel (shown in Figure 5.2-1) is the immediate support and container for the primary sodium and the reactor internal structures. The core support structure makes direct welded connection near the bottom head of the vessel. Other than the core support connections and shipping restraints, the vessel has no attachments and no penetrations. The weight loads of the reactor internal structures, 431 tons, and the core, 84 tons, are transferred to the reactor vessel through the core support attachment. The vessel carries those loads and the 242 tons of contained sodium in tension up to its integral connection with the reactor closure.

A cylindrical shell with an integral hemiellipsoidal shell (the bottom head) make up the reactor vessel. Significant dimensions and some relevant design data are given in Table 5.2-3. Construction of the reactor vessel shall meet the requirements of Section III and Cases N-47 through N-51 of the ASME B&PV Code for Class 1 components. Table 5.2-4 lists service conditions of the reactor vessel and categorizes them in the Code defined service level loadings.

Under abnormal conditions the reactor vessel is subjected to temperatures higher than 800°F and ASME Code Case N-47 is therefore applicable for design. During normal operation, however, the 875°F reactor outlet sodium wets only the vessel liner; whereas the sodium wetting the vessel is at a level ten feet lower and a temperature about 325°F lower.

The drawn-down level between the vessel liner and the reactor vessel reduces temperature differences along the length of the vessel and greatly lessens concerns about creep in the vessel wall.

Service condition pressures are given in Table 5.2-4 and result from the sodium/cover gas volume and temperature changes in the reactor vessel and in the annulus between the reactor vessel and containment vessel.

The overall dimensions of the reactor vessel were established with consideration to the requirements for sodium level, RVACS surface area, refueling operations, and fit within the containment vessel. The need to fit within the containment vessel and maintaining the sodium level above the IHX inlet including potential vessel leak determined the reactor vessel's diameter. The dimensions given in Table 5.2.3 satisfy all requirements.

5.2.2.2 Containment Vessel

Several functions influence the design of the containment vessel. The containment vessel is classed as a nuclear component governed by the requirements of Section III of the ASME B&PVC. Possible service as a container for leakage of sodium from the reactor vessel means that the containment vessel could be subjected to temperature higher than 800°F, so its code nuclear component classification must be Class 1, the only acceptable upgrade from Class MC. Subsection NB of Section III and Code Cases N-47 through N-50 define the applicable requirements for design and construction. For its radiant heat transfer function in the RVACS, its construction will include special measures to provide surfaces with emissivities that meet design requirements. Lastly, the containment vessel flange is part of the support structure for the reactor module. The load of the module is transferred through the flange to the reactor module support subsystem. The compression load on the containment vessel flange is beneficial to maintenance of the seal between flange and closure.

Dimensions and other design data for the containment vessel are listed in Table 5.2-5. The given diameter has been intentionally limited to better fit within overland shipping constraints. The vessel length is slightly greater than that of the reactor vessel, and like the reactor vessel, provides the required RVACS heat transfer surface area. The length and diameter also limit the volume between the containment vessel and reactor vessel. With the volume limitation, sodium leakage from the reactor vessel will not lower the sodium level sufficiently to prevent continuance of flow into the IHX's and retention of the core cooling circuit.

The cylindrical shell of the containment vessel terminates at the top in a deep flange. The flange is bolted and sealed to the reactor closure. The seal between the containment vessel and reactor vessel is an Omega seal. As shown on Figure 5.2-5, the seal is welded to the containment vessel flange and the reactor closure. The argon pressure (at approximately 14 psig) that is present between the two vessels is continuously monitored. If a leak is detected, the reactor will be shut down and the abnormality investigated and corrected. The outer diameter of the flange has machined grooves into which fit tongues on the support segments of the reactor module support subsystem. The loads of the reactor module are carried in shear and bearing by the tongue and groove joints at the containment vessel flange, and are transferred to the seismically isolated HAA floor structure. The bottom head of the containment vessel is hemiellipsoidal with contour like the head of the reactor vessel.

The containment vessel normally operates at temperatures below 600°F, with a variation of about 220°F from the cooler bottom to the top. The normal operating pressure will be about 15 psig. The variations in temperature and pressure resulting from other than normal conditions are presented in Table 5.2-6.

5.2.2.3 Closure Head

The closure head is the "top head" of both the reactor vessel and the containment vessel, and is a major element of the containment boundary. It is designed to operate at comparatively low temperature (200°F-300°F). The low operating temperature is attained by inclusion in the design of 22 horizontal layers of stainless steel plates, the top one of which is 18 inches under the main structural plate of the closure. The plates are 5/8-inch thick and are spaced at 7/8 inches. They are supported on the reactor vessel liner.

The main structural plate of the closure is 12 inches thick and 19 feet-10 inches in diameter. At its outer edge it rests on, and is fastened and sealed to, the 5-1/2 inch wide top flange of the containment vessel. It is penetrated by the central rotatable plug, extensions of two IHX's and four EM pumps, a refueling port, and approximately thirty smaller nozzles and ports for various services and functions. The closure penetrations are shown on Figure 5.2-2 and are specified in Table 5.2-7.

The equipment for which the penetrations are made are, in general, supported by the closure. A notable exception is the load that is applied at the fuel transfer port penetration. Approximately 90 percent of the weight of the equipment attached at this penetration will be supported on exterior structures over the closure. The closure is designed for the normal and seismic loads applied to it by its own weight, the applied equipment loads, and attached shield weights, in combination with the cover gas pressures and service conditions given in Table 5.2-4. The governing design requirement is a 0.25-inch limit for the static load deflection of the closure under normal conditions.

For its function as part of the containment boundary, the closure must not allow leakage from the reactor vessel that exceeds 0.1 percent of the cover gas volume per day. The many penetrations of the closure must, therefore, be tightly sealed. All penetration joints, including the rotatable plug, the IVTM, and the fuel transfer port will have welded seals of some type. Because refueling operations require joint separation at the three aforementioned penetrations, reliable mechanical seals will be used for them, but provisions have been made in the design to make and break the welded seals.

A thin layer of thermal insulation covers the top surface of the closure. The thermal insulation limits the heat flow from the closure into the head access area.

All structural parts of the closure, other than bolting, are constructed of Type 304 stainless steel. The choice of bolting materials depends on function and environment, but is, in general, high alloy steel. The material for non-structural parts depends on function and environment. The structural connection of the closure to the containment vessel flange is made with cap screws over which seal welded covers are installed. The hold down force of the cap screws reduces moment and rotation at the joint, resists the pressure load in the between vessels annulus, and resists possible SSE joint separation forces.

The functions of the closure require that it be considered a Class 1 nuclear component governed by the requirements of Section III of the ASME B&PVC. The application of Code Case N-47 or the related code cases for elevated temperature construction is not necessary, however, so design and construction will be in accordance with Subsection NB. The cover gas retaining function of the closure also establishes its ISI examination requirements for compliance with Division 3 of Section XI of the ASME Code. The structural boundary of the closure allows continuous specified radiation monitoring in the head access area.

5.2.2.3.1 The Rotatable Plug (RP)

The rotatable plug is a non-integral, but mechanically attached, part of the reactor closure governed by the closure's design criteria. Its basic structure is much the same as that of stationary portion of the closure; a 12-inch thick load carrying plate with 22 layers of insulating plate underneath and thermal insulation on the top. The RP is penetrated by six CRD lines, the IVTM port, an ISI port, and a port for a cluster of above-core instrumentation conduits. In addition, the rotatable plug has suspended from its underside the upper internals structure (UIS) and the IVTM.

At the rim of the RP is an upstanding cylinder and flange which interface with a similar structure on the stationary portion of the closure. Within the vertical length of the interface are three graduated steps for blocking radiation streaming through the joint around the RP.

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The lower step also serves as a ledge seal and RP support during reactor operation. The mating flanges over the joint provide the mounting surfaces for the hardware of the bearings and seals of the RP.

5.2.2.3.2 Rotatable Plug Support and Sealing Arrangement

The bearing support and sealing arrangement for the RP are illustrated on Figure 5.2-3. As shown in this illustration, bolting down the rotatable plug during normal operation is by the tightening of tie-down bolts extending down through the top seal ring flange and into the fixed closure flange. The weight of the rotatable plug and the hold-down bolting force are reacted at the ledge of the closure opening on which the plug rests during operation. The sealing function of this ledge seal is to prevent sodium vapor deposition in the annular space above.

The RP primary seal is a pair of inflatable elastomer seal rings which seal against the largest diameter region of the RP riser. This seal is normally activated at all times except refueling. It is leak checkable by monitoring the space between the two seal rings.

The dynamic seal is the pair of inflatable elastomer seal rings near the top of the assembly. The primary function of this seal region is to serve as an argon gas buffered sliding seal during RP rotation for refueling operations. It serves as a backup seal while inflated during reactor operation. The argon gas buffer region between the two inflatable seal rings can be used for seal leakage monitoring.

The three seal regions described above plus the four sets of double O-ring elastomer seals between the bolted flange regions are expected to meet the gas leakage criteria for normal reactor operation. If higher integrity sealing becomes necessary, it could be provided in the form of an inverted U cross-section cover seal welded between the fixed closure head flange and the RP flange. Such a welded cover arrangement is shown on Figure 5.2-4.

In preparation for refueling, the RP hold-down bolts are loosened, the primary seal rings are de-pressurized (retracted) and the RP is raised to provide clearance at the set-down ledge seal region. Three jack assemblies, semi-permanently mounted on the fixed closure and connected to the upper seal ring flanges, provide the necessary lift action. The bellows seal provides for the vertical travel of the RP while preserving the system leak tightness. The top seal ring flange (and thus all intermediate flanges) are again clamped by use of support chocks in the space indicated just below the bellows region.

The various flange sections are individually bolted to one another and thus removable if necessary for part maintenance or replacement. If such a task required removal of the primary seal components, a temporary inflatable ring could be installed in the maintenance seal space provided near the fixed reactor closure flange.

5.2.3 Design Evaluation; Structural Evaluation Summary

A detailed steady state thermal analysis of the structures in the upper sodium pool region of the PRISM reactor was performed using the ANSYS-4.2 finite element computer code. Temperature distributions were calculated for the reactor vessel, liner, and containment vessel over the region bounded by elevation 0'-0" to minus 23'-0" (-276 inches). The condition analyzed represents normal full power operation. The results show that sharp thermal gradients exist in the vicinity of the sodium surface in the annular space between the reactor vessel and the reactor vessel liner (-230 inches). The maximum gradients are in the reactor vessel, and these produce compressive stresses on the inner surface of 32300 and 15400 pounds per square inch in the meridianal and circumferential directions respectively. These elastically calculated thermal stresses are less than the ASME Code (Section III-NB) secondary stress limit of $3S_m$. The stresses also satisfy the fatigue stress limit of the Code. Thus the thermal stresses in the reactor vessel are acceptable. Significant thermal stresses are also present in the reactor vessel liner. The magnitudes of the liner stresses are lower than those in the reactor vessel, and are also acceptable.

Transient conditions in the reactor were considered based on system analyses available to date. Of these, one event was judged to be the most significant from the standpoint of severity of induced thermal stresses. This event is reactor scram along with loss of power to all pumps. This causes the rapid rise of sodium in the annulus region between the reactor vessel and liner, producing a thermal shock to these structures. Therefore, a detailed transient thermal analysis was performed using the ANSYS-4.2 finite element computer code, and utilizing a variation of the analytical model used previously for the steady state case. Temperature distributions were calculated for the reactor vessel, liner, and containment vessel over the region bounded by elevation 0′-0" to minus 23′-0" (-276 inches). The results show that relatively large radial thermal gradients are produced in the reactor vessel and the reactor vessel liner. The axial gradient changes at the mid-walls of these structures are negligible. The maximum gradients are in the reactor vessel, and these produce compressive stresses on the inner surface of 33700 pounds per square inch in both the meridianal and circumferential directions. These stresses satisfy the ASME Code (Section III-NB) secondary stress limit of $3S_m$ and the fatigue stress limit and are therefore acceptable. Significant thermal stresses are also present in the reactor vessel liner. The magnitudes of the liner stresses are lower than those in the reactor vessel, and are also acceptable.

Steady state thermal conditions were also studied in the region of the reactor closure. The most significant thermal stresses found in this region occurs at the junction of the reactor vessel and the closure plate. The magnitudes of the stresses are 5200 psi in the meridional direction, and 1600 psi in the circumferential direction. These values are quite modest, and the potential for fatigue damage and/or ratcheting in this region is judged to be negligible.

The steady state and transient thermal analyses described previously all included the containment vessel in the analytical model, and therefore results were obtained for this structure as well. These show that the stresses produced are less severe than for the reactor vessel, and are acceptable.

Additional thermal transient analyses are expected to be performed for the reactor vessel and closure as system thermal/hydraulic studies are completed in the future. Also, further analyses will be made for the reactor structures subjected to seismic loadings. Based on preliminary studies of these, the current concept for seismic isolation indicates that the induced stresses will not be severe.

5.2.4 Compliance With Codes and Standards

The reactor vessel, containment vessel, and closure head are under the jurisdiction of the ASME Code, Section III, for Nuclear Power Plant Components and 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 temperature, Code Case N-47 will apply.

ASME Class 1, Class MC, or Seismic Category I components shall be designed to withstand the concurrent loadings associated with the Service Level B conditions and the vibratory motion of 50 percent of the Safe Shutdown Earthquake (SSE) or in other words, the Operating Basis Earthquake (OBE). The design limits for this case are specified in NB-3223 and NB-3654 of the ASME Code for vessels and piping. 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 Service Level A conditions, and Service Level D conditions.

For components at elevated temperature, the design stress limits are specified by Paragraph 3223, "Levels A and B Service Limits," and by Paragraph 3225, "Level D Service Limits," of Code Case N-47, 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 Appendix T.

References - Section 5.2

5.2-1 Specification 23A3071, Revision 0, "PRISM Design Requirements," dated March 1986.

TABLE 5.2-1

VESSEL, CONTAINMENT VESSEL, AND CLOSURE HEAD OPERATIONAL REQUIREMENTS

Primary Coolant	Sodium	
Primary Cover Gas	Helium	
Containment Vessel Gas	Argon	
Core Bulk Outlet Temperature	875°F	
Core ΔT	265°F	
Primary Circuit Pressure	100 psig	
Maximum Primary Sodium Temperature		
Service Level A	1100°F	
Service Level B	1100°F	
Service Level C	1200°F	
Service Level D	1300°F	
Reactor Sodium Refueling Temperature	400°F	
Reactor Hot Standby Sodium Temperature	550°F	
Gas Pressures	Primary Cover Gas	Containment Vessel Gas
400°F Refueling	0.55 atma	1.46 atma
400°F Cover Gas Cleanup	0.0 atma	1.46 atma
550°F Hot Standby	0.67 atma	1.71 atma
Normal Operating (875°F)	1.0 atma	1.86 atma
RVACS Conditions	1.54 atma	2.85 atma

TABLE 5.2-2

REACTOR MODULE PRESSURE DROP ALLOTMENTS

	Maximum Allowed Pressure Loss at 100% Flow, psi
Reactor Core, Including Inlet Components	85
Reactor Hot Plenum and Structures	0.5
Intermediate Heat Exchangers	4.0
Reactor Structures (IHX Discharge to Pump Inlet)	1.0
Reactor Structures (Pump Discharge to Inlet Plenum)	5.0
Reactor Core Inlet Plenum, Not Including Core Inlet Components	2.0
TOTAL	97.5

TABLE 5.2-3

REACTOR VESSEL DESIGN DATA

Vessel Outer Diameter	18'-10"
Vessel Shell Length	58'-4"
Main Cylinder Wall Thickness	2"
Bottom Head Thickness	2"
Construction Material	Type 316 Stainless Steel
Design Temperature	900°F
Design Pressure	±20.0 psig

TABLE 5.2-4

REACTOR VESSEL SERVICE CONDITIONS IN COVER GAS REGION

Service Condition	Service Loading Level	Hot Pool Sodium/ Maximum Vessel Temperatures °F	Cover Gas Volume, Ft	Internal Pressure psia*
Refueling; Vessels Closed	A	400/390	2038	0.0 21.5 Ext.
Hot Standby; Vessels Closed	A	550/540	1892	10.0 25.2 Ext.
Normal Operation (N.O.)	A	875/770	1589	12.2 to 17.2 14.7 Nom. 27.3 Ext.
Loss of IHTS, RVACS Cooling	C&D	1155/1130	1226	22.6 41.9 Ext.
N.O., Detected Sodium Leak	D	400/390	-	-
N.O., Undetected Sodium Leak	D	875/770	-	-

* Unless noted

TABLE 5.2-5

CONTAINMENT VESSEL DESIGN DATA

Vessel Outer Diameter	19'-10"
Vessel Lengths	
Main Shell, Including Top Flange	59'-6"
Main Cylinder Wall Thickness	1"
Bottom Head Thickness	1"
Top Flange Size	5-1/2" Wide x 12" Deep
Containment Vessel-To-Closure Seal	1" Dia. Omega Ring
Construction Material	2-1/4 Cr-1Mo Low Alloy Steel
Design Temperature	800°F
Design Pressure	20 psig

TABLE 5.2-6

CONTAINMENT VESSEL SERVICE CONDITIONS

Service Condition	Service Level	Maximum Temperature °F Max.	Pressure psia
Refueling; Vessels Closed	A	235	-
Hot Standby; Vessels Closed	A	320	22.5
Normal Operation	A	445	29.4
Loss of IHTS; RVACS Cooling	C	785	37.7

TABLE 5.2-7

CLOSURE PENETRATION SCHEDULE

Function or Service	Number	Opening Description	Location (Centerline Radius & Azimuth)
Rotatable Plug (RP)	1	Flanged & Stepped 103" Min. Dia.	0
CRD Lines	6	7.5" Dia.	In RP
IVTM	1	≅20 Dia	In RP
Above-Core Instruments	1	4" Dia	In RP
IHX	2	Kidney-Shaped 38" Wide With 131" Arc Length	87" 90° & 270°
EM Pumps	4	39" Dia.	85" 22°,158°,202°,338°
Fuel Transfer Port/ Preheat Exit	1	Flanged & Stepped 12" Min. Dia.	72" 180°
ISI Ports	12	3" Dia. (Coincident With Bolt Holes)	
	1	12" Dia	In RP
Cover Gas Processing	1	1" Dia.	104" 355°
Sodium Processing Return/Preheat	2	3" Dia.	104" 47°, 313°
Flux Monitors	4	3" Dia.	69" 0°,137°,223°
Sodium Processing Supply (With EM Pump)		20" Dia.	In IHX Plug 47°, 313°
Thermocouples	8	1" Dia.	In IHX Plugs & Pump Plugs
Delayed Neutron Detectors	2	2-1/2" Dia.	In IHX Plugs
Sodium Level Gages	5	≅3 Dia.	In Pump Plugs & Refueling Port
Fission Gas Monitor	1	12" Dia.	In IHX Plug

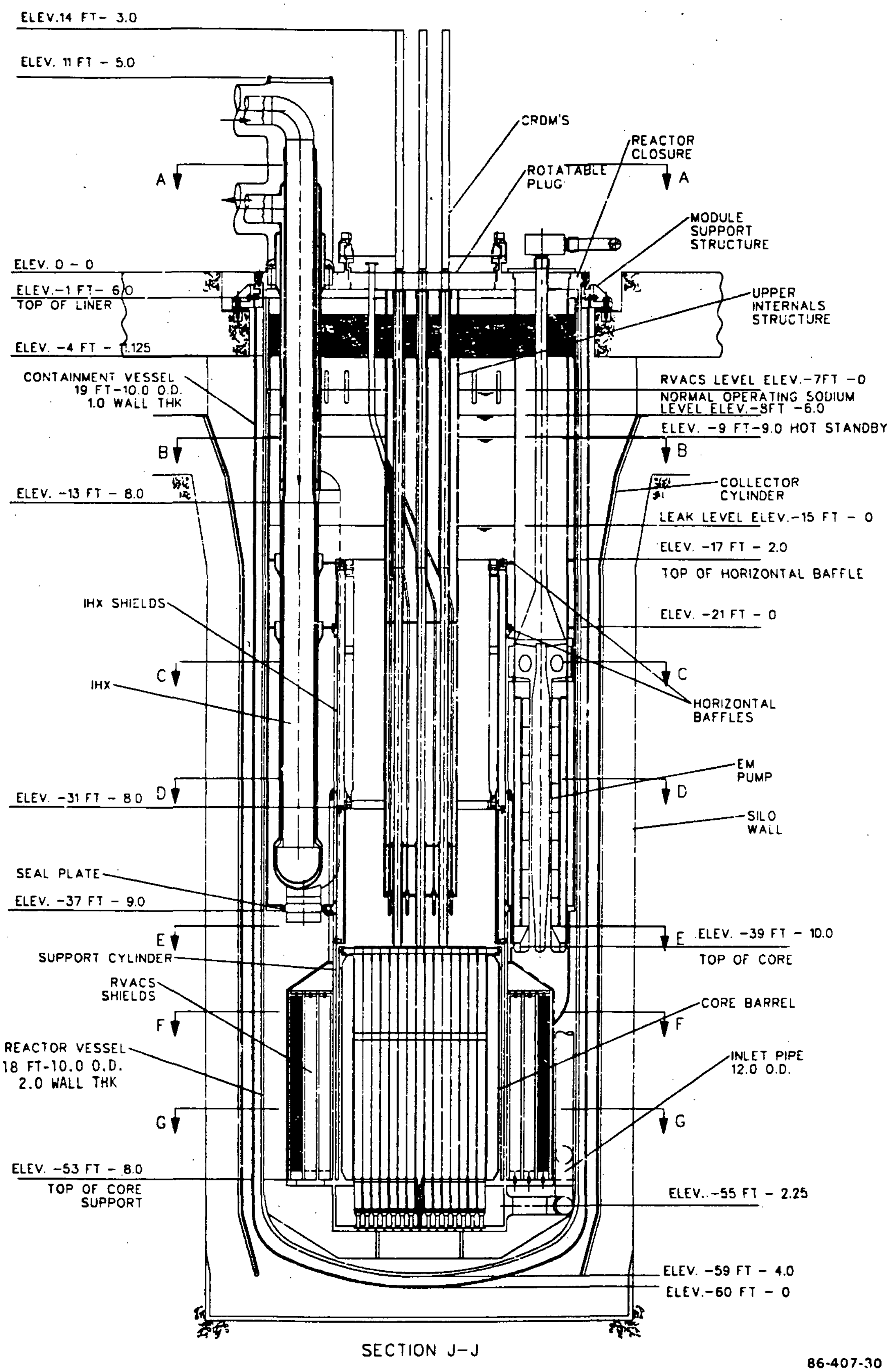

Figure 5.2–1 REACTOR MODULE ELEVATION

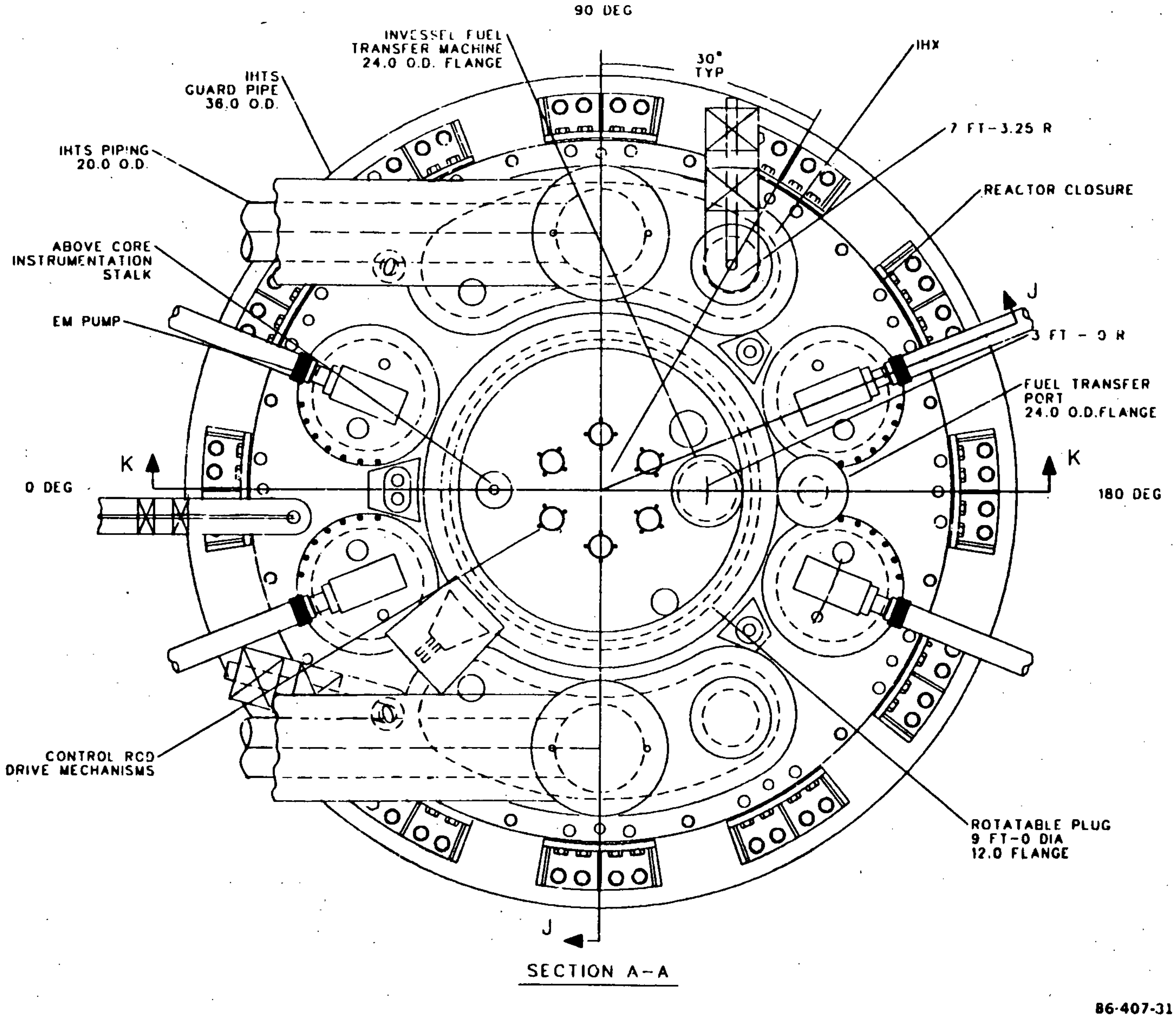

Figure 5.2–2 REACTOR CLOSURE

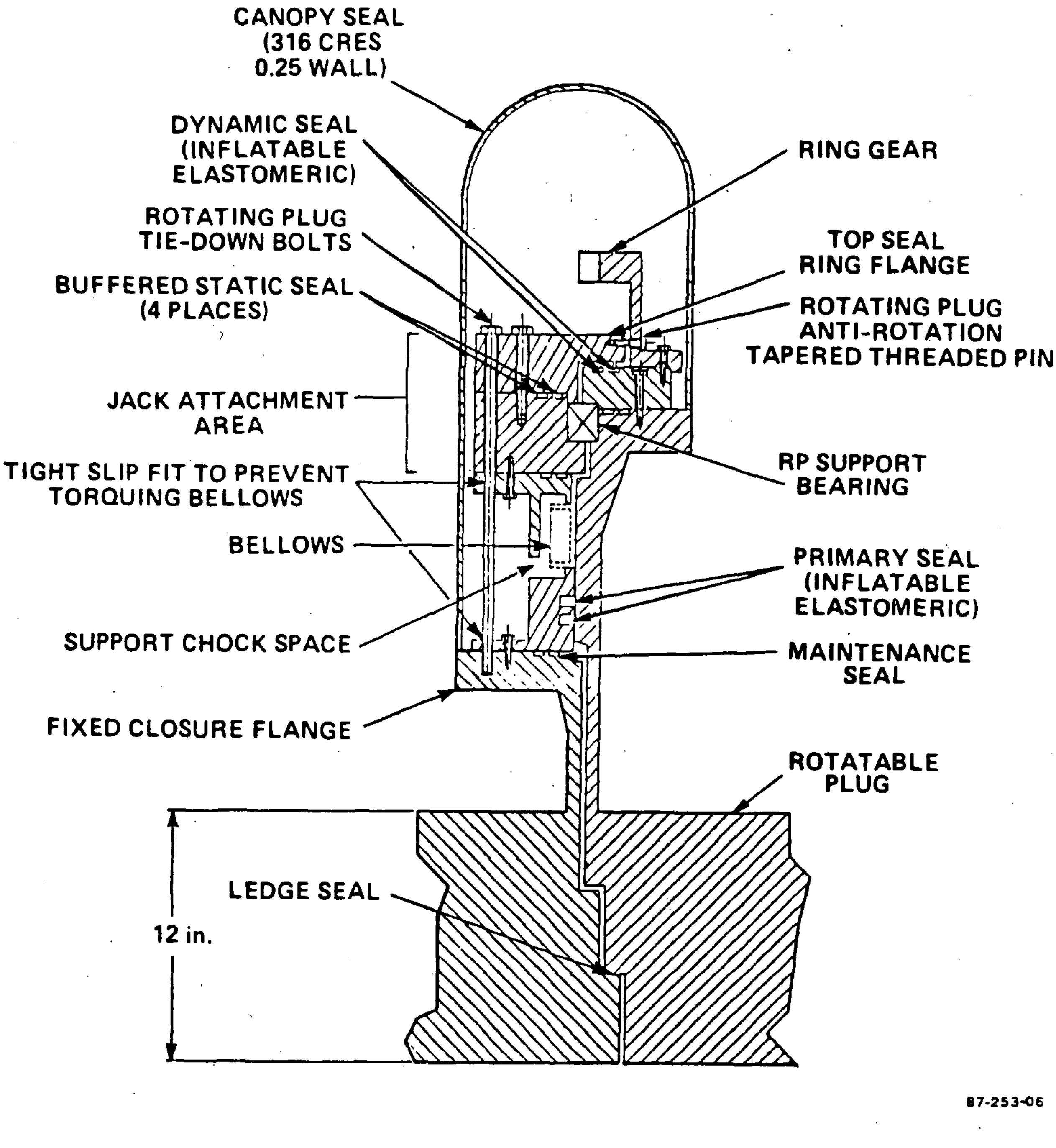

Figure 5.2–3 ROTATABLE PLUG SUPPORT AND SEAL ARRANGEMENT

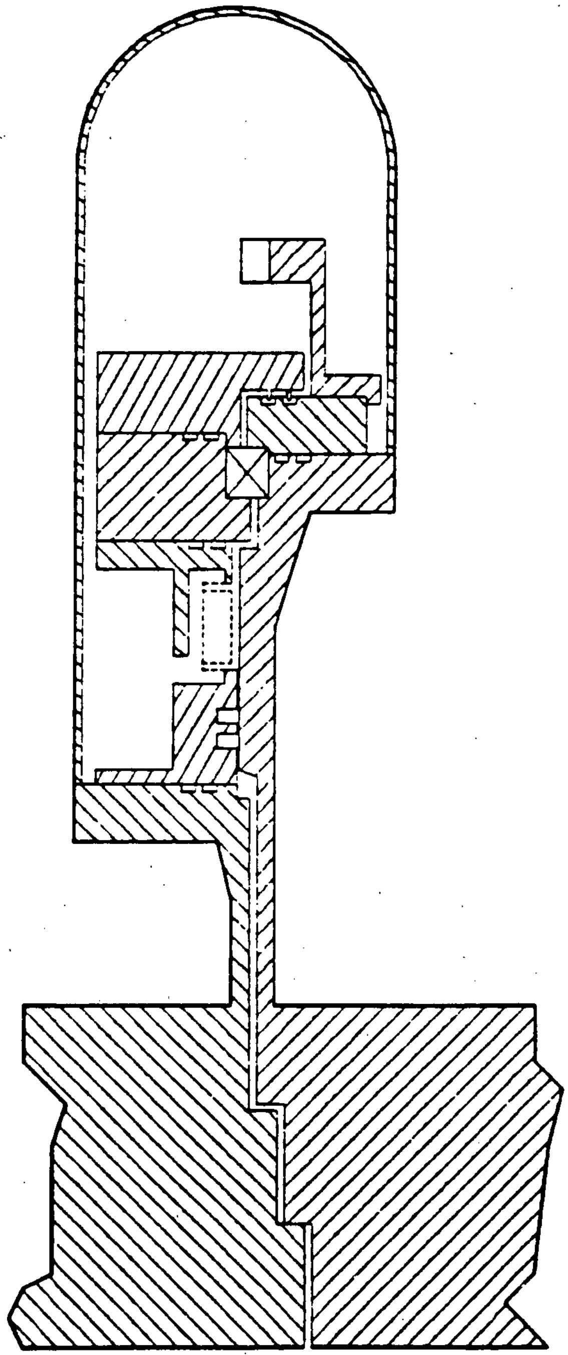

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Figure 5.2—4 ROTATABLE PLUG JOINT WITH WELDED SEAL COVER

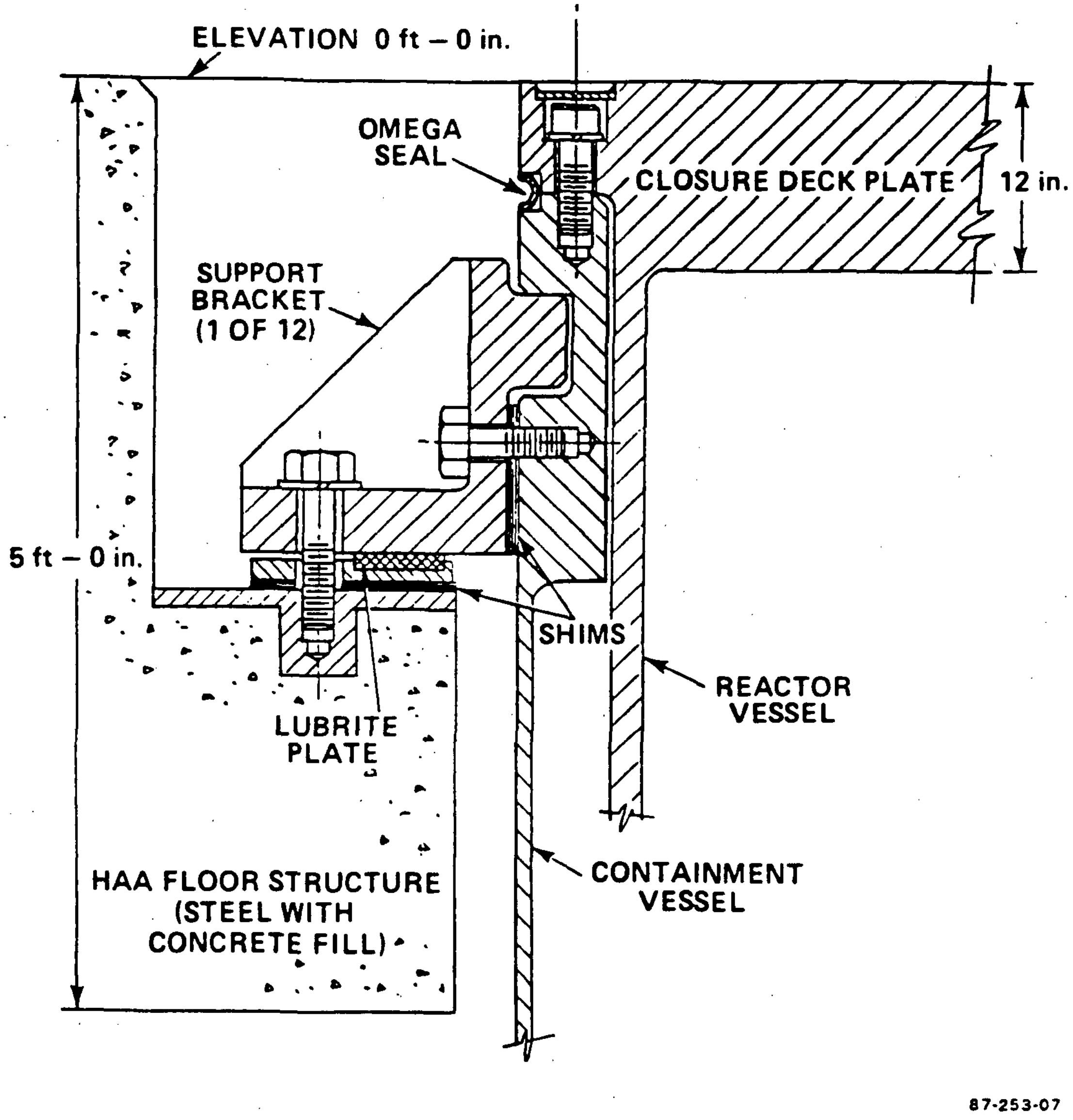

Figure 5.2–5 MODULE SUPPORT ELEVATION

5.3

5.3 Reactor Internal Structures

5.3.1 Design Basis

5.3.1.1 Functional Requirements

1. Provide in-vessel structural support for the core, reactor instrumentation, primary heat removal system equipment, in-vessel piping, fuel transfer equipment, and in-vessel stored fuel.

2. Provide the flowpath for primary sodium inside the reactor vessel for both forced and natural circulation cooling of the core.

3. In conjunction with the reactor vessel, containment vessel, closure head, and PHTS, provide shielding to limit the activation of secondary sodium passing through the intermediate heat exchanger (IHX) and ambient air passing through the reactor vessel auxiliary cooling system (RVACS).

4. In conjunction with reactor vessel, containment vessel, closure head, and PHTS, limit the irradiation levels within the Head Access Area (HAA) such as to permit personnel access during reactor operation.

5. Provide thermal baffles to protect structures and minimize heat losses between hot and cold plena within the reactor.

6. Provide an upper internal structure to support control rod driveline shroud tubes, in-vessel fuel transfer machine guides, and above core instrumentation.

7. Provide for the distribution of coolant to individual core assemblies and near core structure as appropriate.

8. Provide features and/or devices to prevent the hydraulic fluid forces from levitating core assemblies.

9. Limit, in conjunction with the reactor vessel, closure head and supports, the maximum vertical core displacement.

10. Limit, in conjunction with the reactor vessel, closure head and supports, the horizontal seismic deflection and acceleration to within the capability of the control rod drive and the structural and functional limits on the core assemblies.

11. Provide flowpaths to assure adequate and predictable operation of the RVACS during all duty cycle and beyond design basis events.

12. A service life of 60 years shall be used. Those items which cannot be reasonably expected to last the 60-year life of the plant shall be either sufficiently redundant or easily replaceable such that plant availability is not adversely affected. The design shall be based on the operating hours from the design duty cycle specified in Appendix D.

13. In conjunction with the reactor vessel support and the reactor vessel, limit the maximum vertical core deflection relative to the reactor closure to 0.25 inches during SSE.

14. Provide for the support and in-vessel storage of 16 core fuel subassemblies during reactor power operation at locations accessible to the in-vessel transfer machine of the reactor refueling system.

5.3.1.2 Structural Requirements

The reactor internal structures shall be designed to withstand all of the pressures, temperatures, and forces to which they are subjected. The design conditions shall umbrella all their service conditions. Evaluation for structural adequacy shall include duty cycle events and operational basis and safe shutdown earthquakes. The design of the reactor internal structures shall include all fabrication, handling, transportation, and installation loads.

Design Conditions

The loading conditions to be taken into account in designing the reactor internal structures shall include but not be limited to the following: internal and external pressure, weight of the component and its contents, superimposed loads from other components, vibration and seismic loads, reactions at supports, temperature effects, irradiation effects, and the effects of the sodium environment. Design basis pressures and temperatures for the system are shown in Tables 5.2-1 and 5.2-2. These conditions shall be used in conjunction with the plant duty cycle (Appendix D) to establish the thermal and mechanical loading conditions for the reactor internal structures.

Seismic Criteria

The internal structures shall be capable of withstanding the effects of the Operating Basis Earthquake (OBE) without loss of capability to remain functional and to withstand the effects of the Safe Shutdown Earthquake (SSE) without loss of capability to perform their safety functions.

Appropriate analyses are required, using the ground motion inputs below, to define the specific design loads and accelerations for the reactor internal structures.

(a) OBE Conditions

- The OBE horizontal and vertical maximum ground accelerations are 0.15g. The OBE response spectra shall be 0.5 times the SSE values given in Paragraph (b) below.

- Five OBE's, each with 10 maximum peak response cycles, shall be assumed to occur over the design life of the plant.

- Four of these OBE's shall be assumed to occur during the most adverse normal operating conditions (Level A service limit) determined on a component limiting basis. The other one OBE shall be assumed to occur during the most adverse Service Level B determined on a component limiting basis, and at the most adverse time in the event. The low probability for the OBE makes this event a Service Level B loading.

(b) SSE Conditions

- The SSE horizontal and vertical maximum ground accelerations are 0.3g. The SSE response spectra are given in Section 6.8.1 of Reference 5.3-1.

- One SSE, with 10 maximum peak response cycles, shall be assumed to occur over the design life of the plant.

- The SSE shall be assumed to occur during the most adverse Level A or B service conditions determined on a component limiting basis. The probability of the SSE implies a Level D service condition. During and following the SSE, the primary and intermediate pumps are assumed to be functioning at pony motor speed.

Design Criteria

Design and construction of core support structures and designated reactor internal structures shall conform to the ASME B&PV Code, Section III, Subsection NG and Code Case N-201.

Establishment of the limiting values for design stress intensity shall include allowances for any known or predictable degradation of mechanical properties that may occur as a result of irradiation, stress at service temperatures and changes in material properties over the design life.

The reactor internal structures shall be designed as Seismic Category I structures.

5.3.1.3 Material Requirements

The materials of construction of the reactor internal structures components shall be selected on the basis of performance in fast reactor and liquid sodium environments. Constituent elements whose transmutations have long half-lifes shall be controlled to minimize their impact.

1. The effects of environmental conditions such as neutron radiation exposure, temperature, and sodium shall be included in determining the allowable value of material properties used in the design of system components.

2. Material surface in contact with the liquid sodium coolant shall be austenitic stainless steel unless other materials must be used for strength or wear resistance.

3. Surfaces that experience relative motion during operation, installation, or removal shall be made of suitable material combinations or shall be provided with hard-surfaced regions to provide adequate wear properties and to preclude galling or seizing.

4. Appropriate heat treatments and processes shall be utilized during fabrication to minimize sensitization of stainless steel components.

5.3.2 Design Description

The principal function of the reactor internal structures is to provide the mechanical support and restraint of the reactor core. Additionally, these structures also participate in providing restraint for the primary components, providing control and direction of the primary coolant within the reactor system and supporting the in-vessel shielding necessary for biological protection. Those components that comprise the reactor internal structures are shown on Figure 5.3-1. Except in the few cases indicated below, all the internal structures are fabricated from austenitic stainless steel, thus eliminating concerns over oxidation, differential thermal expansion, dissimilar metal welds and post-weld heat treatment.

5.3.2.1 Core Support Structures

The core support structure provides the restraint of the reactor core assemblies necessary to maintain them in their prescribed geometry during all modes of reactor operation. This integrally welded structure is attached to the reactor vessel shell, also by welding, to form a rigid radial beam structure. This approach has large design margins and the added advantage that the consequences of failure in a given member is negligible. The core support is located in the bottom end of the reactor vessel where the operating temperatures is the lowest of the entire system and thermal transients, because of the distance from their source, will be negligible.

The major elements of the core support structure are shown on Figure 5.3-2. These are:

1. The weldment comprised of radial beams and support plates;
2. The primary sodium inlet plenum which contains the receptacles for the subassembly nosepieces;
3. The core barrel and the core restraint rings;

The core support function is subdivided into lateral and vertical restraint. Each sub-function is provided in a different fashion by the core support structures. Figure 5.3-3 shows the restraint subsystem features.

For lateral restraint, the core assemblies are held (1) by their nosepieces in the receptacles, and (2) by the load pads near the top of the assemblies which are surrounded by a core restraint ring attached to the core barrel. The separation of the assemblies is maintained by an intermediate plane of load pads at an elevation above the active core. Positioning of the handling sockets is also maintained by the top load pads. The intermediate load pads above the core are not restrained by former rings attached to the core barrel. Thus, the core assemblies are free to bow as

5.4.2 Design Description

5.4.2.1 Introduction

The PHTS sodium flow path is contained within the reactor vessel. Sodium is routed through the reactor core, the hot pool, the shell side of the IHX, the cold pool, the pumps, the pump discharge piping and the core inlet plenum.

Flow paths in the PHTS are identified in Figure 5.4-1. Sodium from the hot pool enters and flows through the two IHX's where it is cooled. The sodium exits the IHX at its base and enters the cold pool. From there, cold pool sodium is drawn through the fixed shield assemblies into the pump inlet manifold. The four EM pumps intake the cold pool sodium from the manifold and discharge it into the high pressure core inlet plenum through the piping connecting each manifold to the plenum. The sodium is then heated as it flows upward through the core and back into the hot pool.

5.4.2.2 Primary EM Pump

Four submersible EM pumps provide primary sodium circulation through the reactor. The pumps are installed through penetrations in the fixed portion of the closure head into an annular area above the core shared with intermediate heat exchangers. The pump assembly is shown in Figure 5.4-2. Each pump is approximately 40 inches in diameter by approximately 19 feet in length. Primary sodium coolant is drawn from an inlet plenum beneath the pump. This is cold sodium from the IHX which has passed through the fixed core radial shield region.

The pump design configuration is shown in Figure 5.4-3. As depicted on this illustration, sodium enters through a large annular opening at the bottom of the pump. Within the pump, the sodium converges to the tapered inlet section of the pump duct where velocity increases from approximately 30 fps to the design velocity of approximately 50 fps through the remaining 2/3 of the pump duct. The sodium discharge at the top of the pump passes radially outward into a plenum from which it is piped to the core inlet structure. There are three reactor internal structure seal plate interfaces for the piston ring seals of the pump - one seal plate at the pump inlet and two seal plates near the top of the pump forming part of the discharge plenum.

As shown in Figure 5.4-3, the pump stator is located radially outward from the aforementioned pump duct. It is in an inert gas-filled enclosure formed by the outer pump duct wall, the external stator support cylinder, and the end forgings to which these cylindrical sections are welded. The electrical power leads are routed from the stator enclosure, through a conduit across the pump outlet plenum, and into the lifting/ handling structure which extends upward through the reactor vessel closure head.

This pump is self-cooled in that the heat generated by electrical losses in the stator is transferred to the surrounding sodium. Most of this heat energy is transferred through the duct wall into the pumped sodium since that sodium boundary has the best thermal coupling to the heat source. A smaller portion is transferred radially outward through the stator support cylinder. Since all heat losses are transferred into the primary sodium coolant, the adverse effect on overall plant efficiency is minimized.

6. Support of the EM pump's discharge manifolds and lateral support of the pumps;
7. Support of the horizontal baffles.

Structural interfaces associated with each of the above functions are discussed below with the specific components.

5.3.2.3 Fixed Shielding

Radiation shielding is provided within the reactor to limit the activation of secondary sodium flowing through the IHX, to limit the activation of impurities in the air flowing through the RVACS, to provide a radiation environment that accommodates the various neutron flux monitors, and to minimize the secondary fissioning in the stored fuel caused by thermal neutrons in the hot plenum. A small portion of this shielding also provides required irradiation protection for the core barrel structural welds. Other portions of the core support are sufficiently remote to not require dedicated shielding to prevent excessive material property degradation. The fixed shielding is provided in three regions within the vessel: (1) core barrel shielding inside the core barrel, (2) IHX effluent shielding at the core elevation and outside the core barrel, and (3) IHX secondary sodium shields attached to the support cylinder at the IHX elevation level.

Near core shielding, shown on Figure 5.3-2 and Figure 5.3-4, is comprised of four thick cylinders which are supported on the upper core support grid plate. There are two types employed: a) two 8.0 inch thick solid steel, and b) two cylindrical B_4C shields as described below. Both of the steel shields are located immediately outboard of the support cylinder. Beyond these two shield is one of the B_4C shields. These three components are provided primarily to limit the irradiation of any contaminants that might be exhausted from the RVACS with the air, however, they also serve a secondary role of reducing the neutron fluence levels near the bottom of the IHX's. Lateral restraint for these cylinders is through shear pins at the bottom end and through brackets attached to the support cylinder at the top end. Each of these three cylinders are spaced two inches apart and it is through these gaps that the cold sodium returning to the core passes. The remaining B_4C shield is located immediately inside of the core barrel and serves to protect this component from neutron damage. The core barrel shielding is exposed to the highest neutron flux and so has been designed as removable shield assemblies, which can be replaced before problematical deterioration of the B_4C or its two containments.

Above the top of the core and near the IHXs B_4C shield assemblies are provided to limit the activation of the secondary sodium passing through the IHXs and to reduce the amount of fissioning in the stored fuel. This shielding is shown in elevation on Figure 5.3-5 and in plan on Figures 5.3-6 and 5.3-7 for two different sections through the module. They are supported by brackets attached to either the inside or the outside of the support cylinder. Those on the inside surface of the support cylinder and near the core outlet are protected by stainless steel liner plates from the high sodium flows and the rapid temperature changes that will occur in the hot pool.

The design of the B_4C elements of the fixed shielding is based upon the design employed for the control absorber assemblies. For this application, natural B_4C powder is compressed into pellets at 70% of theoretical density. The pellets are loaded into stainless steel pins of approximately 1.0 inch in diameter and sealed with a cap weld. The pins are then mounted in thin walled (0.25 inches) stainless steel cans in a tightly packed array. The cans provide the mechanical support and protection for the pins and also serve as a secondary barrier against the leakage of B_4C into the primary sodium in the event of a pin failure. B_4C cleanup is not provided for because leakage of the sintered pellets is not expected. During the initial fill of the reactor with sodium, the cans will also fill through small holes in their tops and bottoms. During reactor operation the trapped sodium will remain essentially stagnant although it may breath with temperature changes.

The parameter governing the life of the pins is the neutron fluence that they experience over their life and the amount of helium and tritium generated as a result of this fluence. In the case of the most highly irradiated shield element the total 60-year fluence will be less than that seen by a control assembly over its design life. Additionally, the temperatures at the shield locations are less than in the control assemblies so that the service conditions for the shields are easier to accommodate. The

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individual pins are designed with sufficient voids to accumulate the generated gases and do not require venting. Figure 5.3-8 presents the conceptual arrangement of the B_4C shield structurals.

5.3.2.4 Primary Pump Inlet Manifold

Primary sodium is directed to the pump intakes through a manifold structure that surrounds the near core fixed shielding. Cold pool sodium enters this manifold after passing upward from the bottom of the reactor vessel through the four annuli between the fixed shielding cylinders. The boundaries of the manifold region are formed by the support cylinder and the flow guide as shown on Figure 5.3-9. Numerous flow holes in the brackets supporting the tops of the fixed shielding provide the communicating path between the annuli and the manifold. Sodium is collected here by a conical extension of the flow guide. This extension is attached to the support cylinder to close the manifold boundary. The pressure drop through the shield region is small because of the large flow area available.

The conical extension of the manifold has local appurtenances (or nozzles) to interface with the primary pumps. These appurtenances are attached through sleeves to the seal plate and the pumps are provided with piston ring type seals at their bottom end to minimize bypass leakage. The interface with the seal plate is discussed in Paragraph 5.3.2.5 (Reactor Vessel Liner and Seal Plate).

The principal function of this extended manifold design is to enhance the operation of the RVACS. The manifold assures that sodium drawn by natural circulation through the pumps and into the core will always come from the lowest elevation and, thus the coolest region within the reactor vessel.

5.3.2.5 Reactor Vessel Liner and Seal Plate

The reactor vessel liner provides steady state and transient thermal protection for the vessel and forms a portion of the pressure boundary between the hot (outlet) plenum and the cold (inlet) plenum regions within the reactor. This boundary is completed by the seal plate which spans the gap between the liner and the support cylinder as shown on Figures 5.3-1 and 5.3-10. The thermal liner provides support for the horizontal baffles which force thermal stratification in the upper volumes of the cold pool and thus minimize heat transfer between the hot and cold pools. Additionally, the fixed portions of the reactor deck shield and insulation plates are supported by the top of the vessel liner.

The vessel liner is a cylindrical member located 1.5 inches inside the reactor vessel between elevation 1 ft-6 inches and 37 ft-9 inches. It is provided with slots at its top end that under normal operating conditions are always above the sodium level. During reactor operation these slots are approximately 12.0 inches above the hot pool sodium level and the cold pool communicates freely with the annulus between the liner and the reactor vessel at the support plate elevation. The free surface of the sodium within the inner annulus formed by the vessel and the liner will be approximately 10 feet below the hot pool free surface due to the IHX primary pressure drop. This causes the portions of the reactor vessel adjacent to the hot pool to be exposed to cover gas over their entire length thus helping to insulate the vessel and minimize steady state heat losses to the RVACS. The liner also isolates the reactor vessel from rapid temperature changes in the hot pool sodium that result from duty cycle events, thus minimizing the thermal loading on the vessel, its attachment to the reactor closure, and to the containment vessel.

The bottom end of the liner is located at the elevation of the IHX discharge nozzle. Here the horizontal seal plate is welded between the liner and the support cylinder to complete the pressure boundary across the hot and the cold pools. The seal plate has a number of circular penetrations that allow the IHX's, the primary pumps, and the primary piping to pass through while providing a lateral seismic support for these components. The seal plate has been positioned at a vertical elevation that minimizes the size and number of penetrations and eliminates the need to seal on non-circular perimeter of the IHX body. Details of the seal plate are shown on Figure 5.3-11.

During events requiring RVACS for total heat removal, the reactor resident sodium will heat up, expand, and flow through the liner slots. Since the pumps are not operating in this event, the sodium level will be the same on both sides of the liner. A significant fraction of the hot pool sodium bypasses the inoperative IHX's and is cooled through the reactor vessel wall. The cooled sodium sinks to the bottom of the vessel where it is drawn into the pump inlet manifold and directed back to the core to complete the circuit.

5.3.2.6 Pump Discharge Manifold and Seals

There are four EM pumps in the PRISM Reactor Module. These pumps interface with the reactor internals through two manifold assemblies that collect the high pressure pump discharge and distribute it, through pipes, to the core inlet plenum. Each manifold consists of a closed annular (90° arc length) shaped chamber and accommodates two of the EM pumps. Four 12-inch pipes exit the bottom of each discharge manifold and are routed downward to the reactor inlet plenum. Figures 5.3-12, 5.3-13, and 5.3-14 show the EM pump discharge manifold, EM pump outlet pipe routing configurations, and the pump seal/lateral support configurations.

The EM pump is axially supported by a flange which is mounted on the reactor closure assembly. At reactor systems elevations 17 ft-2 inches and 21 feet, the pump passes through the close fitting holes of the horizontal baffle plates. At elevations 21 ft-8 inches and 24 ft-2 inches, the EM pump has double piston ring seals. These two elevations represent the pump junctions to the upper and lower discharge manifold plates. These pump piston ring seals are designed to contain the high pressure within the pump discharge manifold while permitting relative vertical motion between the pump and the manifold plates. At elevation 37 ft-9 inches the pump passes through the seal plate. A cylindrical nozzle is welded between the seal plate and the pump inlet plenum, as shown on Figure 5.3-14 and on 5.3-10. The pump inlet plenum at elevation 39 ft-10 inches, is sealed to the pump using a piston ring seal. Each of the piston rings is made of Inconel 718 material which contacts a stellite hardfaced cylindrical surface.

While designed for the 60-year life of the plant, the pump manifold and pump support design permits removal of the pump by unbolting the pump flange from the reactor module closure head. The piston rings remain with the pump upon removal. This permits a new set of piston rings to be installed upon installation of a repaired or replacement pump.

There are eight 12-inch diameter stainless steel pipes which are routed from the pump discharge manifolds to the reactor inlet plenum. These pipes are designed to accommodate the pump discharge pressure and the differential thermal and seismic movements between their anchor points that arise during prescribed duty cycle events. The pipes penetrate the seal plate at reactor elevation 37 feet-9 inches where an Inconel 718 piston ring interfaces with the pipe periphery to form the seal to the seal plate.

5.3.2.7 IHX Seals and Supports

The support and the seals for the Intermediate Heat Exchanger (IHX) are located at the seal plate (reactor module elevation 37 feet-9 inches), and in the reactor closure.

The vertical support of the IHX occurs at the reactor closure. A mounting flange on the IHX's vertical support cylinder is bolted to the reactor closure as shown on Figure 5.3-15. The support cylinder is welded to the IHX riser pipe which carries all the vertical weight. The lateral support is provided at the reactor closure and at the seal plate as shown in Figure 5.3-11. The IHX primary outlet nozzles (two nozzles per IHX) form the load path from the IHX to the seal plate.

The IHX mounting flange is part of the reactor primary boundary and is sealed at the reactor closure with the use of metallic "O" rings which are compressed by the bolting of the IHX mounting flange. A non-structural seal is welded in place at the flange interface. At reactor module elevations 17 feet-2 inches and 21 feet, horizontal baffles exist, which contain loose fitting kidney-shaped holes for the IHX to fit within. These plates restrict the circulation of hot sodium from penetrating downward into the annulus between the IHX and the reactor support cylinder. At reactor elevation 37 feet-9 inches, the sodium annulus surrounding the IHX is sealed to limit leakage from the hot pool to the cold pool by the seal plate. Inconel 718 piston rings on each primary outlet nozzle of the IHX are used to contact the stellite surfaces of the seal plate. These piston rings form the seal between the hot and cold pools of the reactor module.

The method of supporting and sealing the IHX described above permits complete removal of the IHX by cutting the flange seal weld, unbolting the IHX mounting flange from the reactor closure head and then vertically removing the IHX.

The seal plate geometry is shown on Figure 5.3-11. The seal plate, in addition to separating the reactor module hot and cold pools, acts as the vertical support for the reactor module liner and the annular horizontal baffles. With this design, a radial gap exits between the horizontal baffles and the support cylinder. The location of this radial gap leads to a smaller sodium leakage area then would occur if the horizontal baffle plates were supported by the support cylinder and the radial gap is located at the reactor vessel thermal liner.

5.3.2.8 In-Vessel Fuel Storage

Provisions are made within the reactor vessel for storage of spent fuel assemblies during reactor operation to allow them to decay to power levels sufficiently low as to permit "dry" handling and ex-vessel storage. This is done to simplify the equipment used ex-vessel to remove fuel from the reactor system, transfer it between on-site buildings and to store it. Spent fuel, including failed fuel, is stored within the vessel for a period of one refueling outage. The decay power level of blanket assemblies is sufficiently low and these units do not require in-vessel storage.

The system is designed with 22 storage positions to accommodate both the reference metal core and a backup oxide core storage requirements. Two of the positions are unassigned spares for the reference metal core and five unassigned spares result for the backup oxide core. All storage positions are located in the hot plenum above the top of the core barrel.

They are attached to the top of the support cylinder as shown on Figure 5.3-16. The bottom end of the fuel assemblies in the storage positions are supported in a manner similar to that used in the core matrix. That is, a nosepiece receptacle is provided with internal features identical to the inlet plenum receptacle. The top end of the assembly is supported in a cup utilizing interfacing features on the outlet nozzle.

5.3.2.9 Core Assembly Transfer Station

Transfer of core assemblies into and out of the reactor vessel is accomplished with a straight push-pull type device operating through a fixed port in the reactor deck that is immediately outboard of the rotatable plug. A station is provided directly below this port for the temporary parking of the fuel transfer bucket while core assemblies are being transferred into and out of the bucket. Within the reactor vessel, assemblies are moved into and out of the bucket using the in-vessel transfer machine (IVTM).

The transfer station is shown on Figures 5.3-17 and 5.3-18. It is comprised of: a thimble which is supported from the reactor deck, and of the appropriate interfacing structures and access provisions within the reactor internals.

The transfer thimble, shown on Figure 5.3-17, is a cylindrical component 12 inches in diameter and 42 feet long. It is supported vertically from a flange at the top end which sets on the fuel transfer port nozzle. The lower end is supported laterally by interfacing structures connected to the support cylinder. The thimble has internal tracks which guide the fuel transfer bucket as it is lowered or raised within the thimble. The bi-stem drive mechanism for raising and lowering the bucket also operates within these tracks so as not to obstruct the fuel handling path. Core assemblies are moved into and out of the thimble through a long vertical window designed into its mid-length. While removable from the reactor module, the thimble is designed for the 60-year life of the reactor.

Within the module, a U-shaped slot is provided for the thimble through the horizontal baffles, one of the pump discharge manifolds and the pump inlet manifold. This slot, located in the upper end of the support cylinder, interfaces with the thimble window to allow movement of core assemblies as required. These features are shown on Figures 5.3-17 and 5.3-18.

5.3.2.10 Hot Pool Thermal Insulation

Insulation is provided between the reactor's hot pool and cold pool to reduce the amount of energy bypassing the IHX's. This insulation, which is comprised of various components, is typically used primarily to satisfy some other function. The overall design is shown on Figure 5.3-19. Basically the insulation is formed by the support cylinder and IHX shielding attached to the inside and outside surfaces of the support cylinder and by the horizontal baffles.

Insulation at the vertical boundary of the hot pool is derived principally from the IHX radiation shields. At the elevation range corresponding to the IHX insulation cross-section is comprised of approximately 4.0 inches of B_4C in pins, thin-walled cylinders containing the pins, trapped sodium, and the support cylinder. Between elevations 31 feet-0 inches and the top of the core, the insulation consists of the support cylinder with similar contained B_4C pins on both its inside and outside surface. Additionally, the innermost diameter which is exposed directly to the core outlet transients and to thermal striping will be protected by three stainless steel liner plates. Each of these plates is 0.5 inches thick and is attached to the shielding. These liners increase the effectiveness of the total insulation in a region where hot-to-cold pool temperatures are greatest.

For the horizontal boundaries between the hot and cold pools, horizontal baffles between vessel liner and the support cylinder at elevation 21 feet-0 inches and 17 feet-2 inches serve to stratify the sodium outside of the support cylinder. This stratified sodium will be at a temperature intermediate to the reactor hot pool and cold pool temperatures thus reducing the need for insulation in this region. The total heat transferred through the vertical and horizontal insulators is estimated to be approximately 0.5% of reactor thermal power.

5.3.2.11 Instrumentation and Equipment Support

The reactor structures provide mechanical support for invessel sensors and equipment items that are required by other systems. This support for instrumentation is primarily in the form of drywells with the actual instrument being provided by the requesting system. For other equipment, space in the reactor and penetrations through the closure head and rotatable plug are provided. Table 5.3-1 presents a list of this supported instrumentation and identifies the system or function requiring it. The arrangement for this instrumentation on the reactor closure is shown on Figure 5.3-20. A discussion of the various items supported is given below. Note that instrumentation and equipment that will be located outside of the reactor module (such as the wide range flux monitors) are discussed in Section 7.0.

Core Outlet Temperature Measurements - These measurements will be made in the region directly above the core outlet nozzles and in the IHX inlet region. The above core measurements are attained by temperature monitors contained in drywells supported by the UIS as discussed in Paragraph 5.3.2.12 below. The IHX inlet temperatures are attained by thermocouples contained in drywells that are integral with the IHX. These drywells access the reactor through the IHX support flange, pass downward along the riser pipe, and enter the inlet region after passing through the high pressure secondary sodium dome on the top of the IHX. All these temperature monitors are replaceable.

Sodium Level Measurements - Access for devices to continuously monitor the primary sodium level in the reactor vessel cold and hot pools is provided in each of the primary pump mounting flanges and in the fixed deck. These indicators will sense small changes in the sodium level that will indicate a leak either into or out of the primary system. Additionally, there are continuity detectors in the containment vessel that indicate the presence of sodium in the space between the two vessels. These devices enter the annulus through ports in the deck and terminate in the region below the reactor vessel bottom head. Differential sodium level measurements will be used as an indirect measure of primary sodium flow through the core.

<u>Flux Monitors</u> - Drywells for the in-vessel source range flux detectors enter the reactor deck through three stations located immediately outboard of the rotatable plug flange. The three source range flux detectors are located in-vessel and within five-inch I.D. drywells extending straight downward from the rotatable plug to approximately six inches above the core and within the radial envelope of the upper internal structure (UIS). The three drywells are azimuthally positioned at 120° intervals such that shadow shielding by the control rod neutron absorber material is minimized.

<u>Delayed Neutron Detector</u> - The failed element detection system employs two delayed neutron detectors. Drywells for these items are mounted in the IHX support flange. The drywells pass downward through the flange, the secondary sodium dome, and the upper tube sheet. They then pass through the tube bundle and are terminated at a point above the lower tube sheet.

<u>Fission Gas Monitors</u> - Each module will support one fission gas monitor. This device will be mounted in an IHX support flange as shown on Figure 5.3-20. The IHX provides a penetration with a mounting flange as required by the monitor interface. The monitor will extend downward through the flange and the IHX shield and insulation plates to the cover gas. It does not impact the lower portion of the IHX body.

<u>Initial Fill Equipment</u> - The initial fill of the reactor system involves a preheat phase and a subsequent charging with liquid sodium. The system will be heated by hot gas which is injected through two permanently

installed pipes. These pipes enter the reactor deck through penetrations below the IHTS piping and are routed to the bottom head region of the reactor vessel. The gas is discharged through the fuel transfer port using temporary fittings and, after reheating, returned to the reactor fill pipes. Following heating to the prescribed temperature level, one of the fill pipes is disconnected from the gas supply and connected to the sodium supply. Sodium is then introduced in to the system through this pipe. The progress of the filling operation is monitored with instrumentation installed through the ISI and Maintenance port in the rotatable plug. When the filling process is completed one of the fill lines is capped at the reactor deck and the other is used as the return for the sodium cleanup system.

Core Inlet Temperature and Pressure - The core inlet temperature is measured at the discharge of the primary pump. Thermocouples are installed in drywells that are integral with the pump construction. There is one drywell in each pump. The core inlet pressure is also measured at the pump discharge plena.

Sodium and Cover Gas Purification System - These systems require supply and return lines at the reactor module. A single line for the cover gas is required. For sodium return, one of the sodium fill lines (see above) is used. Access to the reactor for sodium supply is provided through the flange of one of the IHX's. This opening is sufficiently large to accommodate the small EM pump that is installed in the upper regions of the reactor and used to circulate sodium to the cleanup cell. The cover gas system requires a single 1.0 inch line that accesses the reactor through the deck. Here a penetration through the head plate and into the cover gas is provided.

5.3.2.12 Upper Internals Structure

The upper internals structure (UIS) will be attached to the rotatable plug of the reactor enclosure and cantilevered downward into the reactor hot pool. Its bottom end will be located 2.0 inches above the top of the core assemblies during power operation. The principal functions satisfied by the UIS are: (1) lateral support of the control rod drivelines, (2) protection of the drivelines from sodium flow induced vibration, and (3) support of the above core instrumentation drywells.

The UIS is shown in elevation on Figure 5.3-22. Its overall length is 38 feet-8 inches and its outside diameter is 52 inches. Principal features of the UIS are the control rod driveline shroud tubes, the instrumentation drywells including the conduit and ducting used for their support, several horizontal structural plates, the structural cylinder, and insulation and shielding in the region below the reactor closure. Its major interface within the reactor system is with the invessel fuel transfer machine.

Shroud Tubes - There is one shroud tube assembly for each of the six control rod drivelines. Each assembly consists of an upper SS-316 tube, a lower Inconel Alloy 718 tube, and an internal bushing. The tubes are sized and located such that the drivelines pass through the center of the assemblies without contacting the tubes except over a region near mid-elevation where the close fitting guide bushing is located. As shown on Figure 5.3-23, the Inconel Alloy 718 bushing is positioned within the tube by a series of close fitting diameters and a shoulder that supports it in the vertical direction. It is mechanically attached to the SS-316 shroud tube with a SS-316 ring which is pinned and welded to the tube body. The SS-316 tube itself is welded to the horizontal structural plate near elevation 20 feet. From here it extends upward to the reactor rotatable plug and downward to the lower UIS structural plates. The upper end is captured in a nozzle protruding from the underside of the plug. The lower end is similarly captured in the lower Inconel Alloy 718 tube which, as shown on Figure 5.3-24, is mechanically attached to the lower structural plates of the UIS. This arrangement minimizes the thermal interaction between the shroud tubes and the UIS structurals due to differential temperatures and materials and eliminates the need to use dissimilar metal welds. Inconel Alloy 718 was selected for the bushing because of its mechanical wear properties. It was selected for the lower tube to sustain the thermal striping and thermal shock conditions existing near the core outlet.

Instrumentation Drywells - There are twenty drywells routed from the top of the rotatable plug to the region directly above the reactor core. These pass through and are supported by the UIS. Each SS-316 drywell is 0.5 inches in diameter and will carry multiple thermocouples. The lower end of the drywells, which are located approximately 1.5 feet above the core outlet, are contained in heavy Inconel Alloy 718 forgings which provide structural support and thermal protection. The most severe mechanical loads would be the sodium flow induced vibration if not precluded by the post design. The Inconel Alloy 718 material is specified to guard against thermal fatigue that would occur in, say SS-316, due the steady state thermal striping and the thermal shock arising during scram transients.

The drywells are evenly distributed over the core outlet plane so as to provide information on the various core regions. Above this plane they are routed so as to allow them to exit the reactor through a single port thus minimizing the number of required plug penetrations. Between the port and the instrument post the drywells are enclosed in conduits and ducts that facilitate their routing and provide mechanical protection.

UIS Structurals - The principal structural member of the UIS is the 52 inches in diameter SS-316 cylinder that is attached by welding to the rotatable plug and extends down to approximately three feet above the core outlet. This cylinder has a vertical slot through its side that extends for the lower 25 feet of length and provides access for the invessel fuel transfer machine. The cylinder's wall thickness of 1.0 inches was selected to assure adequate resistance to seismically induced displacements thus satisfying requirements on the motions of the control rod drivelines relative to the core. Additional stiffness for the cylinder is obtained from the three horizontal plates that are welded to its walls and, except for the radial refueling machine slots and penetrations for the shroud tubes, span the area inside of the cylinder. The bottom end of the structure is provided with two liners that protect it from the thermal environment at the core outlet. The outermost liner is made from Inconel Alloy 718 and is used for thermal striping protection. The second SS-316 liner, in conjunction with outer liner is used to insulate the structure against rapid temperature changes occurring during scram transients.

5.3.3 Design Evaluation

The adequacy of the reactor internal components to sustain the gravity, pressure and seismic loads has been ascertained through comparison of their structural response with the appropriate ASME Code stress limits and the system functional deformation limits.

The initial PRISM thermal-hydraulic development has focused on the inherency response and functional performance; the thermal environment is not yet characterized sufficiently to permit detailed thermal stress analyses. Therefore, the component response to thermal loads is presently assessed primarily through judgment and comparison with CRBR and LSPB which were designed for considerably higher temperatures and more severe thermal transients. Thermal loads on the components subject to relatively large mechanical loads (core support structure) are expected to be small. Conversely, mechanical loads on the components subject to large thermal loads (outlet plenum components and the vessel thermal liner) are also small. Also, the permissible thermal stress limits are relatively large compared to the primary stress limits. Therefore inclusion of the thermal loads, when they are better defined, is not expected to increase the stress levels beyond the design limits for most of the components. Possible exceptions are some parts of the upper internal structure and the IHX shielding which may be subject to large thermal transient and striping loads because of direct exposure to the core effluent. If larger than expected, these loads may require additional austenitic, Modified 9Cr-1Mo, or Inconel 718 liners which can be easily accommodated in the reference design concept.

5.3.3.1 Design Criteria

The structural design of the reactor internal structures was assessed using the rules of the ASME Code Section III, Subsection NG and the Code Case N-201. The functional adequacy was assessed through clearance analyses. Specifically it was ascertained that seismic loads will not produce interference between the control rod drivelines and their enveloping components, and impact between the reactor thermal liner and the reactor vessel.

5.3.3.2 Load Estimates

Mechanical Loads - Mechanical loads considered in the design evaluation are the gravity, a coolant pressure of 100 psi in the core inlet plenum, and the specified 0.3g SSE free-field ground vertical and horizontal zero period accelerations. Finite element soil-structure interaction analyses were performed for a range of soil parameters covering the PRISM siting envelope to obtain the reactor system seismic loads. These analyses used the Bechtel synthetic design earthquake time-history with response spectra enveloping the NRC Regulatory Guide 1.60 design spectra (Figure 5.3-25), the Bechtel finite element code BSAP, and the analysis model shown in Figure 5.3-26. The soil-structure interaction and the horizontal seismic isolators were modeled as spring elements and the reactor components were modeled as beams, springs, and lumped mass elements. The response spectra calculated at the isolator support were expanded ±15% to obtain design response spectra and used in response spectrum analyses of the structure including the isolators using the ANSYS Code and the model of Figure 5.3-26 but without the soil-structure interaction springs.

The component peak accelerations from the system analyses are summarized in Table 5.3-3. In the horizontal direction, the entire reactor system vibrates essentially as a rigid body at the isolation system frequency of 0.75 Hz. Correspondingly, the component peak accelerations are about the same and can be enveloped by a peak acceleration value of 0.5g. Since the system essentially vibrates at a single frequency, no modal combinations are involved, and the components horizontal SSE response can be calculated in detailed reactor system analyses for a statically applied 0.5g inertial load.

With no vertical seismic isolation, there is a considerable variation in the component vertical accelerations as shown in Table 5.3-2. The peak accelerations for the reactor vessel and the structures supported from the vessel may be enveloped by an SSE peak acceleration of 1.0g. This includes contributions from different modes combined using the 'Square Root of Sum of the Square' approach. Therefore, the component vertical SSE responses can be obtained from a detailed reactor vessel/supported system analysis using a 1.0g inertial load applied statically.

Thus, 0.5g horizontal and 1.0g vertical equivalent static loads are sufficient for evaluation of the reference design for the currently specified 0.3g SSE vertical and horizontal design criteria. To accommodate possible increases in the vertical excitations, the design was evaluated for 1.5g rather than 1.0g equivalent static load. This load was combined with the natural gravity load to give a 2.5g loading used in the analysis.

<u>Thermal Loads</u> - The most significant thermal loads on the internal structures are the thermal striping during normal operation and the thermal shock during transient operation on the components directly exposed to the core exit coolant. This includes the instrument support posts, the control rod driveline/shroud tubes, the UIS lower plate, and the IHX shielding in the outlet plenum. The outlet plenum temperature gradients are not yet well characterized but the PRISM temperatures and thermal transients are milder than the CRBR and LSPB outlet plenum thermal environments. The component striping is also expected to be lower because of the specific thermal-hydraulic design and the larger core/UIS distance. Therefore, the PRISM design which follows the CRBR/LSPB approach of using Inconel 718 and austenitic liners in the outlet plenum will be shown to be acceptable when detailed evaluations are performed.

The thermal loads on remaining structures will vary from essentially isothermal environment of the core support structure to the high temperatures and thermal gradients on the support cylinder in the outlet plenum and the horizontal baffle plates. The temperatures of these components are not yet evaluated in detail but the thermal gradients and stresses for these components are expected to be within allowable limits.

5.3.3.3 Component Evaluation

5.3.3.3.1 Upper Internal Structure (UIS)

The principal loads on the UIS are the gravity, seismic loads, and the thermal loads. As the structure is cantilevered from the closure, the seismic and gravity-induced stresses are the largest at the support and decrease to small levels at the lower end. In contrast, thermal stresses are highest at the bottom plate exposed to the core exit environment and smaller at the upper elevations. Therefore, the UIS lower section design is dictated by thermal loads, and the support structure design is dictated by seismic loads.

UIS Lower Assembly - Mechanical loads are small and the design is dictated by the thermal loads. The outlet plenum environment is not yet characterized sufficiently to permit detailed evaluation. However, the design is expected to be adequate based on a comparison of the expected thermal loads with the CRBR/LSPB design/analysis experience.

The thermal loads during normal operation consist of global temperature gradients and local temperature fluctuations. The relevant global temperature gradients are the radial temperature gradients on the PRISM UIS lower plate which will be comparable to or smaller than those in the LSPB. The resulting thermal stresses, however, will be considerably smaller because of the smaller diameter of the PRISM UIS (52") as opposed to the LSPB UIS (87"). The temperature gradients across the wall and the resulting thermal stresses are easily controlled by the number and thicknesses of the liners which will be finalized when the thermal environment is better defined. The relevant local temperature fluctuations are the thermal striping of the components exposed to the coolant streams from different core assemblies with different temperatures. Test data and the CRBR/LSPB experience suggest that the PRISM usage of Inconel-718 for the lower shroud tubes, instrument posts, and the lower UIS plate liner will be sufficient to sustain the striping loads. Subsequent tests/ evaluations may even permit usage of 316 stainless steel or modified

9Cr-1Mo instead of Inconel 718 for the UIS liner plate because of the relatively large UIS/core separation.

The relevant thermal loads during thermal transients are the thermal shock due to rapid coolant temperature changes and the thermal gradients due to the subsequent coolant stratification in the outlet plenum. The most severe thermal shocks on the UIS components result from the transients associated with reactor trip from full power with minimum decay heat. The CRBR and LSPB UIS have been designed for 25-30°F/sec transients associated with this event. The corresponding PRISM transients are expected to produce ramp rates of 15°F/sec. Therefore, the PRISM UIS thermal shocks should be within the material design limits. The axial temperature gradients due to coolant stratification following reactor scram may be similar in PRISM and LSPB. The resulting thermal stresses will be lower than the thermal stresses due to the axial temperature gradient at the sodium free surface during normal operation. These are expected to be within the design limits.

UIS Support Cylinder - Thermal loads on the UIS support structure are expected to be small except in the regions of abrupt temperature changes such as the sodium free surface and the support attachment. The resulting thermal stresses are not evaluated in detail but are expected to be smaller than the thermal stresses calculated for the reactor vessel and liner summarized in Section 5.2.3.

The response of the reference UIS support cylinder design to mechanical loads is not yet specifically evaluated but its structural and functional adequacy is implicit in the parametric analyses which were performed during the design concept and selection phase. The concept development has focussed on developing UIS support designs with seismic deflections sufficiently small to preclude interference between a driveline and its guiding components which could permanently deform the components and affect the scram performance. The clearances and therefore the allowable deflections and the associated seismic stresses are small. Therefore, the design is deflection- rather than stress-limited.

The permissible SSE deflections were calculated from conservative clearance and misalignment analyses with allowance for the core system assumed to move out of phase with the UIS and all the core assemblies assumed to be piled up in the direction opposite to the UIS deflection. SSE response spectrum analyses were performed to calculate the UIS deflections for the reactor closure SSE response spectra shown in Figure 5.3-27. The spectral loads in the figure were applied in directions parallel to and perpendicular to the refueling machine access slot in the UIS and the deflections in the two directions were combined to obtain the control rod shroud tube SSE deflection.

The SSE deflections and deflection limits calculated for several UIS cylinder diameters and thicknesses and two different reactor isolation frequencies are shown below:

UIS SSE DEFLECTIONS

UIS Support Cylinder		UIS Frequencies	SSE Deflections Isolation Frequency	
Diameter	Thickness	Hz	0.5 Hz	1.0 Hz
72"	0.5"	3.28,5.62	0.29"	0.43"
	1.0"	4.00,7.10	0.19"	0.27"
45"	0.5"	2.36,4.85	0.67"	1.34"
	1.0"	2.96,4.20	0.45"	0.69"
Deflection Limits For Two-Point Contact			0.60"	0.44"

The PRISM isolation frequency (0.75 Hz) and the reference UIS cylinder dimensions (52" diameter, 1" thickness) are within these ranges. While interpolation of the above results suggest that the SSE deflections in the reference design will be close to or may exceed the design limits, the two-point contact limit used in the evaluation leaves considerable room for elastic non-damaging deflections of the components. Also, the driveline design has considerable room for optimization. Finally, the deflections can be decreased by increasing the UIS diameter and/or thickness, or by using alternate stiffer concepts developed during the design development phase.

5.3.3.3.2 Vessel-Supported Structures

The response of the vessel-supported structure to mechanical loads - gravity, a 100 psi inlet plenum coolant pressure, and SSE seismic loads represented as equivalent statically applied loads of 0.5g in the horizontal direction and 1.5g in the vertical direction - was calculated in elastic finite element analyses. Vertical and horizontal seismic responses were calculated in separate analyses. The inlet plenum pressure and the gravitational acceleration were included in the vertical analysis with the 1.5g SSE static load and gravity load assumed to be in phase. The analysis was performed using the ANSYS 4.2 Code. A 180^0 sector was modeled using the thin shell elements for the entire model except for the six sleeves coupling the inlet plenum upper and lower plates which were represented by three-dimensional beam elements.

The components included in the analysis model are shown in Figure 5.3-28 and listed in Table 5.3-3 by their material-ID's in the model to facilitate association of the stress results reported subsequently with different regions of the components. This is necessary because the models for large components were sub-divided into several groups to facilitate different load, thickness, and material property specifications.

In the vertical analysis, the fluid mass was assumed to produce internal pressure in the reactor vessel with magnitude equal to 2.5 times the local static head. The inertial and gravitational loads from the core mass, the shielding and the insulation at the top of the reactor were modeled as pressure loads, nodal forces or increase in the structure density.

In the horizontal analyses, the annular fluid mass was distributed between the bounding cylindrical components. No additional mass was added to account for hydro-dynamic effects which are expected to be relatively small for the purpose of stress calculations, and neglecting the hydrodynamic resistance is expected to give conservative values for the relative displacement between the reactor vessel and the thermal liner.

Maximum tensile principal stress, maximum compressive principal stress and maximum shear stress occur at different locations in the components. Also the maximum and minimum stresses for the horizontal and vertical loading occur at different locations. Finally, when considering simultaneous horizontal loads in two horizontal directions, the maximum stress location for one of the loads will generally coincide with the minimum stress location for the orthogonal load because of the circular symmetry of the structure and the nature of bending stress distributions. The analysis results were not processed to combine local stresses in order to find the actual maximum stress levels required for design evaluations. Instead, the maximum stresses due to vertical loading and two orthogonal loadings were conservatively assumed to occur at the same location and were combined using the 'Square Root of the Sum of the Squares' formula to give the maximum stress values shown in Table 5.3-3. Also, the maximum stresses were found to be larger than twice the maximum shear stresses to be used in ASME Code evaluations. Thus, added conservatism was introduced by comparing the maximum stresses in Table 5.3-3 with the code stress limits.

The ASME Code stress limits the seismic stresses to $2.4S_m$ which are included in Table 5.3-3 as function of temperature. A comparison of the calculated stresses and the stress limits show comfortable design margins in view of the conservatism used in combining the maximum stresses in different directions and the use of 1.5g vertical seismic load instead of the 1g load calculated in the reactor system seismic analyses.

A conservative estimate of OBE stresses may be obtained by multiplying the results in Table 5.3-3 by (1.9/2.5 = 0.76) where 2.5 is the vertical SSE excitation including gravity and 1.9 is the vertical OBE excitation obtained by adding 60% of SSE to gravity. The corresponding maximum OBE stresses will be in the core support structure with values of 28106 x 0.76 = 21360 psi in the center cylinder and 23507 x 0.76 = 17865 psi in the radial beams. Most of these stresses are bending stresses which are less than the ASME Code limit of $1.5S_m$ = 24300 psi at 650°F which is about 40°F above the nominal cold pool and core support temperature.

The duty cycles in Appendix D specifies 5 OBE and 1 SSE with 10 load cycles for each earthquake. This would amount to 60 cycles of relatively small strain amplitudes giving acceptable fatigue damage.

The stresses from the mechanical loads are relatively small in the components at higher elevations where considerable thermal stresses exist. During normal operation, the largest thermal stresses will be in the reactor vessel and the reactor liner at the sodium free surfaces. These stresses, when combined with the mechanical stresses, satisfy the ASME Code secondary stress limit of $3S_m$.

The primary stresses in the high temperature region may be conservatively obtained by dividing the values in Table 5.3-3 by 2.5. The maximum stress value in this case is 17404/2.5 = 6962 psi in the reactor liner. In comparison, the ASME time-dependent stress limits for SS304 for 300000 hours operation at 950° and 900°F are 12200 and 16000 psi, respectively.

5.3.4 Compliance With Code and Standards

Core support structures and other reactor internal structures that are stipulated to be 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 N-201 will apply.

ASME Class CS or Seismic Category I components shall be designed to withstand the concurrent loadings associated with Service Level B conditions and the vibratory motion of 50 percent of the safe shutdown earthquake (SSE) or in other words, the operating basis earthquake (OBE). The design limits for this case are specified in NG-3223 and NG-3233 of the ASME Code for core support structures. The design limits specified in NG-3225 and NG-3235 of the ASME Code for component supports shall not be exceeded when the component is subjected to concurrent loadings associated with the Service Level A conditions and Service Level D conditions.

For components at elevated temperature, the design stress limits, as specified by Paragraph 3.6 of Code Case N-201 shall be used for the OBE load condition and the SSE load condition, respectively.

References - Section 5.3

1. Specification 23A3071, Revision 0, "PRISM Design Requirements," March 1986.

Table 5.3-1

IN-VESSEL INSTRUMENTATION SUPPORTED BY REACTOR STRUCTURES

Sensors	Support Method
Core Outlet Thermocouples	Attached to UIS and integral with the IHX structures
Short Range Na Level	Penetration well through the EM pump support flange
Source Range Flux Detector	Dry wells attached to the support cylinder and core barrel
In-Service Inspection	Gauge rods, benchmark and inspection ports
Design Verification and Acoustic Monitoring	Strain gages, thermocouples, accelerometer, and acoustic monitors mounted to appropriate in-vessel structures
Vessel Preheat Thermocouples and Sodium Fill (Long Range) Level Sensors	Temporary access through the rotatable plug
Delayed Neutron Detectors	Penetration drywell with the IHX tube bundle
Fission Gas Monitor	Penetration in the IHX support flange
Core Inlet Temperature	EM pump structures
IHX Inlet Temperature	IHX structures
Control Rod Position Indicator	Drive mechanisms
Rotatable Plug Position Indicator	Reactor deck
Reactor Vessel Na Leak	Sensor in containment vessel
Core Outlet Pressure	EM pump discharge manifold

TABLE 5.3-2

COMPONENT SSE ACCELERATIONS

Structure/Component	Acceleration, g	
	Horizontal	Vertical
ISOLATOR	0.44	0.19
REACTOR VESSEL	0.46	0.82
IHX	0.49	1.28
PUMP	0.41	0.75
UIS	0.47	0.75
STORED FUEL	-	0.98
HEAD INSULATION	-	1.00
CORE	-	0.90

TABLE 5.3-3 SSE PRINCIPAL STRESSES (SRSS)

Material ID	Component	Stress, psi Calculated	Allowable*
	REACTOR VESSEL		
1	bottom head - lower end	3459	38880
2	- middle part	9273	38880
3	- upper part	17929	38800
4	transition/core support attachment	22789	38800
5	vessel shell	7625	35520
	INLET PLENUM LOWER PLATE		
6	central region	6077	38880
7	outer circle	8160	38880
	INLET PLENUM UPPER PLATE		
8	central region (core assembly holes)	11489	38400
9	in-barrel fixed shielding support	3959	38400
10	core-barrel/support cylinder annulus	6742	38400
11	FIXED SHIELDING SUPPORT PLATE	11955	38880
12	FLOW GUIDE/PUMP INLET MANIFOLD CONE	17566	38400
	SUPPORT CYLINDER		
13	in-core section	7792	36240
15	top of the core to spent fuel support	8657	35040
16	spent-fuel support to top of the insulation	2538	35040
17	cylinder top region	789	35040
14	FIXED SHIELDING TOP SUPPORT PLATE	2299	38400
18	SEAL PLATE	10605	37440
	REACTOR LINER		
19	solid (unslotted) cylinder - lower end	9777	38400
	- upper end	672	35040
20	section with the overflow slots	1105	35040
21	HORIZONTAL BAFFLE PLATES	9776	35040
22	INLET PLENUM CYLINDER	11819	38400
23	CORE BARREL	2652	35040
24	PLENUM PLATE COUPLING SLEEVES	-	
25-26	CORE SUPPORT STRUCTURE	25548	38880

*SS304 ASME SSE stress limits ($2.4S_m$)

temperature	S_m, psi	$2.4S_m$, psi
600 F	16400	39360
650 F	16200	38880
700 F	16000	38400
750 F	15600	37440
800 F	15100	36240
850 F	14800	35520
900 F	14600	35040
950 F	14300	34320
1000 F	14000	33600

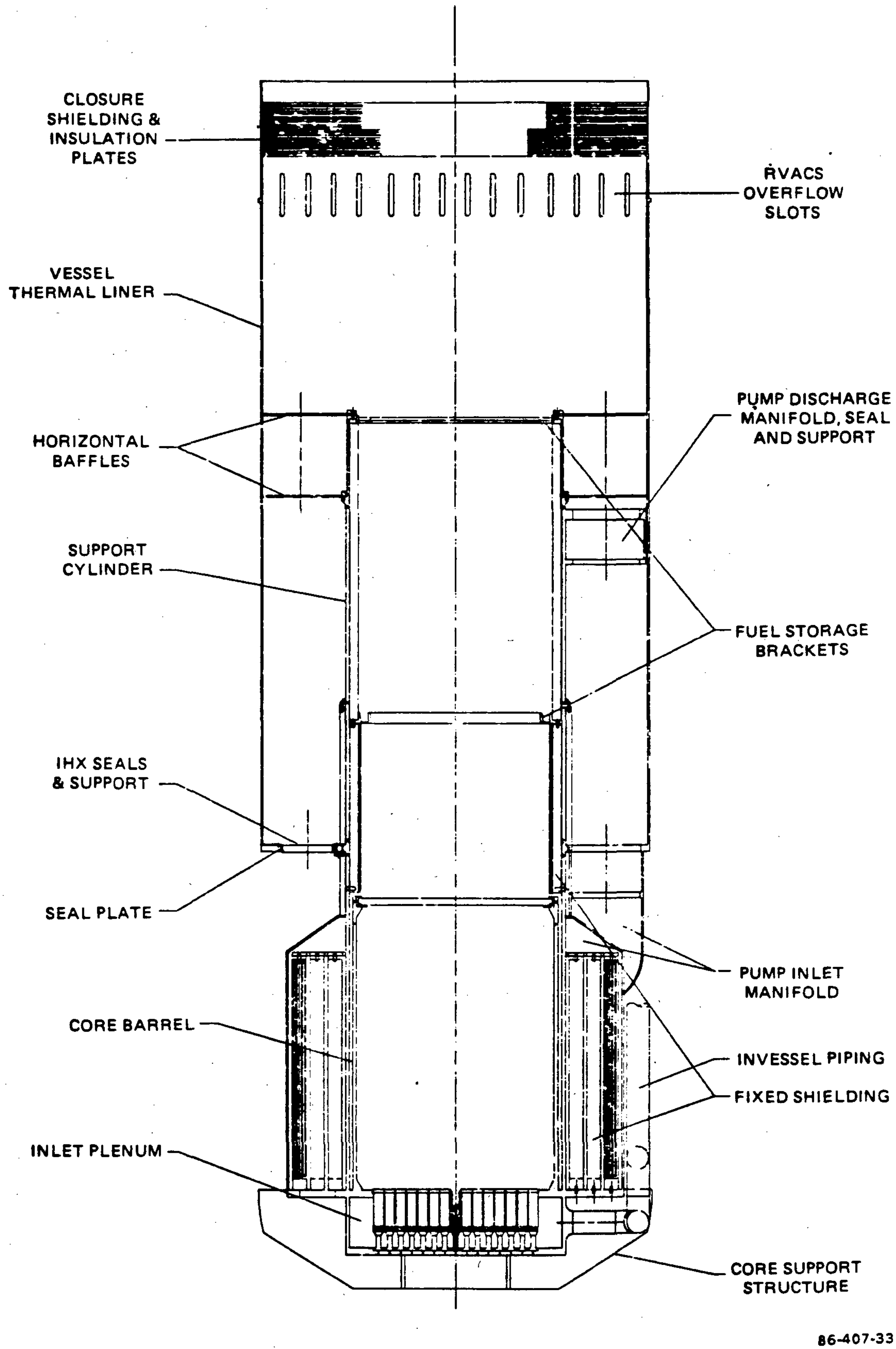

Figure 5.3–1 REACTOR INTERNAL STRUCTURES

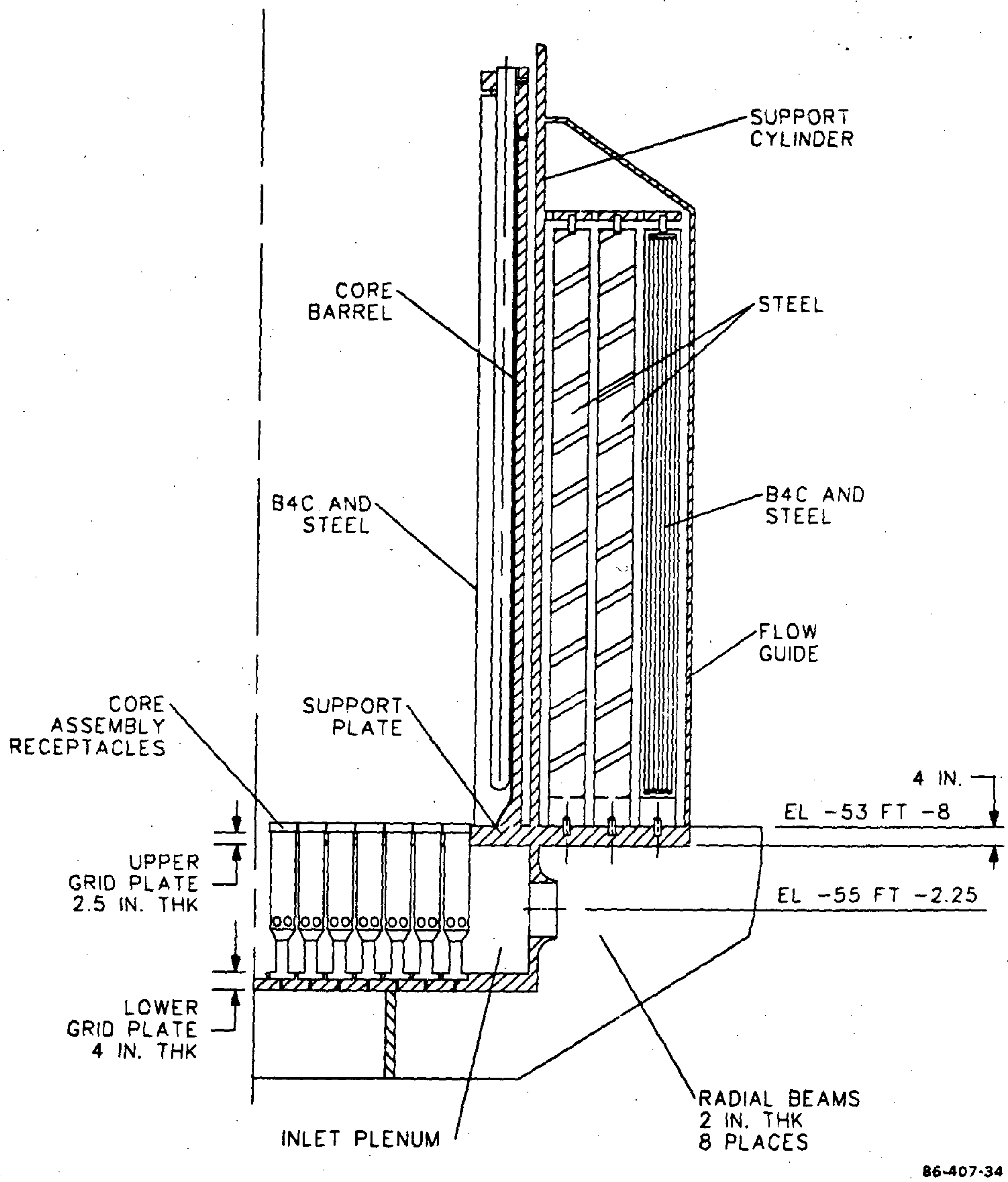

Figure 5.3–2 CORE SUPPORT STRUCTURES

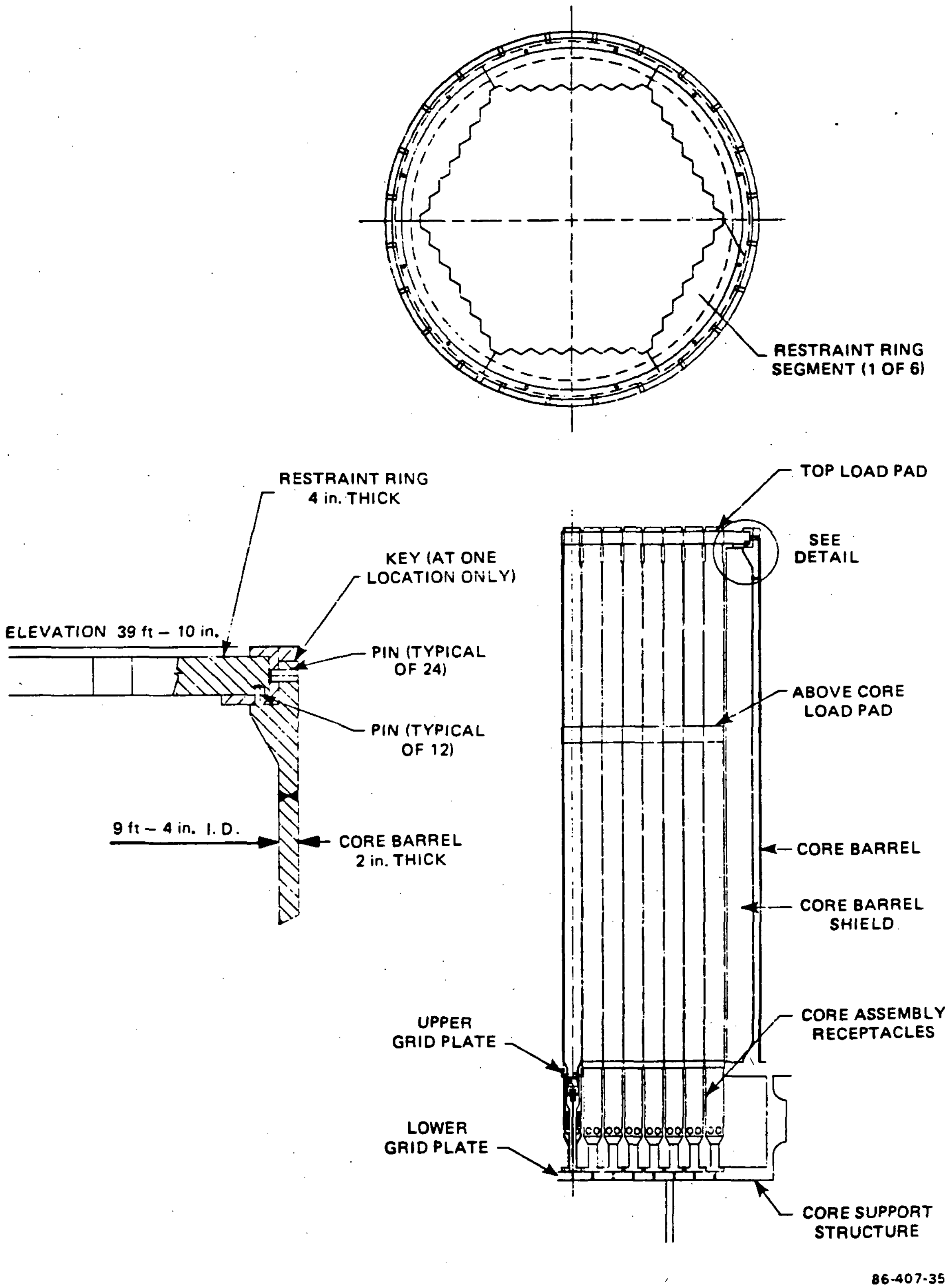

Figure 5.3–3 CORE RESTRAINT

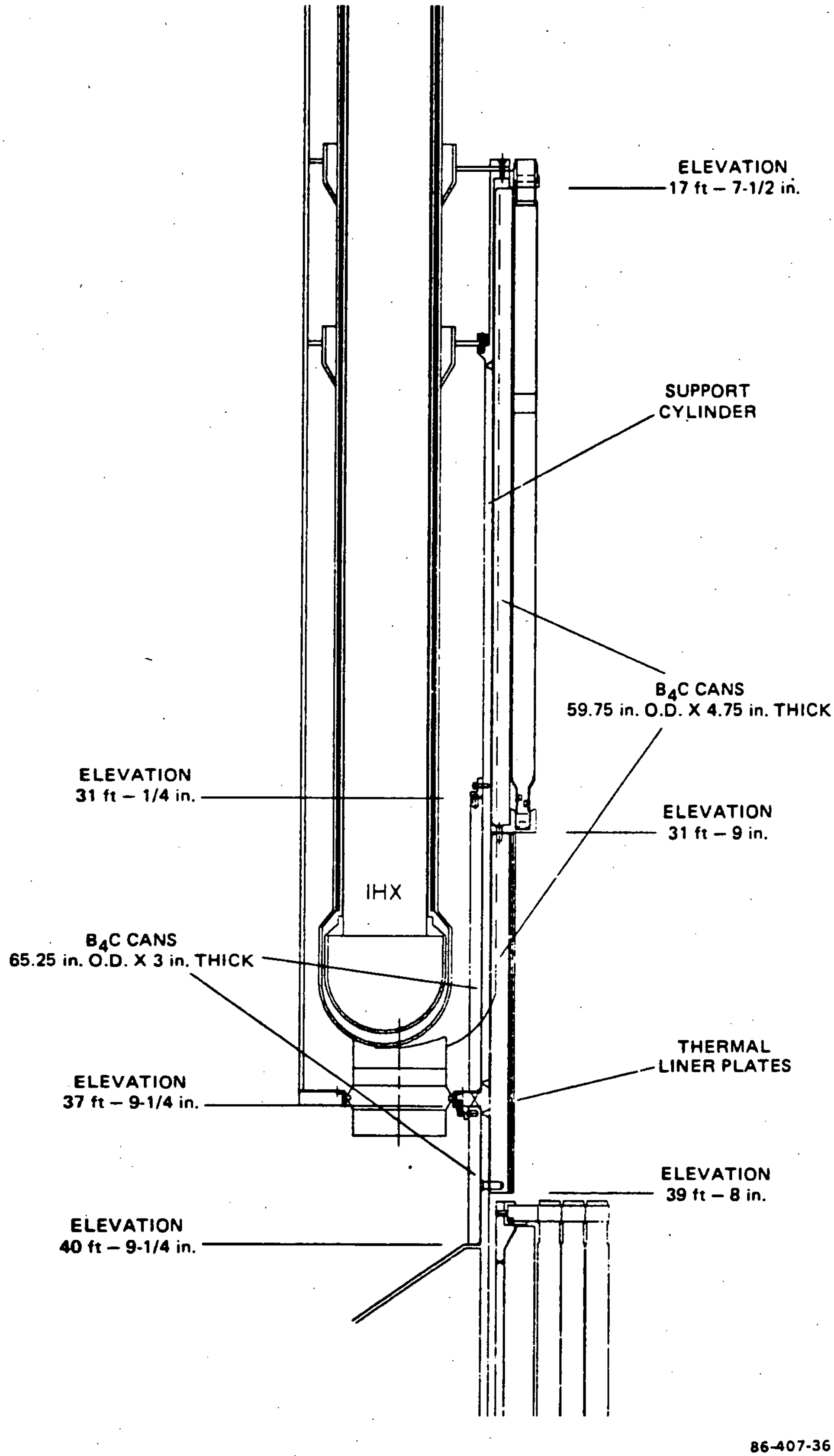

Figure 5.3—4 IHX SHIELDING ELEVATION

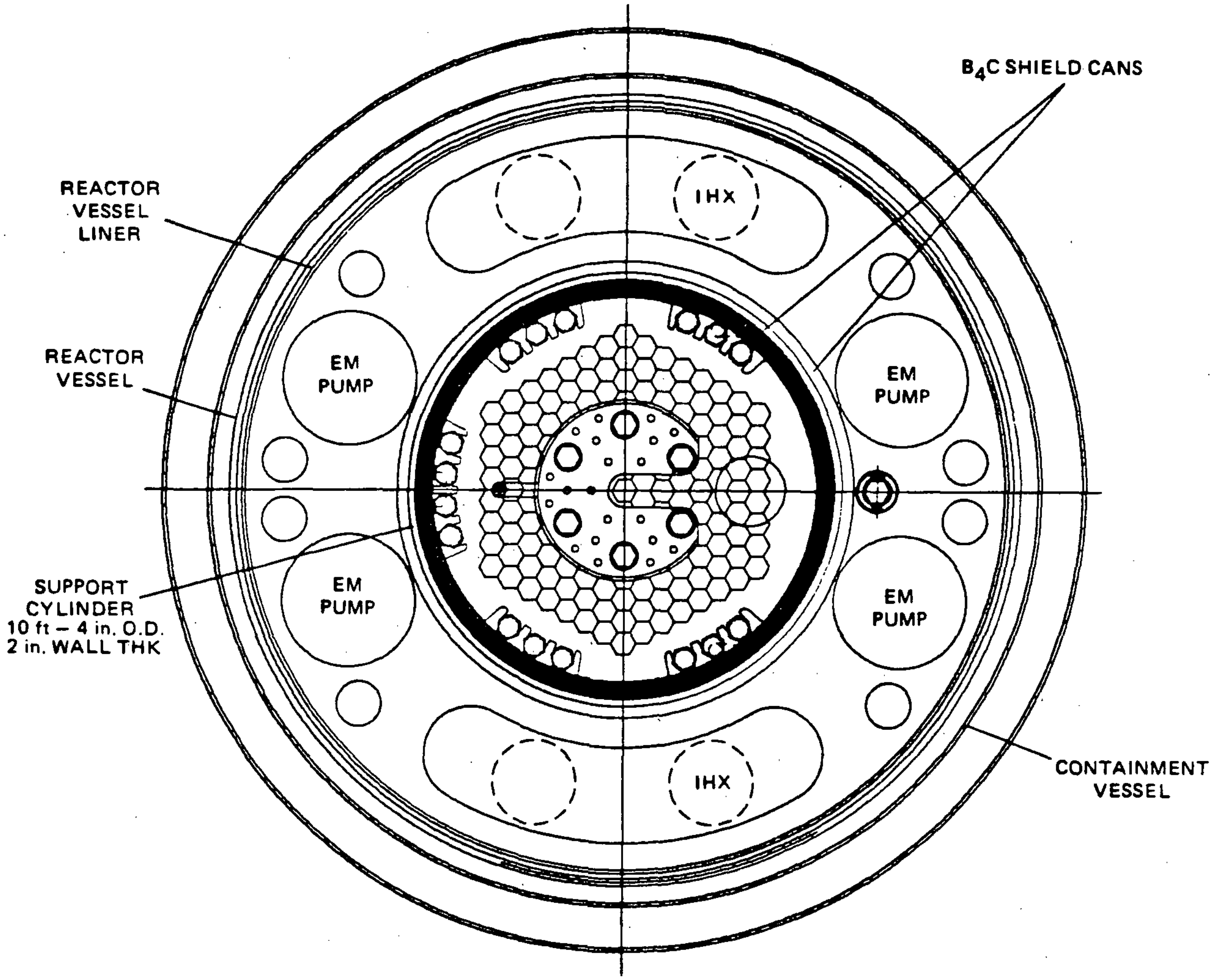

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Figure 5.3–5 IHX SHIELDING PLAN, ELEVATION -30 ft

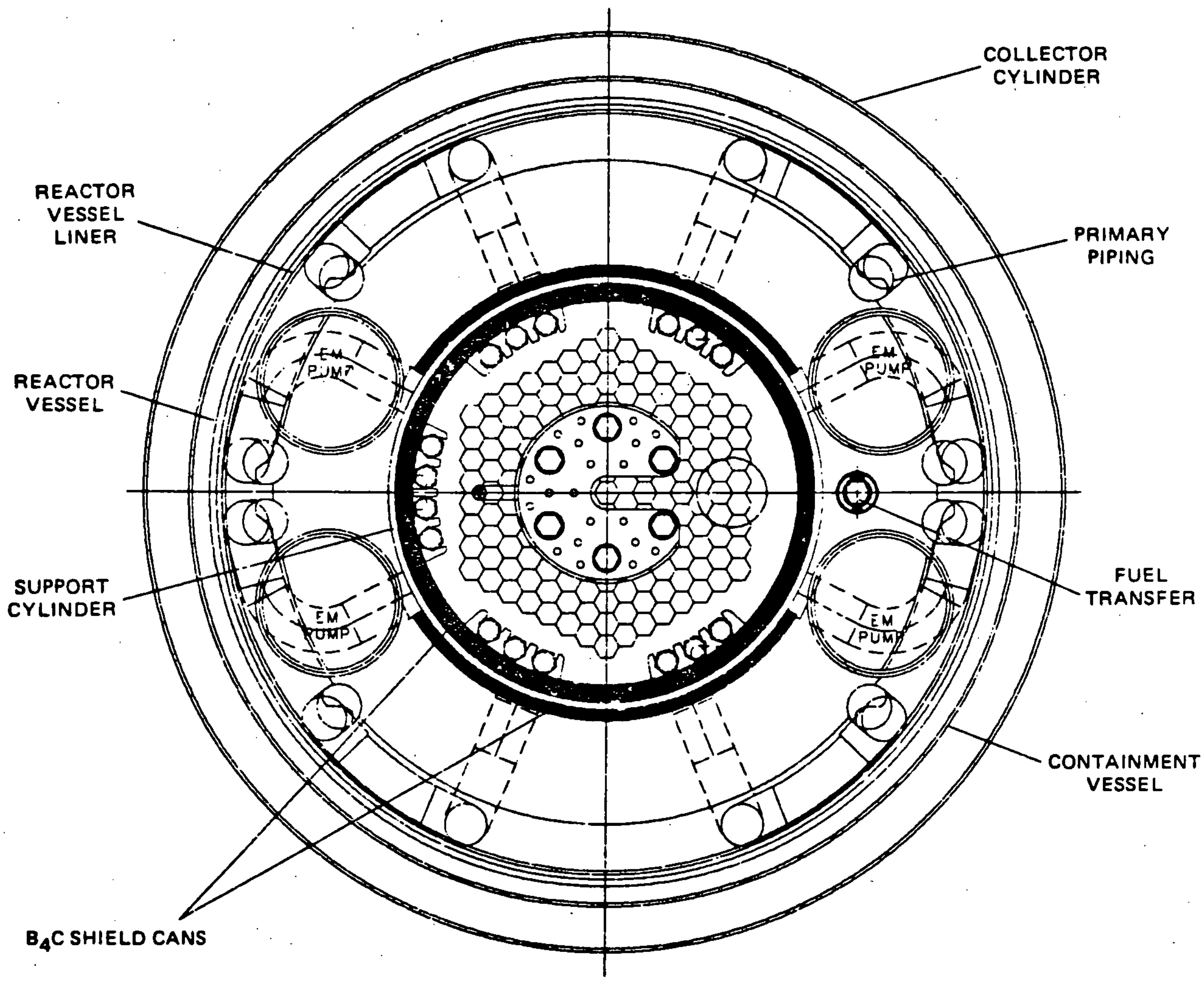

Figure 5.3–6 IHX SHIELDING PLAN, ELEVATION -39 ft

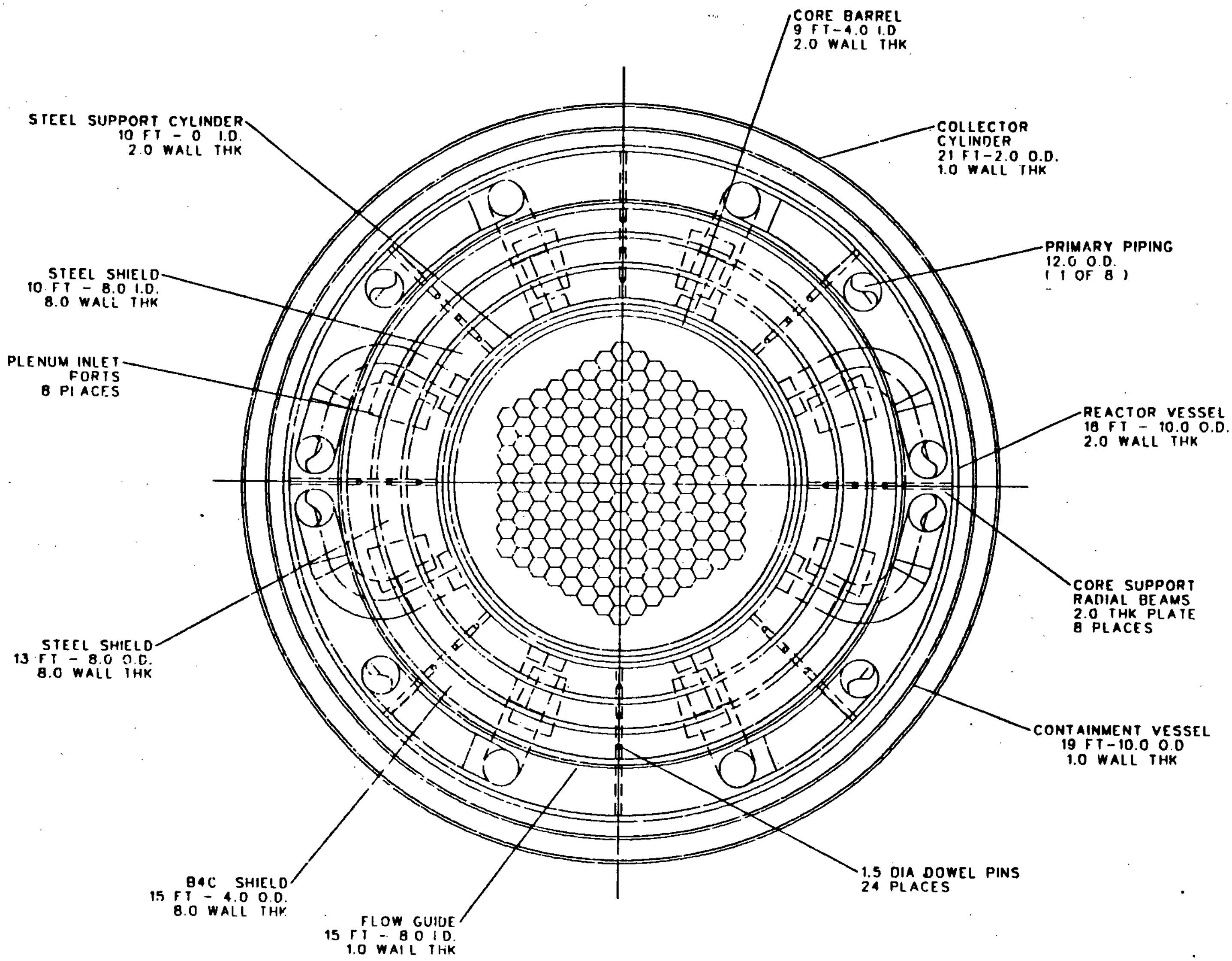

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Figure 5.3–7 RVACS SHIELDING PLAN

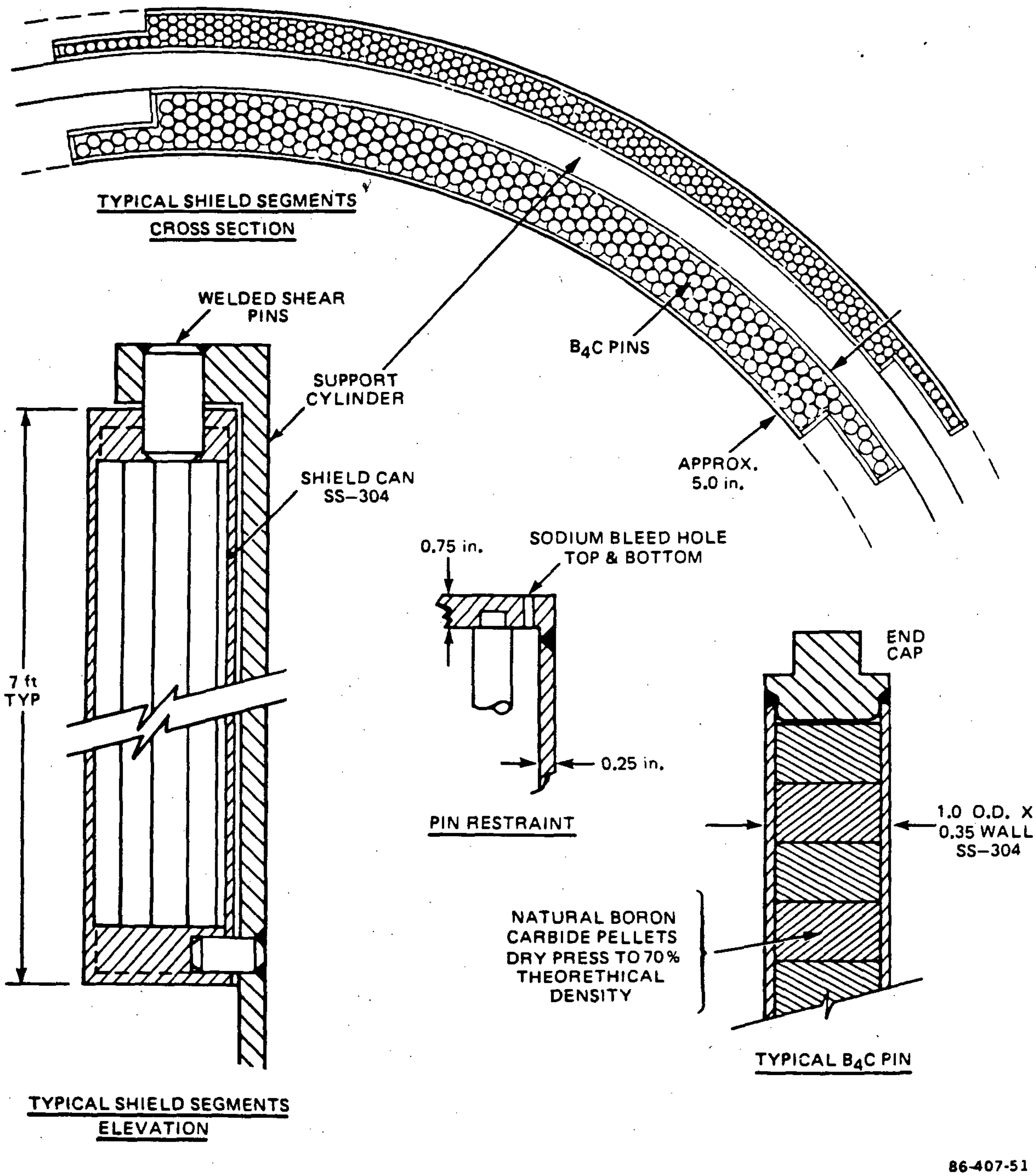


Figure 5.3–8 B_4C SHIELDING DETAILS

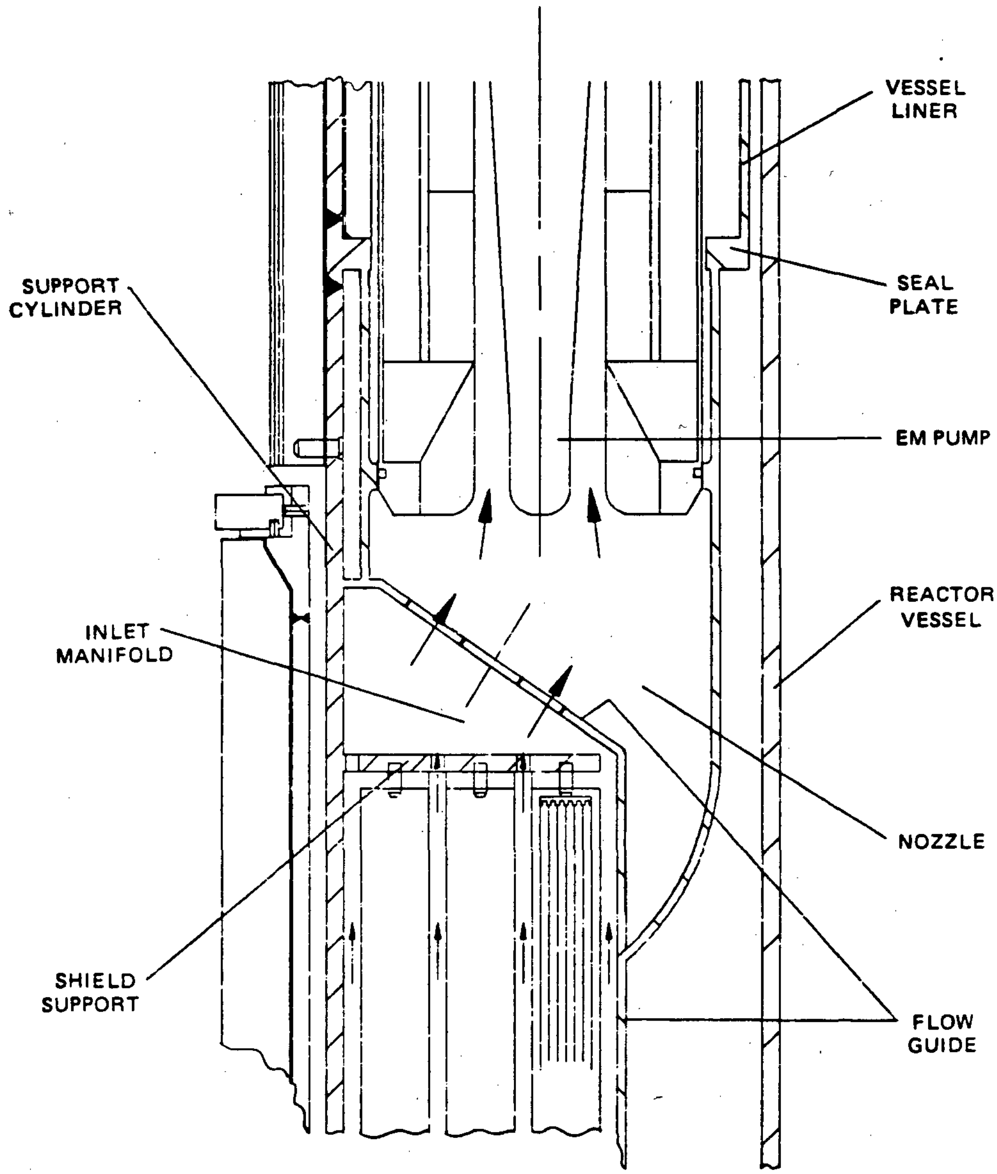


Figure 5.3–9 PUMP INLET MANIFOLD

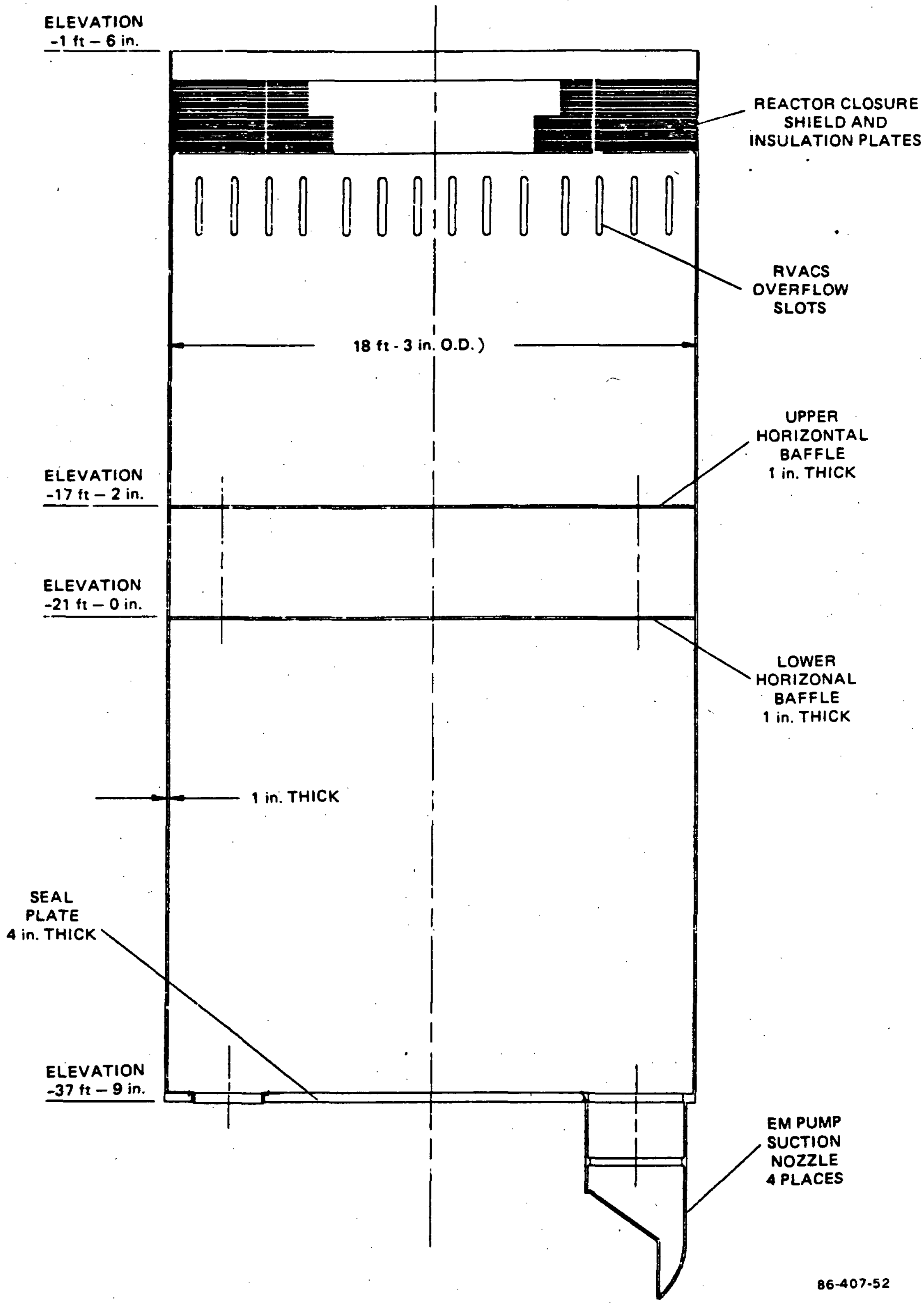
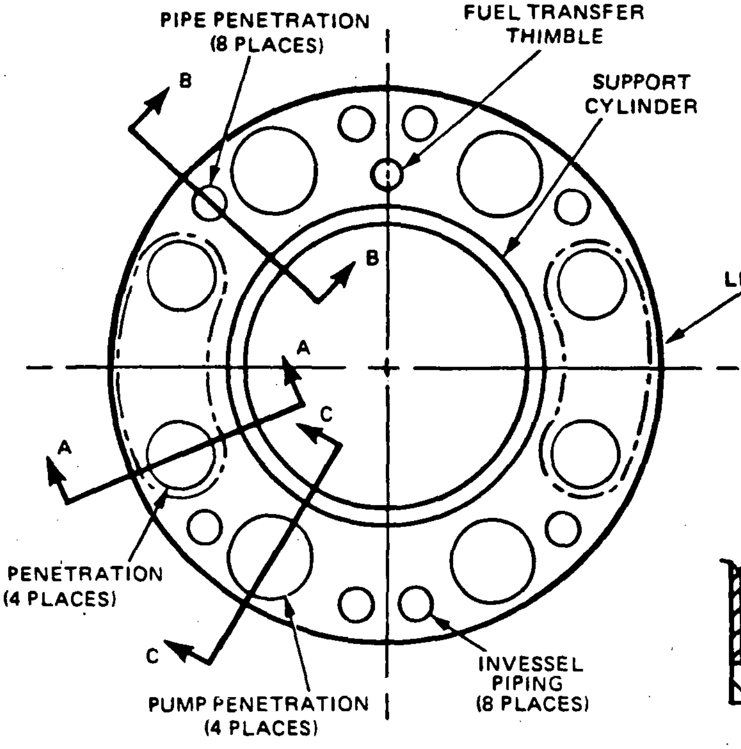
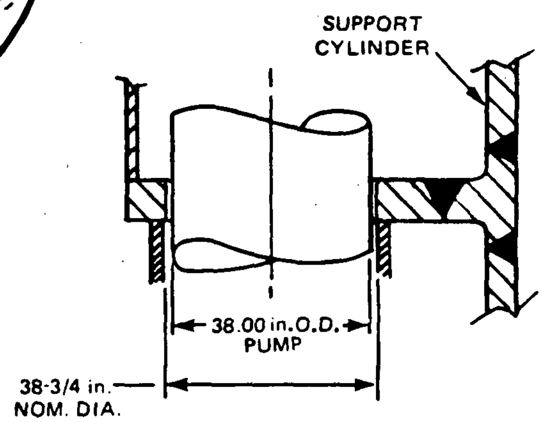


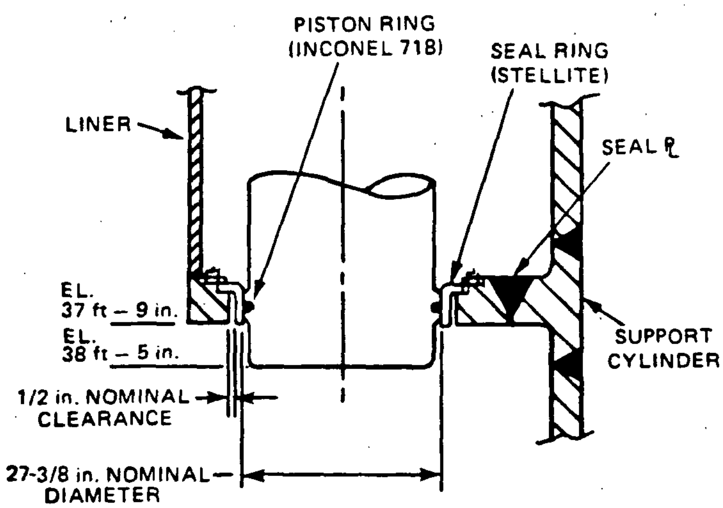
Figure 5.3–10 REACTOR VESSEL LINER AND SEAL PLATE



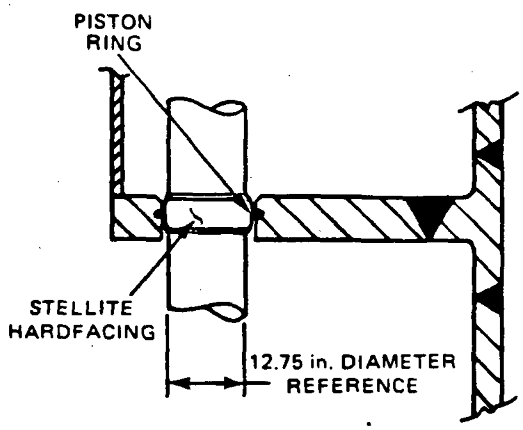
SEAL PLATE
APPROX. EL. 37 ft



SECTION C–C
TYPICAL SEAL PLATE CONFIGURATION
FOR PUMP PENETRATIONS



SECTION A–A
TYPICAL SEAL PLATE CONFIGURATION
FOR IHX PENETRATIONS



SECTION B–B
TYPICAL SEAL PLATE CONFIGURATION
FOR PIPE PENETRATIONS

86-407-53

Figure 5.3–11 SEAL PLATE DETAILS

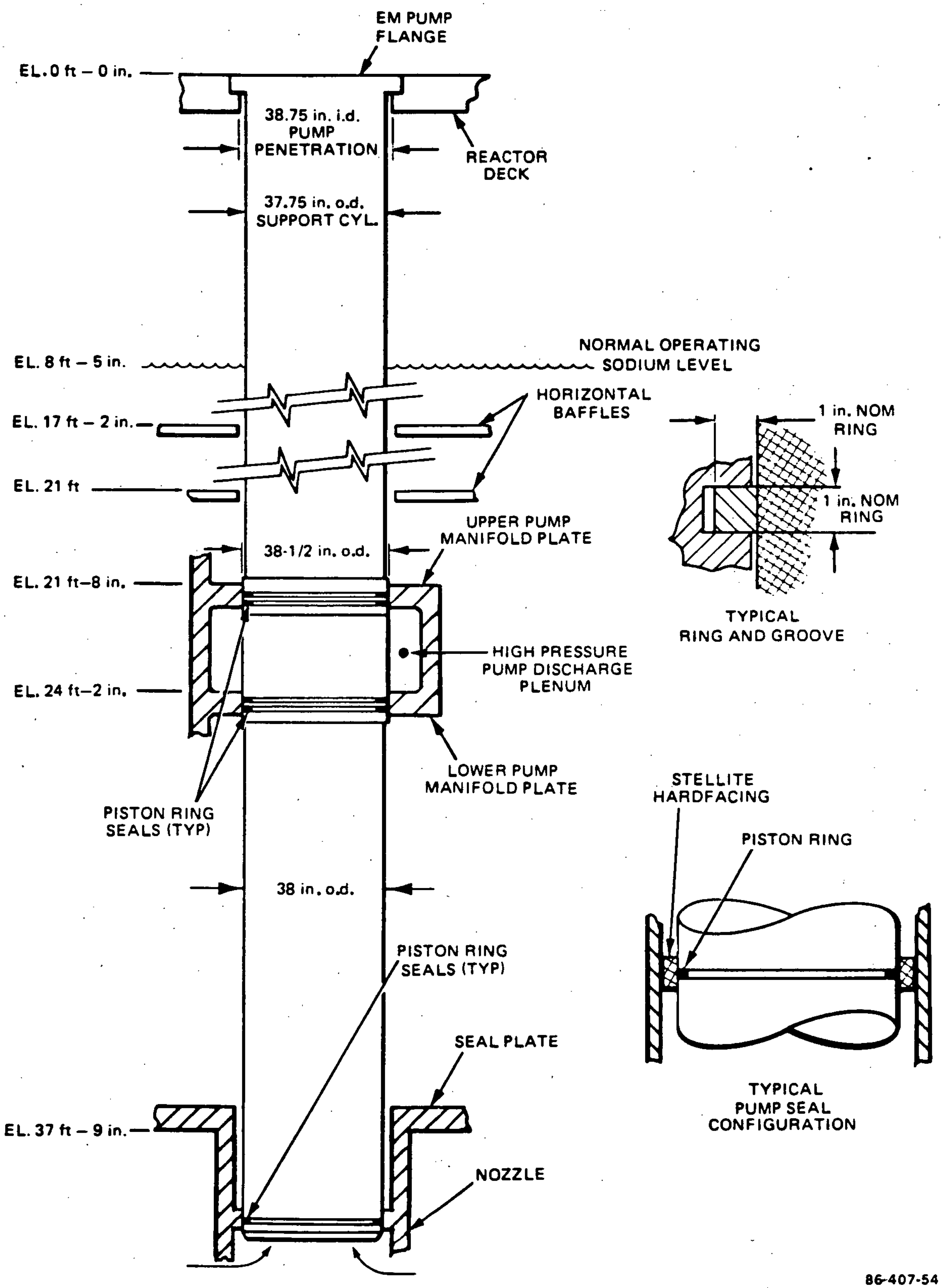
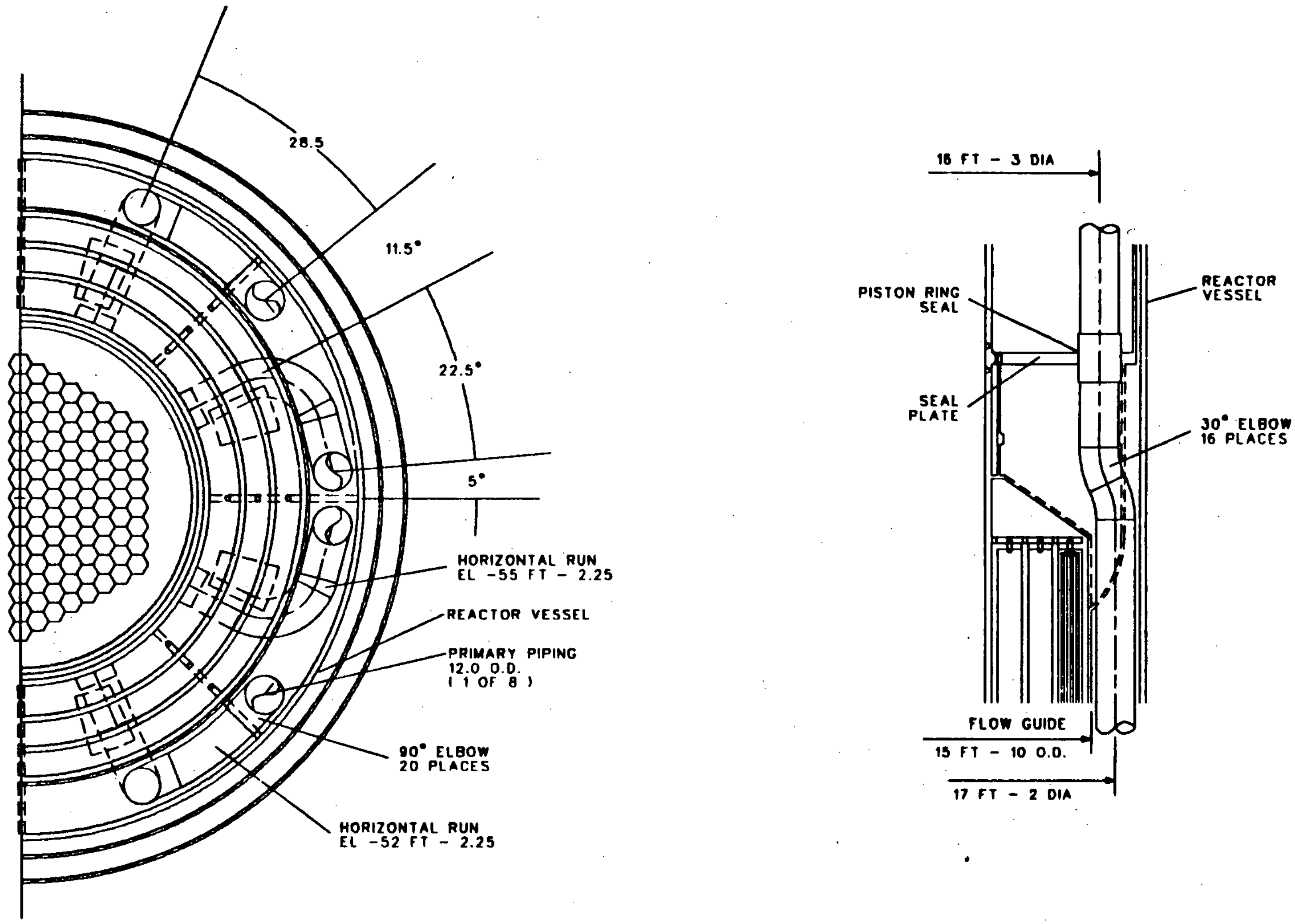


Figure 5.3–12 EM PUMP SEALS AND SUPPORTS



SECTION G–G

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Figure 5.3–13 INVESSEL PIPING ROUTING

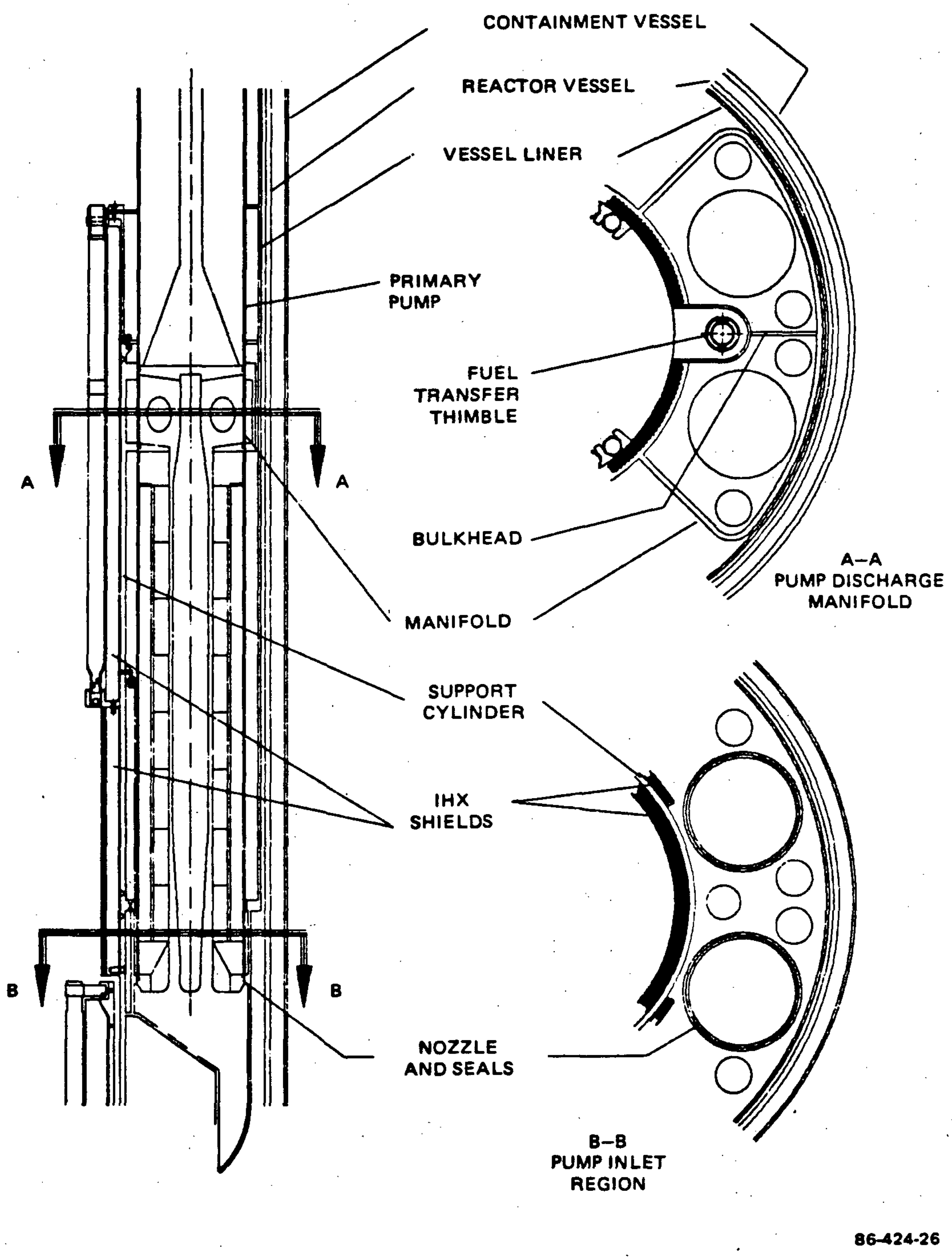

Figure 5.3–14 PRIMARY – PUMP MANIFOLDS

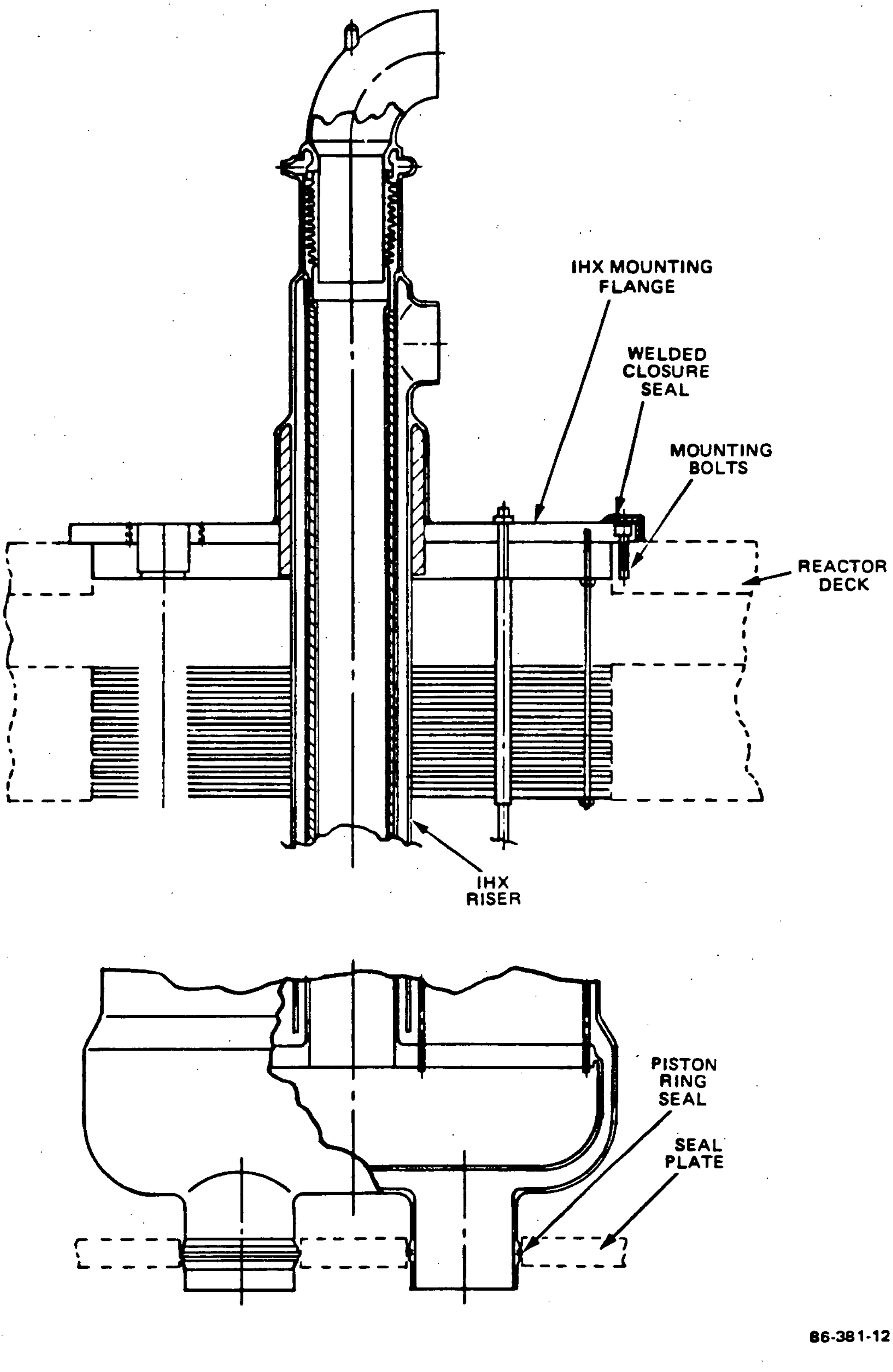

Figure 5.3–15 IHX SEALS AND SUPPORTS

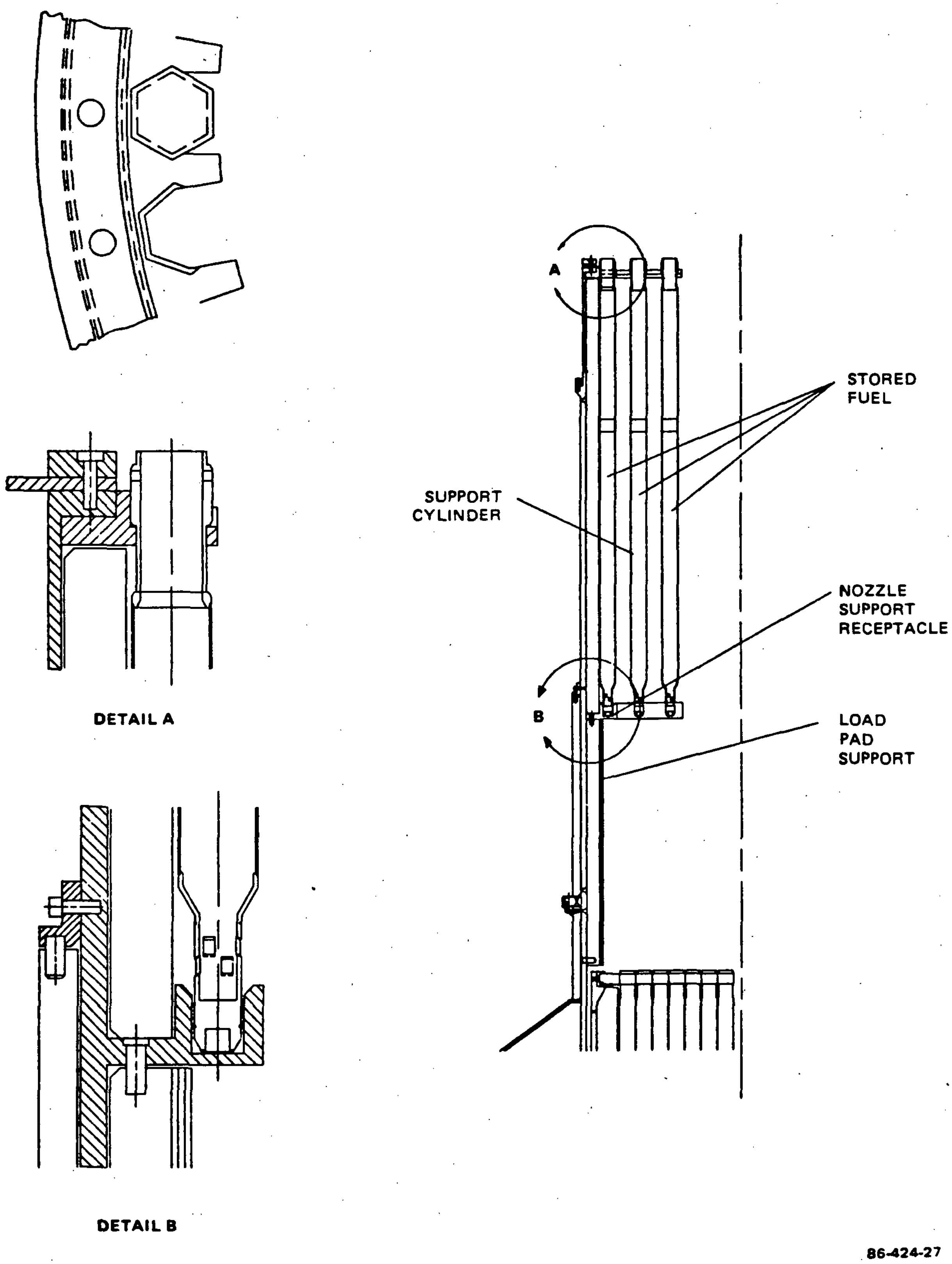

Figure 5.3–16 INVESSEL FUEL STORAGE

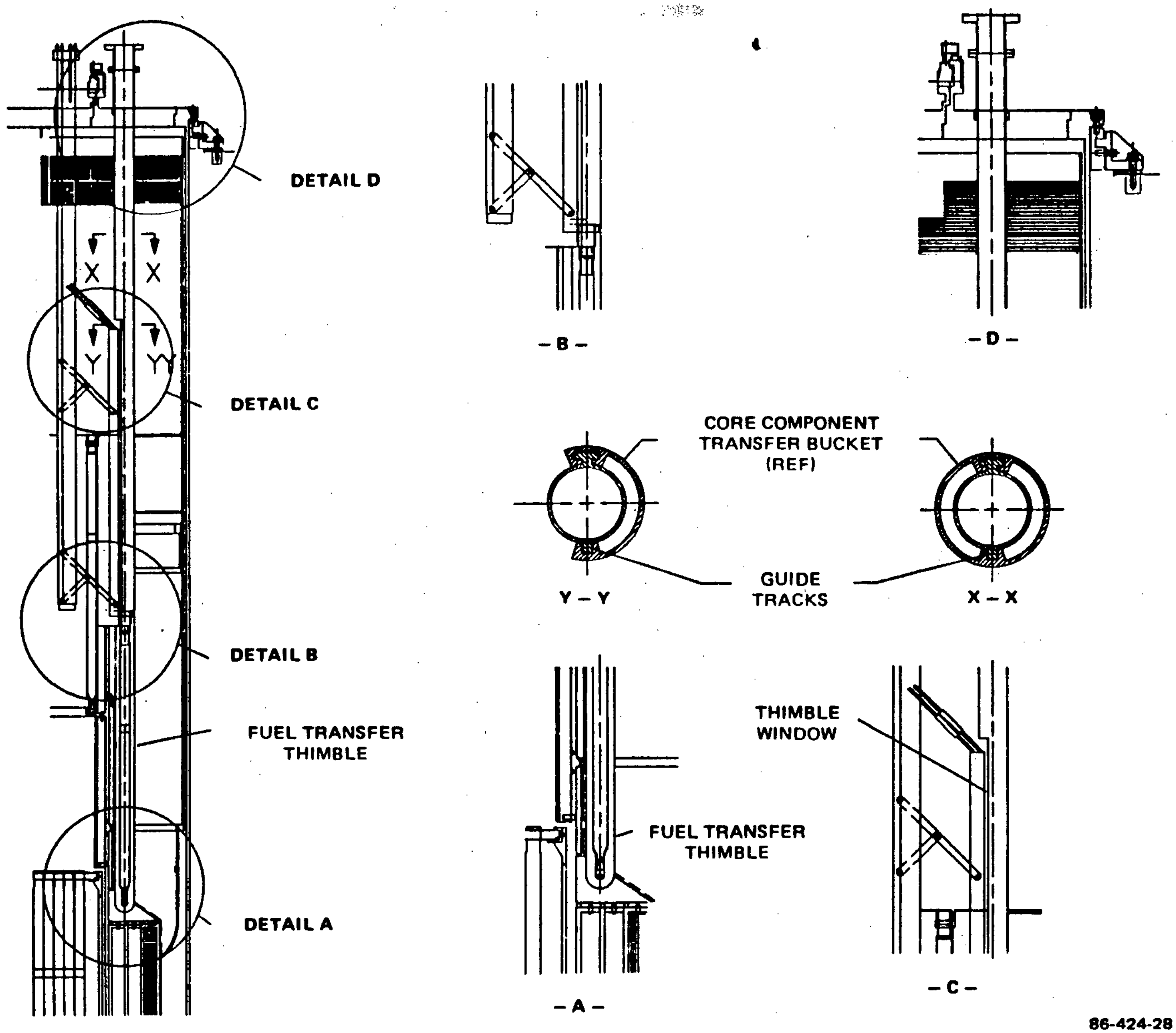

Figure 5.3—17 FUEL TRANSFER THIMBLE

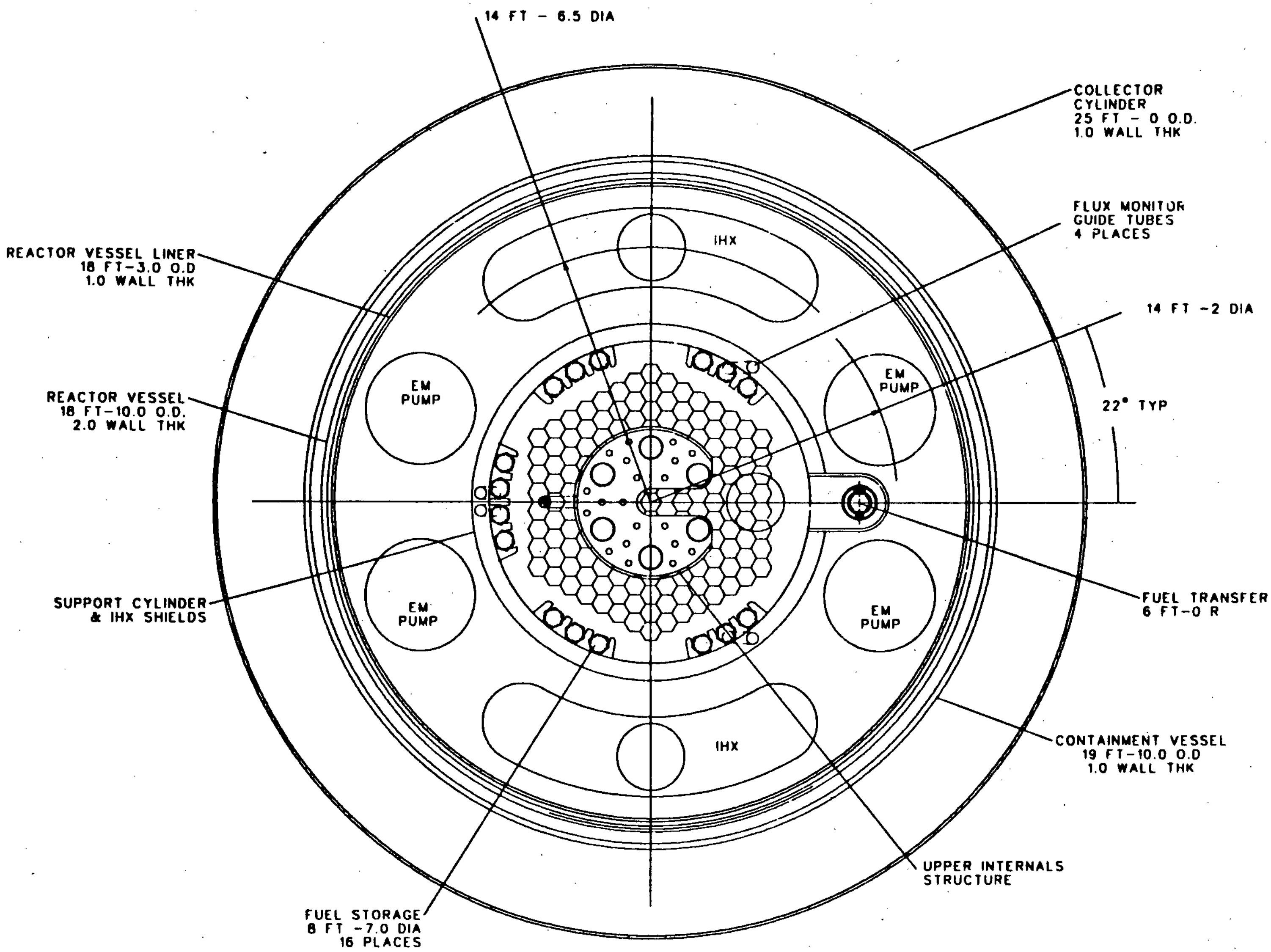

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Figure 5.3—18 REACTOR SYSTEM INTERNALS PLAN

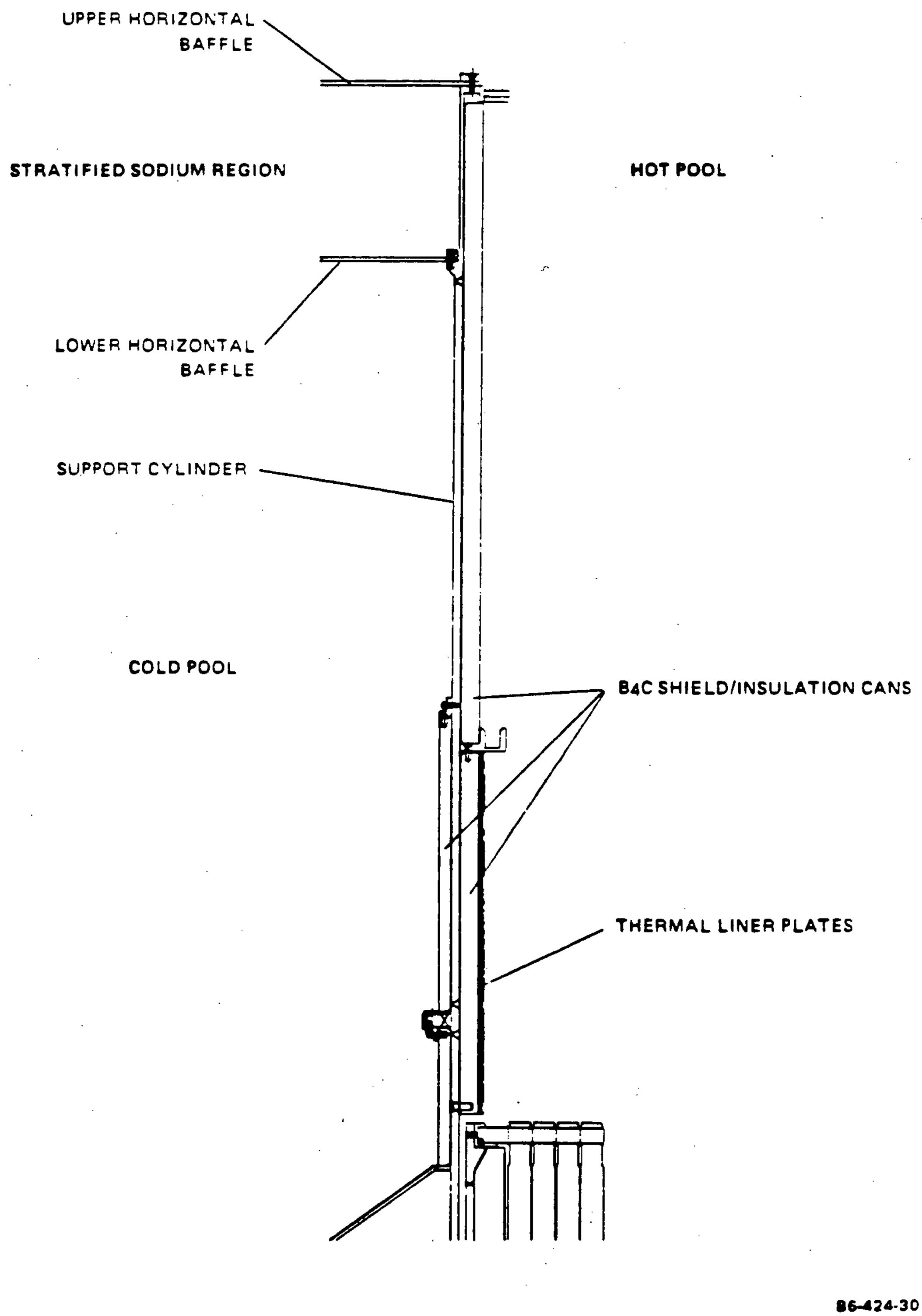

Figure 5.3–19 HOT POOL – COLD POOL THERMAL INSULATION

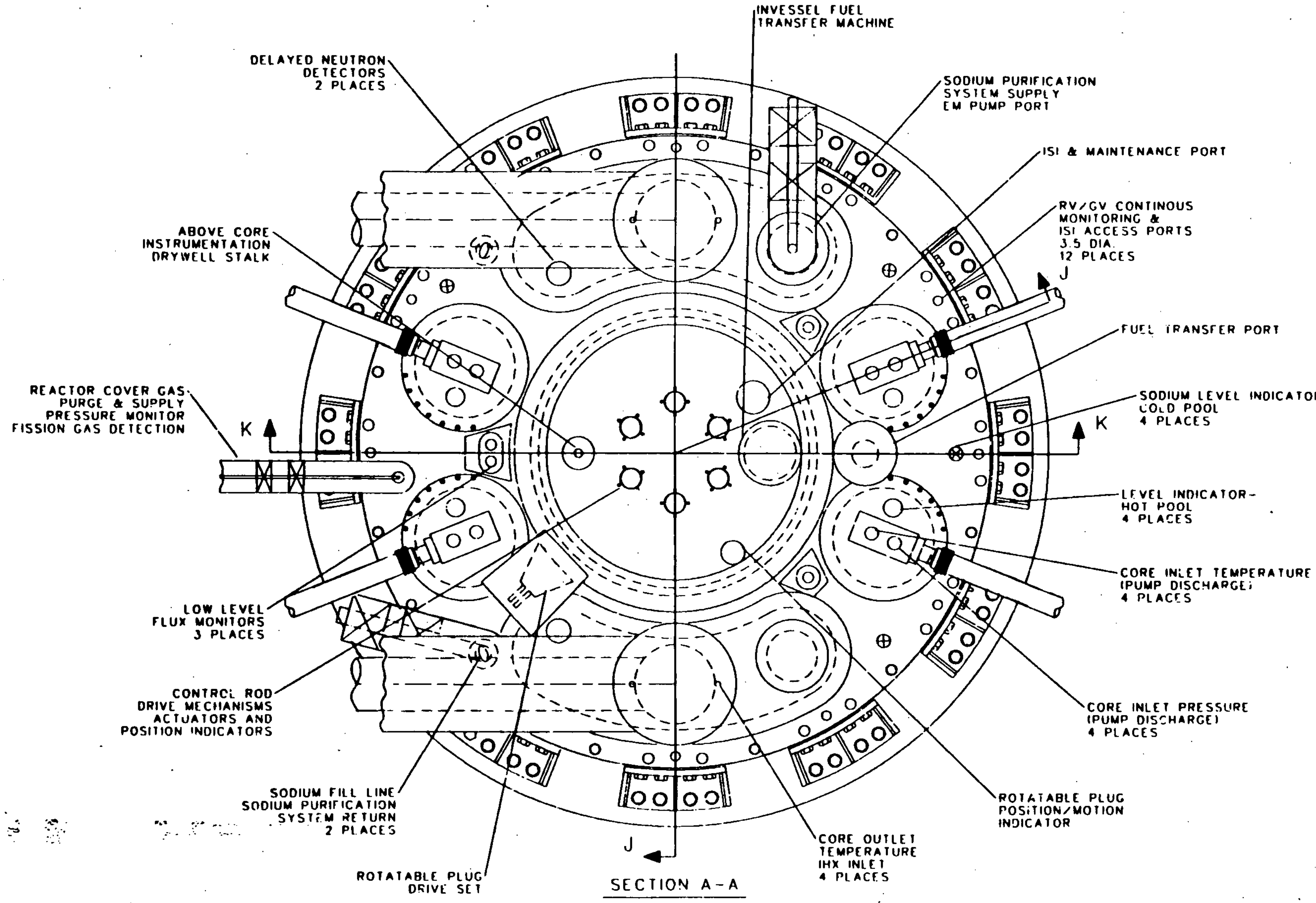

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Figure 5.3–20 CLOSURE MOUNTED EQUIPMENT AND INSTRUMENTATION

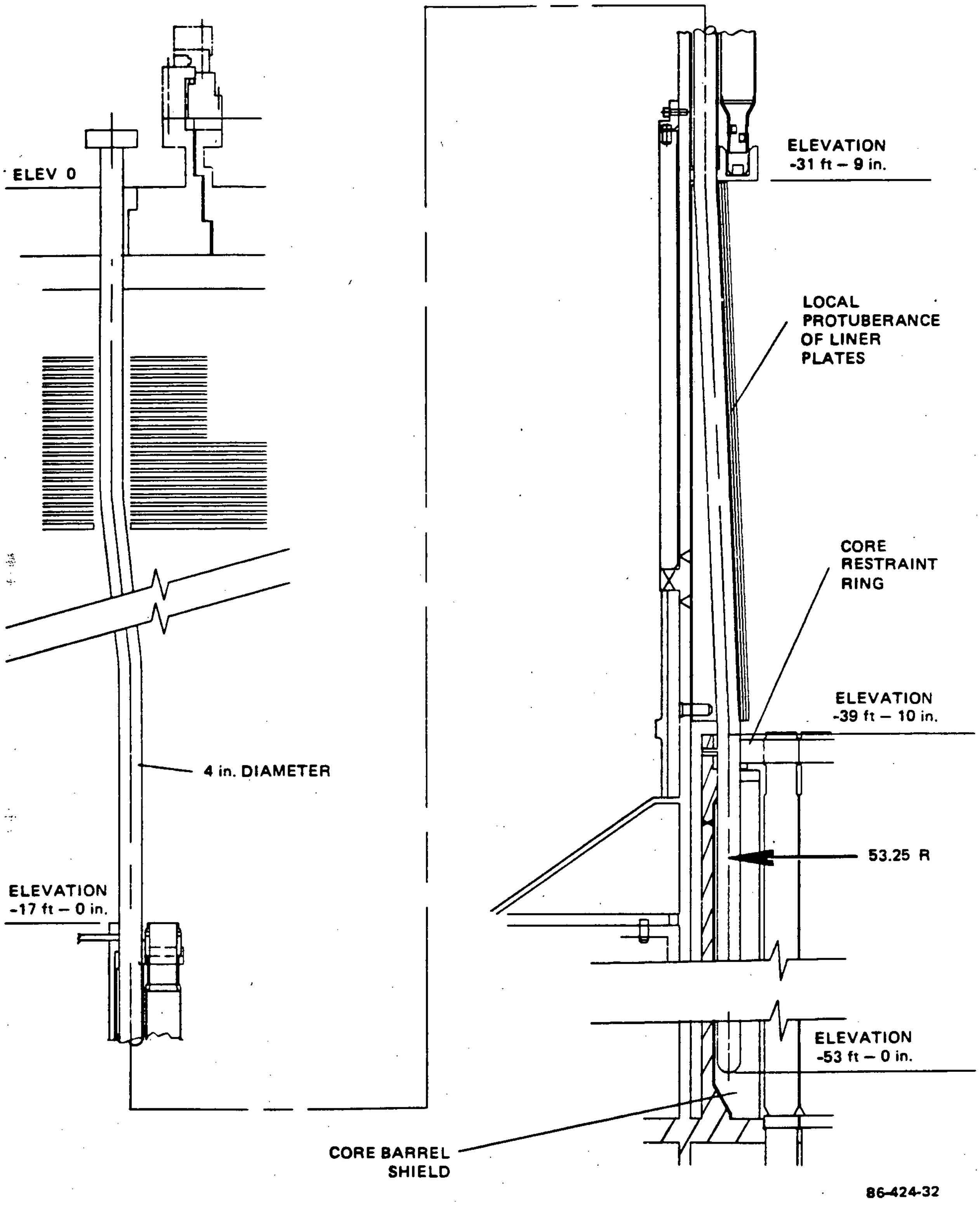


Figure 5.3–21 FLUX MONITOR GUIDE TUBES

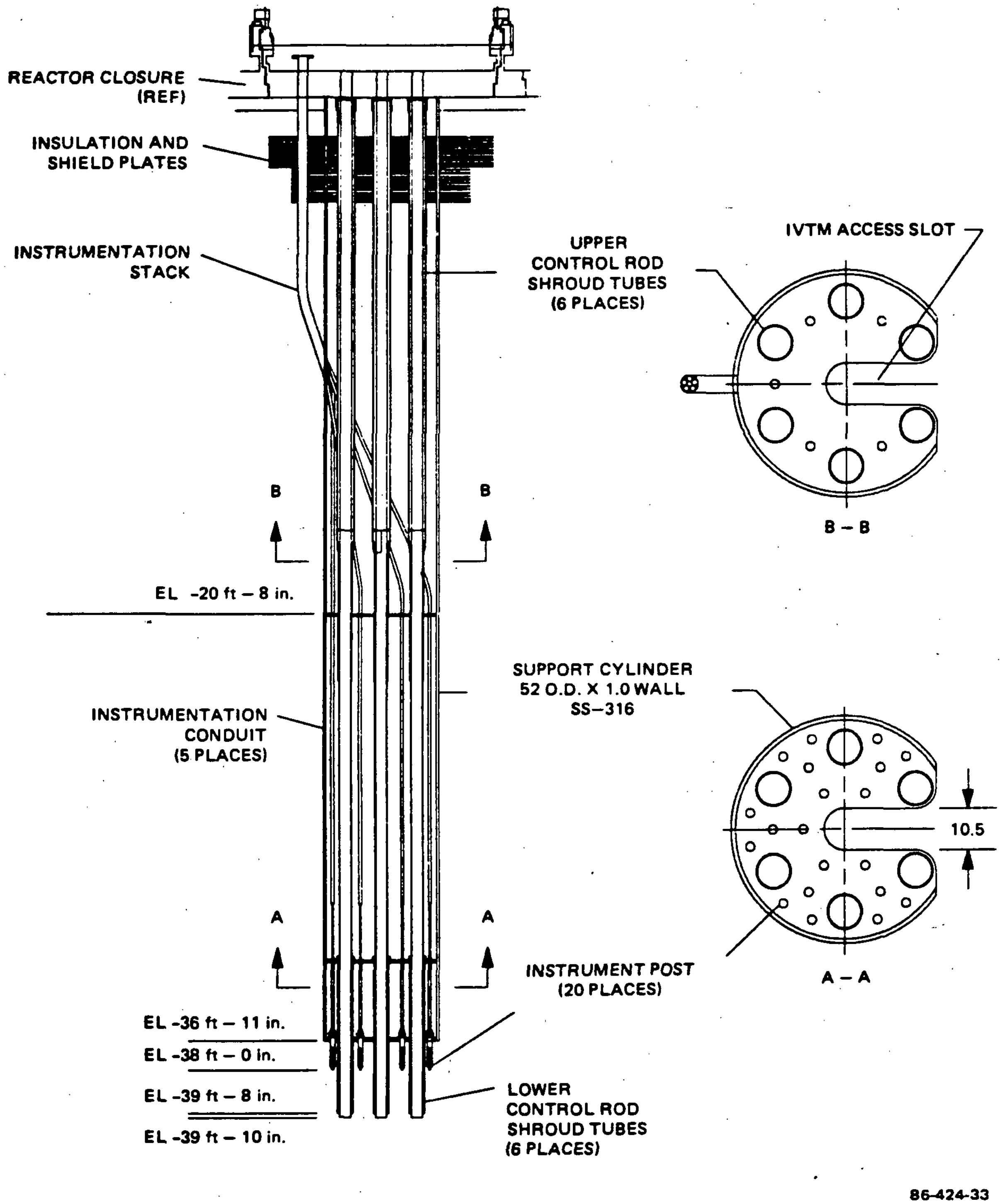


Figure 5.3–22 UPPER INTERNALS STRUCTURE

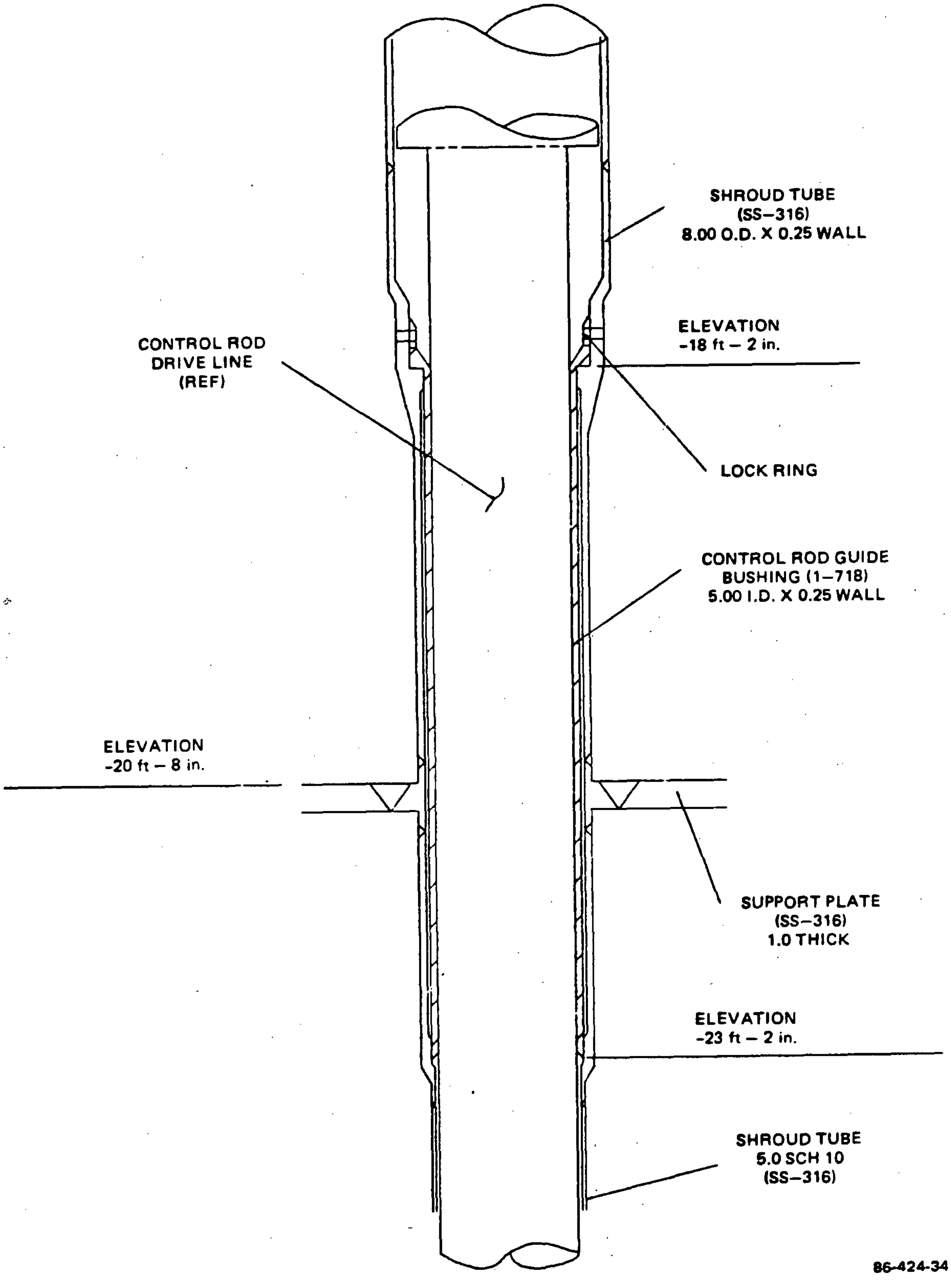


Figure 5.3–23 UPPER SHROUD TUBE DETAILS

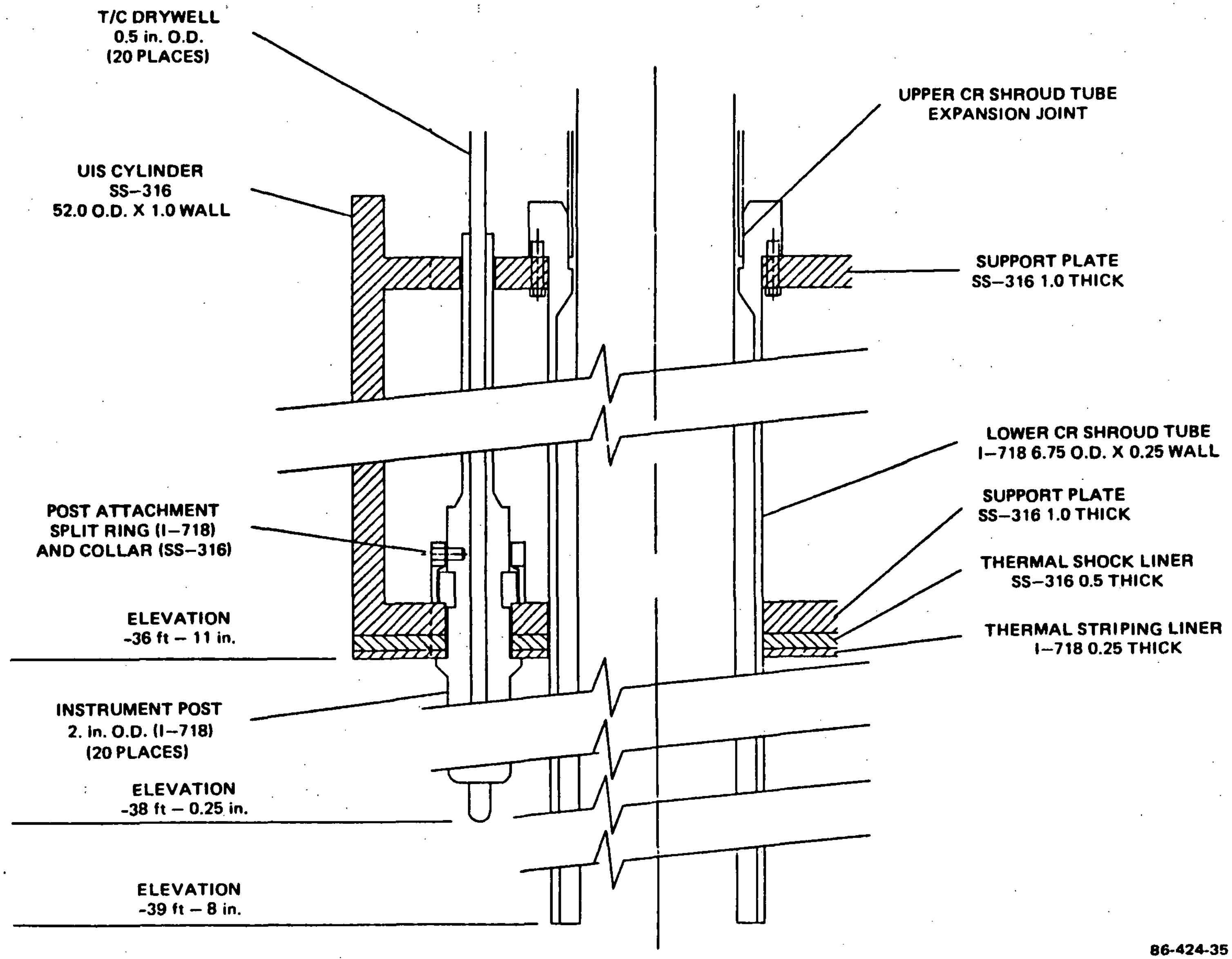


Figure 5.3–24 UPPER INTERNALS STRUCTURE DETAIL

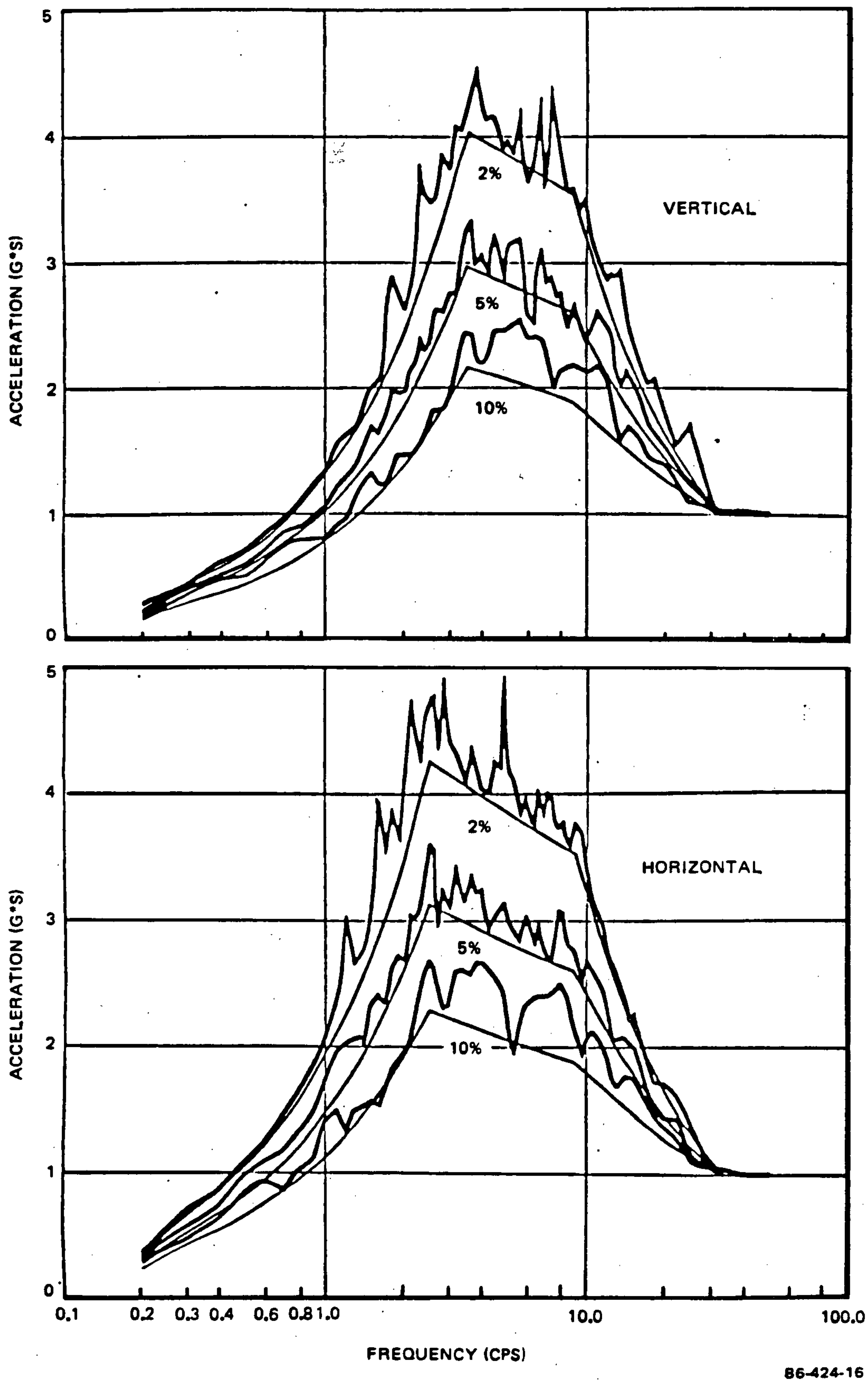


Figure 5.3–25 PRISM FREE-FIELD ANALYSIS/DESIGN SPECTRA (SSE)

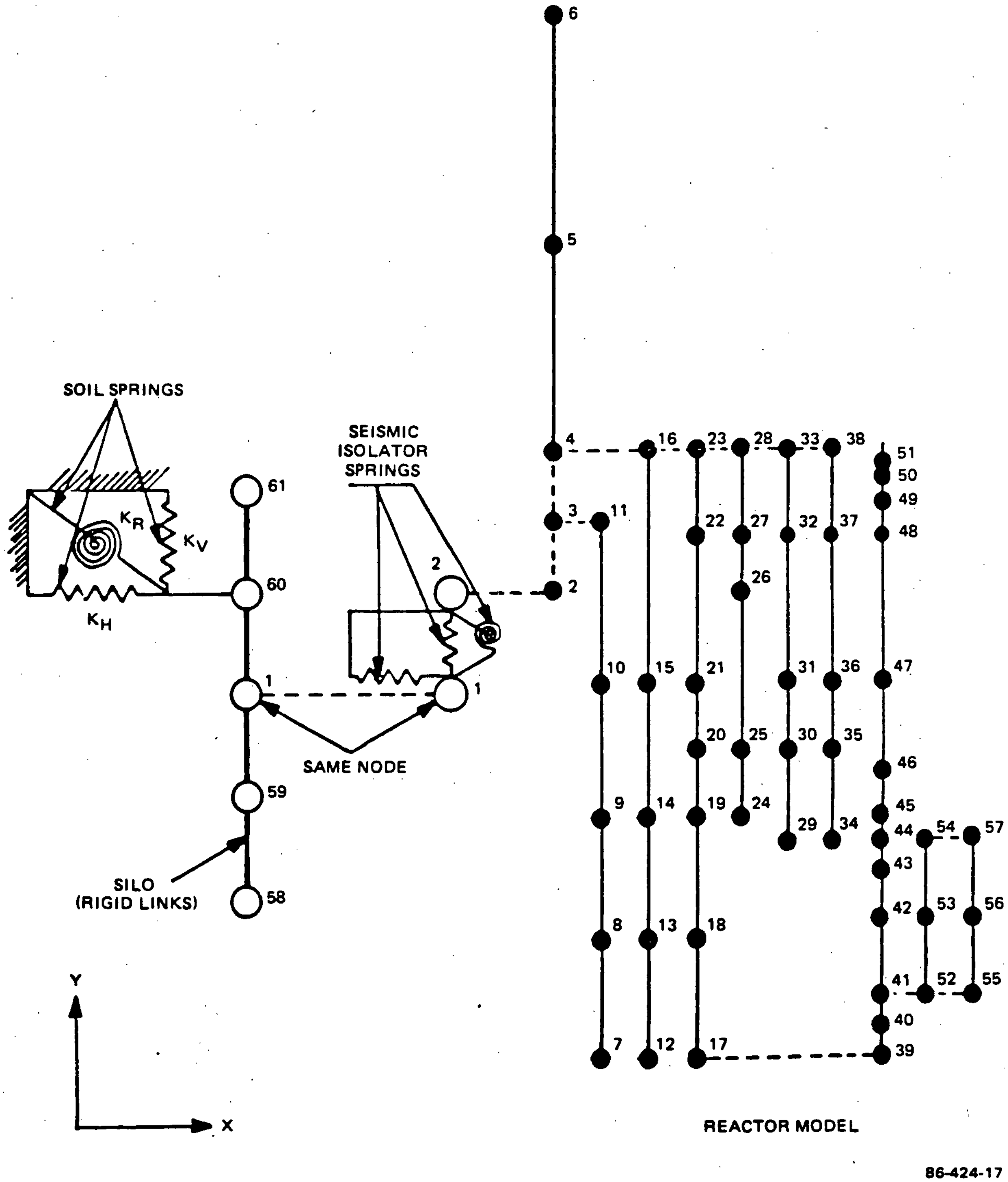


Figure 5.3–26 SOIL-STRUCTURE INTERACTION ANALYSIS MODEL

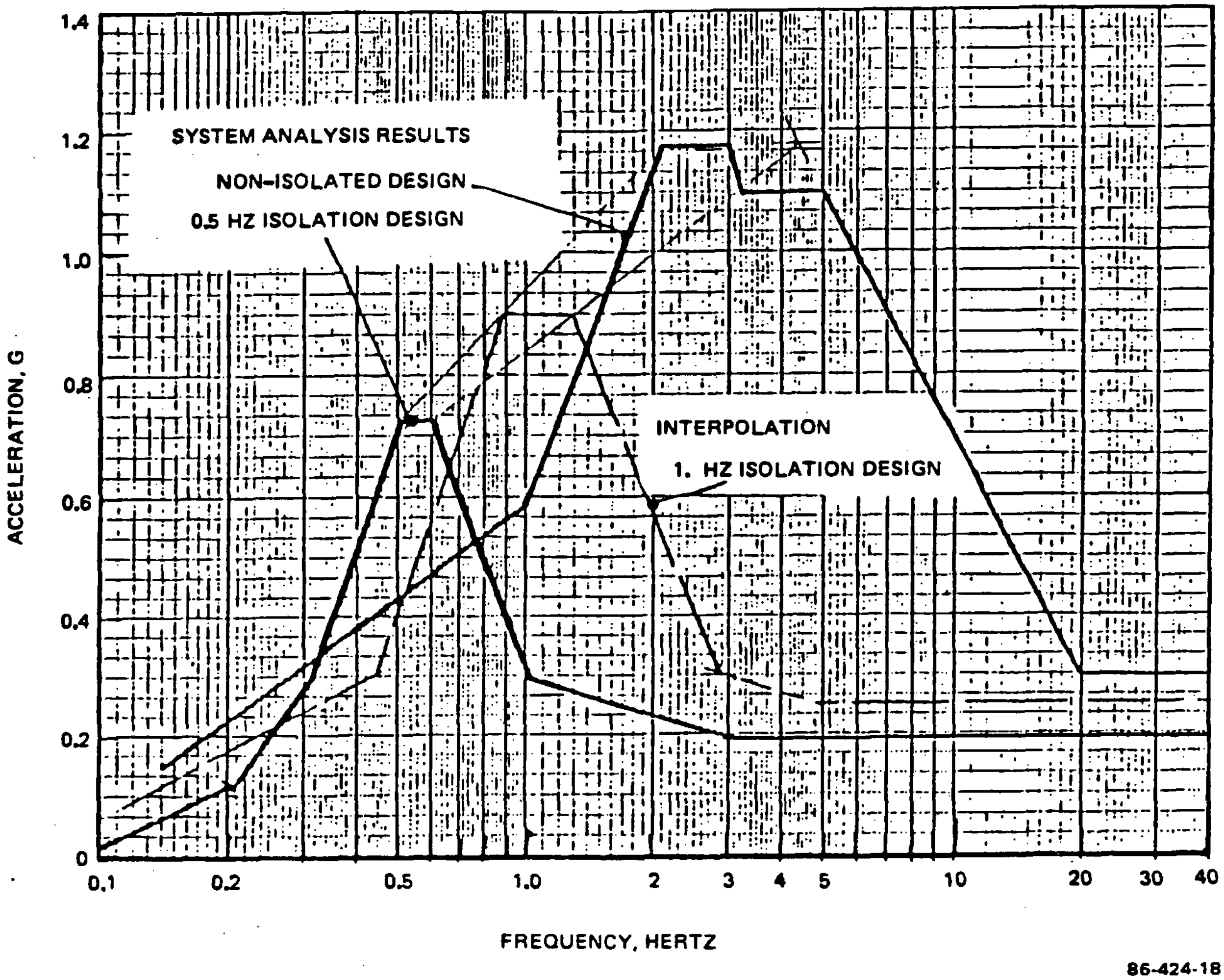


Figure 5.3–27 UIS SUPPORT (ROTATABLE PLUG) HORIZONTAL SSE RESPONSE SPECTRA USED IN THE UIS RESPONSE SPECTRUM ANALYSES

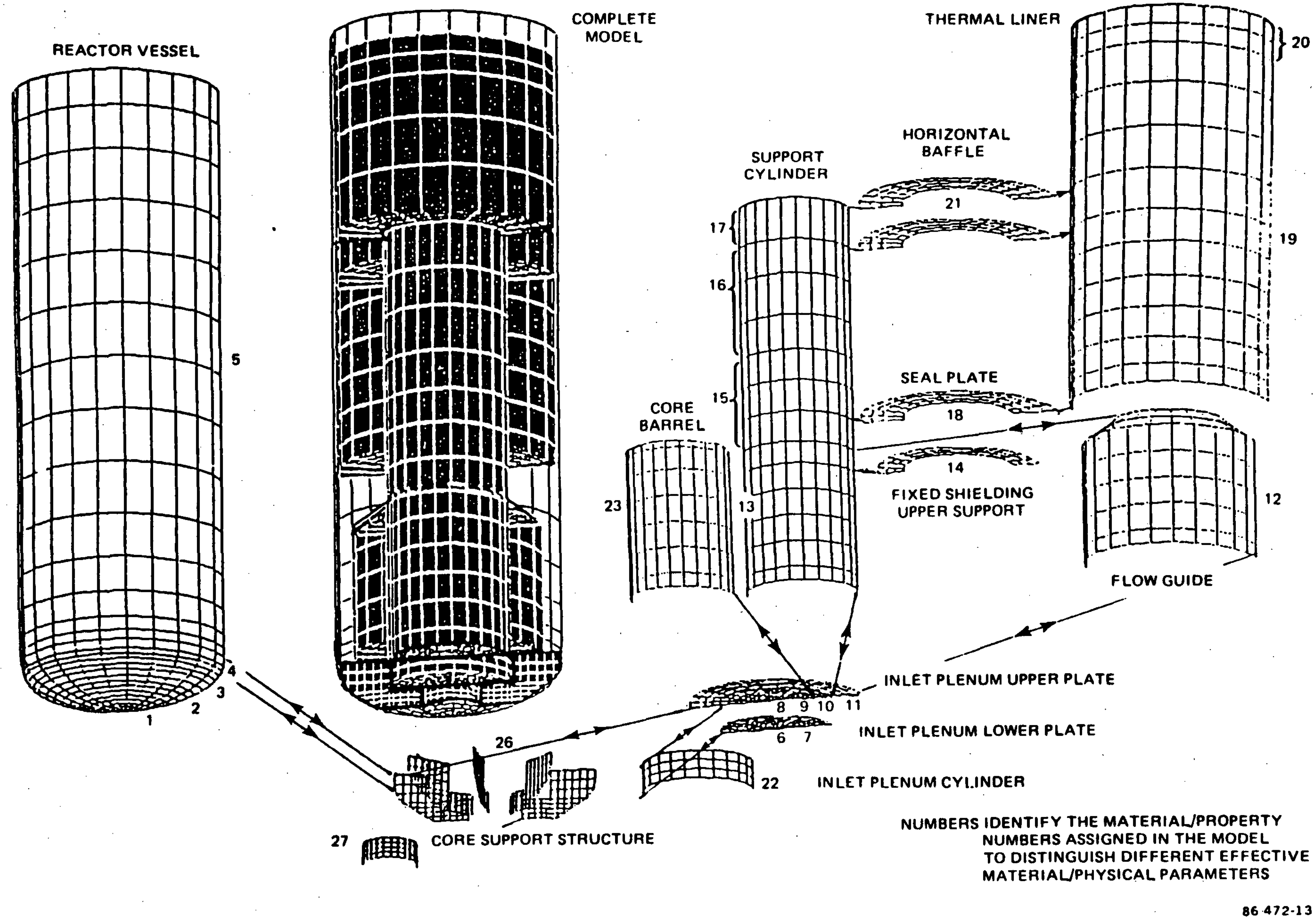


Figure 5.3–28 REACTOR VESSEL/SUPPORTED STRUCTURES ANALYSIS MODEL

5.4

5.4 Primary Heat Transport System (PHTS)

5.4.1 Design Basis

5.4.1.1 Functional Requirements

1. Transport reactor core generated heat to the IHTS sodium.

2. Provide a primary sodium flow rate for controlling reactor temperature conditions within limits which preclude damage to the reactor vessel, fuel, and reactor internals during normal reactor operating and shutdown conditions.

5.4.1.2 Design Life

A service life of 60 years shall be used as a basis for all PHTS components. Those items which cannot be reasonably expected to last the 60-year life of the plant shall be either sufficiently redundant or easily replaceable such that plant availability is not adversely affected. Each component shall be designed based on the operating hours and event frequency from the design duty cycle specified in Appendix D.

5.4.1.3 Configuration and Feature Requirements

Configurations and essential features that apply to the primary heat transport system of the reactor system are as follows.

1. The primary heat transport system shall be comprised of more than one primary pumps and more than one intermediate heat exchangers.

2. The primary heat transport system shall be located within the boundaries of the reactor system.

3. The intermediate heat exchangers shall be designed to have primary sodium on the shell side.

4. The vertical support of the IHX's shall be from the tops of the units.

5. The primary pumps and the IHX's shall be designed to be replaceable in the event of component failure.

5.4.1.4 Operational Requirements

1. The primary heat transport system (PHTS) shall accommodate a plant operational power range of 25 to 100 percent of rated power (425 MWt) with up to 10.0 percent step load changes at a maximum 60.0 percent per minute ramp load changes over the 10 percent step.

2. The PHTS shall be operable from the central control room under normal power or decay heat removal conditions.

3. The PHTS shall be operable with 0.1 percent failed fuel.

4. Components of the PHTS shall be designed to minimize radiation rates to ALARA requirements.

5. The PHTS shall be designed to the thermal hydraulic design conditions shown in Table 5.4-1.

6. The PHTS shall have the capability to transport reactor-generated decay heat effectively to the IHTS during all normal conditions while maintaining an adequate flow for controlling reactor temperature conditions within limits which preclude damage to the reactor vessel, fuel, and internals.

7. The PHTS shall support operation in a hot standby condition.

8. The primary pump total developed head requirement shall be 120 feet static head of sodium at the thermal-hydraulic design flow.

9. PHTS shall have the capability for heating the reactor sodium to hot standby temperature using pump power.

10. During normal plant operations, including refueling, the minimum temperature of the sodium in the PHTS shall be 375°F.

11. The PHTS piping and components shall have the capability to be heated in a dry, gas-filled or sodium vapor containing condition from ambient (70°F) to 450°F, outer surface temperature. The maximum heatup rate shall be dictated by stress considerations.

    The overall PHTS in conjunction with the IHTS and SGS shall be capable of removing heat from the reactor and supplying steam at 543°F and 975 psig at the turbine throttle.

12. The insulation of components and piping shall be designed to limit the outside surface temperature of the insulation to 125°F maximum during hot functional testing with 610°F sodium in the system and with an ambient HAA temperature of 100°F.

13. The PHTS shall be designed to mitigate the effect of thermal transients on other subsystems and components.

14. The layout of the PHTS shall be such that the primary circuits have approximately equal transport times and pressure losses for any given flow.

15. Leakage of primary sodium into the intermediate sodium loops shall be precluded by the IHX barrier. In addition, in the event of a leak, the IHTS pressure being higher than the PHTS pressure will allow leakage only from the intermediate to the primary loops.

16. The primary pumps shall provide sufficient inherent coastdown to prevent transient peak core outlet temperatures from exceeding 1300°F during unprotected (without scram) events while providing coastdown

flows which limit the core outlet bulk transient rate to less than -15°F/sec during duty cycle events.

5.4.1.5 Structural Requirements

The PHTS shall be designed to withstand all of the pressures, temperatures, and forces applicable to its components. The design conditions refer to the steady state which umbrella all service level conditions and service level limits. The PHTS duty cycle events include the steady-state and transient conditions for which the system shall be structurally designed. Structural evaluations of the system and its components shall include the duty cycle events and the operational basis and safe shutdown earthquakes. The design of the PHTS shall include all fabrication, handling, transportation, and installation loads.

5.4.1.6 Design Conditions

The loading conditions to be taken into account in designing the PHTS components shall include but not limited to the following: internal and external pressure, weight of the component and its contents, superimposed loads from other components, sodium/water reactions, vibration and seismic loads, reactions at supports, temperature effects, irradiation effects, and the effects of the sodium environment. Design basis pressures and temperatures for the system are shown in Tables 5.2-1 and 5.2-2. These conditions shall be used in conjunction with the plant duty cycle (Appendix D) to establish the thermal and mechanical loading conditions for the PHTS components.

5.4.1.7 Seismic Criteria

All structures and components of the PHTS shall be capable of withstanding the effects of the Operating Basis Earthquake (OBE) without loss of capability to remain functional and to withstand the effects of the Safe Shutdown Earthquake (SSE) without loss of capability to perform their safety functions.

Appropriate analyses are required, using the ground motion inputs below, to define the specific loads and accelerations for the primary heat transport system.

5.4.1.7.1 OBE/Plant Condition Load Combinations

1. The OBE horizontal and vertical maximum ground accelerations are 0.15g. The OBE response spectra shall be 0.5 times the SSE values given in Paragraph 5.4.1.7.2.

2. Five OBE's, each with 10 maximum peak response cycles, shall be assumed to occur over the design life of the plant.

Four of these OBE's shall be assumed to occur during the most adverse Service Level A operating conditions determined on a component limiting basis. The other one OBE shall be assumed to occur during the most adverse Service Level B condition determined on a component limiting basis, and at the most adverse time in the event. The low probability for the OBE makes this event a Service Level B loading.

5.4.1.7.2 SSE/Plant Condition Load Combination

1. The SSE horizontal and vertical maximum ground accelerations are 0.3g. The SSE response spectra are in Section 6.8.1 of Reference 5.4-1.

2. One SSE, with 10 maximum peak response cycles, shall be assumed to occur over the design life of the plant.

This SSE shall be assumed to occur during the most adverse Service Level A limit or Service Level B loading event determined on a component limiting basis. The probability of the SSE implies a Service Level D event. During and following the SSE the primary and intermediate pumps are assumed to be functioning at pony motor speed.

5.4.1.8 Design Criteria

1. Design of the PHTS components which comprise the sodium boundary shall conform to the component codes and standards given in Table 5.4-1.

2. Establishment of the limiting values for design stress intensity shall include allowances for any known or predictable degradation of mechanical properties that may occur as a result of irradiation, stress at service temperatures and changes in material properties over the design life.

3. The PHTS shall be designed as Seismic Category I structures.

5.4.1.9 Material Requirements

The materials of construction of the PHTS components shall be selected on the basis of performance in fast reactor and liquid sodium environments. Constituent elements whose transmutations have long half-lives shall be controlled to minimize their impact.

1. The effects of environmental conditions such as neutron radiation exposure, temperature, and sodium shall be included in determining the allowable value of material properties used in the design of system components.

2. Material surface in contact with the liquid sodium coolant shall be austenitic stainless steel unless other materials must be used for strength or wear resistance.

3. Surfaces that experience relative motion during operation, installation, or removal shall be made of suitable material combinations or shall be provided with hard-surfaced regions to provide adequate wear properties and to preclude galling or seizing.

4. Appropriate heat treatments and processes shall be utilized during fabrication to minimize sensitization of stainless steel components.

5.4.1.10 Quality Assurance Requirements

1. Quality Assurance programs complying with ANSI/ASME NQA-1 shall be established and implemented during the design, development, fabrication and installation phases.

2. Safety-related items shall be identified, and appropriate records of the design, fabrication, erection and testing of safety systems shall be maintained throughout the life of the plant.

3. The Quality Assurance program for non-safety related structures, systems, and components shall be consistent with ANSI/ASME NQA-1 and/or the objective of energy generation at minimum costs.

5.4.2 Design Description

5.4.2.1 Introduction

The PHTS sodium flow path is contained within the reactor vessel. Sodium is routed through the reactor core, the hot pool, the shell side of the IHX, the cold pool, the pumps, the pump discharge piping and the core inlet plenum.

Flow paths in the PHTS are identified in Figure 5.4-1. Sodium from the hot pool enters and flows through the two IHX's where it is cooled. The sodium exits the IHX at its base and enters the cold pool. From there, cold pool sodium is drawn through the fixed shield assemblies into the pump inlet manifold. The four EM pumps intake the cold pool sodium from the manifold and discharge it into the high pressure core inlet plenum through the piping connecting each manifold to the plenum. The sodium is then heated as it flows upward through the core and back into the hot pool.

5.4.2.2 Primary EM Pump

Four submersible EM pumps provide primary sodium circulation through the reactor. The pumps are installed through penetrations in the fixed portion of the closure head into an annular area above the core shared with portion of the closure head into an annular area above the core shared with intermediate heat exchangers. The pump assembly is shown in Figure 5.4-2. Approximate size and weight of each pump is approximately 40 inches in diameter by approximately 40 feet long, weighing about 18 tons. Primary sodium coolant is drawn from an inlet plenum beneath the pump. This is cold sodium from the IHX which has passed through the fixed core radial shield region.

The pump design configuration is shown in Figure 5.4-3. As depicted on this illustration, sodium enters through a large annular opening at the bottom of the pump. Within the pump, the sodium converges to the tapered inlet section of the pump duct where velocity increases from approximately 30 fps to the design velocity of approximately 50 fps through the remaining 2/3 of the pump duct. The sodium discharge at the top of the pump passes radially outward into a plenum from which it is piped to the core inlet structure. There are three reactor internal structure seal plate interfaces for the piston ring seals of the pump - one seal plate at the pump inlet and two seal plates near the top of the pump forming part of the discharge plenum.

As shown in Figure 5.4-3, the pump stator is located radially outward from the aforementioned pump duct. It is in an inert gas-filled enclosure formed by the outer pump duct wall, the external stator support cylinder, and the end forgings to which these cylindrical sections are welded. The electrical power leads are routed from the stator enclosure, through a conduit across the pump outlet plenum, and into the lifting/ handling structure which extends upward through the reactor vessel closure head.

This pump is self-cooled in that the heat generated by electrical losses in the stator is transferred to the surrounding sodium. Most of this heat energy is transferred through the duct wall into the pumped sodium since that sodium boundary has the best thermal coupling to the heat source. A smaller portion is transferred radially outward through the stator support cylinder. Since all heat losses are transferred into the primary sodium coolant, the adverse effect on overall plant efficiency is minimized.

The center iron, providing a magnetic boundary for the "air-gap" (pump duct) flux, is also in an inert gas-filled enclosure. This enclosure is formed by the pump duct wall (on the outside) and an internal support cylinder. The center iron assembly is installed in the central region of the pump near the end of the fabrication sequence and thereby is an integral part of the pump as installed in the reactor vessel.

Each EM pump has a safety-related (IEEE Class 1E) solid-state power supply and is thereby controllable over its full flow range. The power controller is shown schematically on Figure 5.4-4. Pump coastdown flow, shown on Figure 5.4-5, is provided by a synchronous converter that is motoring on the system during normal operation. Also, while motoring during normal operation, the synchronous converter provides power factor correction for the system to counteract the inductive load of the EM pump.

The general design conditions and special requirements for the solid state system are:

1. EM pump rating: 1165 kW at 1110V and 20Hz.
2. Power factor: 63 percent.
3. Startup: from no load to full power on 30-second ramp.
4. The system is to allow for occasional pump flow adjustment so that all pumps are delivering the same flow.
5. Emergency Coastdown: minimum coastdown requirements are shown in the form of a curve on Figure 5.4-5.
6. Emergency coastdown is initiated after 1/4-second delay to prevent spurious reactor trips due to ac line transients.
7. During emergency coastdown no more than two pumps out of four per reactor are allowed to fail.
8. After emergency coastdown automatically continue to operate at seven percent rated flow from off-site or on-site power if available.
9. Components required for emergency coastdown are to satisfy Class 1E requirements.
10. After loss of one EM pump the reactor is tripped and all remaining pumps are coasted down.

Upon loss of the normal power supply, the stored kinetic energy in the synchronous machine is utilized for coastdown of the EM pump.

The power conditioning unit includes a solid state rectifier section, a dc reactor link and a load commutated inverter (LCI). The LCI represents a solid state system which is based on proven control technology. It is widely used in the industry for adjustable speed drive systems. The LCI system is highly reliable and is designed with flexible operating characteristics, which allows variation of both frequency and voltage over a wide range. The power conditioning unit is rated at 1500 kVA with output of 1110 V, 3 phase, 20 Hz. The LCI unit is provided with the required control, protective, monitoring and alarm circuits. A built-in diagnostic feature facilities trouble shooting and reduces repair time significantly.

The synchronous machine is rated at 2000 kVA, 1110 V, 3 phase, 20 Hz. It is provided with a fast acting excitation system, which is capable of controlling excitation and voltage of the machine over a wide range. The inertia of the machine is believed to be capable of providing the required coastdown energy for the EM pump. A flywheel may be added if necessary.

The electrical system includes all required power and control devices. Power circuit breakers are also provided as indicated in Figure 5.4-4. In order to ensure a very reliable coastdown operation, the synchronous machine is designed and qualified as Class 1E. A circuit breaker is provided for isolation of the Class 1E parts of the system from the non-Class 1E. Although desirable, the circuit breakers need not be Class 1E. However, the system has to be analyzed and designed to accommodate this option. Also, the system may be designed such that no active excitation control is required during the emergency pump coastdown. However, this may result in a system which is less efficient during normal operation.

Power to the EM pump is normally supplied from the 4160 V, 3 phase, 60 Hz ac distribution system through a dedicated input transformer and solid state power conditioning unit. This unit is also utilized to supply power to the EM pump during startup and normal shutdown operation.

An auxiliary synchronous machine supplies coastdown power to the EM pump when the normal power supply through the power conditioning unit is lost. This machine is connected in parallel with the EM pump. Normally it is running unloaded, in an overexcited mode of operation, supplying the reactive power requirements of the EM pump and power conditioning unit.

5.4.2.3 Intermediate Heat Exchanger

There are two intermediate heat exchangers (IHX's) in the PRISM reactor module. The IHX's are in the primary heat transport subsystem along with the four EM pumps. The purpose of the IHX is to transfer the heat from the primary radioactive sodium to the intermediate non-radioactive sodium. The two IHX's are located above the reactor core in the annular region between a reactor support cylinder and the reactor vessel wall. All components of the IHX are constructed of austenitic stainless steel. The design conditions for the IHX are shown on Table 5.4-2.

The IHX design consists of an upper and lower tubesheets separated by straight tubes, with a central downcomer and riser for incoming and outgoing intermediate sodium, respectively. The IHX cross section is kidney shaped. This IHX shape was selected to optimize the IHTS piping arrangement exiting the reactor system and to most effectively use the reactor space available to minimizing the overall diameter of the reactor vessel. The IHX design layout, tube bundle details and an isometric drawing of the IHX are shown on Figures 5.4-6, 5.4-7, 5.4-8.

Each IHX is rated at 212.5 MWt for a total rating of 425 MWt for the modules. The IHX is designed for operation in the vertical position within the primary sodium pool as it is supported by, and hangs from, the reactor vessel top head. The support arrangement is such that by cutting a seal weld and removing bolts at the IHX flange to top head junction, the IHX can be completely removed from the reactor module.

Primary sodium from the hot sodium pool enters the IHX at an elevation just below the upper tubesheet. The sodium flows downward through the IHX on the shell side. The primary sodium flows around and through the tube support plates into a plenum below the lower intermediate sodium channel head. This lower plenum has two outlet nozzles where the primary sodium exits into the cold sodium pool.

The cold incoming intermediate sodium flows down the central downcomer and enters the lower intermediate sodium channel where it splits into two streams. Each stream then flows upward through the lower tubesheet, inside the tubes, and through the upper tubesheet. The intermediate sodium exits the top of the upper tubesheet at the intermediate sodium upper channel. It then flows into the annular space between the concentric external riser cylinder and internal downcomer cylinder. This sodium leaves the IHX through the intermediate outlet nozzle for use in the intermediate heat transport loop.

The IHX downcomer is of double-wall design in order to minimize regenerative heat transfer between cold incoming and hot outgoing intermediate sodium. A thermal barrier of an additional stainless steel cylinder gas has been provided in the annulus between the double walls of the downcomer above the tube bundle region. This barrier is necessary to protect the double walls from the temperature gradients that exist, as well as to minimize regenerative heat transfer effects on the intermediate side.

The upper end of the IHX has a shield plug which contains the thermal and radiation shielding material. This plug is an integral part of the IHX. The upper tubesheet is fixed and is supported from the riser cylinder, intermediate sodium upper channel, and the outer shell. The lower tubesheet is floating and is supported by the tubes.

The tube bundle for each IHX consists of two individual tube bundles located on either side of the downcomer. The two tube bundles are structurally integrated at a single upper tubesheet at the top and a single lower tubesheet at the bottom. The tube bundle contains 2139, 5/8 in. x 0.035 in. minimum thickness straight tubes per IHX. The tubes are arranged on a 1.031 in triangular pitch. The tubes will be end-welded to the tubesheets and then explosively expanded into the tubesheet holes. The explosive forming will use the same techniques developed by Foster Wheeler on the CRBRP-IHX.

The tube support plates, which follow the tube bundle contour, are located, alternatively, at the inside (radially close to the IHX inner shell) and the outside (radially close to the IHX outer shell) of each bundle of the IHX. The support plates are perforated to accommodate the tubes and will also contain flow holes which have been sized to control the cross flow sodium velocities. The support plate tube holes will be sized to provide adequate clearances. The support plates will be spaced to control flow-induced tube vibration and tube buckling. The clearances between the support plates and the shell will be controlled and minimized by providing machined fits between the support plates and the shell. The sodium cross flow in the IHX tube bundle is achieved by the use of these support plates. The sodium cross flow mitigates the potential for thermal imbalance in the tube bundle.

Expansion bellows are located at the top of the downcomer assembly (upper end) of the IHX. These bellows absorbs the differential thermal growth between the tube bundle and the IHX downcomer. The bellows were sized for an internal design pressure of 150 psi and an axial displacement of 1.125 in. In order to accommodate the sodium-water reaction pressure of 1000 psi, the bellows are reinforced with backing rings and a backing cylinder.

5.4.3 Design Evaluation

5.4.3.1 Steady State Thermal Hydraulic Analysis

The three-dimensional thermal hydraulics code COMMIX was used to analyze the PRISM reactor under steady state conditions. The following are brief descriptions of the COMMIX Code, the model representation of PRISM, and the analysis results for steady-state. More detailed information about the COMMIX analysis results are presented in Reference 5.4-2.

5.4.3.1.1 Description of COMMIX Computer Code

COMMIX is a series of computer codes for the analysis of LMFBR systems components having three-dimensional and transient capability which represents state-of-the-art methodology. The code is designed to accommodate both forced and natural circulation flow conditions. An effective viscosity turbulence model combined with distributed resistances is used to provide a simplified but adequate approximation of turbulent transport phenomena. The standard governing equations for conservation of mass, momenta, and energy are solved as an initial-value problem in time and a boundary-value problem is space. Modeling is also provided to describe heat generation within and transfer from solids distributed in the fluid domain.

5.4.3.1.2 Model Description

The PHTS and surroundings have been modeled using the cylindrical geometry option in COMMIX-1AR. Symmetry considerations showed that a 90° sector was sufficient, spanning the space from the centerline between two adjacent pumps to the centerline of one intermediate heat exchanger (IHX). The radial extent of the model is bounded by the centerline of the reactor vessel (RV) and the collection cylinder (CC), which is the outer boundary of the air riser portion of the reactor vessel auxiliary cooling system (RVACS); the air downcomer of the RVACS was not modeled, as it is separated from the riser by an insulating barrier. Axially, the model begins at the bottom of the RV and extends to the bottom of the overflow slots in the reactor vessel liner (RVL). The hemispherical shape of the bottom of the RV was ignored. Although multiple fluids are allowed, they must be separated by fixed nonporous boundaries; this precludes modeling the free surface motion of the sodium levels in the main hot pool and in the space between the RVL and the RV; these regions were assumed to be filled with sodium and the cover gas space was not modeled.

Several views of the resulting geometry are shown in Figures 5.4-9 through 5.4-12. An elevation view is provided in Figure 5.4-9, showing two azimuthal locations: the right-hand side of the figure is the centerline of the pump duct and the left-hand side is in the IHX. All major geometry boundaries (shown as solid lines) are displayed by these two views except for the pipes which run from the pump outlet plenum down to the core inlet plenum. Figure 5.4-10 provides a plan view at the bottom of the core inlet plenum. Figure 5.4-11 is a plan view at axial level just below the pump outlet plenum. This shows the location of the pump duct and the return pipes. Figure 5.4-12 is a plan view at axial level just above the primary sodium inlet to the IHX. The blank cells in this region correspond to the pump support; material in this region is not modeled.

5.4.3.1.3 Steady State Results

The steady-state solution is obtained by performing a "transient" calculation with constant boundary conditions. A calculation has been performed for steady state using the COMMIX model. Various portions of the velocity distribution are shown in Figures 5.4-13 through 5.4-17. The overall character of the axial velocity distribution is shown in the elevation view in Figure 5.4-13. The flow is upward through the pump in cells (6,3, 8-11) to its outlet plenum. After going down through the pipes (not shown in this figure) the flow enters the inlet plenum at K=2, mixes and travels upward through the several assembly regions in cells (1-4, 1-8, 4-7). The flow field is significantly affected by the presence of the bottom support plate of the UIS, which is above the core exit at K=8. As shown in Figure 5.4-14, most (97 percent) of the core exit flow is diverted to cells at J=1, which is the refueling slot, and to cells at I=4, which is the space between the UIS bottom plate and the fuel storage rack. The rest of the flow field in the UIS region shows circulation patterns with small velocities. The presence of the UIS baffle plates appears to prevent the hot coolant exiting the driver and radial blanket assemblies from streaming directly upward to the top of the hot pool.

At the top of the hot pool, the flow is diverted toward the IHX-side of the model (J = 6-8) from all other regions as shown in Figure 5.4-15 and the previous two figures. Referring to Figure 5.4-13, the flow then travels downward through the IHX tubes, exiting into the cold pool at axial level K=8. The flow then redistributes to other radial and azimuthal cells as it continues to flow downward in the annulus between the fixed shielding and the RV. At the bottom of the fixed shielding (K=3) the flow turns radially inward and then begins its upward travel through the fixed shielding. At the top of the fixed shielding (K=6), the flow is diverged azimuthally from the IHX-side (J = 6-8) to the pump-side (J = 1-5) of the model as shown in Figure 5.4-16 and then exits into the pump inlet plenum region. This completes the circuit for the primary flow path.

Figure 5.4-17 shows the velocity distribution in the sodium in the gap between the RVL and the RV; the left half of this figure is for I=8 (adjacent to the RVL), the right half is for I=9 (adjacent to the RV), the center corresponds to J=1 (pump side), and the left and right edges both correspond to J=8 (IHX side). There are two types of natural convection circulation patterns present: in the lower portion, the flow is upward along the IHX and downward along the pump which is cooler than the IHX; in the upper portion, the flow is upward along the RVL and downward along the RV which is cooler than the RVL.

The major features of the temperature distribution are shown in Figure 5.4-18. The temperature in the cold pool, fixed shields, pump, piping, and core inlet plenum is uniform at 610°F (321°C). The temperature reaches a maximum of 900°F (482°C) at the top of the fuel in the driver assemblies in cells (1, 1-8, 5). A significant amount of fluid mixing occurs just above the top of the assemblies at K=8, this reduces the radial temperature difference to only 32°F (18°C), which is substantially below the 283°F (157°C) value at the top of the fuel segment at K=5. The additional mixing induced by the UIS further reduces this radial temperature difference to 9°F (5°C) at K=14, which corresponds to the top of the fuel storage rack, as shown in Figure 5.4-19. The majority of the hot pool is well mixed and fairly uniform in temperature at 874°F (468°C). Referring to Figure 5.4-18, one can see the decrease in the temperature of the primary sodium as it flows downward from the hot pool (K=17) to the cold pool (K=8) through the IHX (on the side with J=7).

As shown in Figure 5.4-20, the temperature distribution in the sodium between the RVL and the RV is fairly uniform both radially and azimuthally; the primary variation is axial, with a minor azimuthal variation near the axial level corresponding to the location of the pump outlet plenum (K = 12-15).

In summary, results for the steady-state analysis show that the sodium is well mixed and near isothermal conditions in the hot and cold plena. Preliminary evaluations of the ANL water tests data show that the plena are as well mixed as the COMMIX predictions indicate. The flow field is significantly affected by the presence of the bottom support plate of the UIS. The flow is diverted to the refueling slot of the UIS and the annular region between the UIS and the fuel storage rack. The mixing in the region above the core exits and the UIS bottom plate appears to be very effective. The presence of the UIS also results in a more direct flow path to the IHX with generally low flow velocities at the hot pool free surface.

The COMMIX analysis results confirm the existence of natural circulation loops within the 1.5-inch annular gap between the reactor vessel liner and the reactor vessel. These circulation loops are created by uneven reactor vessel liner temperatures set up by the presence of the IHX (hot) and the primary pumps (cold) near the liner. Circumferential sodium temperature differences of 25°F and more are set up in the gap sodium. The existence of these circulation loops indicates that heat transfer is effective through this gap.

The results of the steady-state analysis shows that there are no unusual or unexpected conditions that are of concern to the designer.

5.4.3.2 Intermediate Heat Exchanger

5.4.3.2.1 Analytical Methods for the Intermediate Heat Exchangers

The intermediate heat exchanger shall be constructed as Class 1 components in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components.

The construction of parts and components for design temperatures exceeding 800°F shall be in accordance with Code Cases N-47 through N-51.

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 PRISM 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 the purchaser.

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:

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

2. Determine the overall pressure loss characteristics.

3. 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 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.

Vibration analyses of the tube bundle structural design covering peak velocities over the range of flows from zero to 100 percent of design flow rate shall be performed. 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 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. Since the long term maximum operating temperature of the PRISM IHX is 875°F, the creep damage factors in the austenitic stainless steel IHX components are expected to be small. Engineering analysis will focus its attention on those PRISM IHX components which experience the maximum temperatures and loadings. The upper tubesheet, upper tubesheet-to-shell junctions, upper tubesheet plenum, and the riser pipe to upper plenum junction are all regions which need to be evaluated for creep-fatigue damage.

Two areas of the PRISM IHX, in which the seismic loads in themselves must be considered, are the riser-to-upper tubesheet plenum junction and the upper portion of riser/support cylinders IHX mounting flanges junction. Both these areas experience larger overturning moments due to the horizontal portion of the seismic motion.

For those portions of the PRISM 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 PRISM IHX elastic and simplified inelastic analysis will be used in the design phase and fabrication release phase. The simplified inelastic analysis involves the use of two dimensions, thick-walled cylinder of finite length, thin shell cylinder programs 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 IHX will be modeled using cam and shell-type three-dimensional finite elements.

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

1. Upper Tubesheet Assembly
2. Riser Pipe to Upper Plenum Junction
3. Upper Portion of Primary Shell

Component damage accumulation will be accounted for using the methods contained in Code Case N-47 of the ASME Boiler and 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 faulted condition allowables of Code Case N-47.

5.4.3.2.2 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. Areas of low flow, or pockets, or corrosion entrapment shall be avoided to the maximum extent possible.

The IHX shall be sized to dissipate 215 MWt for the full load process temperatures and flows given below:

Parameter	Primary	Intermediate
Inlet Temperature (°F)	875	540
Flow Rate (Full Flow (lbs/hr)	8.94×10^6	9.15×10^6
Pressure Drop (Nozzle to Nozzle) (psi)	$4 \pm 20\%$	$19 \pm 20\%$
Inlet Pressure (psig)	15	110
LMTD (°F)	71.0	

The IHX nominal overall heat transfer coefficient is 1785.5 Btu/hr-ft^2-°F for the temperature and flow conditions given above. This yields a required heat transfer area of 5786 ft^3 (or a length of 16.53 feet for each of the 2139 tubes which have an outside diameter of 0.625 inch and a nominal wall thickness of 0.0385 inch). The actual effective tube length is 17.58 feet. This increase in length (5.4%) accounts for the uncertainties in tube and shell side heat transfer coefficients and tube wall thickness as well as allowances for tube plugging. The actual tube length (between tubesheets) is 19.25 feet versus the effective length of 17.58 feet. The difference includes partially inactive entrance and exit regions and 6-3/4 inch inactive length through the support plates. This sizing will assure that under the most pessimistic conditions, the unit will transfer 215 MWt with the primary temperatures of 875°F (inlet) and 607°F (outlet), and the intermediate temperature of 540°F (inlet) and 800°F (outlet) at the flow rates given above. The implication of this oversizing is that the primary temperatures will not have to be as high as 875/607°F when the plant is operating at rated power. The nozzle-to-nozzle pressure drops at rated flow are 4 psi nominal for the shell (primary) side and 19 psi nominal for the tube (intermediate) side.

The intermediate heat exchanger is located above the reactor core and therefore the thermal center of the IHX tube bundle is placed at an elevation higher than the core midplane. This arrangement promotes natural circulation of the primary sodium, thus helping cool down the reactor core, particularly at low flows and reactor scrams.

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 IHX is designed to use tubes with 0.035 inch D wall thickness (i.e., 0.035 inch min.). Allowing for 0.001 inch corrosion on either side of the tube wall, and 0.0005 inch for scratches on the surface, the minimum available wall thickness for analysis is 0.028 inch.

Analysis per ASME code for 150 psi design pressure at 900°F requires a minimum wall of 0.003 inch for internal pressure. The available wall thickness of 0.028 inch would permit an external pressure of 430 psi (per code) with an inherent safety factor of 3. Actual calculated collapse pressure is therefore 1287 psi for the minimum inches available wall thickness of 0.028 inch. Therefore, there is ample margin over the specified primary sodium design pressure.

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.

5.4.3.3 EM Pump Performance Characteristics

The pump characteristic curves of head, at various voltage and system pressure versus volume flow are plotted in Figure 5.4-21. The curves pivot about the synchronous flow rate which is determined by the driving frequency. The frequency can be adjusted along with the voltage to provide an operating point which is both efficient and stable at the design point of 120 psi and 10,500 gpm (42,000 gpm for all four pumps).

The information is a based on an equivalent circuit analysis of a preliminary pump design. The equivalent circuit computer program for the annular EM pump was developed from basic equivalent circuit equations found in any classical electrical machinery text.

Referring to Figure 5.4-22, the developed pressure in the pump is,

$$P = Bjc \qquad (1)$$

where B = magnetic induction

j = instantaneous current density

c = length of duct

The voltage required is,

$$V = (\rho_f j + v_f B)b \qquad (2)$$

where ρ_f = electrical resistivity of fluid pumped

v_f = velocity of the fluid

b = width of the fluid in duct

The volume flow rate is,

$$Q = v_f ab \tag{3}$$

and the power required is,

$$W_o = Bjv_f(abc) \tag{4}$$

where a = height K:+id in duct

The high currents induced within the fluid itself, flow in closed loops about the center line contained within the annular duct. The magnitude of the force on each element of fluid is determined from the product of the flux density and current density in each element as shown by Equation 1. The resulting currents in the fluid interact with the traveling magnetic field to produce force on the fluid in the direction of motion of the field. The principle of operation is identical to that of the polyphase induction motor, which is used widely in industry and utilities for motor applications in five to multi-thousand hp sizes.

Hydraulic Pressure Drop - Hydraulic pressure drop in an EM pump is caused by entrance and exit losses and the viscous drag imparted to the fluid at the duct boundaries. Entrance and exit losses are usually expressed in terms of the velocity head,

$$Pe = h\frac{\rho v2}{2} \tag{5}$$

where h = number of velocity heads

ρ = mass density of fluid

The number of velocity heads depends upon the details of the entrance and exit conditions and may be approximated for a particular configuration by reference to various publications.

Viscous loss is normally expressed in terms of the friction factor δ, the ratio of duct length to hydraulic diameter L/D, and velocity head. In a form of the Fanning equation,

$$P_{\mu} = 4 \ \delta \ \frac{L}{D} \frac{\rho v^2}{2} \qquad (6)$$

With the usual flow conditions, friction factor is a function of Reynolds number N_R, duct surface conditions, and duct curvature. When a conducting fluid flows through a magnetic field, circulating electrical currents flow within the fluid and introduce an additional body force on the fluid that influences the velocity distribution across the duct. This tends to modify the friction factor and higher values of pressure drop are associated with higher velocity gradients at the boundaries.

Cavitation Considerations - Cavitation is generally associated with centrifugal pumps and propellers and has been the subject of extensive analyses and experimentation in performance or physical damage to mechanical parts, or both.

Reasonable correlations of net positive suction head (NPSH) with cavitation inception and cavitation damage have been obtained for centrifugal pumps but similar criteria for EM pumps have not been well established. The duct passage of an EM pump consists, essentially, of an inlet section, an inlet transition region in which the fluid velocity is increased in magnitude and may be changed in direction, an outlet transition region leading to the pump outlet, and a pump outlet section. Thus, flow in EM pumps can be considered as similar to flow through venturi type duct passages. Some insight concerning the cavitation performance of EM pumps may be developed by considering cavitation test data on venturi passages.

Cavitation inception data on venturi type passages are generally presented in relation to the cavitation number, given by the expression,

$$\sigma = \frac{P - P_v}{1/2 \, \rho v^2} \qquad (7)$$

where P is the ambient static pressure in the free stream, P_v is the vapor pressure, ρ is the mass density of the liquid, and v is the velocity in the free stream. Data on venturi cavitation tests give values of 0.08 to 0.17 for sonic cavitation and 0.06 of visible cavitation. The flow at the entrance to the pumping section of an EM pump may be less uniform than that in such test section, since it may be affected adversely by fringing electromagnetic fields and a less favorable transition configuration. Thus, the range of cavitation numbers sited above probably represents a minimum for EM pumps and typical values, even for carefully designed pumps, may be considerably higher. No evidence of cavitation damage has been reported for the several induction pumps tested. However, comprehensive tests are needed to provide sound criteria for the design of EM pumps for applications requiring low NPSH.

Selection of Materials - The classes of materials which directly affect the design and construction of electromagnetic pumps for liquid metals include:

- Duct Materials
- Magnetic Materials
- Electrical Conductors and Insulation

Duct Materials - The electrical resistivity of the duct material significantly influences the design of EM pumps. The duct material is electrically in parallel with the pumped fluid; and should have high electrical resistance. Thus, to minimize losses, metallic duct walls should have electrical and mechanical requirements. The use of 304 type stainless steel has been used for sodium applications in this temperature range (600°F), however, there may be improved superalloys which would be advantageous to consider. Candidate materials will be investigated in the future because of the significant influence on pump performance.

Magnetic Materials - The principle of electromagnetic pumping requires that the duct and fluid be a nonmagnetic portion of the pump's magnetic circuit. Thus, electromagnetic pumps are inherently large air gap devices. Since the m.m.f. required for the nonmagnetic gap is much greater than that necessary for the rest of the magnetic circuit, there is little advantage in using high permeability magnetic materials.

The Curie temperature of a magnetic material is the first criterion for selection. The nickel-iron alloys appear to have a sufficiently high Curie temperature for the operating range in this application and would cause an inherent pump shutoff above 1200°F which would be advantageous in RVACS operation without scram scenarios. The final choice of magnetic materials will be based on a future study but their appears to be many state-of-the-art materials to make a selection from.

Electrical Conductors and Insulation - Since conductors are rarely used alone, it is difficult to treat conductors and the insulation separately. The service conditions, the manufacturing processes employed, and the overall electrical insulation must be considered.

Selecting electrical conductor materials for EM pumps requires a knowledge of the electrical resistivity and strength characteristics of basic conductor materials. Where needed, consideration must then be given to modifying these characteristics using metallic claddings, coatings, or metallurgical variations. For the polyphase pumps, the conductor strength should be sufficient for unsupported leads and connections. The thermal conductivity of the conductor is important to the transfer of heat within the windings.

The dispersion hardened coppers exhibit strengths at elevated temperatures and apparently have a lesser tendency to grain growth on extended exposure to high temperature. In the EM pump application, in which protective atmospheres minimize or eliminate the oxidation problem and assist in cooling the electrical conductors, beryllia or zirconium dispersion hardened copper and, to a lesser extent, clad copper appear to be the best choices. The electrical conductivities of conductors of these types are not seriously reduced from that of pure copper.

Electrical conductor insulations for the EM pump must operate at 1000°F (hot spot) in an inert atmosphere and provide adequate insulation at a maximum of 20 volts turn to turn after shaping the coil. Candidate insulation materials have been screened by testing small samples and narrowed to two candidates which are presently being tested at 1380°F in air with 1500 volts applied. The candidates are:

Mono-Aluminum Phosphate with White Mica Tape
Mono-Aluminum Phosphate with Amber Mica Tape

Test specimens insulated with Mono-Aluminum Phosphate & Amber Mica Tape are also being aged at 1345°F and energized with 1500 volts to establish a second point on an Arrhenius curve in an attempt to determine the probable life of the insulation at the service conditions at or below 1000°F.

Full size pump coils are being made to develop the process for applying the insulation system and these coils will be qualification tested at accelerated aging test first as individual coils and later as a part of a stator segment to get the influence of mechanical interaction with the stator laminations.

Successful testing of high temperature insulation is most important aspect of the EM pump in this application.

5.4.4 Compliance With Codes and Standards

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. 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 temperature, Code Case N-47 will apply.

The primary system ASME Class 1 or Seismic Category I components shall be designed to withstand the concurrent loadings associated with the Service Level B conditions and the vibratory motion of 50 percent of the Safe Shutdown Earthquake (SSE) or in other words, the Operating Basis Earthquake (OBE). The design limits for this case are specified in NB-3223 and NB-3654 of the ASME Code for vessels and piping. 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 condition.

For components at elevated temperatures, the design stress limits specified by Paragraph 3223, "Levels A and B Service Limits," and by Paragraph 3225 of Code Case N-47 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 Appendix T of the Code Case.

References - Section 5.4

5.4-1 Specification 23A3071, Revision 0, "PRISM Design Requirements," March 1986.

5.4-2 Hunsbedt, A. (Editor), "Thermal Hydraulic Analysis," General Electric Company Report GEFR-00782, August 1986.

TABLE 5.4-1

COMPONENT CODES AND STANDARDS

PHTS Component	Code
IHX	ASME Section III, Class 1, NB 3200 & Code Case N47
EM Pump Primary Boundaries	ASME Section III, Class 1, Code Case N47

TABLE 5.4-2

PRISM IHX: CODE SIZING

Component	Size	Remarks
Shells and Cylinders		
o Outer Shell	0.75 in.	
o Shell Stiffener	0.375 in.	
o Shell Lower Toroidal Head	0.375 in.	
o Intermediate Lower Channel Head	1.0 in.	
o Intermediate Upper Channel Head	1.0 in.	Sized for Internal SWR Pressure
o Downcomer	0.375 in.	Sized for Internal SWR Pressure
o Thermal Sleeve	0.625 in.	Sized for External SWR Pressure
o Riser	1.0 in.	Sized to Support IHX Weight and Seismic Loads
Tube Bundle		
o Upper Tubesheet	6.0 in.	30% Ligament Efficiency
o Lower Tubesheet	5.0 in.	30% Ligament Efficiency
o Tubes (2139 per IHX)	0.625 O.D.	Sized for Internal SWR Pressure
	0.035 Min.Th'k	(1) Buckling Under Axial Compression
		(2) Flow Induced Vibration
o Support Plates (6)	0.75 in.	

Based on:

(1) Internal Pressure (150 psi)

(2) Axial Compression (1.125 in.)

(3) Bellows Will Be Cold Worked

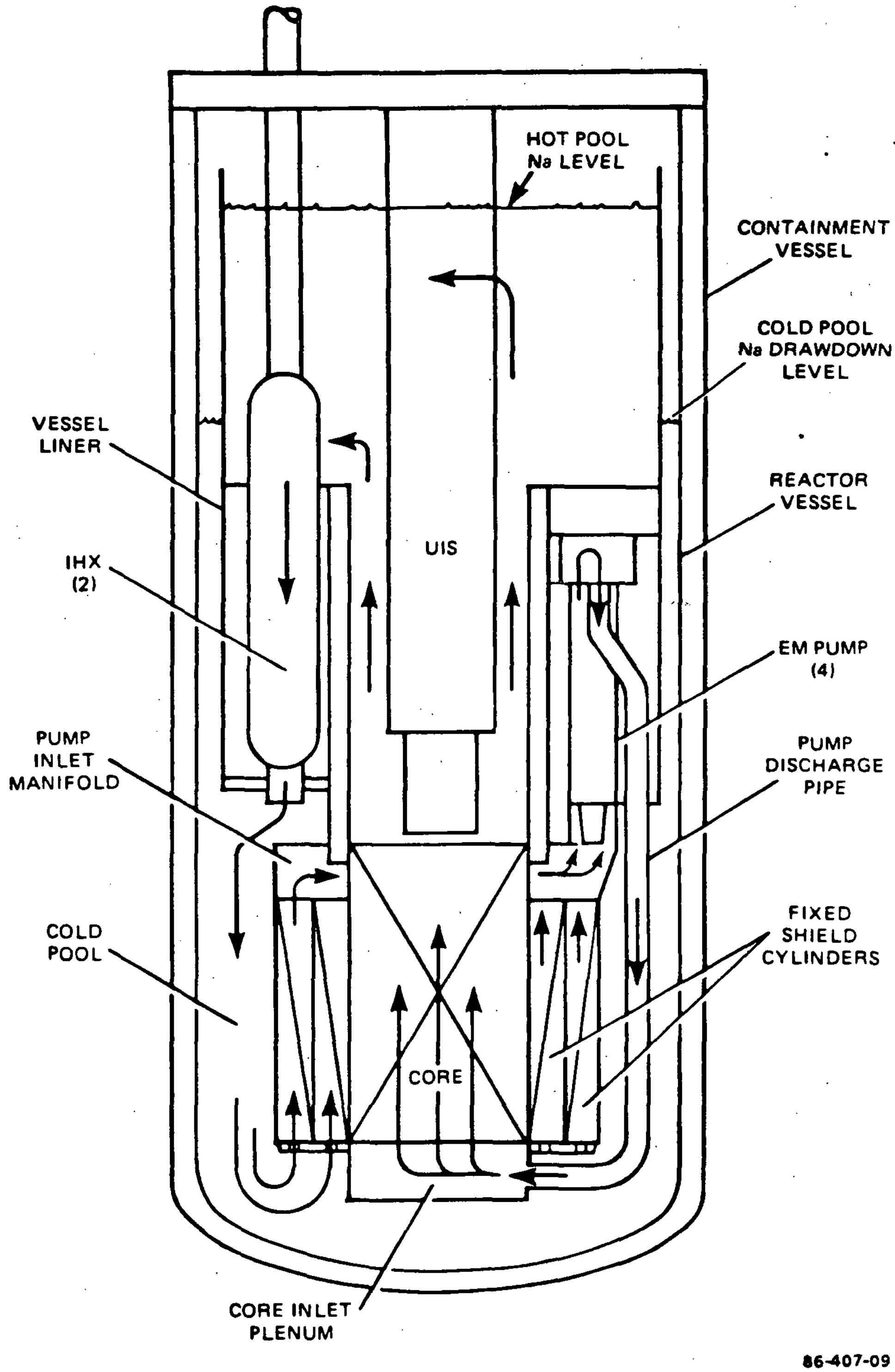


Figure 5.4–1 NORMAL SODIUM FLOW PATH

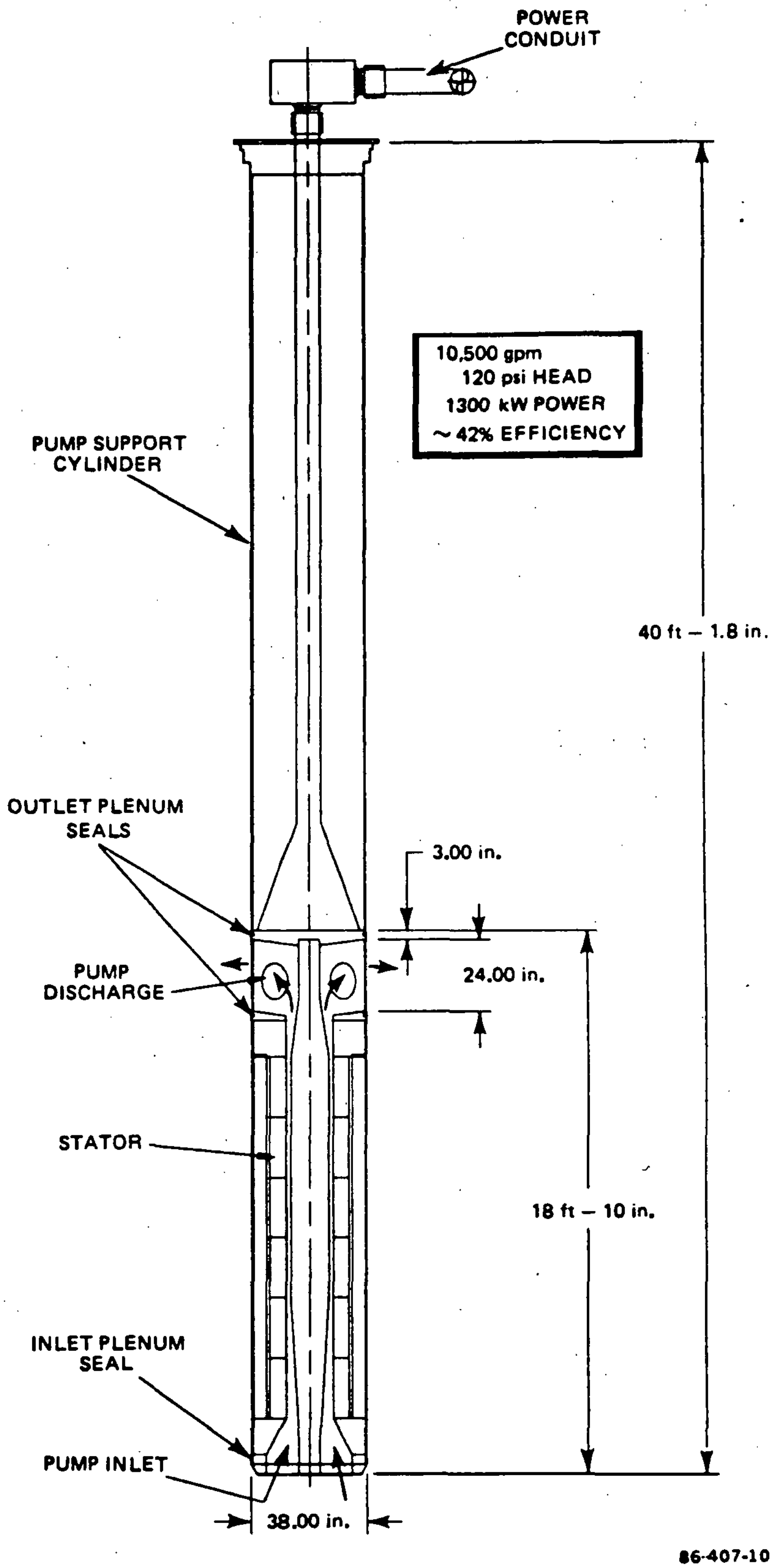


Figure 5.4–2 PRISM PRIMARY EM PUMP ASSEMBLY

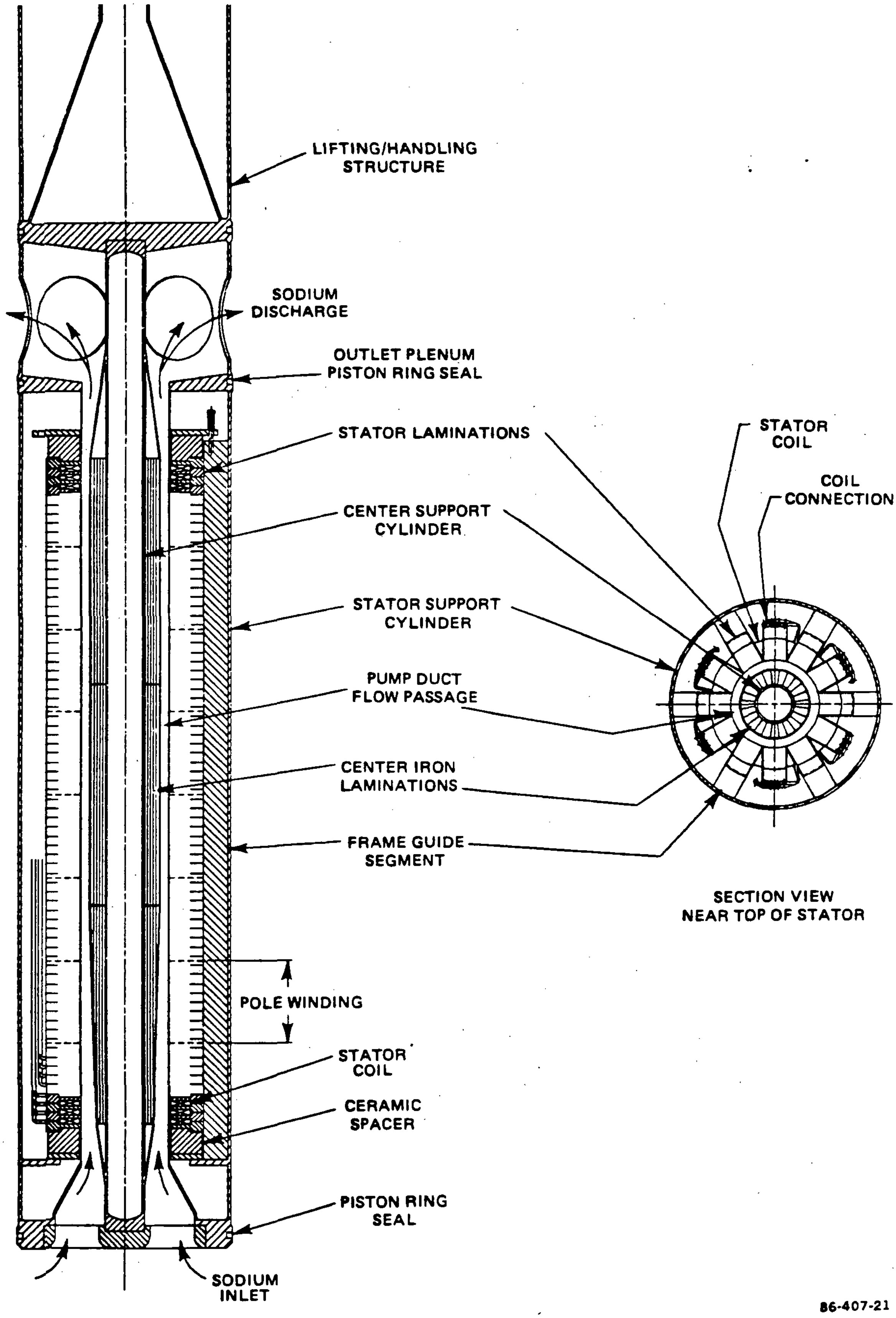


Figure 5.4–3 PRISM PRIMARY SODIUM EM PUMP

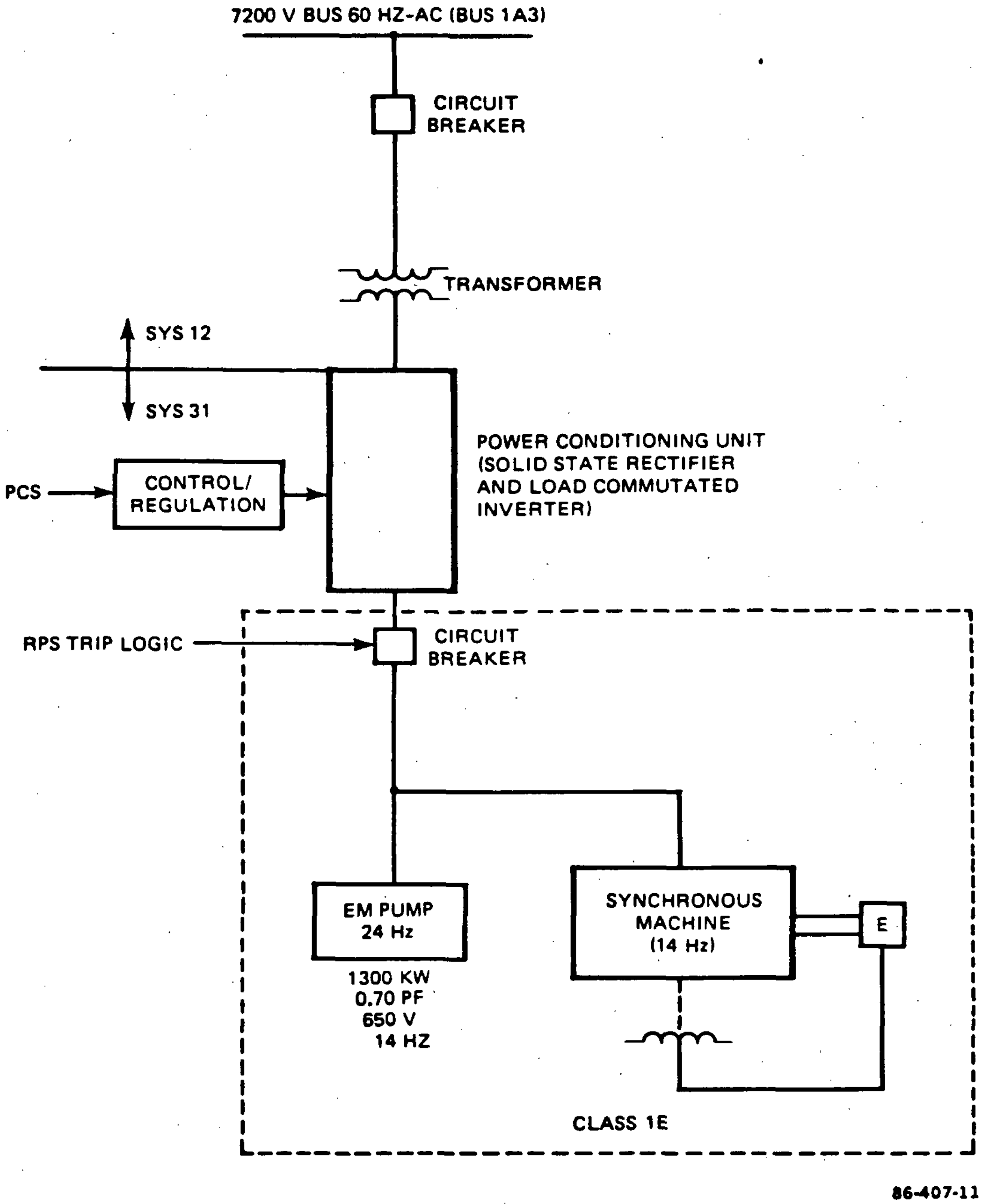


Figure 5.4—4 EM PUMP POWER SUPPLY

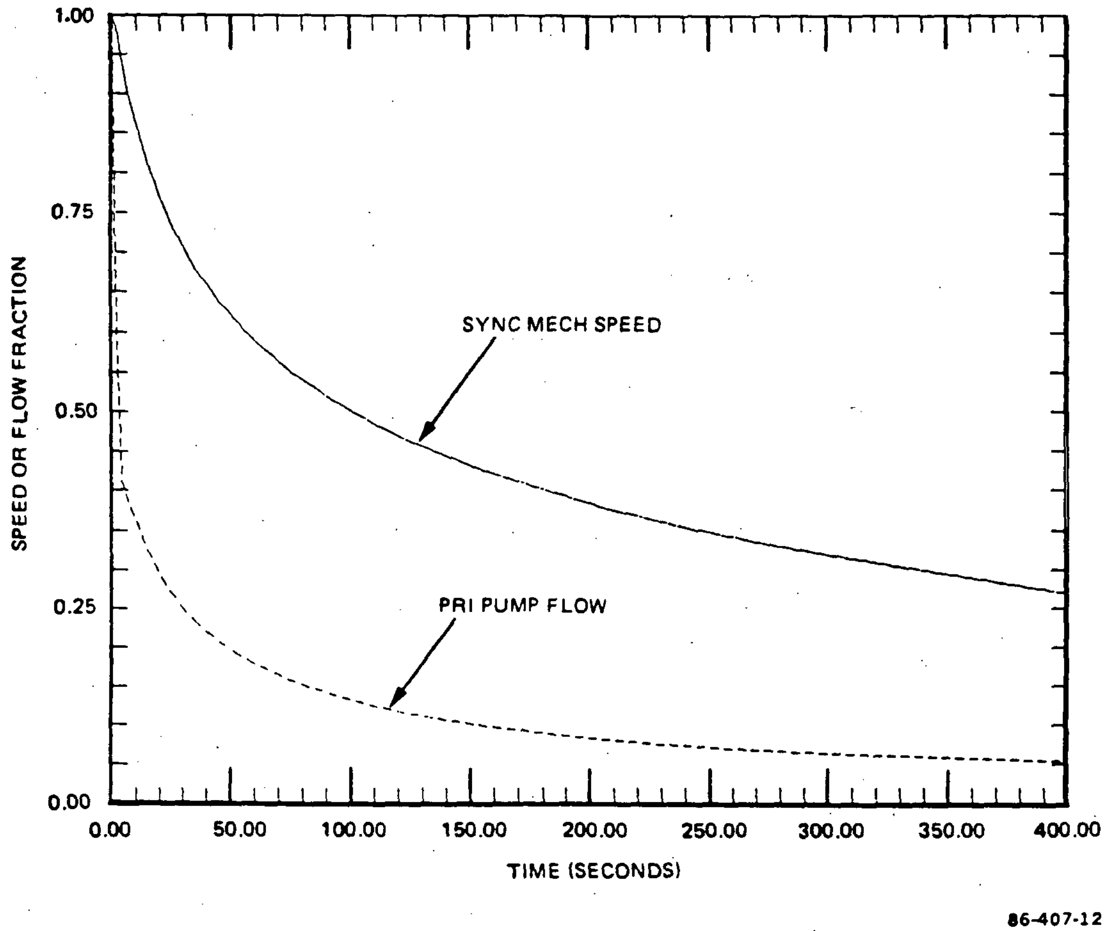


Figure 5.4–5 GENERATOR SPEED FRACTION AND PUMP FLOW FRACTION

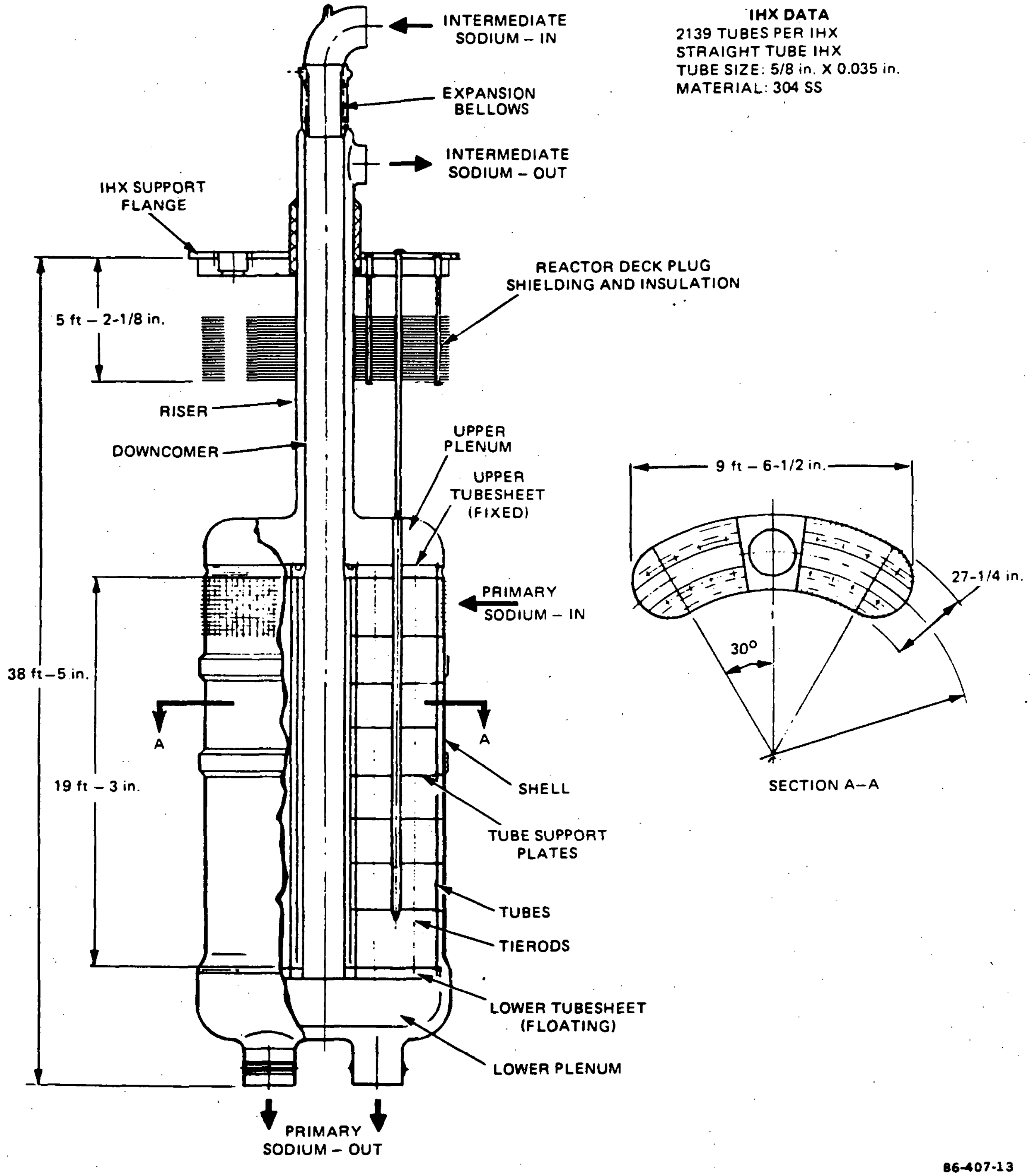


Figure 5.4–6 PRISM INTERMEDIATE HEAT EXCHANGER (IHX)

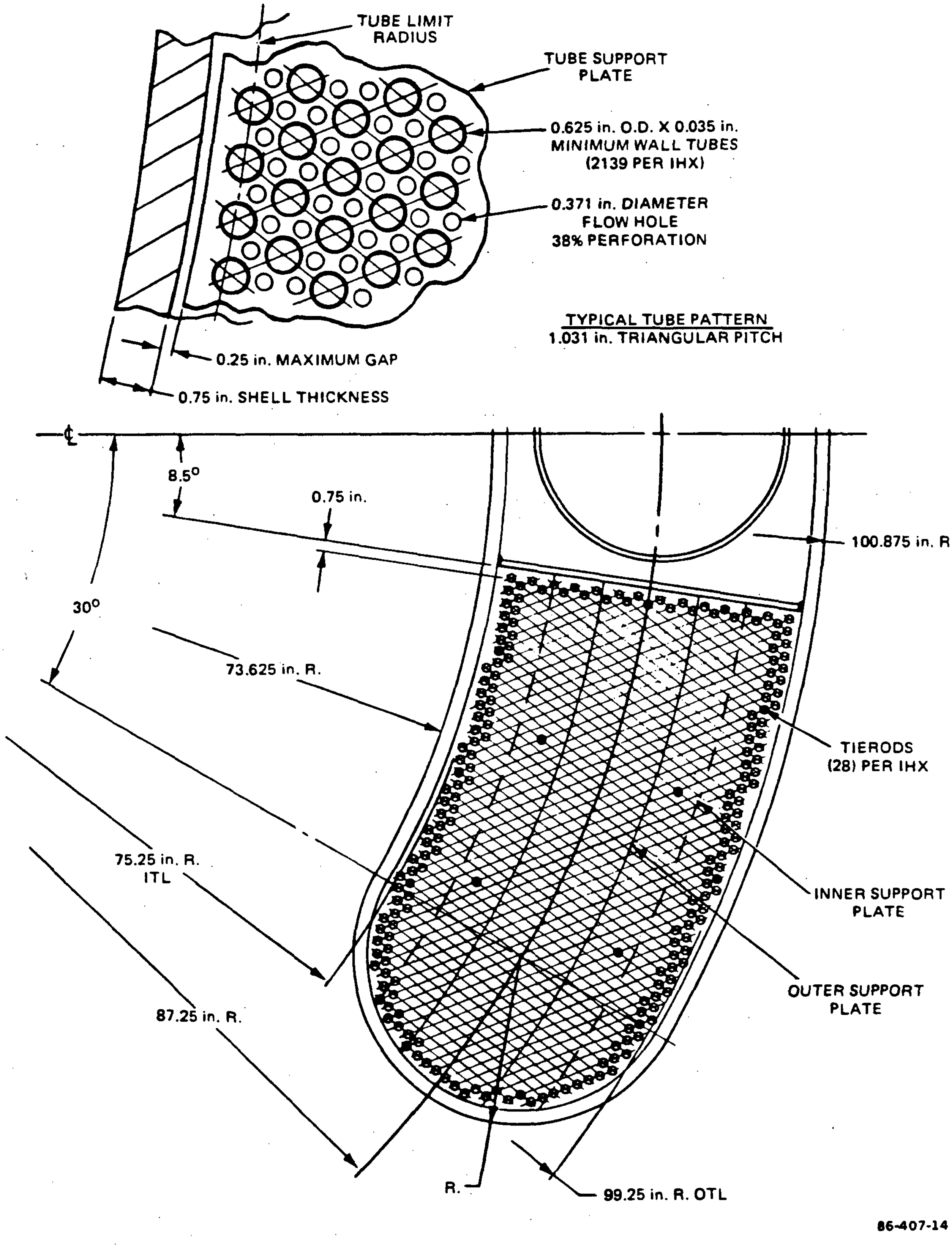


Figure 5.4–7 IHX TUBE BUNDLE DETAILS

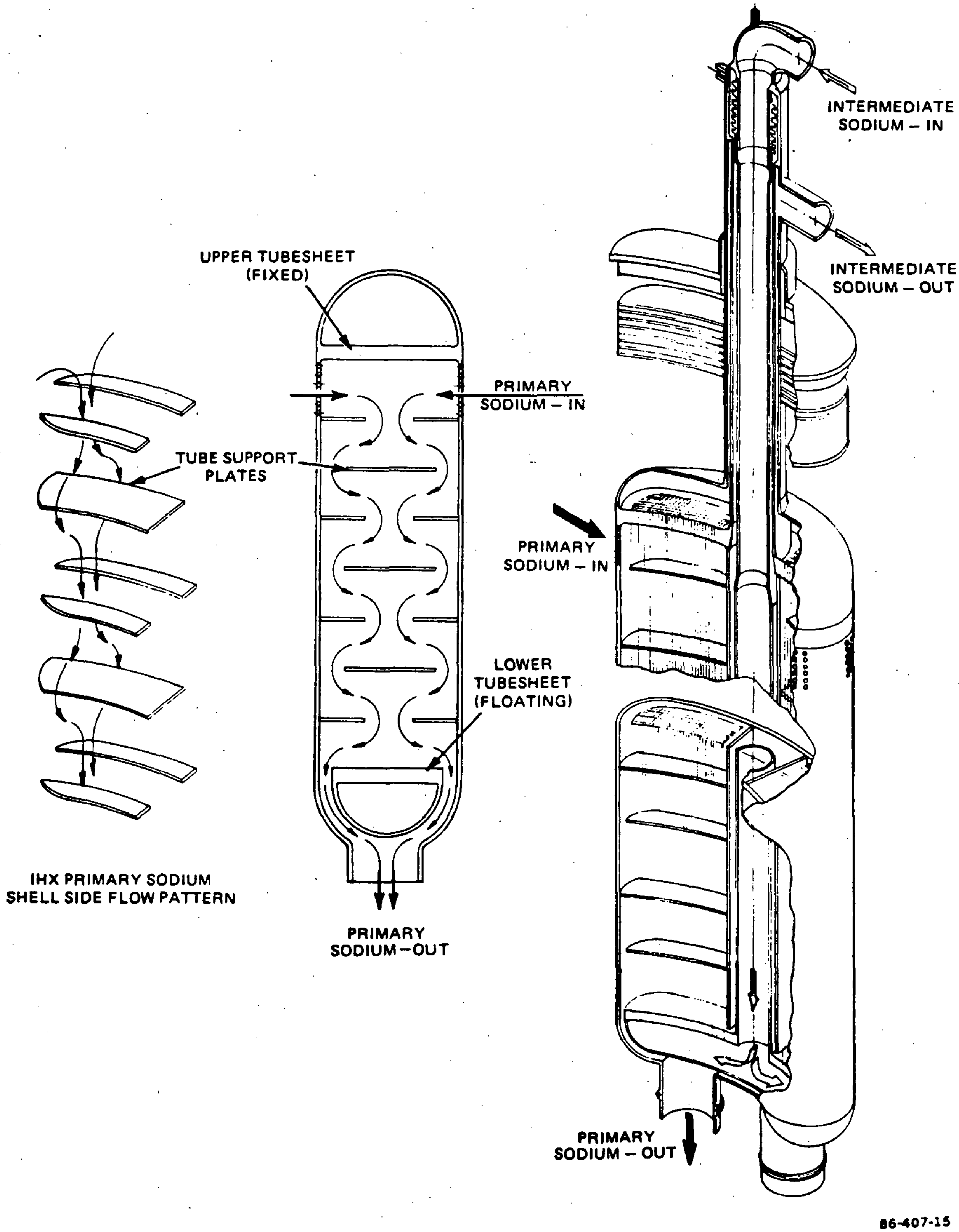


Figure 5.4–8 IHX ISOMETRIC

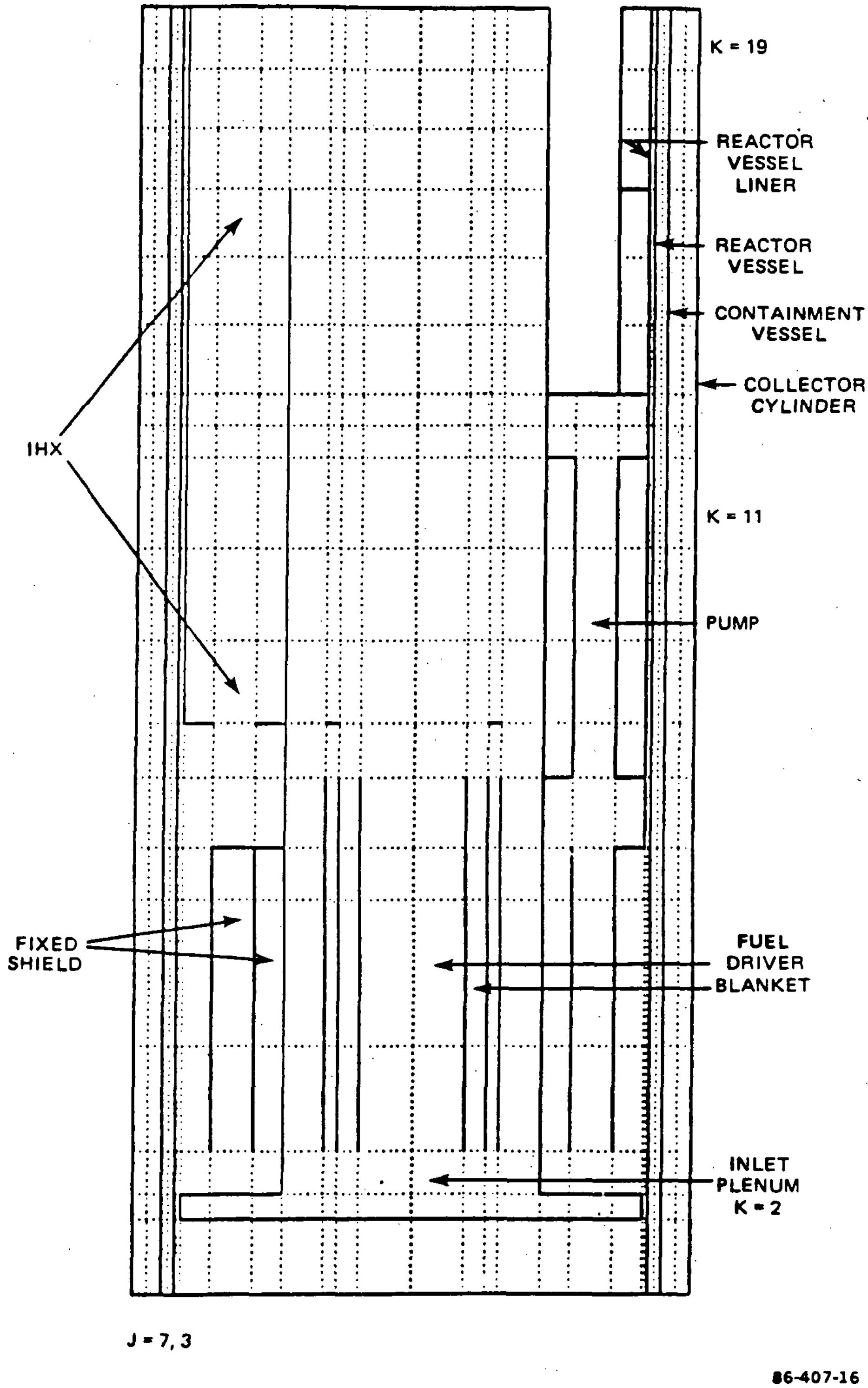


Figure 5.4–9 ELEVATION VIEW OF COMMIX MODEL

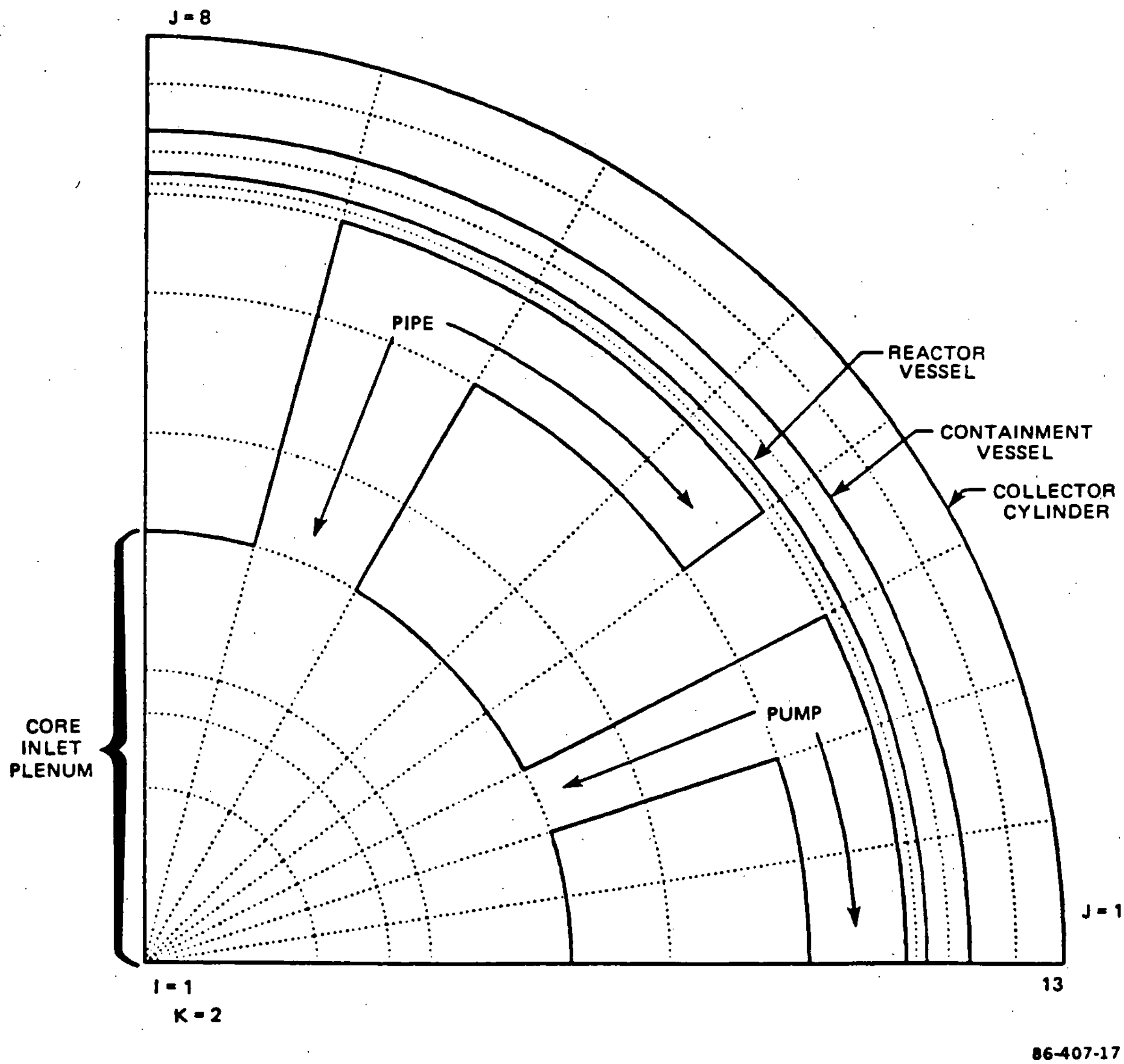


Figure 5.4–10 PLAN VIEW OF COMMIX MODEL AT INLET PLENUM ELEVATION

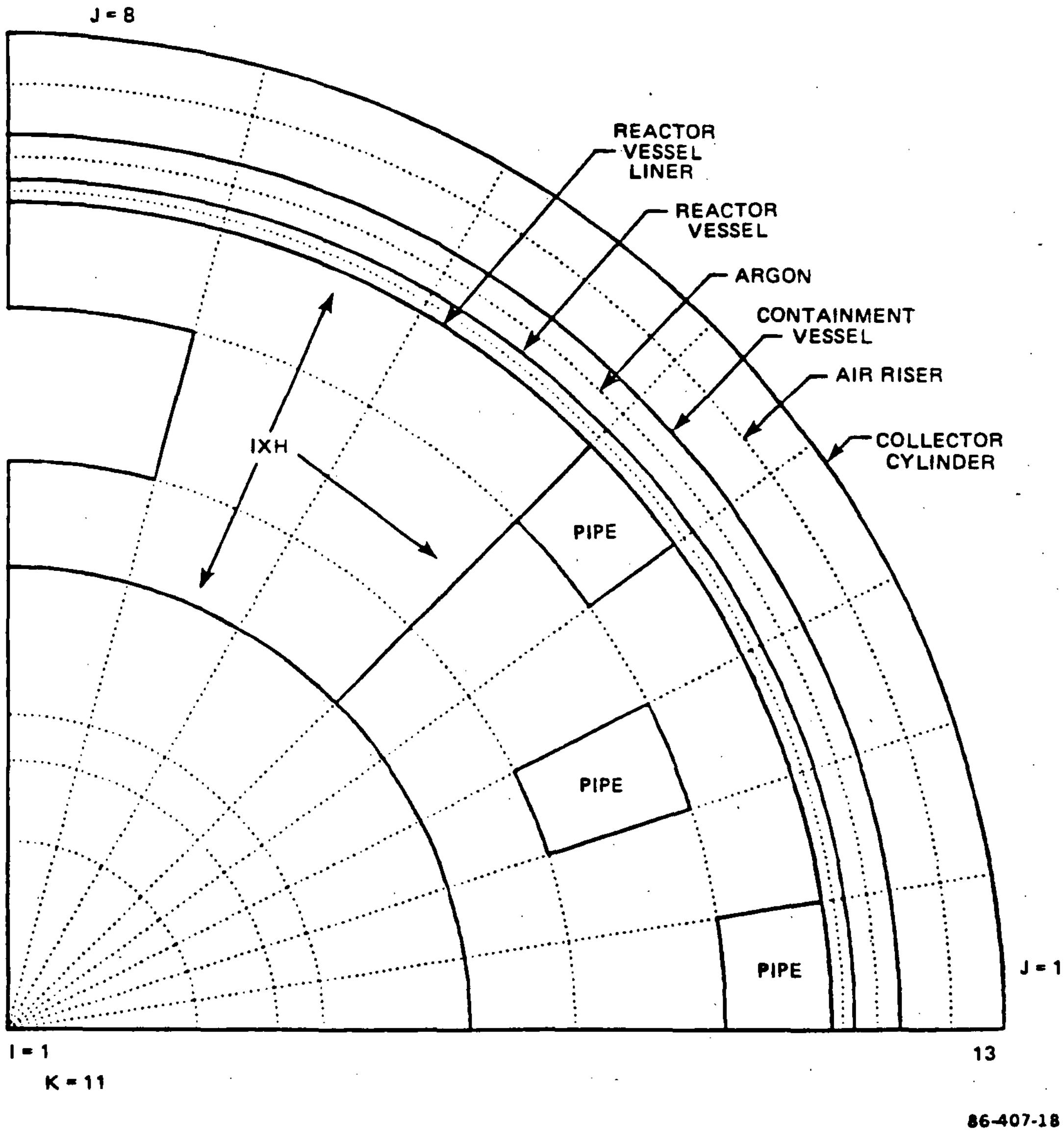


Figure 5.4–11 PLAN VIEW OF COMMIX MODEL JUST BELOW PUMP OUTLET PLENUM

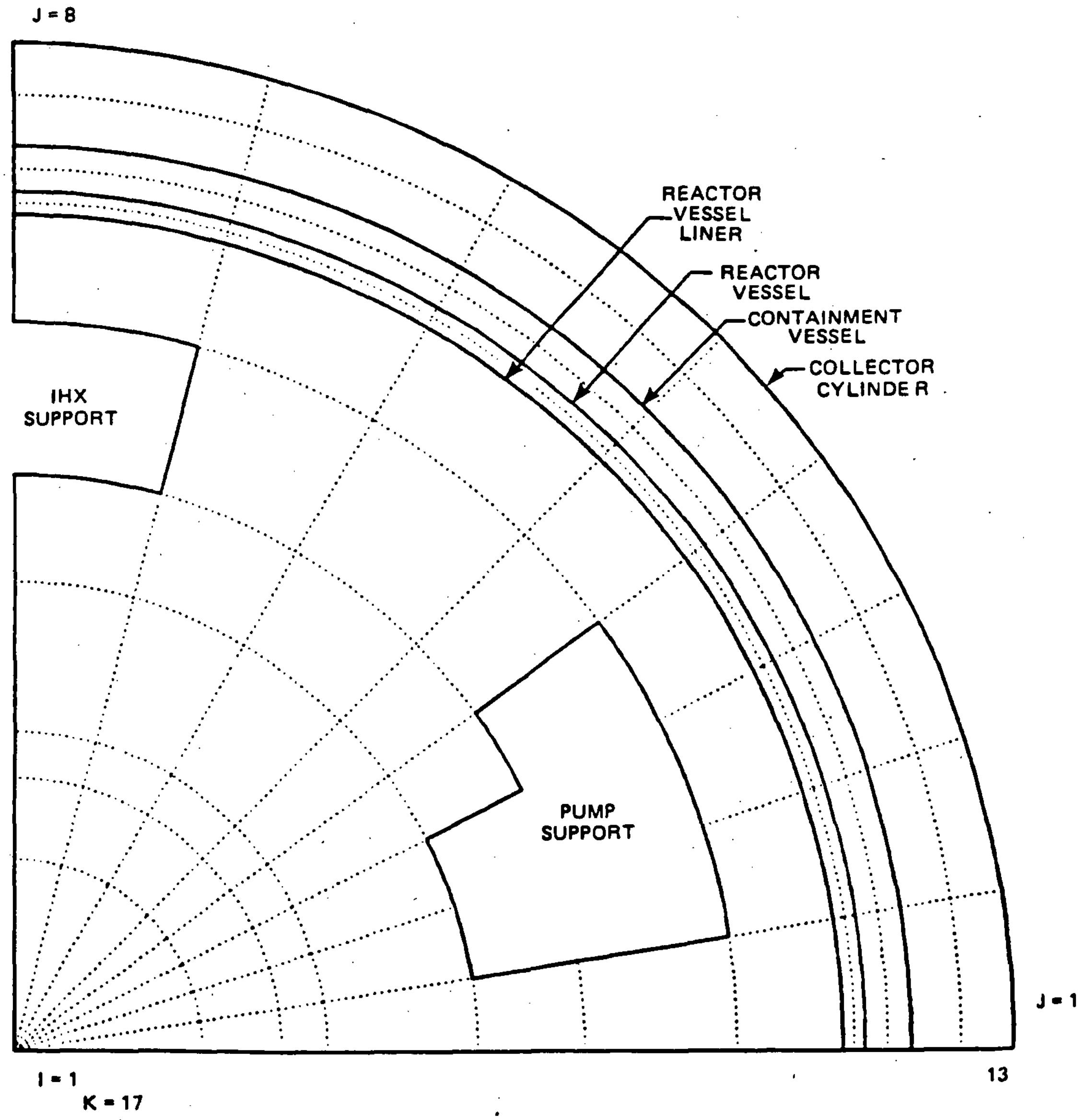
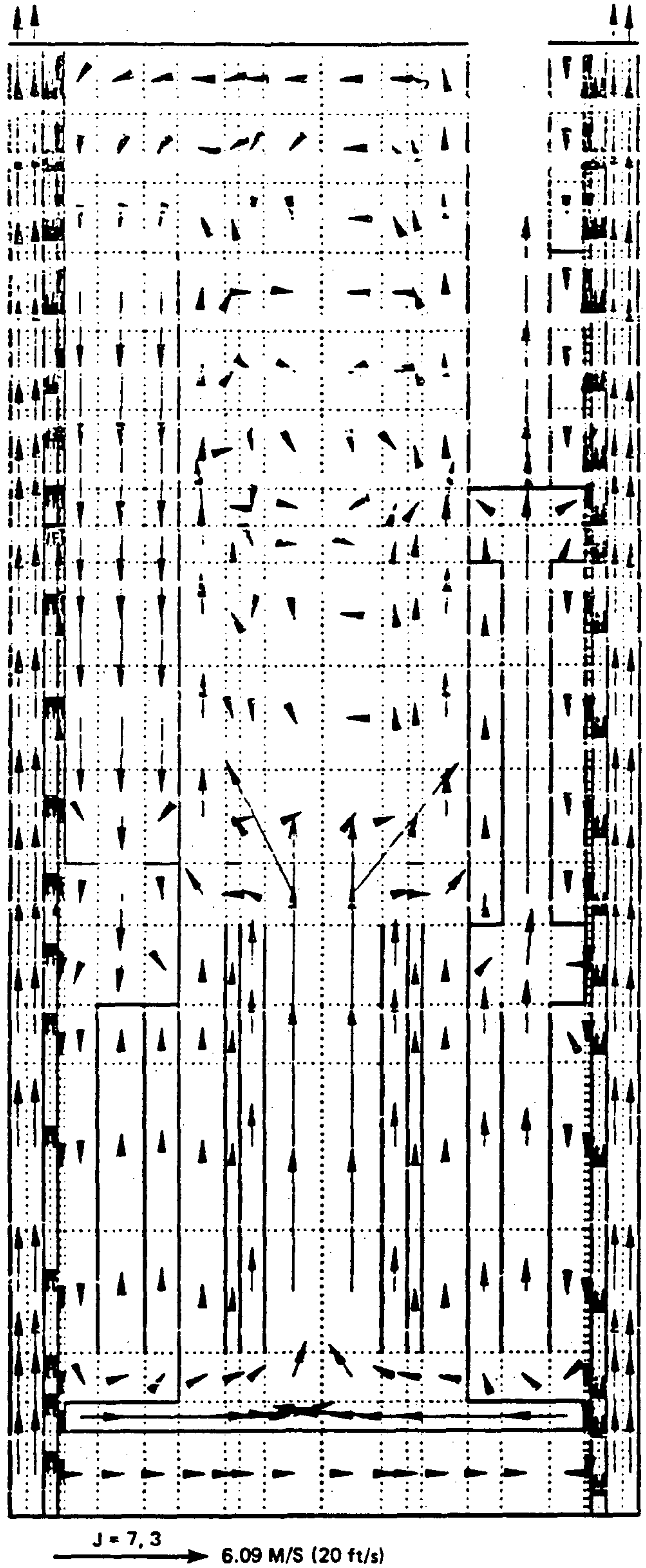


Figure 5.4–12 PLAN VIEW OF COMMIX MODEL JUST ABOVE IHX INLET



86-407-22

Figure 5.4–13 AXIAL VELOCITY DISTRIBUTION CALCULATED USING COMMIX AT J = 3 AND 7 FOR STEADY STATE

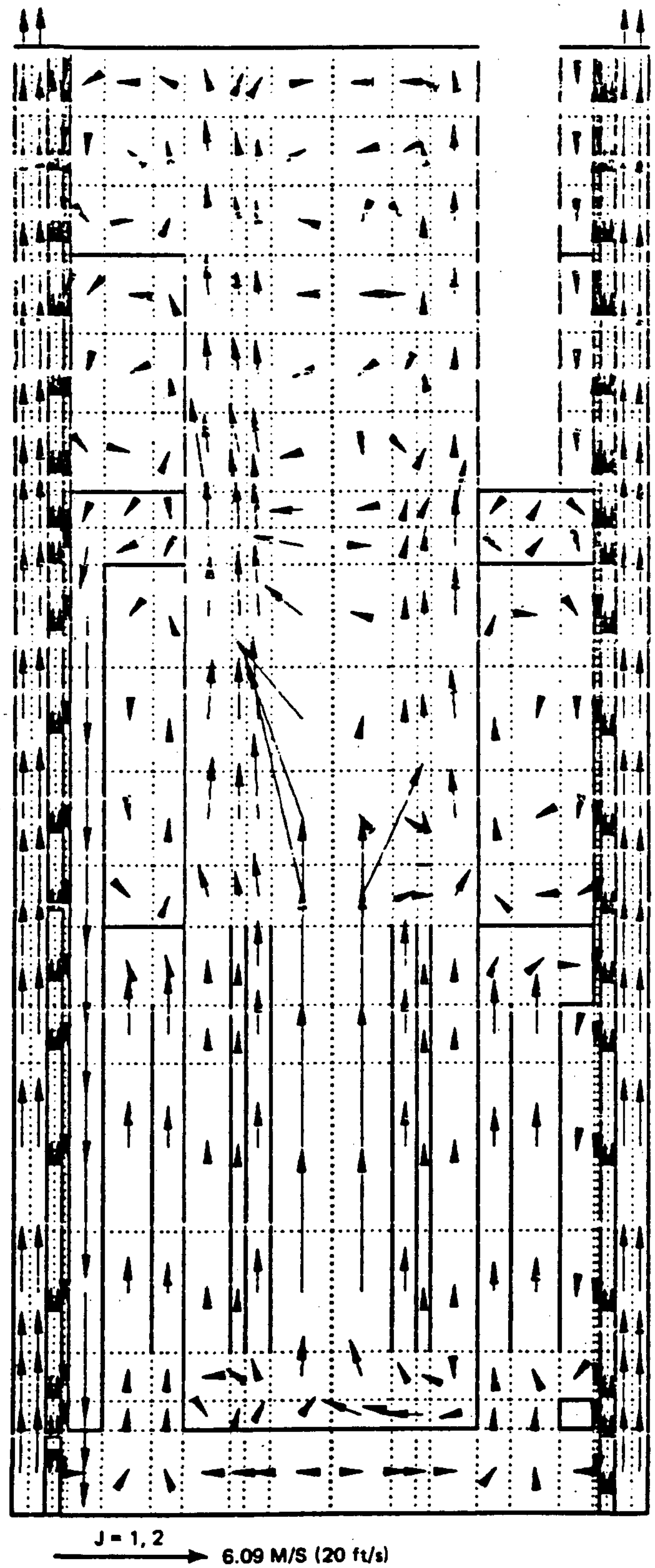


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Figure 5.4–14 AXIAL VELOCITY DISTRIBUTION CALCULATED USING COMMIX AT J = 1 AND 2 FOR STEADY STATE

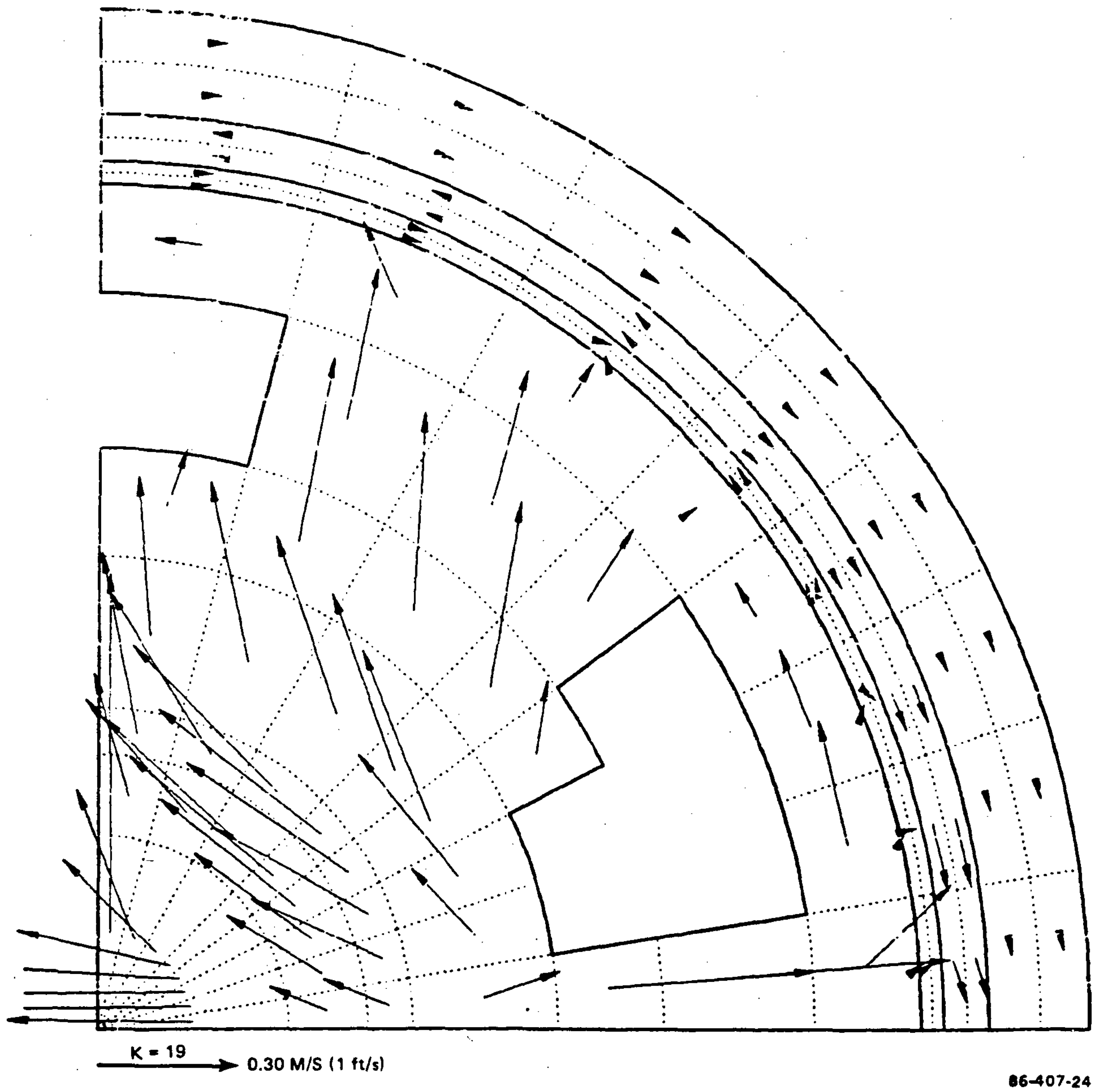


Figure 5.4–15 LATERAL VELOCITY DISTRIBUTION CALCULATED BY COMMIX AT TOP OF HOT POOL FOR STEADY STATE

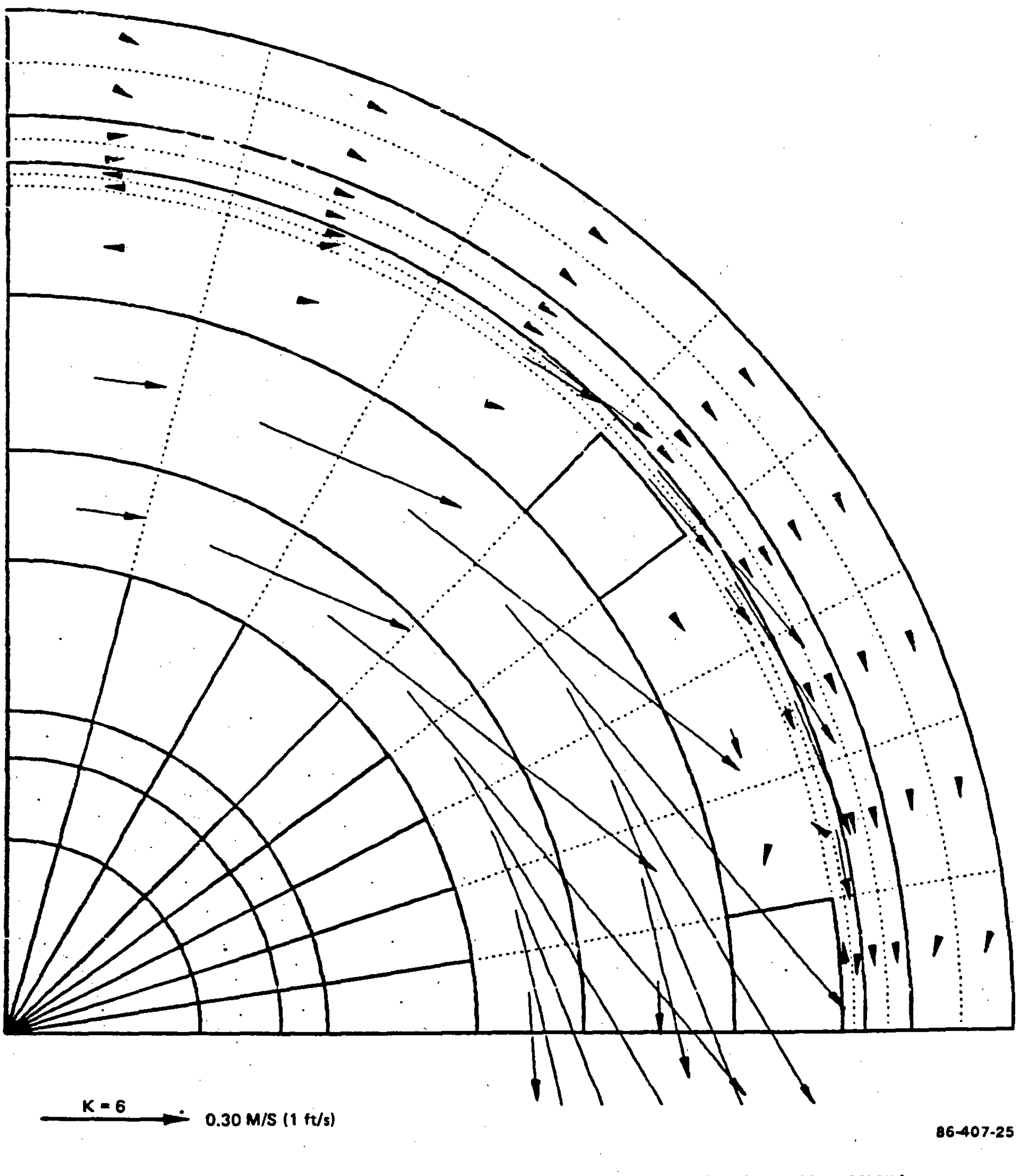


Figure 5.4–16 LATERAL VELOCITY DISTRIBUTION CALCULATED BY COMMIX AT TOP OF FIXED SHIELDS FOR STEADY STATE

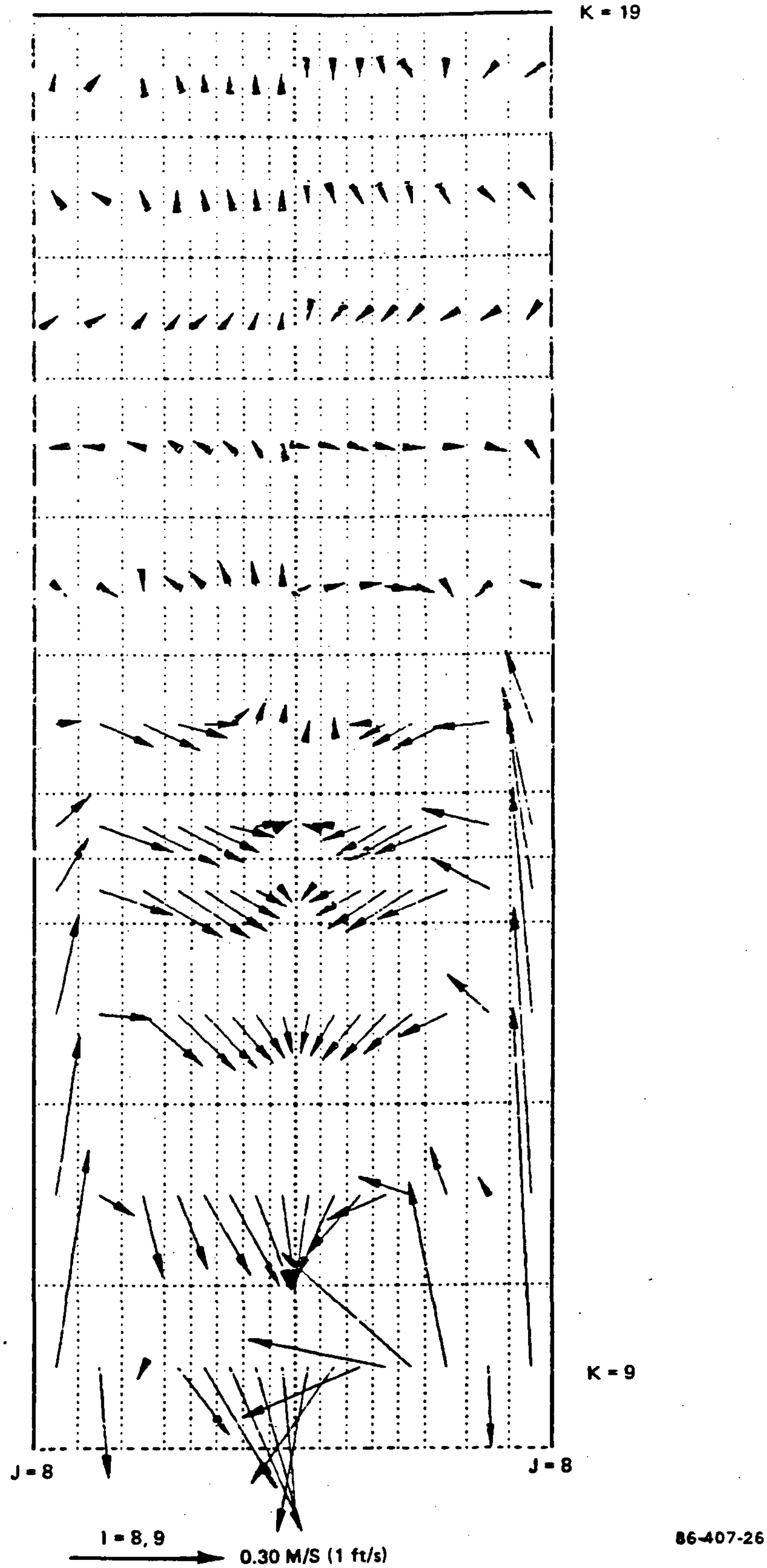
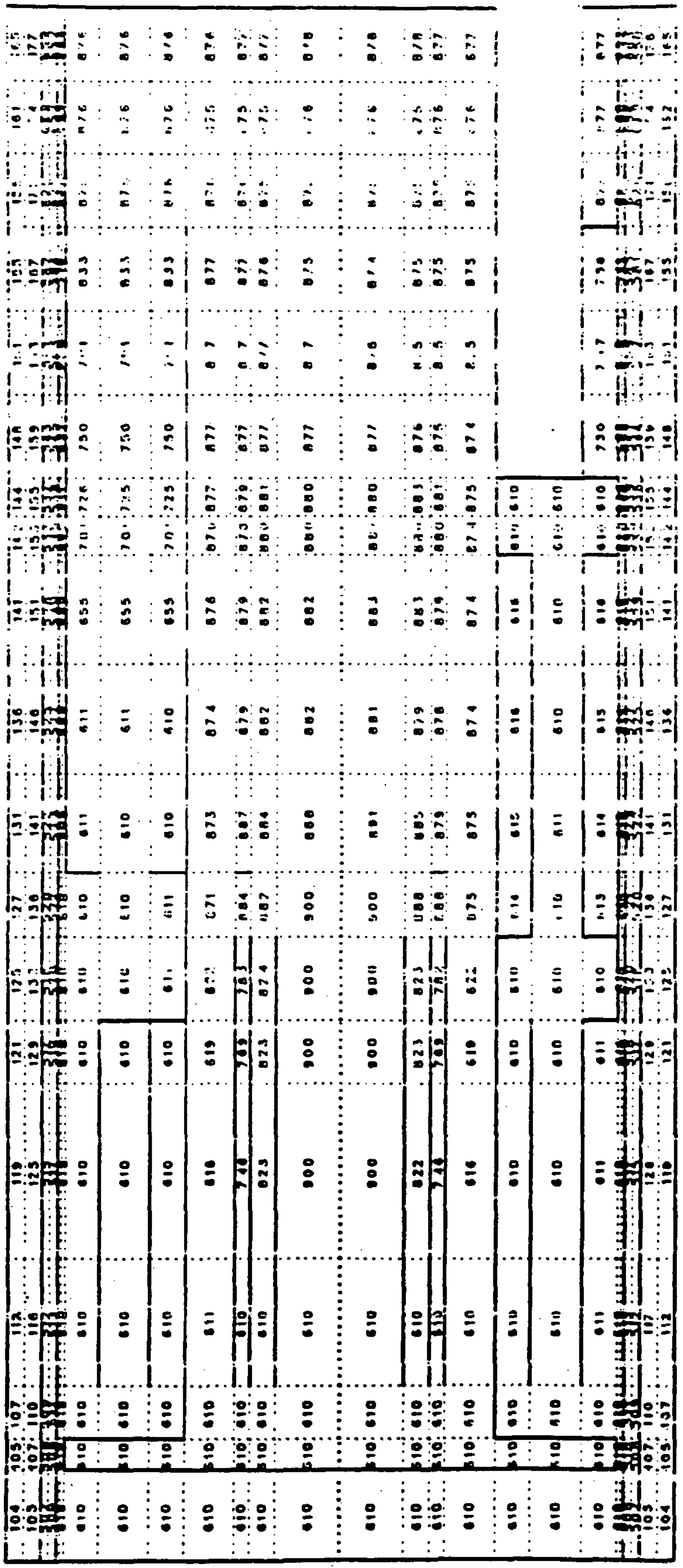


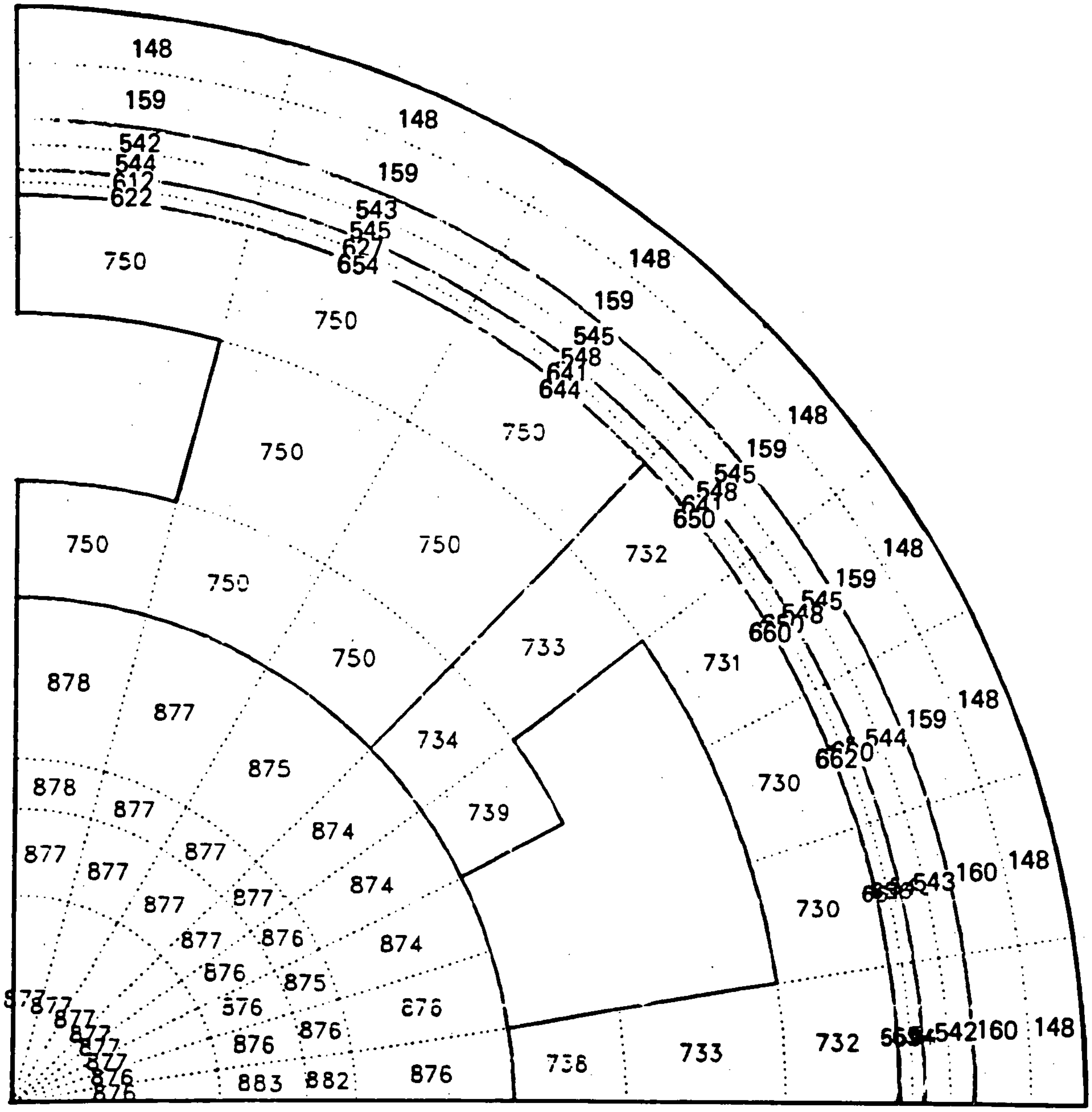
Figure 5.4–17 VELOCITY DISTRIBUTION CALCULATED BY COMMIX IN RVL/RV GAP AT STEADY STATE



J = 7, 3

Figure 5.4–18 ELEVATION VIEW OF TEMPERATURE DISTRIBUTION (°F) CALCULATED BY COMMIX FOR STEADY STATE

86-407-27



K = 14

86-472-20

Figure 5.4–19 TEMPERATURE DISTRIBUTION (°F) CALCULATED BY COMMIX AT TOP OF FUEL STORAGE RACK FOR STEADY STATE

J = 8

I = 8,9

610	610	610	610	610	622	670	72[illegible]	838	864	8[illegible]6
609	608	609	612	617	634	675	719	842	861	8[illegible]6
610	609	612	620	629	644	652	713	847	859	8[illegible]6
610	611	615	624	634	650	63[illegible]	714	844	861	8[illegible]5
611	613	617	628	637	656	680	714	844	861	8[illegible]6
612	614	619	629	636	662	678	711	846	860	8[illegible]6
613	616	620	628	634	661	677	710	847	860	8[illegible]6
615	618	620	625	630	659	677	711	847	860	8[illegible]6
614	617	620	625	631	654	674	712	845	858	864
612	615	620	628	636	656	674	712	845	858	8[illegible]4
610	613	619	630	640	656	675	713	845	859	8[illegible]4
610	611	616	627	637	650	678	714	844	859	864
610	610	613	622	632	647	681	714	844	859	8[illegible]3
610	609	610	616	624	641	680	714	845	859	8[illegible]4
609	608	608	609	611	627	674	719	842	860	8[illegible]4
610	610	609	609	609	612	662	721	839	861	8[illegible]4

J = 8

K = 9

K = 19

Figure 5.4–20 TEMPERATURE DISTRIBUTION (°F) CALCULATED BY COMMIX IN RVL/RV GAP FOR STEADY STATE

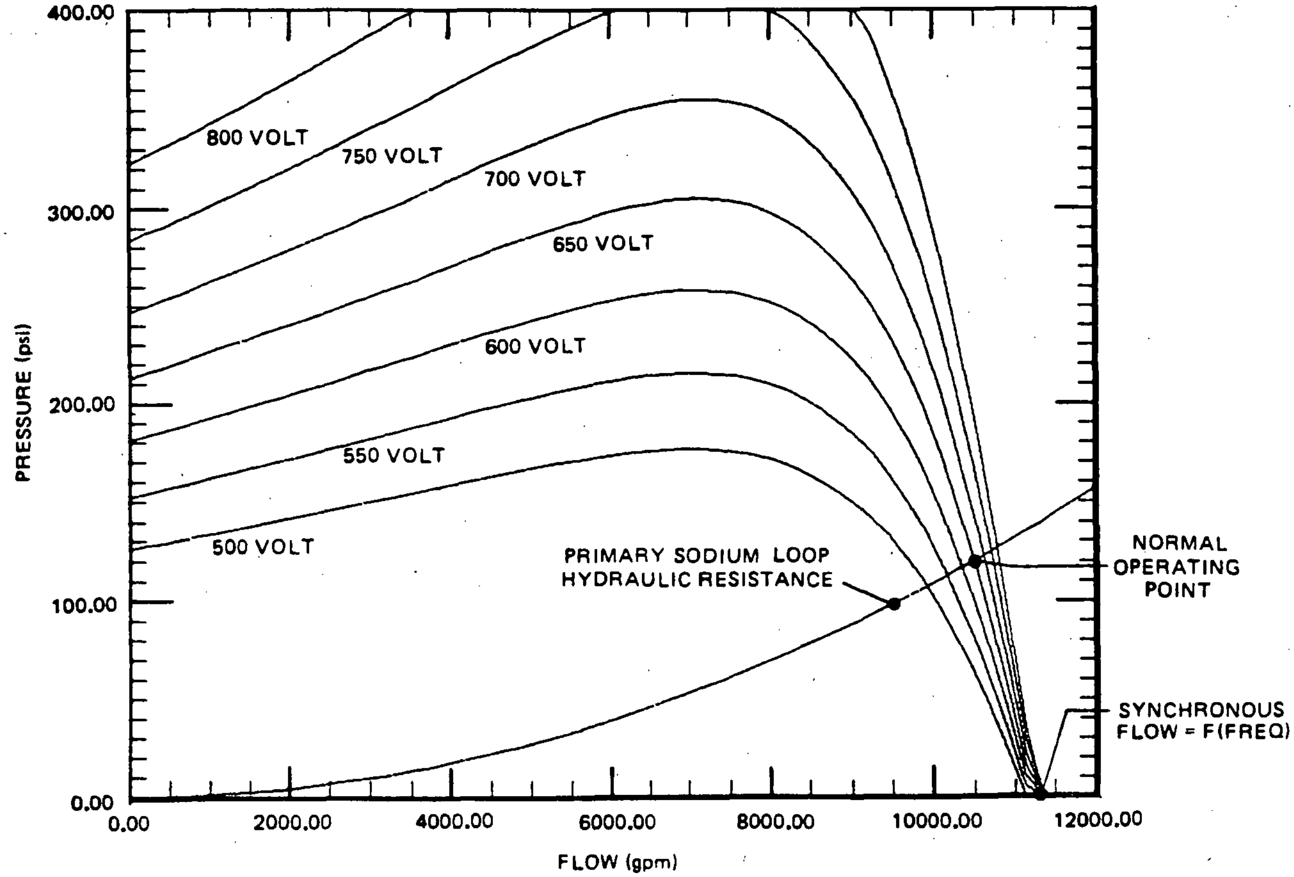


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Figure 5.4–21 PRIMARY EM PUMP PERFORMANCE CURVES

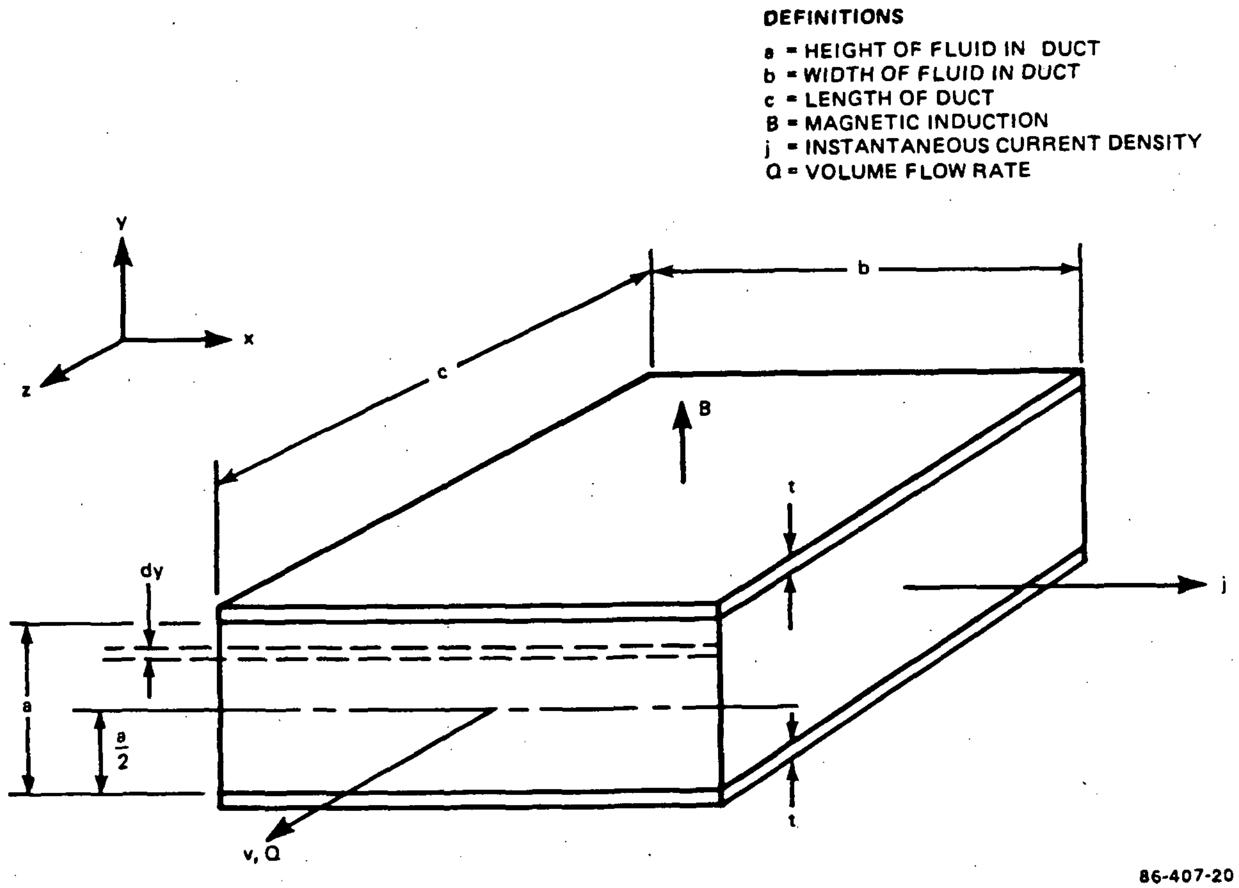


Figure 5.4–22 PUMP DUCT ELEMENT

5.5 Intermediate Heat Transport System

The intermediate heat transport system (IHTS) transfers reactor-generated heat from the primary heat transport system (PHTS) to the steam generator system. The IHTS performs this function during normal power operation, during shutdown (decay heat removal), and under upset conditions. The system in the standard plant is comprised of nine identical heat transport loops operating in parallel; one for each reactor module. Each IHTS loop is thermally coupled to a PHTS by an intermediate heat exchanger (IHX) and to the steam generator system by a steam generator. Non-radioactive sodium is circulated by the IHTS pump located in the cold leg and transports heat from the IHX to the steam generator. The IHTS extends from the IHX tube side (IHTS) outlet nozzle to the steam generator shell side inlet nozzle and from the steam generator shell side outlet nozzle to the IHX tube side (IHTS) inlet nozzle. Each loop contains a sodium pump and sodium expansion tank. The expansion tank allows the IHTS to function as a closed loop system without need for sodium makeup or removal to accommodate thermal expansion of the sodium coolant.

The IHTS also includes the sodium drain piping, the sodium/water reaction pressure relief subsystem (SWRPRS), and vent piping from the steam generator. The SWRPRS provides pressure relief and gas venting capability to mitigate the effects of a sodium/water reaction if a tube failure should occur in the steam generator. The IHTS piping is protected from missiles within the reactor head access area and a protective barrier is provided around the pipes in this area to mitigate the effects of a sodium leak. The IHTS piping and components in the steam generator building (SGB) are located in separate, unshielded, accessible cells. A schematic diagram of the IHTS is shown in Figure 5.5-1.

5.5.1 Design Basis

The IHTS transfers reactor generated heat from the intermediate heat exchanger (IHX) of the PHTS to the steam generator system under all normal and upset operating conditions. Specific functions include:

1. The IHTS shall transport reactor generated heat from the primary heat transport system to the steam generator system while providing an adequate flow rate for maintaining reactor temperature conditions within limits which prevent damage to the reactor vessel, fuel, and reactor internals for all normal and upset events.

2. The IHTS shall provide constant sodium flow over the operating power range of 0 to 100 percent reactor thermal power.

3. The IHTS shall transfer decay heat from the PHTS to the steam generator system under all normal and upset conditions.

4. The IHTS shall provide a high integrity containment of sodium coolant.

5. Each IHTS loop shall be capable of plant loading and unloading in accordance with the plant duty cycle given in Appendix D.

6. The IHTS shall provide capability for drain of the sodium coolant (not including IHX inventory).

7. The IHTS shall prevent the pressure generated by a sodium-water reaction from reaching a value which would cause damage to the IHX.

8. There shall be no isolation valves in the IHTS main loop piping.

9. The IHTS shall control the solid, liquid and gaseous products of a large sodium-water reaction so that the solids and liquids are contained in an appropriate tank and the gas (hydrogen) is released to the atmosphere in a safe manner and burned.

5.5.1.1 Process Requirements

1. Each of the IHTS loops shall be thermal hydraulically designed to remove 430 MWt from the intermediate heat exchangers (IHX) with IHTS temperatures of 540°F at the IHX inlet, 800°F at the IHX outlet, 800°F at the steam generator inlet, and 538°F at the steam generator outlet, with each loop flow rate of 18 x 10^6 lb/hr. There shall be one IHTS loop per reactor module (i.e.; one pump, one steam generator, one expansion tank).

2. The IHTS shall have the sodium pump located in the cold leg of the loop.

3. The IHTS shall support operation in a hot standby condition. Sodium flow shall be maintained at a nominal rate of 10% of full flow during pony motor operation. The IHTS cold leg temperature shall be a nominal 550°F and the IHTS hot leg temperature shall be consistent with flow and decay power requirements.

4. The IHTS shall have the capability to transport reactor-generated decay heat from the PHTS to the steam generator system by pony motor or natural circulation operation when the SGS is available and by pony motor to the SGACS when the steam generator is not available.

5. The IHTS shall have the capability to heat up dry gas filled or sodium vapor-containing piping and components from ambient (70°F) to 450°F, outer surface temperature, within 120 hours maximum. The maximum heat-up rate shall be dictated by stress considerations. Heating shall be accomplished by constant heat input rates over the heat-up cycle unless stress considerations require changes in the heat input rate during the cycle.

6. The IHTS shall have the capability for heat-up from 400°F to 750°F at 50°F/hr maximum.

7. The IHTS shall be capable of maintaining the sodium coolant at a minimum of 375°F during normal plant operations, including refueling. The nominal refueling temperature is 400°F.

8. The IHTS piping normally not filled with sodium during plant operation shall have the capability for heat-up from ambient (70°F) to 750°F, outer surface temperature, using heaters. The heaters shall be capable of maintaining the IHTS piping internal temperature at 650°F during all normal and off-normal plant operation.

9. The IHTS piping and components in the steam generator building shall be insulated to limit the surface temperature of the insulation to 140°F at rated power operation or hot functional testing with an ambient building temperature of 100°F.

10. The IHTS piping and components within the head access area shall be insulated to limit the surface temperature of the insulation to 140°F at rated power operation or hot functional testing with 750°F sodium in the system and an ambient temperature of 100°F.

11. The IHTS thermal design shall ensure that the concrete to equipment support interface will not exceed 150°F.

12. The IHTS shall have the capability of draining the IHTS sodium (except the IHX) by pumping into the sodium dump tank at a rate of 120 gpm minimum.

13. The IHTS shall include vent piping to provide a positive vent from the steam generator or other IHTS high points to the sodium expansion tank or other system high point.

14. The horizontal runs of sodium piping shall be sloped a minimum of 1/8 inch per foot. Drains shall be located at all low points. All high points shall be vented.

15. Sodium/water reaction pressure relief subsystem (SWRPRS).

The process design requirements for this subsystem are:

a. Structural and sizing margin shall be provided in the design of the SWRPRS by requiring the SWRPRS design to accommodate a design basis leak consisting of one equivalent double- ended guillotine tube failure followed at one-second intervals by two additional equivalent double-ended guillotine tube failures without damage to the IHX or loss of sodium containment capability of the IHTS boundary. In addition, the IHX, IHTS, and SGS are designed for 1000 psig under faulted conditions for SWR without steam isolation and blowdown. The SWRPRS rupture disk and sodium dump tank shall be located as close as practical to the steam generator.

b. The hydrogen released by a large sodium-water reaction shall be separated from liquid material and released to the atmosphere and burned.

c. Liquids and solids which are removed from the IHTS during operation of the SWRPRS shall be contained within a vessel under inert atmosphere (sodium dump tank).

d. The SWRPRS shall be activated automatically when one or more sets of SWRPRS rupture disks are burst.

e. The SWRPRS shall be purged with nitrogen gas following a burst of one or more sets of SWRPRS rupture disks and water/steam blowdown of the steam generator.

f. Capability shall be provided for monitoring of sodium leakage into the space between the rupture disks.

g. A rupture disk assembly will be installed to by-pass the two valves installed in the inert gas equilization line connected between the IHTS sodium expansion tank and the sodium dump tank. The rupture disk shall provide relief capability for intermediate size water leaks, and its rupture shall initiate water side isolation and blowdown of the steam generator.

h. An argon gas atmosphere shall be maintained in the SWRPRS components up to the atmospheric seal (rupture disk) in the stack discharge line.

5.5.1.2 Structural Requirements

All IHTS piping and fittings shall be designed to ANSI B31.1, Power Piping Code as a minimum. System vessels shall be designed to the ASME Boiler and Pressure Vessel Code, Section VIII as a minimum.

5.5.1.2.1 Steady State Structural Requirements

1. The structural design for the IHTS components and piping shall be based upon the pressures and temperatures given in Table 5.5-1, Structural Design Parameters.

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

3. Establishment of the limiting values for design stress intensity shall include allowances for any known or predictable degradation of mechanical properties that may occur such as a result of irradiation, stress at service temperatures, and changes in material chemistry over the design life.

4. Penetrations, weld joints, and discontinuities shall exhibit smooth transitions to minimize stress concentrations. Welds shall be full penetration butt welds and located at low stress regions and shall be of a design which will permit radiography of all joints during fabrication.

5. The natural frequencies of all components shall be designed, where possible, to avoid resonance with all expected pump driving frequencies. Where this is not possible, the component design shall ensure that structural damage will not occur as a result of resonance.

6. All sodium or sodium vapor pressure boundary welds shall be surface examined by the dye penetrate method and shall be radiographed in accordance with procedures and acceptance standards of the applicable Code.

7. All liquid metal components and assembled systems shall be subjected to a helium leak test in accordance with the procedures and acceptance standards of the applicable Code.

8. The IHTS piping in the reactor head access area shall have guard piping to protect the head access area from sodium spillage in the event of a pipe leak.

9. Structural design shall provide for IHTS piping and component heat-up from ambient to cold standby (refueling) (400°F) and hot standby (540°F) temperatures.

5.5.1.2.2 Transient Structural Requirements

1. The IHTS shall be designed to accommodate duty cycle events resulting from the normal and upset conditions.

2. The IHTS shall be designed for 1000 psig and design temperature under faulted conditions to accommodate the pressures imposed by a major sodium/water reaction caused by a design basis leak in the steam generator of three double ended guillotine tube failures and/or failure of the water dump system.

3. The rupture disks of the SWRPRS shall be designed to withstand the effects of the operating basis earthquake.

4. Design requirements for the rupture disks shall be sufficient to assure that each disk will rupture at a differential pressure low enough to protect the IHTS and IHX from the sodium water reaction overpressure and high enough to maintain system operability under design conditions.

5.5.1.2.3 Natural Phenomena

The IHTS piping, components and associated controls and instrumentation shall be designed for seismic and other natural phenomena in accordance with the Uniform Building Code (UBC). The IHTS shall be designed to remain operable following and Operating Basis Earthquake (OBE).

Five OBE's, with 10 maximum peak response cycles each, shall be assumed to occur over the design life of the plant. Four OBE's shall be assumed to occur during the most adverse normal operation, and one during the most adverse upset operation on a component and design limit basis.

5.5.1.2.4 Materials of Construction

1. Construction materials for the IHTS piping and components wetted by sodium or sodium vapor shall be Type 304 or 316 austenitic stainless steel, carbon steel or 2-1/4 Cr-1 Mo ferritic steel.

2. The SWRPRS piping to the flare tip shall be carbon steel. The flare tip shall be stainless steel.

3. The Sodium Dump Tank shall be low alloy carbon steel.

4. Material of construction for the rupture disks in the SWRPRS system shall be Inconel 600, except that coated aluminum may be used for the SWRPRS atmospheric vent line rupture disk.

5. Materials having equivalent corrosion resistance and mass transfer properties may be used in such areas as valve seats, bearings, or other areas requiring special materials due to mechanical functional requirements.

6. Factory fabrication and assembly shall be utilized to reduce field construction and field QA/QC labor and thus, reduce the overall cost of the plant. Fabrication and assembly at the site shall be minimized. Equipment shall be modularized to the extent feasible to minimize cost.

7. The IHTS shall be designed to minimize the probability and effect of fires. Non combustible and heat resistant materials shall be used whenever practical.

8. The IHTS insulation materials shall be compatible with sodium at the maximum expected sodium temperature such that they do not add significantly to the potential reaction between sodium and air and the resulting reaction products shall not degrade the plant materials. Off-gases from the insulation shall be chemically non-reactive with other plant materials.

5.5.1.3 Configuration Design Requirements

1. The IHTS shall consist of nine independent cooling circuits, each designed to transport 430 MWt from the IHX to its respective steam generator.

2. Those portions of the IHTS loop within the head access area shall be located in a guard pipe which precludes sodium spills in the access area. Piping shall be sloped from the IHX upward to the steam generator and to the IHTS pump for venting.

3. Those portions of the IHTS loop piping and components outside of the head access area shall be located in a protective enclosure in an air environment. This enclosure shall have liners and/or catch pans as appropriate, to minimize the effect of sodium leakage on the concrete and structures. Piping penetrations to the confinement cell shall be provided with rigid seals to isolate the cell atmosphere from that in the pipe tunnel.

4. All IHTS sodium piping and components (except the IHX) shall be drainable; either by gravity or pumping. Positive means of preventing accidental drainage shall be provided.

5. There shall be no permanent direct piping connections between the PHTS and IHTS.

6. Leakage of PHTS sodium into the IHTS sodium loops shall be prevented by the IHX barrier, and in the event of a leak, by maintaining the IHTS pressure at least 10 psi higher than the PHTS pressure at all points in the IHX. This differential pressure condition shall exist during all normal operational modes.

7. The IHTS shall be designed to minimize the potential of large scale sodium fires. Major consideration shall be given to minimization of the system sodium volume and cover gas pressure and for rapid water dump of the steam generator system in the event of a sodium water reaction. Provision shall be made for sodium drain and containment in the event of accidents.

8. The IHTS piping and components shall be insulated to limit heat losses from the system.

9. The IHTS shall be capable of accommodating reconnection to the reactor module or replacement module.

10. All IHTS piping and equipment containing sodium shall be equipped with accessible sodium leak detectors of sufficient sensitivity to detect sodium leaks soon enough to minimize damage due to corrosion or fire. All IHTS valves shall have a leak detector installed between the bellows stem seal and the backup stem seal.

11. Design and layout of the IHTS shall be consistent with the maintenance requirements.

12. Radioactivity in the IHTS sodium shall be limited to 6×10^{-11} Curies/cm△ sodium by neutron shielding and Reactor System arrangement. This sodium activity corresponds to a piping surface dose rate of 0.2 mrem/hr.

13. Each IHTS sodium loop shall contain one free surface centrifugal pump located in the cold leg (between the steam generator outlet and the IHX inlet).

14. To minimize the probability of a sodium-water reaction in the event of a sodium leak or water line break, water lines shall not be routed in the vicinity of sodium-containing equipment except as necessary.

15. All IHTS liquid metal piping and components shall utilize full penetration butt welded connections. The welds shall be radiographed.

16. The IHTS arrangement and plant building design shall be such that sodium spills, leaks, or fires will not propagate and cause damage to other IHTS loops and will not prevent plant shutdown and effective decay heat removal.

17. Structures supporting the IHTS sodium-containing equipment shall be designed such that damage resulting from the design basis sodium fire will not result in failure or collapse of the supports for the sodium-containing equipment.

18. No single failure shall permit the contamination of IHTS sodium by halogenated inorganics or organics, mercury or water.

19. The IHTS design shall minimize gas entrainment. The sodium cover gas shall be argon.

20. The IHTS shall be designed to limit radiation exposure to the plant personnel to less than 50 Man Rem per year.

21. The IHTS, including the IHTS expansion tank and pump, shall be designed to operate under all power operating conditions and hot standby condition assuming a loss of 300 gallons of sodium inventory per year due to the valve leakage in the sodium drain sub-system.

5.5.1.4 Decay Heat Removal Requirements

The IHTS shall be designed such that decay heat removal can be accomplished utilizing the normal heat removal system by either of the following two methods:

1. IHTS pony motor operation providing the motive force for sodium flow: This capability shall be available following any upset condition from maximum power operation. For these events, sufficient coolant flow shall be provided to ensure that corresponding fuel design limits are not exceeded.

2. Natural circulation providing the motive force for sodium flow: This capability must be available following any upset condition and shall limit temperatures such that damage to the reactor or plant is within limits acceptable to an emergency event.

5.5.1.5 Design Safety Requirements

1. The IHTS shall be designed to meet the maintenance requirements and radiation exposure limits using Federal Regulations 10DFR20 and 10CFR50, Appendix I (ALARA).

2. Sodium spill insulation.

3. Na/H_2O SWRS.

5.5.1.6 Maintenance Requirements

1. Routine maintenance activities requiring plant shutdown shall be scheduled for performance during plant refueling. The refueling interval shall be every two years.

2. The principal design maintenance requirement shall be to provide access space for maintenance and otherwise facilitate contact maintenance of all system, components and piping. Direct access for contact maintenance to the maximum extent practicable shall be provided; radial clearance of 30 inches minimum shall be provided from pipe or component surfaces where maintenance is anticipated.

3. Capability shall be provided for remote maintenance when contact maintenance is not practicable. However, provisions for remote maintenance shall not preclude contact maintenance. This capability applies to the IHTS in the head access area only.

4. The structures containing IHTS equipment shall be designed to permit the installation and support of temporary shielding, if needed, for the protection of maintenance personnel.

5. The IHTS design shall provide for drainage and fill of any individual sodium loop. All components shall be drainable by gravity or pumping.

6. The design of the IHTS mechanical pump internals shall permit their removal for inspection, maintenance, or replacement.

7. IHTS mechanical pump shall be designed to permit seal maintenance and replacement without main drive motor removal.

5.5.1.7 Instrumentation and Control Requirements

1. Process instrumentation and controls shall be provided to monitor and control the IHTS over the full power range during all normal and off-normal operating conditions. Instrumentation and control tolerances shall be such that structural design limits of the IHTS components and piping are not exceeded. The IHTS preheater controls shall regulate heater input without introducing damaging thermal stresses.

2. Redundant instrument sensors, heaters, leak detectors, and other auxiliary items shall be provided where access to specific areas of the IHTS will be limited or prohibited by expected radiation levels and where specific design features may prohibit convenient access for maintenance.

3. Instrumentation shall be designed for in-place verification of instrument performance with minimum disturbance.

4. Control variables shall be incorporated into the design to provide for stable operation of the IHTS including preheating to maintain system temperatures. The IHTS shall be designed for base load operation requiring no sodium flow control.

5. Instrumentation shall be provided for activation of systems which dump water or relieve the pressure in the IHTS in the event of water-to-sodium leaks and provide signals to purge during and after a sodium-to-water reaction.

5.5.2 Design Description

5.5.2.1 Overall Description

The intermediate heat transport system for each reactor module consists of piping and components required to transport 430 MWt of reactor generated heat from the PHTS of that module to the steam generator system (SGS). The system (see Figure 5.5-1) is comprised of a piped loop thermally coupled to the PHTS by the IHX located in the reactor vessel and to the steam generator located in the steam generator building (SGB). Intermediate sodium is circulated by a constant speed mechanical pump (located in the cold leg of the loop) through the tube side of the IHX and the shell side of the steam generator. A permanent magnet (PM) flowmeter located in the cold leg provides the flow monitor function for the loop.

The IHTS is a closed loop system with an expansion tank and argon cover gas to accommodate thermally induced system volume changes. The arrangement and relative elevation of the IHTS piping and components are designed to promote natural circulation for decay heat removal. The initial natural circulation rate following shutdown from normal full power operating conditions is 7 percent normal flow.

The IHTS is designed to support operation over a range of 0 to 100% rated power and to support decay heat removal during all normal and upset operating conditions.

The IHTS also includes the sodium water reaction pressure relief subsystem (SWRPRS). This subsystem consists of rupture disks, sodium dump tank (SDT) and vent stack with flare tip and ignitor.

Auxiliary systems connected to the IHTS main loop necessary to meet operation requirements include sodium fill/drain system, steam/ water-to-sodium leak detectors, recirculating cold trap system, trace heating, thermal insulation, sodium-to-gas leak detection and instrumentation.

The intermediate sodium pump circulates non-radioactive sodium from the tube side of the IHX's located in the reactor vessel to the shell side of the steam generator evaporator located in the SGB and back to the IHX. The IHTS loop for each reactor module is completely independent with no direct interconnections with other IHTS loops. Also, there are no permanent piping connections between the PHTS and IHTS loops. The IHX tube walls provide a passive barrier for isolation of the activated primary sodium from the intermediate sodium. As a precaution against the very unlikely event of an IHX tube leak, the IHTS pressure is maintained at least 10 psi greater than the PHTS within the IHX. Also radiation detectors monitor the IHTS for indications of contamination by radioactive sodium.

The building arrangements further enhance loop separation by providing separate silos for each intermediate sodium loop. The building structure provides protection for all loops against the effects of the natural phenomena and for each loop against the failure of another loop.

The portions of the IHTS in the head access area (HAA) consist of main loop hot leg and cold leg piping between the IHX and the head access area wall penetration. Guard piping is used to prevent intermediate sodium leakage into the head access area in the event of a sodium pipe break.

The SGB has thermally insulated steel catch pans on the floors to protect against sodium spills. The air environment is maintained under normal conditions between 50° to 100°F. The IHTS in the SGB is accessible during normal plant operation.

Outside the head access area the IHTS consists of 20 and 30 inch diameter main loop hot and cold leg piping from the cell penetrations to the steam generator and pump. The sodium pump provides the flow and is located in the cold leg. All major components, auxiliary piping, and interconnecting services are located in the SGB. A permanent magnet flowmeter is located on the cold leg of each loop. The location of this flowmeter in the pipe tunnel is dictated by the necessity for a sufficient run of straight pipe upstream and downstream of the flowmeter. The PM flowmeters on the main loop piping of each loop are used to evaluate plant performance. The component arrangements and pipe routings in the SGB are identical for each loop. The main IHTS hot coolant flow exits the IHX, passes through the head access area penetration, and flows through thermal expansion loops and bellows to the steam generator inlet. The cold sodium exiting from steam generator enters the intermediate coolant pump suction. The flow from the pump discharge returns through the other access area penetration and completes the circuit through the IHX.

Several auxiliary and interfacing systems are required to satisfy the operational needs of the IHTS. A sodium expansion tank is used in the IHTS to provide a closed loop system. The expansion tank outlet is connected to the main loop pipe just upstream of the pump inlet. The expansion tank receives sodium flow from the cold trap return.

Hydrogen leak detectors are located on the main loop piping at the steam generator outlet to allow early detection of a steam generator tube failure. A continuous sodium flow to the cold traps is provided from the system low point located on the SG shell. The cold trap loop flow is also connected to the sodium plugging indicator and sampling system. Sodium from these systems is returned to the IHTS expansion tank. Both supply and return lines to the sodium processing system are equipped with double isolation valves.

The cold trap EM pumps are employed to fill the IHTS with sodium through the expansion tank at a maximum rate of 120 gpm. Fresh sodium is normally pumped from railroad tank cars into the SDT and then pumped into the IHTS after a system charge of sodium has been unloaded. As an alternate, fresh sodium may be pumped directly into the IHTS from the tank cars.

The cold trap EM pumps are also used to empty the IHTS (except the IHX inventory) by pumping the sodium from the IHTS low point at the SG outlet to the SDT or to railroad tank cars. The SDT may be used to store the IHTS sodium inventory during IHTS maintenance or during IHTS filling. The intermediate purification system provides the capability to transfer the sodium during these operations. The sodium may be cold trapped during transfer to and from the IHTS or while in the IHTS or SDT.

The IHTS closed loop configuration requires cover gas volumes to accommodate system volume changes due to thermal expansion. These gas volumes, provided in the IHTS expansion tank and the pump tank, are interconnected. The pressure, approximately 15 psig, is used to assure a 10 psi minimum positive pressure differential across the IHX from the IHTS to the PHTS side.

Sodium inductive level probes are utilized in both IHTS Expansion Tank and pump tank. Numerous temperature and pressure instrumentation are are provided on the IHTS piping.

All IHTS piping and components including the SDT are thermally insulated and trace heated. The thermal insulation of IHTS piping and components is designed to limit the surface temperature of the insulation to 140°F at rated power operation or hot functional testing with an ambient temperature of 100°F.

IHTS piping and components will be examined by direct viewing during periods of plant shutdown. During periods of prolonged shutdown which require loop draining and cooldown, more detailed examinations of individual welds will be conducted as required following insulation removal.

Sodium-to-gas leak detection is provided on all IHTS piping and components. Sodium aerosol or contact-type detectors monitor the insulation to pressure boundary annulus on all equipment. Additionally, sodium valves are equipped with contact type detectors to monitor for valve stem leakage, and major components have cable or spark plug detectors to monitor for collection of a pool of sodium underneath the components.

All materials for the IHTS piping and components are specified to minimize corrosion and erosion and ensure campatibility with the environment. The piping and fittings which make up the loops are of Type 304 SST.

Type 316 SST is used for high strength needs. Trimetallic joints are used to make a transition from ferritic to austenitic steel piping with Alloy 800H (Ni-Fe-Cr) pipe used as an intermediate coupling.

The IHTS expansion tank is constructed of Type 304 austenitic stainless steel. All parts of the intermediate coolant pump in contact with the sodium coolant are Type 304 and Type 316 austenitic stainless steels except for bearings and special parts. The sodium valves in the IHTS are constructed primarily of Type 304 austenitic stainless steel. Other materials in contact with the coolant are special materials such as the Stellite valve seats. The SDT is constructed of low alloy steel SA533 or equivalent which has good impact properties. All other auxiliary piping, flow restricting orifices, branch connections, and instrumentation connections are either constructed of the same material as the major component to which they are connected, or have a suitable trimetallic transition joint.

5.5.2.2 IHTS Piping

The IHTS piping connecting the IHX, steam generator, pump and expansion tank are shown in Figure 5.5-2. The pipes connecting the reactor vessel are located in the IHTS pipeway running from the reactor closure to the steam generator building. The use of articulating bellows joints (see Figure 5.5-3) in both cold and hot leg piping results in a simple piping layout. Within the HAA, the pipes are 20 inches in diameter. Outside the HAA, the 20-inch diameter piping lines from each IHX are headered into a single 30-inch diameter line for both the cold and hot legs. Three 30-inch gimballed bellows and short expansion loops are used to accommodate thermal expansion, seismic anchor displacement and design loadings. The joints allow rotation of the piping in both horizontal and vertical directions and allows a compact piping layout. The pipe support and snubber arrangements for the cold and hot leg pipes are illustrated in Figure 5.5-4. The IHTS pipe layout is designed to accommodate the thermal expansion of components and pipes and the relative movement of the seismically isolated reactor module.

Rigid supports and spring hangers are used to support the piping as depicted in Figure 5.5-5 and 5.5-6. Rigid supports restrain piping movement in either vertical or lateral directions. Spring hangers carry the dead loads only. Energy absorbers resist seismic and other dynamic loads and displacements. They consist of a system of energy absorbing flexible plates which act as elastic springs to allow movement during thermal expansion of the piping system. Under dynamic loading conditions energy is absorbed by controlled yielding of the plates. A typical energy absorber is shown in Figure 5.5-7.

5.5.2.3 Sodium Pump and Pump Drive

The intermediate sodium pump is a vertically oriented, single stage, double suction, free surface, centrifugal pump driven by a constant speed 4000 hp, air cooled induction motor. An auxiliary pony motor drive provides low flow (10%) capability for decay heat removal and other low power or standby conditions.

Design and performance of the intermediate coolant pump and pump drive (see Figures 5.5-8 and 5.5-9) to meet operational requirements are:

1. Capability to provide a flow of 41,000 gpm at a 314 ft static head and 540°F.

2. A main drive motor of 4000 hp @ 1,750 RPM.

3. Capacity in the pump tank to accommodate a free surface level change of six feet.

4. Pony motor capable of providing 10% design flow with automatic changeover from main motor to pony motor during coastdown to give uninterrupted flow.

5. Pony motor alternate power supply from plant gas turbine generator.

6. An impedance to flow between the sodium tank region of the pump and the hydraulics of the pump is provided to decrease the sodium level differences between the pump tank and the expansion tank to dampen potential flow oscillations following a pump trip and during sodium/water reactions, and to impede the rapid sodium drain from the pump tank following a loss of IHTS sodium event (e.g., pipe break).

7. One sodium level probe to facilitate fill and drain operations and to monitor sodium operating levels. Note that a second level probe thimble is provided for probe calibration in place.

8. A two-inch nozzle located above the normal operating sodium level provides a connection to the gas equalization line. This gas equalization line provides a common gas path between the pump and IHTS expansion tank gas spaces, thereby maintaining equal gas pressures in the two tanks.

9. A structure, which is an integral part of the steam generator building, supports the pump. The main drive motor is supported by a motor mount which is an integral part of the pump. A coupling connects the motor with the pump shaft and pump impeller. The pump shaft axial loads and the upper shaft radial loads are accommodated by the main motor bearings.

10. The pump shaft seal and cover gas schematic is shown in Figure 5.5-10. It is composed of a double, oil lubricated, rotating face seal in conjunction with a labyrinth gas seal. Lubricant contamination of the sodium is prevented by employing a slinger and argon gas purge of the labyrinth seal.

11. The pump is designed for seal maintenance and replacement without main motor removal. The rotating shaft seal is designed to have a minimum maintenance interval of 10,000 hours. The pump internals are removable for inspection and servicing without disturbing the pump tank or system piping.

5.5.2.4 IHTS Expansion Tank

A sodium expansion tank (Figure 5.5-11) is located in the SGB near its corresponding intermediate pump. The expansion tank, which is fabricated from 304/316 stainless steel, consists of a 12 ft-6 inch diameter by 17 ft high vertical, cylindrical vessel that is closed at each end by an elliptical head. A steel structure supports the expansion tank and connects its skirt to the steam generator building structure.

The expansion tank accepts the return flow from the IHTS sodium purification system and mixes it with the cold leg sodium to provide a uniform temperature flow to the cold leg piping via the expansion tank outlet piping.

The flow enters the expansion tank through a nozzle on the upper head. This nozzle has an extension tube dipping below the normal operating sodium-gas interface level to minimize gas entrainment. The IHTS sodium level varies as a result of system temperature changes, pump drawdown, and system transients. The expansion tank is sized to prevent sodium level excursions from exceeding the minimum and maximum allowable pump levels. Two drywells in the tank are provided for sodium level probes which monitor the expansion tank sodium level. The expansion tank has a nozzle for connecting the gas equalization line to the sodium dump tank. Another nozzle connects to an argon gas line, which connects to the argon supply and pump tank.

5.5.2.5 Sodium Water Reaction Pressure Relief Subsystem

The sodium water reaction pressure relief subsystem is a passive system which only becomes operational in the event of a large sodium/water reaction within the steam generator. In the event of a large sodium/water reaction the subsystem protects the sodium side of the steam generator, the IHTS and the IHX from overpressure by the use of rupture discs on the steam generator inlet. In the event of a sodium/water reaction, sodium and/or sodium/water reaction products expelled through the rupture discs are directed by sodium water reaction pressure relief subsystem piping to the sodium dump tank (SDT) where separation of liquid, solid and gaseous products takes place. The gaseous reaction products are then vented via a flare stack to the atmosphere.

In order to reduce the amount of water which may be admitted to the IHTS in the event of large sodium/water reaction, the steam/water side of the system is isolated and rapidly drained by the Water Dump Subsystem. Water dump valves are located at the inlet to the steam generator. Water dump piping directs the water/steam from the water dump valve to a water dump tank where the flashed steam is vented to the atmosphere. These valves are automatically actuated by failure of the system rupture disks.

Intermediate size steam or water leaks, up to approximately 2 lbs/sec can be accommodated without failure of the SWRPRS main rupture disks by relieving IHTS system pressure at the expansion tank through relief of the expansion tank/SDT vent line rupture disk. These rupture disks also initiate water side isolation and blowdown of the steam generator modules in the affected loop.

The tube side of the IHX which contains the intermediate sodium, the IHTS pressure boundary and the steam generator shell are designed for 1000 psig. This design pressure, which is equal to the steam pressure, is sufficient to withstand the pressure resulting from a sodium-water reaction caused by a multi-tube failure in the steam generator. Pressures above 1000 psig will exceed the steam pressure and prevent steam/water flow into the damaged steam generator; thereby, stopping the sodium-water reaction. Pressures of this magnitude can only occur if there is a failure in the steam system isolation and/or water dump system.

The SWRPRS is connected to the IHTS to provide overpressure protection in the event of a sodium water reaction. Two, six-inch diameter rupture disks in series on the IHTS expansion tank provide overpressure protection against steam generator water leaks of less than 3 lb/sec. Two twenty-eight inch diameter, rupture disks in series located on the upper shell of the steam generator provide overpressure protection against steam generator water leaks greater than 3 lb/sec. The rupture disks are connected to and discharge into the sodium dump tank (SDT). The SDT is a vertical tank designed to contain the entire sodium inventory of an IHTS loop and to separate the gaseous reaction products from the liquid/solid products in the event of a sodium/water reaction (see Figure 5.5-12). The SDT is located along side of the steam generator unit to minimize the piping between the SDT and SG. Liquid/solid reaction products and displaced sodium are contained in the SDT. Gaseous reaction products are released and burned through a stack with flare tip and ignitor connected to the SDT.

5.5.3 Design Evaluation

Preliminary hydraulic analysis has been performed to establish the system configuration, pipe sizes, vessel sizes and pump heads. Conceptual design of the intermediate sodium pump and drive has been completed by a pump supplier to establish the pump operating characteristics, size, weight, configuration and electrical power requirements.

Preliminary piping stress analysis has been performed on the IHTS hot leg piping to assure that it can be designed within the Code allowable stress limits.

A preliminary study was made to determine the maximum pressure buildup in the IHTS as a function of the number of SG tube failures during a SWR without steam generator system isolation and blowdown. The calculations indicate that the IHTS pressure peaks at about 850 psig even if all steam generator tubes have failed. Thus, it is unlikely that the IHTS pressure will reach the faulted design pressure of 1000 psig.

5.5.4 Tests and Inspection

The intermediate heat transport system and its components shall be designed to permit periodic inspection and testing to access their structural and leak tight integrity.

5.5.4.1 In-Service Inspection

In-service inspection will be performed on IHTS sodium components in compliance with the ASME Boiler and Pressure Vessel Code, Section XI, Division 3, "Rules for In-service Inspection of Nuclear Power Plant Components." This code requirement is not mandatory for the IHTS as it applies to safety related systems designed to Section III of the Code; however, the intent of the Code will be met and any deviations noted.

Prior to initially filling the liquid metal system with sodium, the complete system, components, and appurtenances shall have successfully passed a pneumatic leakage test at 1.2 times the system design pressure and at ambient temperature. The components under test shall be examined at the test pressure by a helium leak test or bubble test. The pressure boundary, all pressure boundary welds, attachments and supports shall be examined during the leakage test. All snubbers and valves shall be tested. Dimensional verification shall be made of the components and piping positions when the plant is a) cold and empty, b) preheated and empty, c) preheated and filled with liquid metal and d) during hot function testing.

In-service inspection consists of scheduled inspections and tests during plant operation and during outages for maintenance. The inspection interval shall be approximately ten years (see Code for specific schedules).

In-service inspection of the IHTS pressure boundary consists of continuous leak detection monitoring and visual inspection. Leak detectors shall be calibrated prior to installation and once every three years. The system will be visual inspected during the in-service leak test.

Integral attachments and integral supports to the IHTS pressure boundary shall be visually examined 100% every inspection interval.

One-third of all non-integral component supports shall be visually inspected every inspection interval. Non-integral component supports include, a) mechanical connections to the components and building structures, b) weld connections to the building structures, c) weld and mechanical connections at intermediate joints in the supports, d) guides and stops, e) spring type, and f) constant load supports.

All snubbers rated at 50 Kips or greater shall be tested during each inspection interval and ten percent of all snubbers rated at less than 50 Kips shall be tested during each inspection interval.

Prior to operation above 5% of rated power and subsequent to shutdown for refueling, maintenance, or in-service inspection, the pressure of the system shall be raised such that the pressure in all points of the system is not less than that seen at 100% of full power operation. The data from each leakage monitor shall be collected and evaluated for a period sufficiently long to detect any trend prior to nuclear operation. A visual examination of the pressure boundary shall be made during these in-service leakage tests.

If a component is replaced, added, or altered during the service lifetime, the preservice examination requirements for the component, alteration or replacement, and the attaching welds shall be performed. System leakage test shall be performed as a minimum at full power operation pressure.

Sodium valves are used in the IHTS to isolate the IHTS from the intermediate sodium processing system. Argon valves are used to isolate the expansion tank cover gas from the SDT. These valves are remote operated from the plant control room. The remote position indicator will be checked against the actual valve position once every two years. Each service has a set of two valves in service to allow part stroke testing every 3 months and leak testing every two years. The valves will be full stroke tested and fail safe tested during each cold shutdown (but not more often than every three months).

A continuous radiation leak detector is provided on the hot leg piping from the IHX to detect the unlikely leakage of radioactive primary sodium into the IHTS.

Duplex rupture disks on the expansion tank and steam generator provide IHTS over pressure protection. The volume between the double disk is continuous monitored for sodium leakage through the upstream disk.

5.5.4.1 Component Functional Test

The key IHTS component that requires functional testing is the sodium pump. The system is normally operating; therefore, pump performance is regularly monitored during the course of plant operation. Pump performance on pony motors can be measured during refueling and hot shutdown when the main motor is disengaged. Pump parameters monitored include; 1) sodium flow rate, 2) pump developed head, 3) fluid temperature, 4) discharge and suction pressure, 5) shaft vibration, 6) pump sodium level, 7) pump cover gas pressure, 8) seal vibration, 9) bearing temperature, 10) pump speed and 11) pump seal oil level.

TABLE 5.5-1

STRUCTURAL DESIGN PARAMETERS

	DESIGN PRESSURE (psig)	FAULTED* PRESSURE (psig)	DESIGN TEMPERATURE (°F)
Main Loop Piping			
IHX to Steam Generator	300	1000	850
Steam Generator to Pump	300	1000	650
Pump to IHX	300	1000	650
Components			
Pump: Casting	300	1000	650
Shaft Seal	300	1000	150
Expansion Tank	300	1000	650
Rupture Disk: SWRPRS	300	-	850
Expansion Tank	300	-	650
Sodium Dump Tank	200	-	850
Piping			
Drain Piping (Upstream of 1st Valve)	300	1000	650
Vent Piping	300	1000	850
Pump Inlet to Expansion Tank Piping	300	1000	650

*Faulted conditions are for SWR with <1 hour duration at design temperature

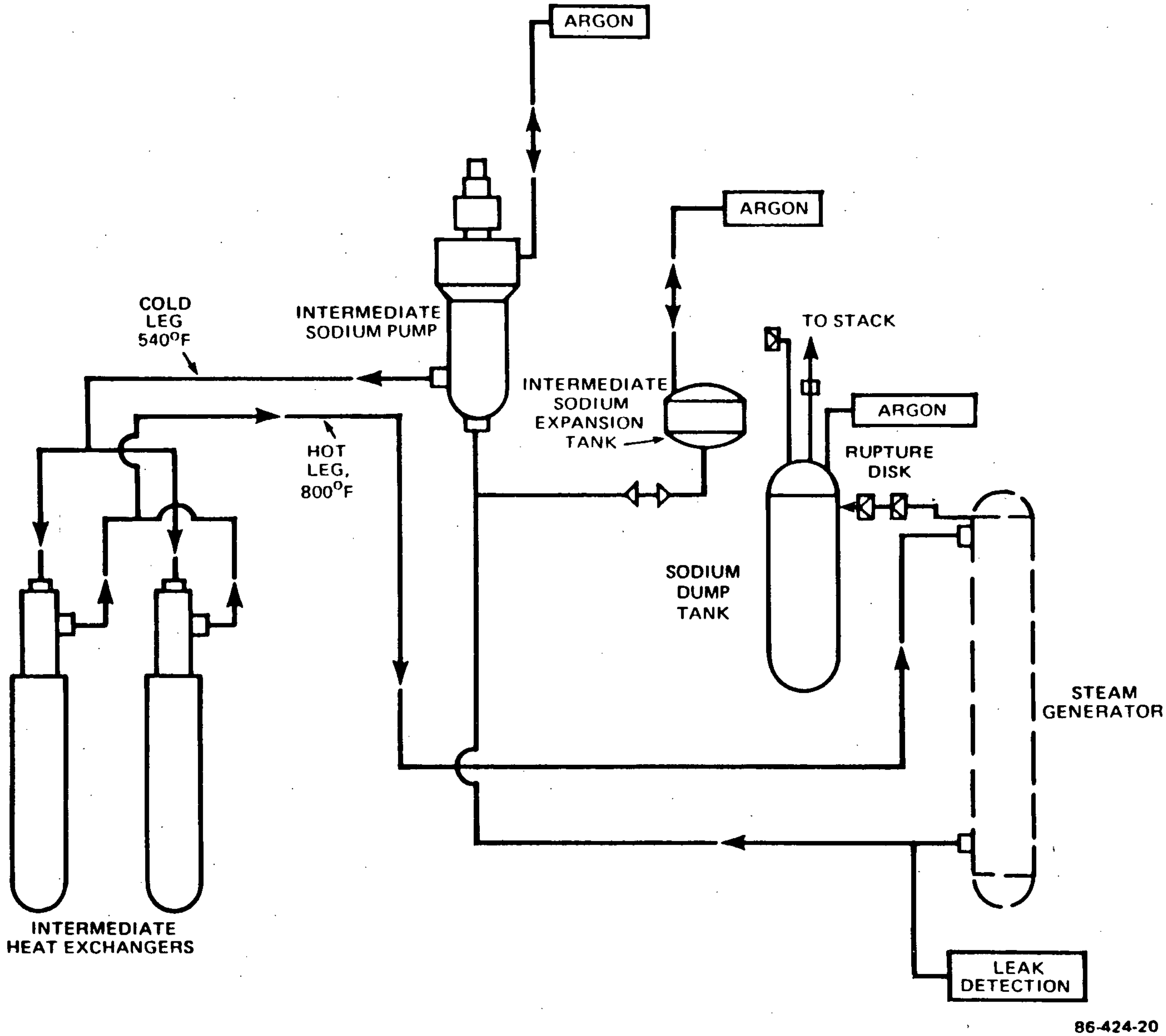


Figure 5.5–1 PRISM INTERMEDIATE HEAT TRANSPORT SYSTEM FLOW DIAGRAM

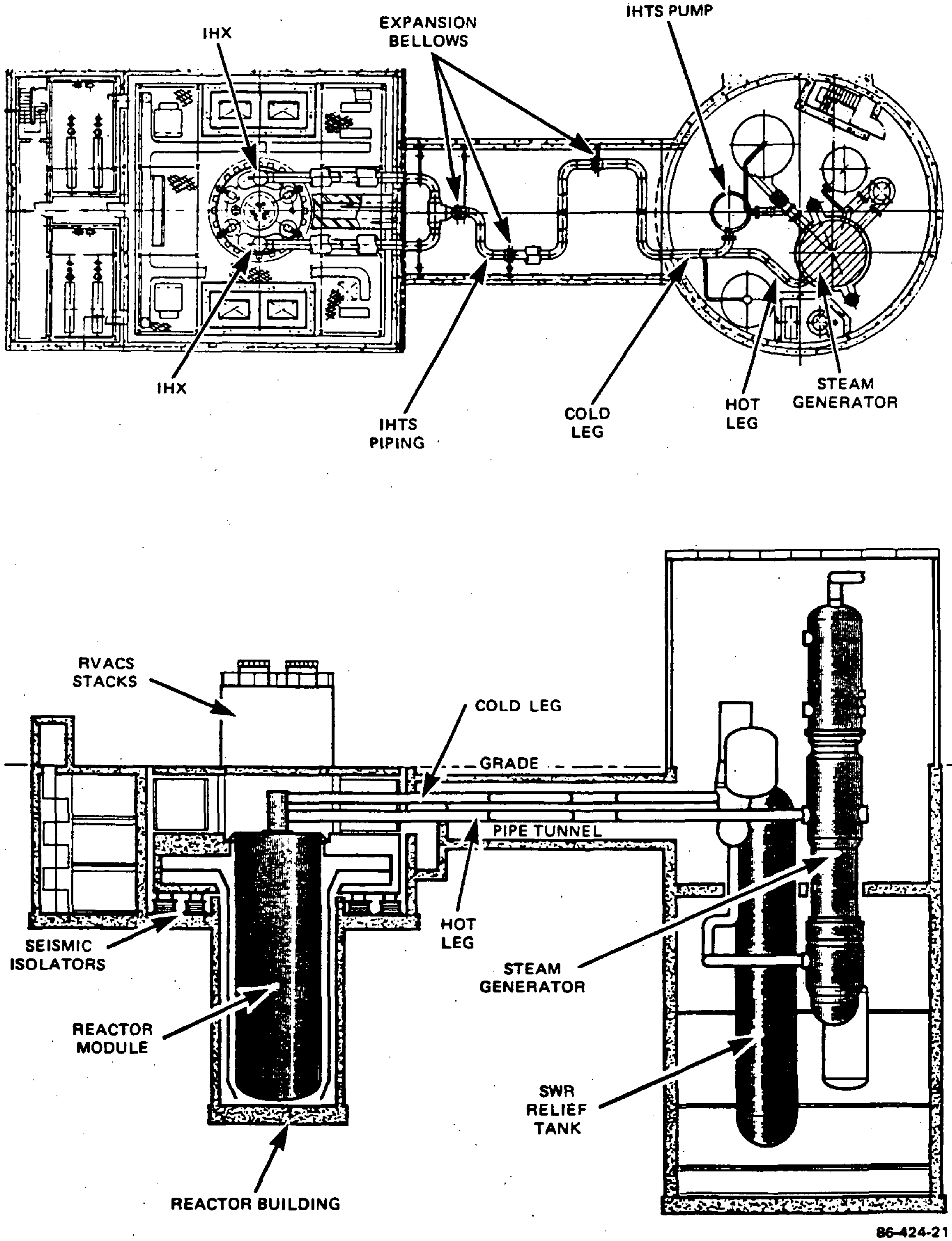


Figure 5.5–2 IHTS PIPING AND COMPONENTS ARRANGEMENT

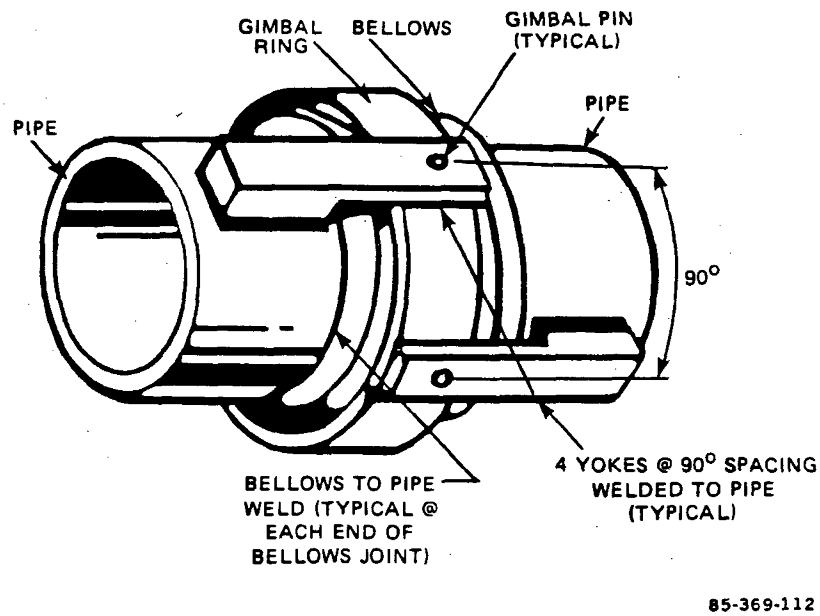


Figure 5.5–3 IHTS PIPING GIMBALLED BELLOWS

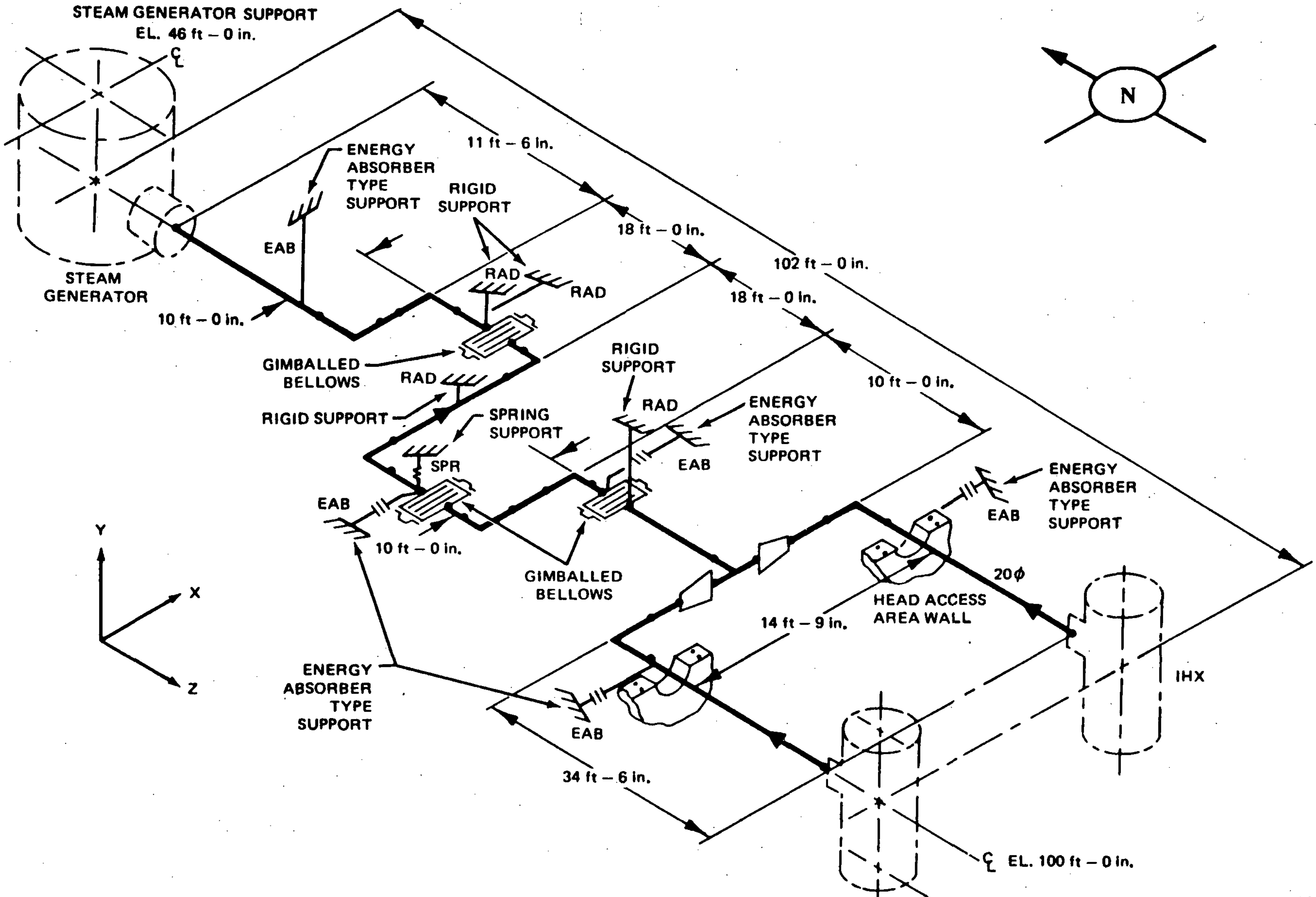


Figure 5.5–4 ILLUSTRATION OF PRISM IHTS HOT LEG PIPING SUPPORTS

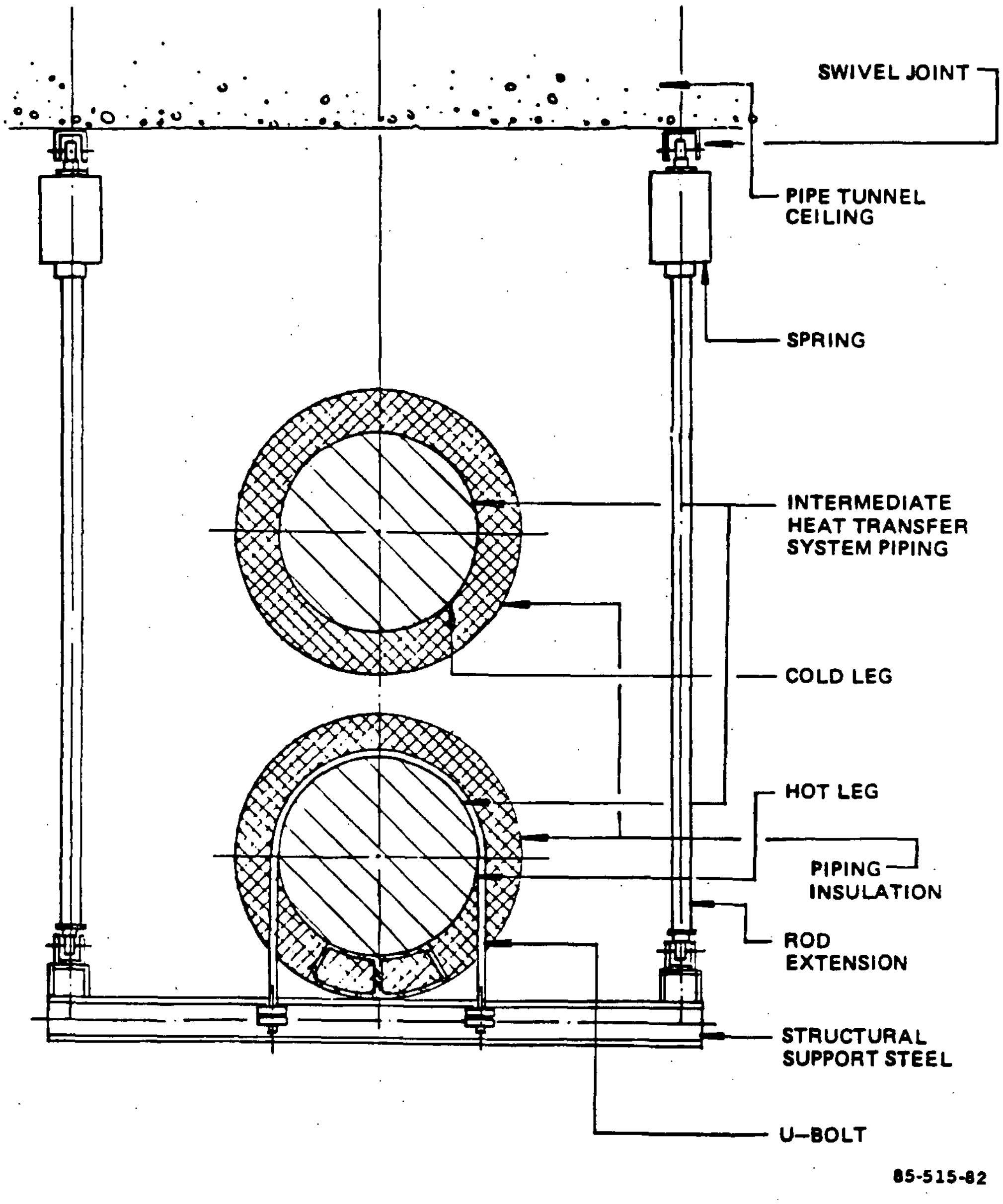


Figure 5.5–5 **TYPICAL SPRING HANGER**

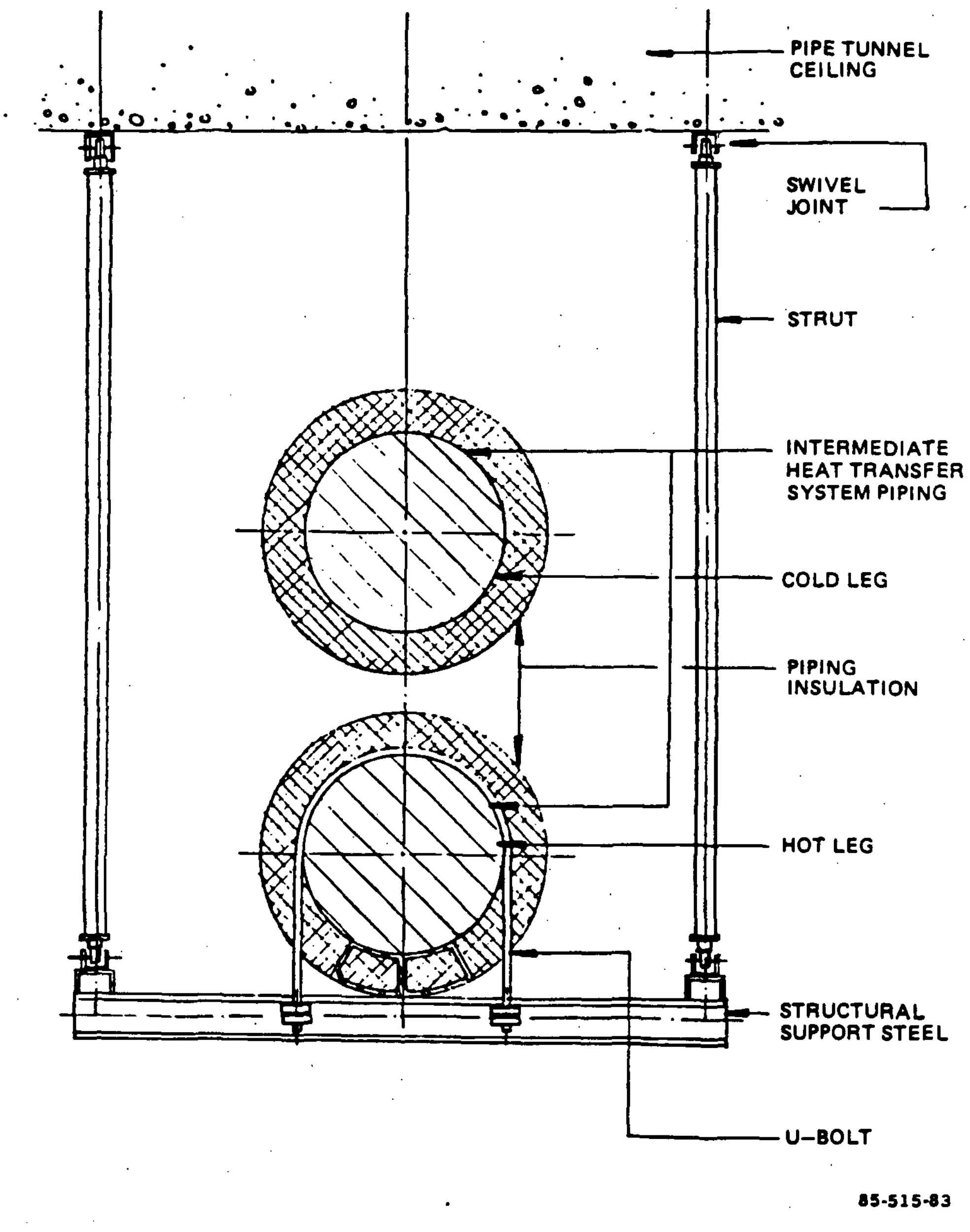


Figure 5.5–6 TYPICAL RIGID SUPPORT

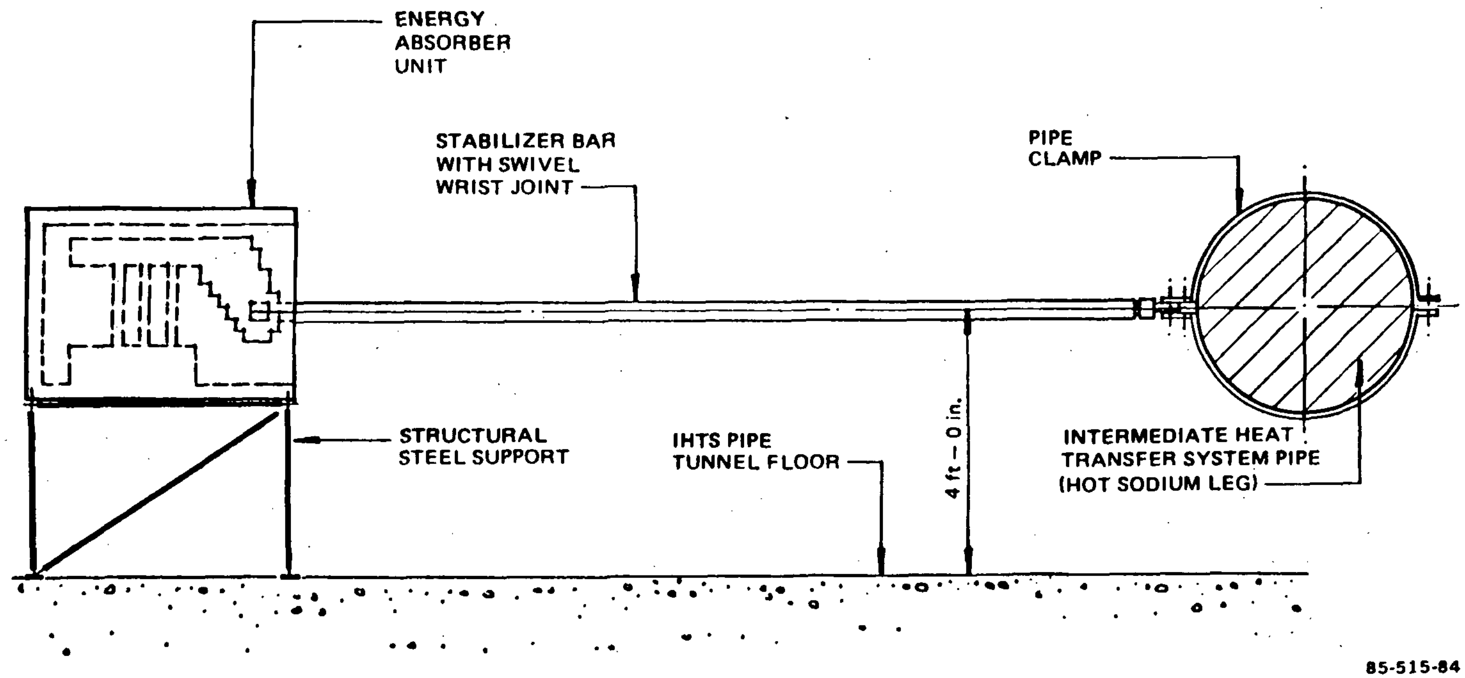


Figure 5.5–7 TYPICAL ENERGY ABSORBER

PUMP CHARACTERISTICS

FLOW	41,000 GPM
SODIUM TEMPERATURE	540°F
TOTAL HEAD	120 PSI
AVAILABLE NPSH	72 ft @ 540°F
REQUIRED NPSH	61 ft @ 540°F
PUMP SHAFT POWER	4000 HP
PUMP SPEED	1400 RPM
PUMP MATERIAL	304 AND 316 SST

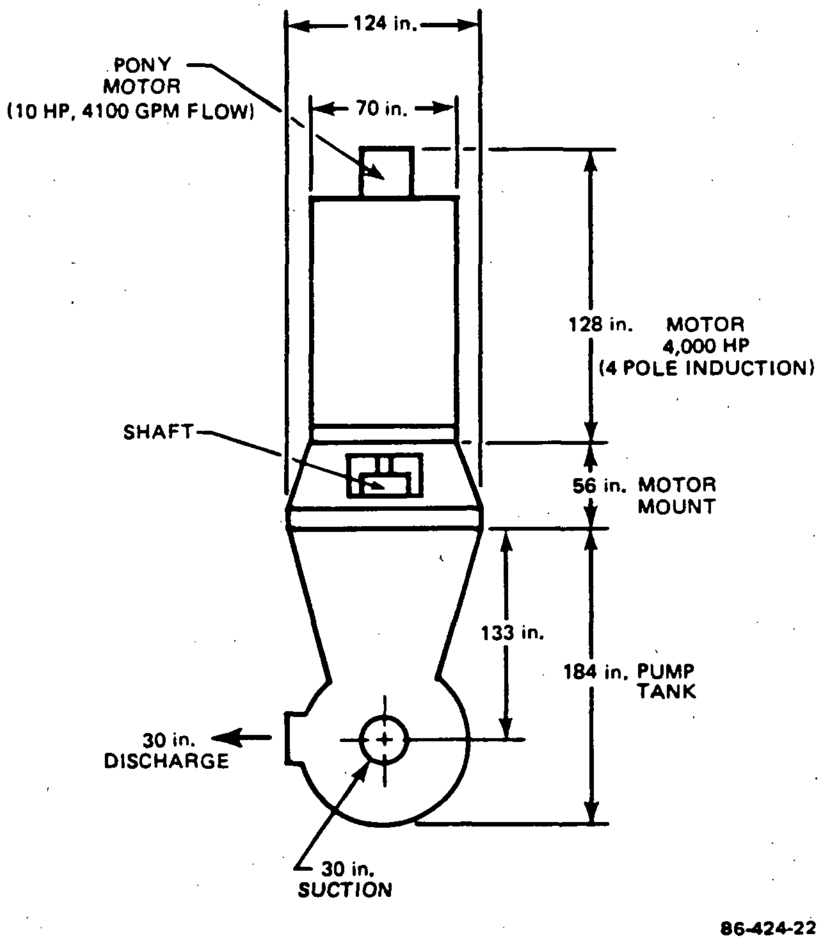


Figure 5.5–8 PRISM IHTS PUMP AND MOTOR

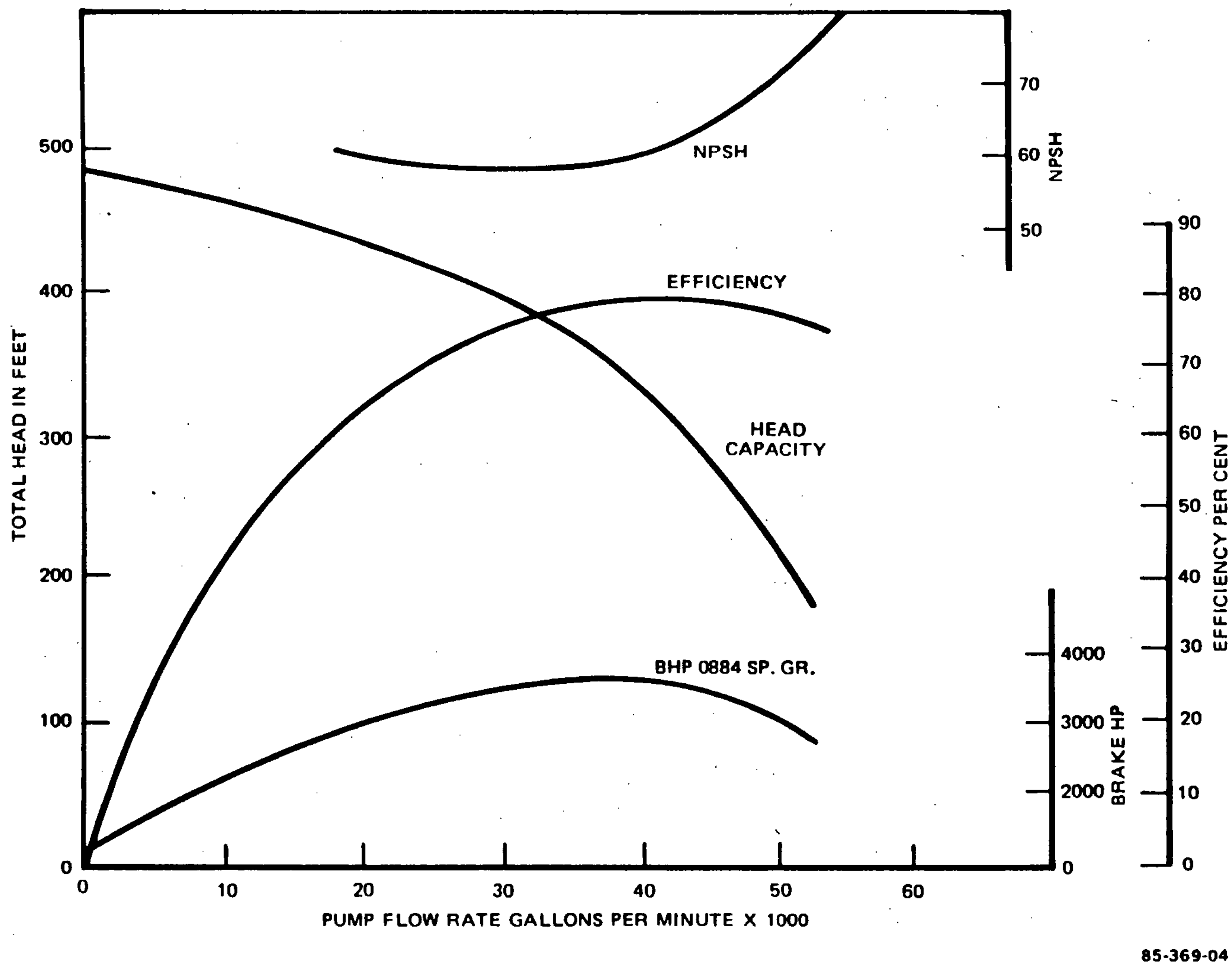


Figure 5.5–9 IHTS PUMP PERFORMANCE

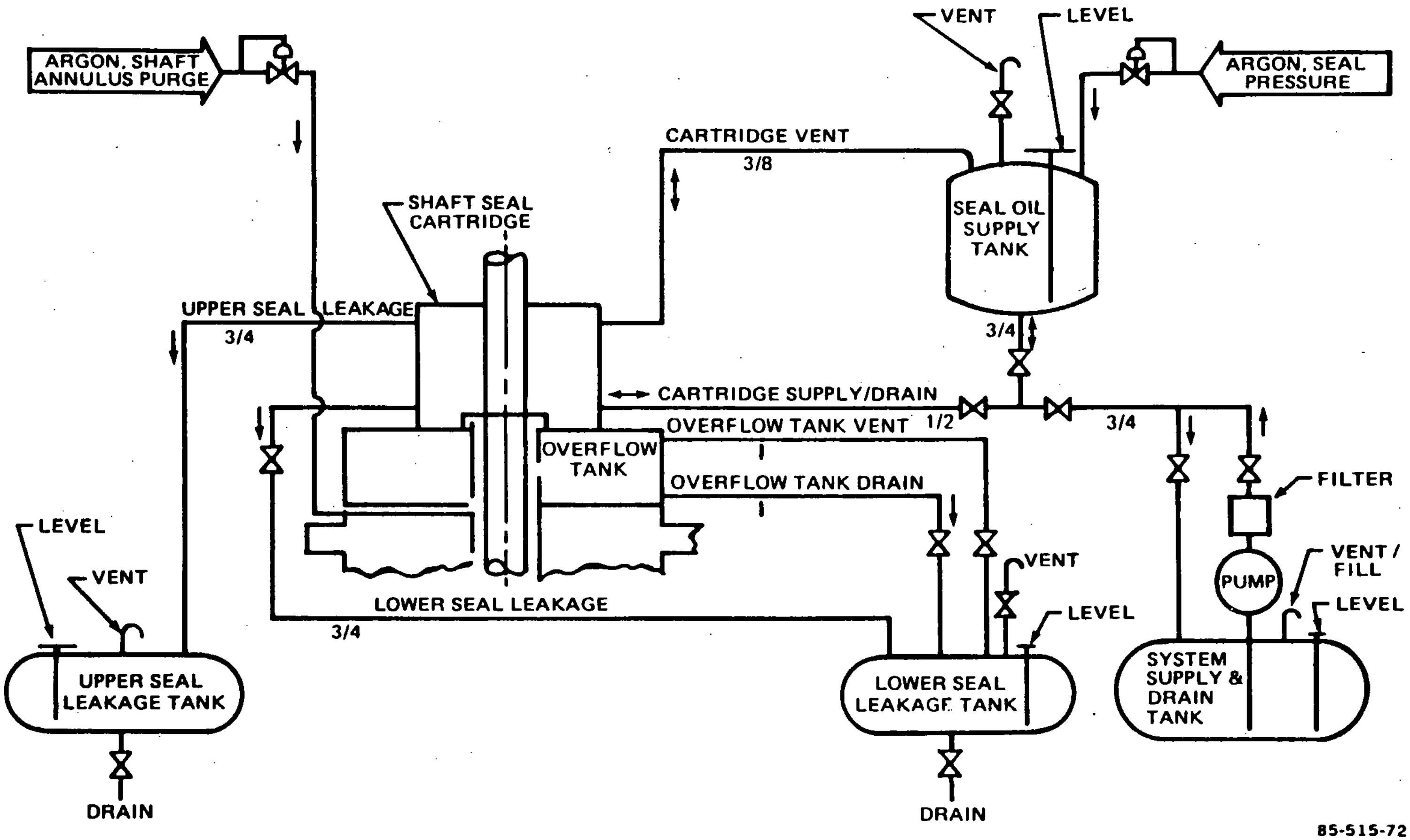


Figure 5.5–10 INTERMEDIATE PUMP SHAFT SEAL OIL AND COVER GAS SYSTEM SCHEMATIC

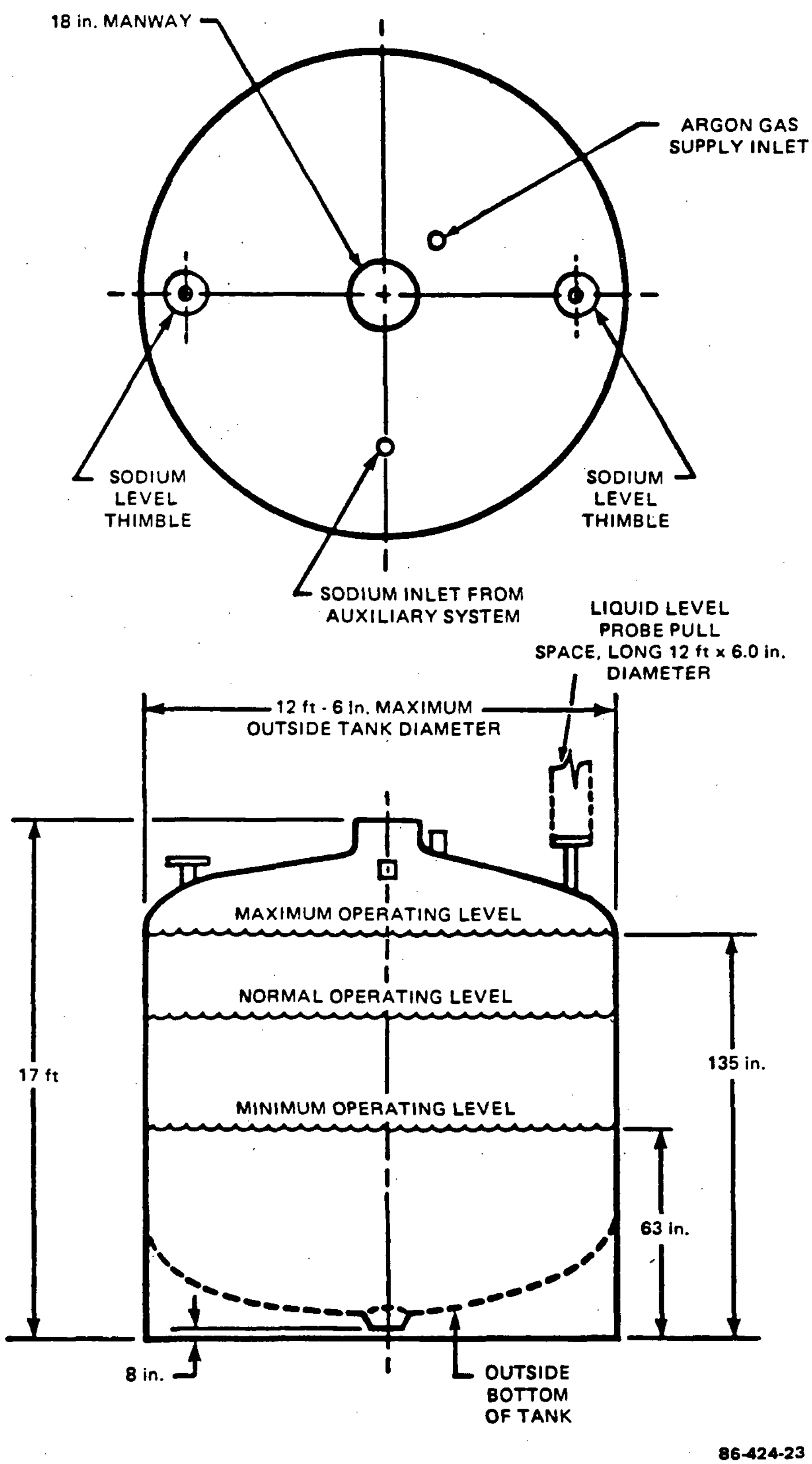


Figure 5.5–11 PRISM IHTS EXPANSION TANK

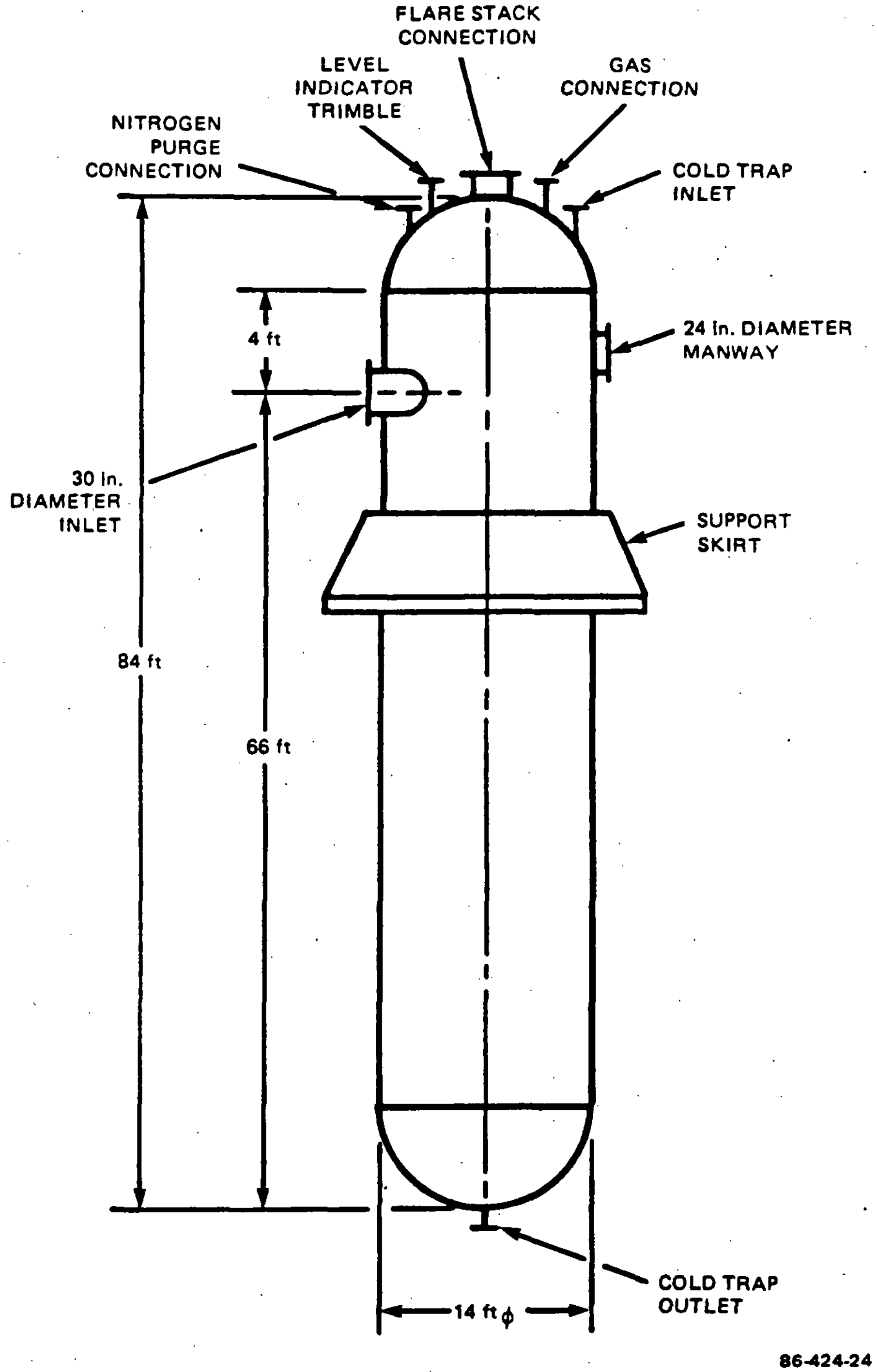


Figure 5.5–12 PRISM IHTS SODIUM DUMP TANK

5.6 Steam Generator System

5.6.1 Design Bases

The functions of the steam generator system (SGS) are to:

1. Convert subcooled water into saturated steam by transferring heat from the Intermediate Heat Transport System (IHTS) sodium to water/steam.

2. Provide dry and clean saturated steam at the temperature, pressure, and flow rate required by the turbine over the 25 to 100 percent operating range. In addition, provide appropriate steam flows for turbine warm-up and initial turbine loading to 25 percent.

3. Cool the IHTS sodium to the temperature levels required for safe reactor cooling during transient and steady state conditions and be capable of cooling the IHTS down to refueling temperature of 400°F. Transients resulting from normal, upset and emergency conditions described in the plant duty cycle shall allow continued operation.

4. Provide a high integrity pressure boundary to assure separation between the sodium and water/steam in the steam generator.

5. Provide rapid blowdown of the steam/water side of the steam generator to mitigate the consequences of a sodium-water reaction and provide the capability of gas backfilling to prevent backflow of sodium into the water/steam side.

6. Provide the capability to isolate the SGS from the main steam system and the feedwater system.

7. Prevent the steam/water side pressure from exceeding a safe value.

8. Provide an auxiliary reactor decay heat removal subsystem to cool the IHTS in the event of loss of the steam generator water/steam coolant.

9. Provide for blowdown as required to maintain water chemistry control.

10. Detect water-to-sodium and steam-to-sodium leaks in the SG and, to the maximum extent possible, identify the approximate size of the leak.

11. Separate sodium and water/steam piping and components as much as practicable to minimize the potential of sodium/water reactions caused by high pressure steam/water line ruptures.

5.6.1.1 Process Requirements

1. There shall be one independent steam generator system for each of the nine reactor modules. The SGS shall generate 432 MWt of steam at full power operation. The SGS shall be capable of operating over a load range of 25 percent to 100 percent of the design power level.

2. The SGS shall be designed for the following thermal/hydraulic requirements at full load conditions:

Intermediate Heat Transport System	
Load per loop	432 MWt
Sodium hot leg temperature	800°F
Sodium cold leg temperature	540°F
Sodium flow rate	18.33×10^6 lb/hr
Steam/Feedwater System	
Steam cycle	Saturated
Steam rate	1.8×10^6 lb/hr
Steam pressure	1000 psia
Steam temperature	545°F
Feedwater temperature	420°F
Recirculation water flow	2.2×10^6 lb/hr
Blowdown rate	2% of steam rate
Tube type	Double-wall

3. The SGS shall be a recirculating system designed for stable plant operation and easy plant control. The ratio of recirculation water mass flow to steam mass flow shall be 1.2/1.

4. The SG DNB margin shall be more than 25 percent to account for uncertainties in flow distribution and temperature variation within the steam generator. The SG water side pressure drop shall be less than 30 psi.

5. The steam drum shall be designed to supply dry saturated steam to the turbine. The steam drum outlet moisture carryover shall be less than 0.1 percent. The steam drum shall provide sufficient water inventory to prevent SG dryout during plant transients.

6. A steam drum blowdown of 3.6×10^4 lb/hr shall be provided for steam impurity control.

7. Heat transfer margins shall be provided to account for uncertainties to the extent that there will be a 95 percent confidence level that the unit will meet the performance requirements.

8. The SG shall be designed for uniform flow distribution within the unit to minimize thermal imbalance and eliminate hot spots. Appropriate baffling and flow distribution rings shall be used to direct the fluid flow.

9. The SGS shall be capable of maintaining the system pressure within ASME code limits with steam-by-pass to the condenser following a turbine trip.

10. The SGS shall be capable of maintaining system pressure within ASME code limits following a reactor trip when steam cannot be delivered to the main and auxiliary steam systems.

11. The SGS shall be designed to remove plant sensible heat and reactor decay heat load from the intermediate loop following reactor module shutdown from full power operation. An auxiliary cooling subsystem (ACS) shall also be provided to remove reactor decay heat and cool the system to 400°F within three days in the event of loss of steam/water cooling.

12. The SGS shall be capable of maintaining hot standby conditions of 545°F, 1000 psia.

13. The SGS shall have the capability of being preheated from ambient temperature to refueling temperature of 400°F at 10°F/hr maximum with the steam generator empty of sodium.

14. The SGS shall be designed for the following sodium purity:

	Material Basis	Water/Steam Leak Detection Basis*
Oxygen, ppm (max)	10	2.0
Hydrogen, ppm (max)	0.8	0.2
Plugging Temperature, °F	350	∇300

*Higher values are permissible during startup.

15. The design requirements of the hydrogen leak detection system are:

  a. Provide sensitive and responsive leak detection instrumentation which will detect leaks in the steam generator.

b. Provide alarm functions for low, intermediate, and high leak detection levels. The response time to alarm shall be consistent with state-of-the-art capabilities.

c. Provide a means to initiate timely plant shutdown to allow SG maintenance and repair operations to be initiated with minimum delay.

16. The design requirements for the water dump system are:

a. Capability shall be provided for reducing the pressure of the steam/water side of the steam generator to a value which is above the sodium side pressure in the module to minimize entry of sodium into the steam/water subsystem after a sodium-water reaction.

b. Capability shall be provided to remove the liquid on the water side of the steam generator during a sodium water reaction to reduce the amount of water available for reaction with the sodium.

c. A storage tank shall be provided to store the water which is dumped from the steam generator during a sodium-water reaction.

d. The steam/water side of the steam generator shall be pressurized or purged with nitrogen immediately following the water blowdown process. The pressure and the purge rate shall be sufficient to prevent significant amounts of sodium from entering the steam/water side of the SG.

e. Sufficient redundancy of equipment shall be provided to assure activation of the water isolation and dump system.

5.6.1.2 Structural Requirements

5.6.1.2.1 Steady State Structural Requirements

The SGS shall be designed according to the applicable requirements of Section VIII of the ASME code. The following general requirements shall be used for the SGS design:

Sodium Side:

Design pressure	300 psig
Design Temperature	850°F

Steam/Water Side:

Design pressure	1100 psig
Design temperature	850°F

Material:

Low alloy ferritic steel and carbon steel

Design Life:

Sixty years

The SGS components shall be designed for critical loadings during normal operating and transient conditions. The sodium side of the steam generator will be designed for a faulted condition of 1000 psig at 900°F for a SWR.

5.6.1.2.2 Materials of Construction

Constructions materials for the leak detection subsystem piping and components shall be Type 304 or 316 austenitic stainless steel. The materials of construction for the sodium-air boundary and sodium-water boundary of the evaporator shall be 2-1/4 Cr-1 Mo. The materials of construction for the steam generator system piping and components wetted by water or steam, except for the steam generator, shall be carbon steel or chrom-moly steel. Materials having equivalent corrosion resistance and mass transfer properties may be used in such areas as valve seats, bearings, or other areas requiring special materials due to mechanical or instrument functional requirements. ASME Code Section II shall be utilized in selecting materials of construction.

5.6.1.2.3 Natural Phenomena

The steam generator system and components shall be designed for seismic and other natural phenomena in accordance with the Uniform Building Code (UBC).

The SGS piping, components and associated controls and instrumentation shall be designed to withstand the effects of the Operating Basis Earthquake (OBE), and remain functional during and after an event. The OBE horizontal and vertical design ground acceleration shall be 0.15g. Five OBE's, with ten maximum peak response cycles each, shall be assumed to occur over the design life of the plant. Four OBE's shall be assumed to occur during the most adverse normal operation, and one during the most adverse upset operation determined on a component and design limit basis.

5.6.1.2.4 Pipe Break Criteria

For a postulated pipe break event in the SGS and assuming a single active component failure, the following criteria shall be met:

1. The effects of a pipe break in any SGS loop must be restricted to that loop and must not impair operability of the other SGS loops.

2. Inside the SGS loop where the pipe break occurs, damage to "critical components" shall be eliminated through the use of pipe restraints, protective barriers, etc.

3. Pipe breaks shall not cause failure of equipment which functions to mitigate the consequences of the pipe break event. Effects of a pipe break in any SGS loop shall not in turn, lead to an uncontrolled

sodium water reaction and result in hydrogen gas release into a cell of the affected SGS loop.

5.6.1.3 Configuration Design Requirements

The SGS shall consist of one independent steam generation loop for each nuclear module of the nine module plant. Each loop shall consist of one steam generator, a steam drum, a recirculation pump, and other necessary components and associated equipment.

The loops shall be arranged so operating conditions or casualty events in one loop will not affect the operation of the others. However, the normal operation of the steam generator loops shall be coordinated to supply saturated steam to the turbine from each loop at the flow rate, pressure, and temperature called for by the Plant Control System.

The steam generator system shall be designed to commercial/industrial standards (e.g., non-nuclear safety grade requirements).

Factory fabrication shall be utilized to the maximum extent possible to reduce field construction and field QA/QC labor and thus reduce the overall cost of the plant.

The steam generator for each SGS loop shall be located at an elevation above that of the IHX. The elevation difference shall be sufficient to provide natural circulation of the IHTS under all shutdown heat load conditions to provide adequate core cooling.

The steam drum for each SGS shall be elevated relative to the steam generator to provide natural circulation of the water side recirculation loop at a rate which is sufficient to remove normal reactor decay heat.

Pipe sizes are chosen so fluid velocities at 100 percent plant power condition will not exceed the following nominal values:

1.	Water	20 fps
2.	Water/Steam Mixture	50 fps
3.	Saturated Steam	125 fps
4.	Sodium	30 fps

The headers and piping from the steam drum to the recirculation pump inlet and the elevation difference between the drum and pump shall be designed to provide adequate NPSH and to avoid pump cavitation under the expected operating conditions.

Remote operated isolation valves shall be provided to permit SGS isolation from the main steam system, feedwater system and blowdown system.

The SGS piping and components shall be insulated to limit heat losses from the system. Sufficient insulation or other provisions shall be made on the SGS piping and components to ensure that surfaces with which personnel may come in contact shall not exceed 140°F at full power operation or hot functional testing at 750°F with an ambient temperature of 100°F. The insulation materials shall not add significantly to the reaction between sodium and air. Where in-service inspection requirements dictate and component geometry permits, the insulation construction shall permit easy removal.

The design life of all major components in the SGS shall be 60 years. Other components may have a design life of less than 60 years if they can be replaced at appropriate intervals with no significant effect on plant availability.

5.6.1.4 Design Safety Requirements

The SGS is not a nuclear safety related system. The following design safety requirements are for investment protection.

The SGS shall provide a sodium-water leak detection subsystem to minimize the potential for large sodium water reactions. The leak detection subsystem shall provide the following features:

1. Tap locations as close as practicable to the steam generator and interconnecting piping to minimize detection time.

2. An EM pump to assure a continuous and constant sodium flow.

3. Heaters and controls for maintaining a sodium temperature up to 875°F at the membrane of the meter. A regenerative heat exchanger shall be employed to reduce the required heat input and to aid in maintaining close temperature control of the leak detector membrane due to transient temperature/flow changes in the system being sampled.

4. The in-sodium hydrogen detector shall be a pumped diffusion membrane type.

5. Redundant leak detectors shall be provided. A minimum of two leak detectors shall be operational whenever the system is filled with sodium and water/steam.

All steam generator system components and piping shall be drainable. Positive means of preventing accidental drainage shall be provided.

The SG shall be designed for pressure release and rapid water-side dump in the event of a sodium water reaction. The water-side blowdown of the steam shall be less than 60 seconds.

5.6.1.5 Maintenance Requirements

The SGS shall be designed for low maintenance requirements. System arrangement shall consider constructibility, maintenance, removal and replacement of large major components.

The SG shall be designed to permit in-place leak location and subsequent plugging of faulty tubes. The steam generator modules shall be arranged in the steam generator building so that each steam generator can be removed and replaced without disturbing the other major components in the building. In addition, space shall be provided for removal of the water/steam lower head from the steam generator in place.

The components of the steam generator system shall be arranged with sufficient space and accessibility to perform in-place maintenance on each component. Sodium components shall be designed to permit sodium drainage for maintenance. The horizontal runs of all water and steam piping shall be sloped a minimum of 1/8 inch per foot, and drain valves shall be located at the low points to facilitate draining of the piping and headers.

The leak detector module shall not be designed for maintenance during operation.

In-service inspection shall be performed on the components constructed in accordance with ASME Boiler and Pressure Vessel Code Section VIII to provide continuing assurance that they are safe. In-service inspection requirements shall meet the intent of those given in Section XI of the Code.

Insofar as practicable, all welds on the main sodium and steam/water piping shall be accessible for inspection after insulation removal. Space and access shall be provided for specified ISI and maintenance operations. All control system instrument sensors shall be replaceable without disassembly of major components or redundancy shall be provided.

5.6.1.6 Instrumentation and Control Requirements

Instrumentation and control equipment for the SGS, in conjunction with the heat transport instrumentation system shall provide the following functions:

1. Measurement of process system variables (e.g., pressures, temperatures, flows and liquid levels) for monitoring of system and component operation.

2. Control of system components (e.g., valves, pumps) needed to maintain process variables within operational and structural design limits over the full range of operating conditions.

3. Activation of systems which isolate, dump, or relieve the water/steam pressure in the SGS in the event of water-to-sodium leaks. Provide signals to purge during and after a sodium-to-water incident.

4. Monitoring of parameters for evaluation of the performance characteristics of the steam generator.

5. Provide for the data handling, display and annunciation for the detection of a water-to-sodium leak in the steam generator modules.

6. Provide signals to identify a rupture of a water/steam line within the Steam Generator Building.

5.6.2 Design Description

5.6.2.1 Overall Description

The steam generator system (see Figure 5.6-1) provides independent steam generation capability for each of the reactor heat transport system loops. The steam generator thermal-hydraulic design conditions at full power are shown in Figure 5.6-7. The steam generator system is comprised of the following subsystems:

1. Steam Generator and Water/Steam Subsystem
2. Leak Detection Subsystem
3. Water Dump Subsystem
4. Auxiliary Heat Removal Subsystem

The steam generator subsystem obtains 1.89×10^6 lb/hr at 420°F feedwater from the feedwater system and produces 1.85×10^6 lb/hr of saturated 990 psia steam. The feedwater enters the steam drum where it is internally mixed to subcool the saturated water from the steam generator. The subcooled water is then circulated by the recirculation pump from the drum to the steam generator. The ratio of the water recirculated to steam generated is 1.2/1. In the steam generator tubes, the subcooled water is preheated and partially vaporized. The saturated water and steam exiting from the steam generator tubes then flows to the drum where separators, internal to the drum, separated the water and steam. A small percentage (2%) of the saturated water is then drained (before the saturated water is mixed with the incoming feedwater) from the drum for water chemistry control and returned to the feedwater and condensate system. The saturated steam then flows through dryers, internal to the drum, and on to the turbine.

In order to reduce the amount of water which may be admitted to the IHTS in the event of large sodium/water reaction in the steam generator, the system is isolated and emptied by the water dump subsystem. Water dump valves are located on the piping at the inlet to the steam generator. Water dump piping directs the water/steam from the water dump valves to a dump tank where the flashed steam is vented to the atmosphere. Double isolation valves in series are located in the feedwater line to each steam generator, in the blowdown line, and in the steam line from each steam drum. These valves are closed automatically in conjunction with the blowdown in the event of a large sodium/water reaction.

The steam generator leak detection subsystem monitors the sodium at the steam generator main sodium outlet. Monitoring is by three redundant hydrogen meters which provide signals relative to the hydrogen concentration level in the sodium. In the event of a water-to-sodium leak, changes in the hydrogen concentration levels are detected. The resulting signals are conditioned, transmitted, and displayed. Off-normal conditions are annunciated.

The steam generator and water steam subsystem is comprised of the following major components.

1. Steam Generator
2. Steam Drum
3. Recirculation Pump
4. Water/Steam Piping
5. Valves

Feedwater from the feedwater system flows through a remotely actuated stop valve, a flow control valve (a parallel start-up valve is provided) a flowmeter, another remote actuated stop valve, and a check valve; it then enters the steam drum.

The steam drum mixes the feedwater with water separated from the saturated water/steam mixture and directs it to the recirculation pump. Continuous steam drum blowdown flows from the steam drum, through two isolation valves and a flowmeter to the blowdown flash tank. The subcooled water from the steam drum flows through a downcomer to the recirculation pump.

Recirculation water is pumped through the tube side of the steam generator. Sodium flowing counter-current on the shell side heats the water to saturation temperature and vaporizes 83 percent of the water to steam at full power. The mixture of saturated water and steam exists from the steam generator and flows to the integral steam drum.

Saturated steam exits from the steam drum through a single nozzle to the 24-inch diameter steam piping system. The steam line contains a flowmeter, two remote operated stop valves, and a check valve. Seven safety/relief valves are mounted on the steam drum for system over-pressure protection. The steam from three reactor modules is combined in a common header prior to entering the turbine.

Water from the feedwater line, from the blowdown line, and from the drum drain line, is transported through sample lines to a monitoring station for measurement of specific feedwater and recirculation water impurities.

During the preheat operation, the steam generator system is required to be heated to 400 ±25°F before filling the IHTS with sodium in order to prevent the sodium from freezing. This is accomplished by circulating gradually heated water through the steam generator system and the tubes of the steam generator. In this manner the steam generator system and the tubes of the steam generator can be heated at the rate of 10°F/hr. However, the shell of the steam generator will only be heated at a rate of about 3°F/hr due to the shell thickness and thermal resistance between the heated tubes and the shell. Sodium filling of the IHTS can be initiated when the steam generator shell is at 350°F.

The steam generator is designed to be cooled by natural circulation of air over the exterior shell surface to provide an alternative method of reactor decay heat removal. This auxiliary cooling system (ACS) is required to supplement RVACS (both are described in Section 5.7) and reduces the time required to cool the reactor system and IHTS to hot standby temperature following a reactor shutdown when the steam system is not available. The ACS is not a safety-related system, but rather, is used to increase plant availability by avoiding the slow plant cool down which would occur if only RVACS were used. The ACS consists of a steel shell enclosing the steam generator so as to form a six-inch wide natural convective, air, cooling annulus. Louvers are provided to initiate air cooling.

5.6.2.2 Steam Generator

There is one steam generator per reactor module. The steam generator is a vertically oriented, shell-and-tube counterflow heat exchanger with water/steam on the tube side and sodium on the shell side. The tubes are straight and of double-wall construction with the tubes prestressed during fabrication to assure contact. The tubesheets are fixed and a convoluted shell expansion joint provides for differential thermal expansion between the shell and tube bundle. The double-wall tubes provides improved reliability by significantly reducing the probability of a tube leak. Double fillet welds are used to joint the tubes to the tube-sheet (front and back face fillet welds); thus, there is a double barrier between the sodium and steam/water throughout the steam generator. Figures 5.6-2 and 5.6-3 show the general arrangement of the steam generator.

Sodium entering the shell side at the inlet nozzle is uniformly distributed around the tube bundle by a selectively perforated cylindrical shroud. It is then uniformly distributed to the tube bundle by a preferentially drilled radial flow hole arrangement in the upper support plate. Crossflow baffles maintain sodium flow velocities to provide acceptable flow distribution and heat transfer. With the exception of the 304SS protector tubes, the construction material is 2-1/4 Cr-1 Mo.

The steam generator is designed to transfer 432 MWt from the intermediate heat transport system to the steam system to produce 1.85×10^6 lb/hr of saturated steam at 543°F and 990 psia. The tube bundle contains 1838 double-wall tubes, 60 ft length, 1.25 inch O.D. on 2.375 inch triangular pitch. The recirculation ratio is 1.2 to 1.0 with a blowdown rate of 2 percent. The circulation ratio was chosen to insure operation without departure from nucleate boiling (DNB). This mode of operation results in efficient heat transfer with a relative uniform heat flux. A water side mass flux of 95 lb/sec/ft^2 was chosen to provide reasonable velocities and sufficient margin on critical heat flux and avoid departure from nucleate boiling.

A double-wall tube was selected for the steam generator because of the reliability offered with the double boundary separating the steam and sodium. The outside diameter of the outer tube is 1.25 inch and the inside diameter 1.096 inch. The outside diameter of the inner tube is 1.096 inch and the inside diameter is 0.946 inch. Both tubes are sized for full design pressure. The total wall thickness of both tubes is 0.152 inch. The double-wall tube is prestressed such that contact at the interface is maintained and separation of the tubes does not occur during operation.

Steam generator design requirements and thermal/hydraulic design sizing parameters are shown in Tables 5.6-1 and 5.6-2 respectively. The total heat transfer area in the tube bundle is 33,080 ft^2. Note that

6.5 ft long protector tubes are used at the sodium inlet region of the bundle to limit the heat flux in this region. Nominal 100 percent power temperature and heat flux profiles are shown in Figure 5.6-4.

5.6.2.3 Steam Drum

The steam drum is mounted directly on the upper head of the steam generator. A general arrangement of the steam drum design is shown in Figure 5.6-5. The pressure boundary is made of a two cylindrical shell courses and elliptical head. The shell is made from a 5.75 inch plate formed in 180-degree shell segments and the hemispherical head is made from plate formed as a complete head. The cylindrical portion of the shell is penetrated by two 24-inch inspection ports, one 6-inch blowdown nozzle, one 12-inch feedwater nozzle, one 2-inch sampling nozzle, four 2-inch water level indicator nozzles, one 2-inch drain nozzle, and one 16-inch return nozzle. The elliptical head is penetrated by one 24-inch saturated steam outlet nozzle.

Major internal features of the steam drum are shown in Figure 5.6-5 and consists of the following: a steam riser plenum made up of five 19-inch diameter cylindrical risers, a separator deck with five centrifugal steam separators; a bank steam dryer situated on a support grid/drain system; a distribution ring for receiving feedwater from the feedwater nozzle and thoroughly mixing this fluid with the downward flowing recirculating water and a chemical addition/fluid sampling ring which connects to the sampling nozzle.

Access to the steam drum internals is provided through two 24-inch diameter inspection ports located at an elevation just above and just below the separator cans. Once the inspection ports are removed, the separators and other internals are readily accessible for maintenance.

The primary structural load path for supporting the bulk of the steam drum internals is through the separator can deck. The separators are seismically supported at the top and bottom of the separator can deck using bolted junctions and oversize holes. This arrangement provides extremely rigid support for the internals while allowing freedom for differential thermal expansion.

The entire operational weight of the steam drum is carried through the connection to the steam generator tubesheet. The dry vessel weight is approximately 170 tons.

Feedwater, at the rate of 1.89×10^6 lbm/hr, enters the steam drum through a single 12-inch diameter feedwater nozzle at 420^0F and is directed to the feedwater ring. This subcooled fluid then flows circumferentially around the ring exiting through sets of opposing elbows. These elbows are located on the upper side of the feedwater ring and serve as sparging devices to assure adequate mixing of the feedwater with the recirculation water which is flowing downward through the annular downcomer plenum. In addition to providing highly efficient sparging action, these elbows are so located to ensure that the feedwater ring remains filled during periods of no feed flow to prevent water hammer upon resumption of feedwater flow.

Saturated steam/water mixture (83% quality at 543°F) from the steam generator enters the steam drum at the rate of 2.22×10^6 lbm/hr into the riser plenum, providing for thorough mixing of the incoming two-phase mixture. Flow is then upward through the five 19-inch diameter risers. The mixture then passes through the five steam separators and approximately 80 percent of the total flow emerges from the upper end of these devices as relatively high quality steam. The steam continues to flow upward and through the peerless type steam dryers. Upon exiting from the bank of dryers, the steam has been dried to a quality of 99.9 percent or better and leaves the steam drum through the 24-inch steam outlet nozzle at the rate of 1.85×10^6 lbm/hr under a pressure of 1000 psia.

The liquid portion of the incoming two-phase flow from the evaporators is, for the most part, removed by the centrifugal action of the fixed vane steam separators and is allowed to drain down onto the separator can deck. The remaining liquid (that which was removed in obtaining the 99.9 percent quality steam mentioned above) is removed by impingement on the corrugated plates and vanes in the steam dryers and is drained back to the separator can deck through eight drain lines. Thus, essentially saturated liquid flows into the steam drum water plenum from which the continuous drain is taken. It is within the downcomer plenum that the saturated liquid is mixed with the feedwater to produce subcooled recirculation water. This subcooled water exits the steam drum through the 16-inch return nozzle and is pumped to the steam generator lower head.

5.6.2.4 Recirculation Pumps

There is one constant speed circulation pump provided for each loop to circulate water from the steam drum to the evaporator at 1.2 times rated steam flow. The pump handles 5300 gpm of water at 440°F and is driven by an electric motor.

Maintenance of the pump shaft mechanical seals is planned every 36 months or when there is excessive water leakage past the seals. To facilitate maintenance, the seals are designed to be removed as an assembled cartridge without removing the drive motor.

5.6.2.5 Leak Detection Subsystem

Any leakage of water/steam into the sodium stream increases the hydrogen and oxygen concentration in the sodium. The leak detection subsystem utilizes the measurement of hydrogen concentration in the liquid sodium stream as the method of leak detection. The measurement of hydrogen in sodium is accomplished by allowing hydrogen to diffuse through a thin nickel membrane, one side of which has a high vacuum held by an ion pump.

Three hydrogen detectors are included in each loop. These detector modules are installed on the main sodium piping at the steam generator outlet.

The leak detection modules have the following capabilities and characteristics:

	In-Sodium Hydrogen Meter
Concentration Range	0.04-2 ppm by weight
Minimum Sensitivity	6 ppb (3% in a background of about 200 ppb)
Maximum Module Response Time	$\leq$30 sec
Minimum Replacement Time	4 hours
Service Life	Diffusion Membrane - 5 years Structural Parts - 30 years
Maximum Drift Rate	$\pm$10 ppb/day

Replacement of critical components will be accomplishable during planned shutdowns. Critical components are those outside of the sodium boundary most susceptible to failure, replacement of vacuum system components and electronic parts.

Response time of the detection module is defined as the time for sodium transported from the inlet of the module to time when the hydrogen in the sodium is sensed at the instrumentation readout.

The module is designed to allow replacement of the diffusion membrane without removal of the module from the loop. This is accomplished by establishing a frozen sodium seal in the piping or drainage of the IHTS.

In order to have adequate coverage, redundant detectors are provided to accommodate detector malfunctions during operation. A minimum of two operational hydrogen meters is required to ensure leak detection coverage any time steam/water and sodium are present across the boundary of the steam generator tubes. The purpose of this requirement is early detection in the event of a water/steam to sodium leak.

5.6.2.6 Water Dump Subsystem

A water dump subsystem is provided in each loop to accept and store the water from the steam generator when rapid depressurization of the steam generator is required. Two quick opening 10-inch dump valves in parallel, which are part of the steam generator and water/steam subsystem, are provided at the inlet to each steam generator. The water dump valves exhaust to the water dump 10-inch piping which directs the water/steam mixture (water plus flashed steam) to a water dump tank where the water is temporarily stored, and the flashed steam is vented to the atmosphere. The water dump tank overall dimensions and major nozzles are shown in Figure 5.6-6.

5.6.2.7 Auxiliary Cooling Subsystem

The auxiliary cooling subsystem (ACS) is required to supplement RVACS and reduce the time required to cool the reactor system and the IHTS to hot standby temperature (540°F) following a reactor shutdown when the steam system is not available. The ACS consists of a shroud around the steam generator which provides a six inch wide cooling annulus. The steam generator can be cooled by natural circulation of air over the exterior shell surface upon actuation of inlet and outlet louvers. Inlet air is taken from the SGB and exhausted through a stack to the atmosphere. The ACS is not a safety system, but rather, is used to increase plant availability by avoiding the long plant cool down (60 to 70 days) which would occur if only RVACS were used. With ACS the plant can be cooled to 540°F after shutdown from full power in about three days. The ACS has a rated heat removal capacity of 2 MWt. A more detailed description of the ACS is given in Section 5.7 with a sketch of the ACS shown in Figure 5.7-6.

5.6.3 Design Evaluation

Preliminary thermal-hydraulic analysis has been performed to establish the steam generator system configuration, pipe sizes, vessel sizes and pump head. Conceptual design of the steam generator and steam drum has been completed by a heat exchanger supplier to establish the size, weight and configuration. Preliminary thermal-hydraulic analysis has been performed to determine the decay heat removal capability of the ACS.

5.6.4 Tests and Inspection

The steam generator system and its components shall be designed to permit periodic inspection and testing to access their structural and leak tight integrity.

5.6.4.1 In-Service Inspection

In-service inspection will be performed on the Steam Generator System components in compliance with the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-service Inspection of Nuclear Power Plant Components". This code requirement is not mandatory for the Steam Generator System as it applies to safety related systems designed to Section III of the Code; however, the intend of Section XI will be met and any deviations noted and justified. In-service inspection of water/steam components will be performed in compliance with Division 1, Class 3 Rules for light water cooled plants and in-service inspection of sodium components will be performed in compliance with Division 3, Class 3 Rules for inspection of sodium cooled plants.

Pre-service inspection and tests will be performed prior to placing the system into service. Prior to initially filling the liquid metal system with sodium, the complete system shall have successfully passed a pneumatic leakage test at 1.2 times the system design pressure and at ambient temperature. The components under test shall be examined at the test pressure by a helium leak test or bubble test. The steam/water systems shall be hydrostatically tested at 1.25 times the design pressure at ambient temperature. The pressure boundary, all pressure welds, attachments and supports shall be examined during these leakage tests. All snubbers and valves shall be tested. Dimensional verification shall be made of the components and piping positions when the plant is, (a) cold and empty, (b) preheated and empty, (c) preheated and filled with coolant and (d) during hot functional testing.

In-service inspection consists of scheduled inspections and tests during plant operation, and during outages for maintenance. The inspection interval shall be approximately 10 years (see code for specific schedules).

Prior to operation above 5 percent of rated power and subsequent to shutdown for refueling, maintenance or in-service inspection, the pressure of the sodium system shall be raised such that the pressure in all points of the system is not less than that seen at 100 percent of full power operation. The data from each continuous leak detection monitor shall be collected and evaluated for a period sufficiently long to detect any trend prior to nuclear operation. The steam/water system shall be given an in-service leakage test at normal operating conditions once each inspection interval. A visual examination of the pressure boundary with the thermal insulation in place shall be made during these in-service leakage tests.

If a component is replaced, added or altered during the service lifetime, the preservice examination requirements for the component, alteration or replacement, and the attaching welds shall be performed. System leakage test shall be performed as a minimum at full power operation pressure.

Continuous sodium-to-air leak detection monitoring is provided for the sodium system pressure boundary. Leak detectors shall be calibrated prior to installation and once every three years.

Integral attachments and supports to either the sodium or steam/water pressure boundary shall be visually examined with thermal insulation removal 100 percent each inspection interval.

One third of all nonintegral sodium component supports and 100 percent of all nonintegral water/steam component supports shall be visually examined with the insulation removed each inspection interval. Nonintegral component supports include: (a) mechanical connections to the components and building structures, (b) weld connections to the building structures,

(c) weld and mechanical connections at intermediate joints in the supports, (d) guides and stops, (e) spring type hangers and and (f) constant load supports.

All snubbers rated at 50 kips or greater and 10 percent of all snubbers rated at less than 50 kips shall be tested during each inspection interval.

Main steam feedwater and blowdown isolation valves will be full stroke exercised and fail safe tested every cold shutdown. The valves will be part stroke tested every three months. The valves will be leak tested and the remote position indication checked every two years.

Main steam, feedwater and blowdown check valves will be full stroke exercised every cold shutdown.

The main steam safety/relief valves will be tested each five year period in accordance with ANSI/ASME OM-1, "Requirements for In-service Performance Testing of Nuclear Power Plant Pressure Relief Devices", Parts A and B for ASME Class 2 and 3 pressure relief devices.

The water dump subsystem shall be functionally tested once each inspection interval at normal operating conditions. The pressure boundary shall be visually examined after the test with thermal insulation in place. Visual examination shall include integral attachments and supports

5.6.4.2 Component Functional Tests

The major components that require functional testing are the snubbers, isolation valves, check valves, safety relief valves and water dump valves. These components will be tested as described in Section 5.6.4.1.

TABLE 5.6-1

PRISM STEAM GENERATOR
DESIGN REQUIREMENTS SUMMARY

- Steam Generator Power - 432 MWt
- One Unit Per Loop
- Recirculating System for Stable Plant Control
- Design Code, ASME Section VIII, Division 2, ANSI B31-1
- Saturated Cycle 1000 psi Steam
- Provide Blowdown to Maintain Water Chemistry Control
- 95 Percent Confidence Level on Performance Requirements
- Achieve Hydraulic Stability Under all Loads
- Design Life - 60 years
- Tubeside Design Conditions
  - Design Temperature - 850°F
  - Design Pressure - 1100 psig
- Shellside Design Conditions
  - Design Temperature - 850°F (900°F Faulted)
  - Design Pressure - 300 psig (1000 psig faulted)
- Material - Low Alloy Ferritic Steel (2 1/4 Cr - 1 Mo)
- Rail Shippable

TABLE 5.6-2

PRISM STEAM GENERATOR
THERMAL/HYDRAULIC DESIGN SIZING DATA

PARAMETER	VALUE
Nominal MWt	432
Water Flow Rate (lbm/hr)	2.22×10^6
Water Side Mass Flux (lbm/sec-ft^2)	95
Circulation Ratio	1.2:1
Mode of Recirculation	Forced
Recirculation Water Inlet Temperature (°F)	440
Recirculation Water Inlet Pressure (psia)	1036
Feedwater Temperature (°F)	420
Steam Outlet Temperature (°F)	543
Steam Flow Rate (lbm/hr)	1.85×10^6
Water/Steam, ΔP (psi)	25
Steam Exit Pressure (psia)	990
Sodium Flow Rate (lbm/hr)	18.33×10^6
Sodium Inlet Temperature (°F)	800
Sodium Outlet Temperature (°F)	539
Sodium Side ΔP (psi)	15
LMTD (°F) (Subcooled/Boiling)	66/118
Overall Heat Transfer Coefficient (Btu/hr-ft^2-°F)	410/537
Pinch Point T (°F)	41.3
Steam Generator, Dry Weight (Tons)	430
Steam Drum, Dry Weight, (Tons)	170
Tube Material	2-1/4 Cr-1 Mo
Tube Construction	Prestressed Double Wall
Outer Tube o.d./i.d. (in./in.)	1.25/1.096
Inner Tube o.d./i.d. (in./in.)	1.096/0.946
Number of Tubes	1838
Tube Length (ft) (Total/Active)	60/55
Protector Tube o.d./i.d. (in./in.)	1.52/1.27

TABLE 5.6-2 - (Continued)

PRISM STEAM GENERATOR
THERMAL/HYDRAULIC DESIGN SIZING DATA

PARAMETER	VALUE
Protector Tube Material	304 SS
Protector Tube Length (ft)	6.5
Triangular Pitch (in.)	2.375
Overall Sizing Margin (%)	28
Plugging Margin (%)	3
Fouling Margin (%)	15
Uncertainties Margin (%)	13
*Maximum i.d. Heat Flux (Btu/hr-ft^2)	153,000
Heat Transfer Area (ft^2)	33,080

*Maximum Heat Flux in clean condition is 183,000 Btu/hr-ft^2

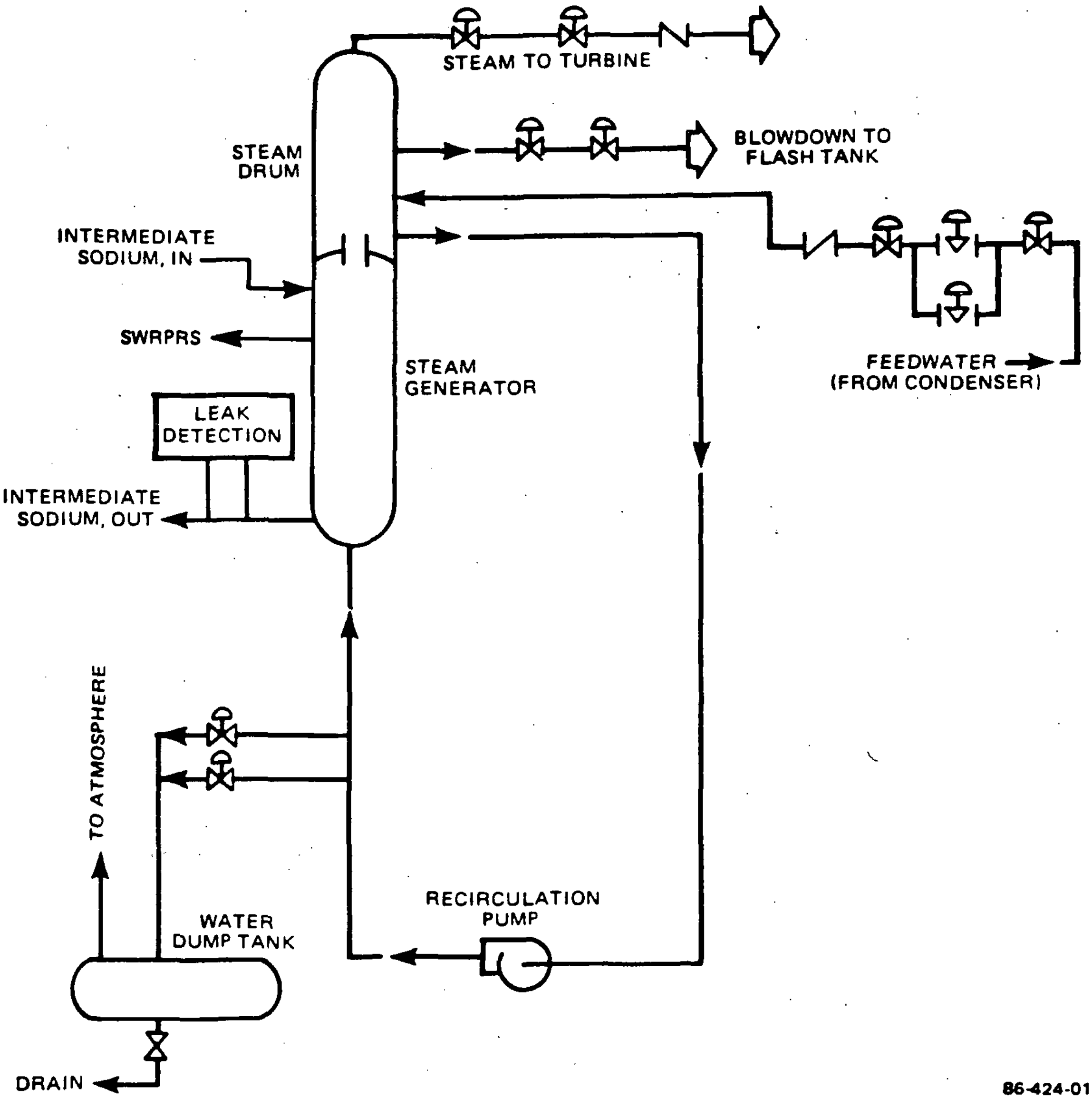


Figure 5.6–1 PRISM STEAM GENERATOR SYSTEM FLOW DIAGRAM

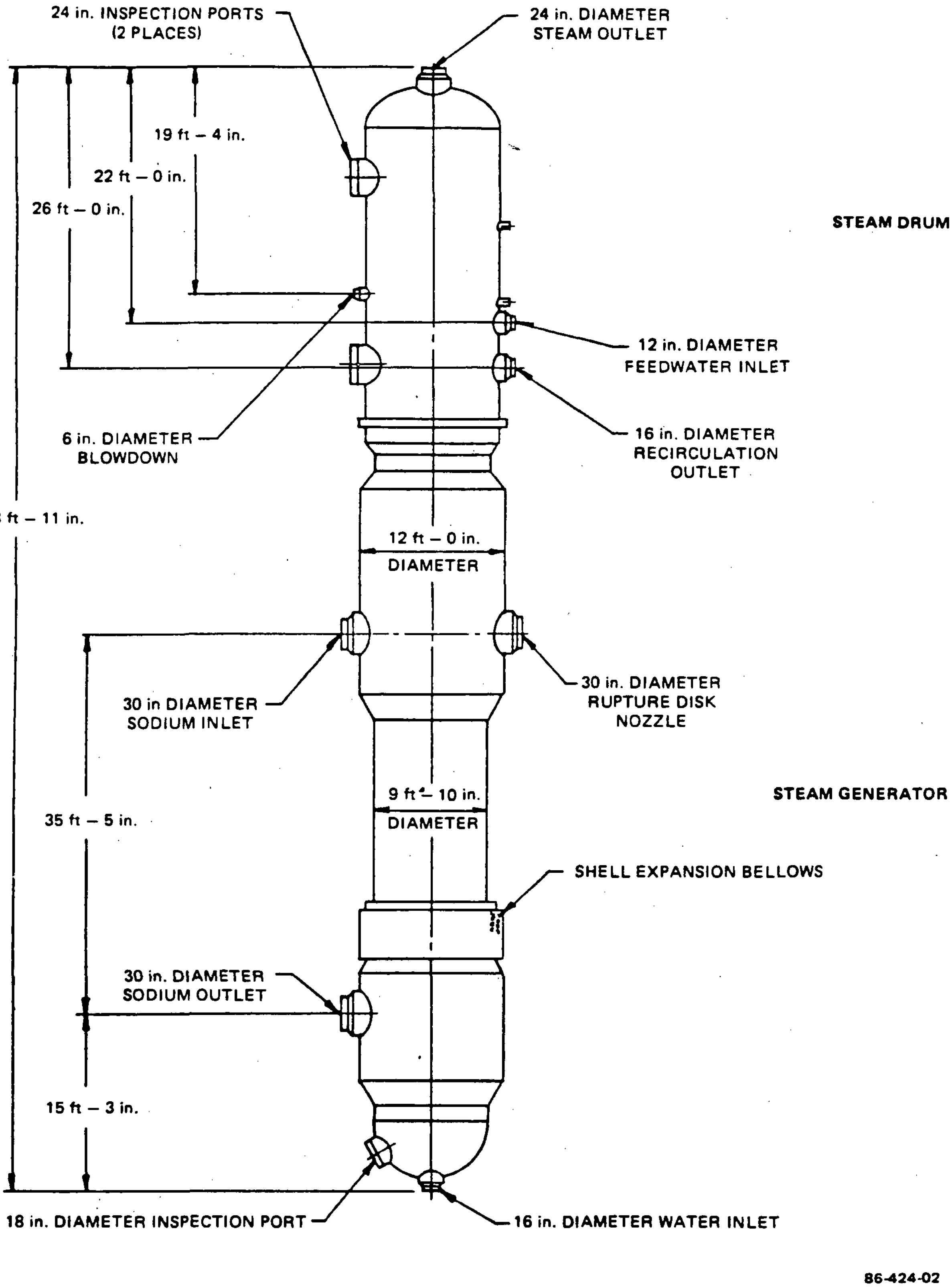


Figure 5.6–2 PRISM STEAM GENERATOR WITH INTEGRAL STEAM DRUM

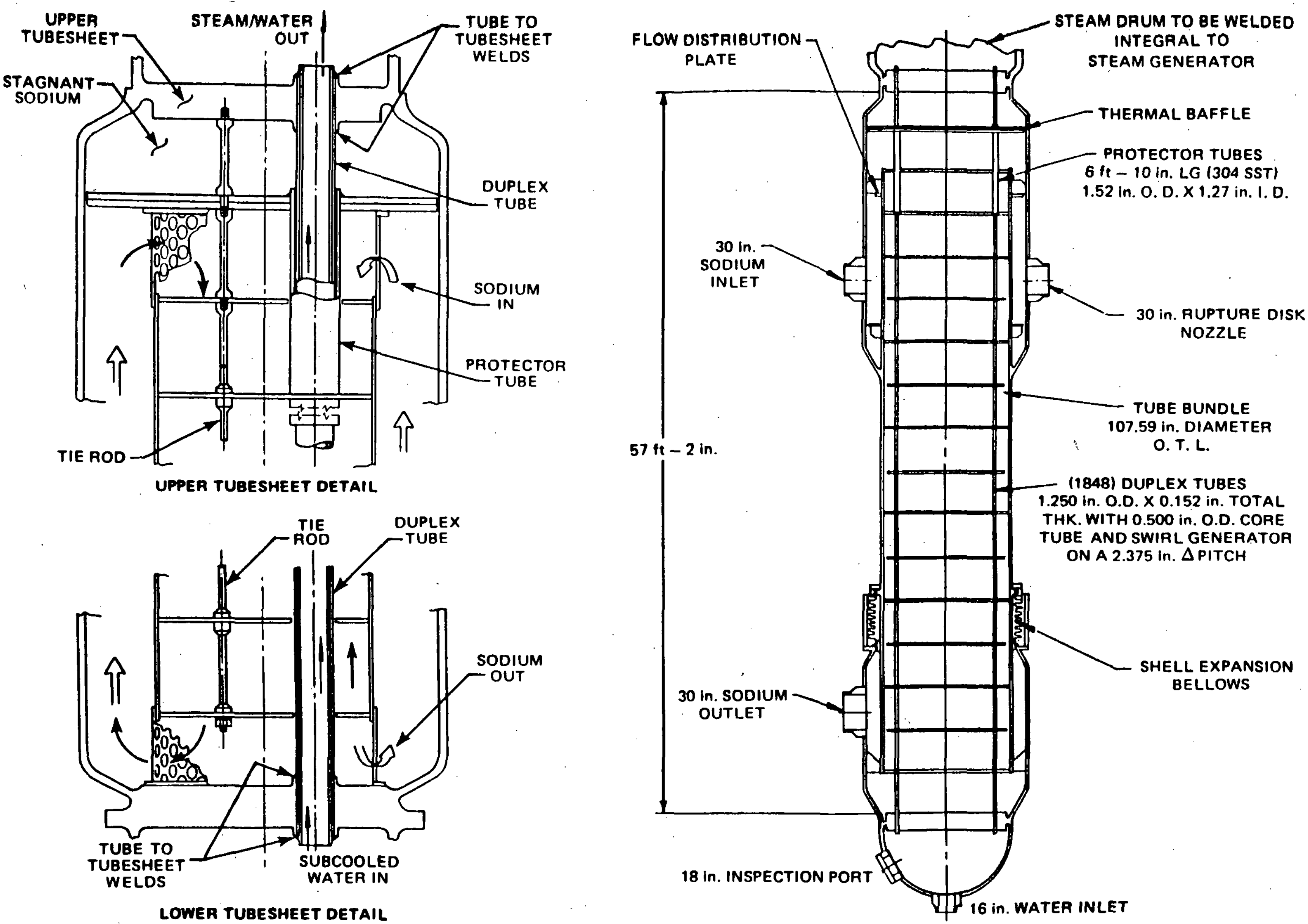
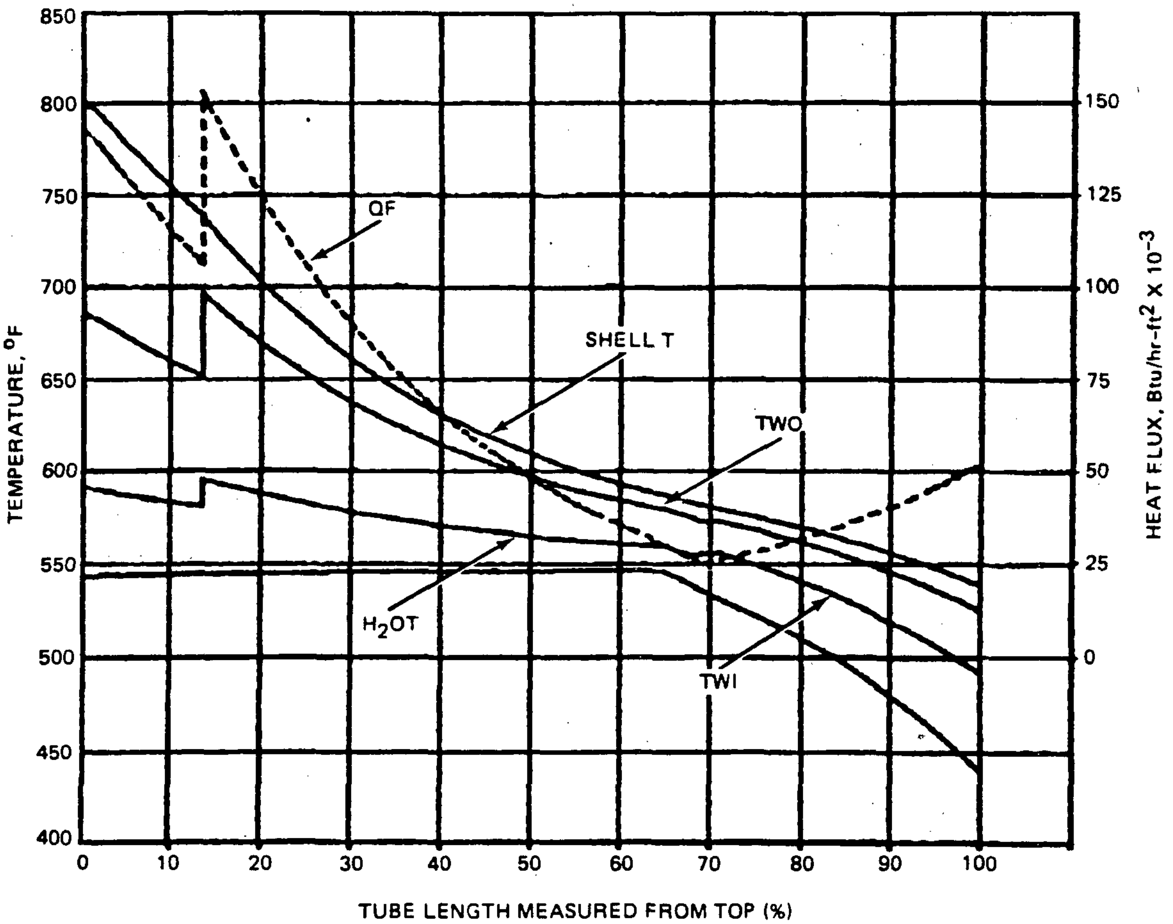


Figure 5.6–3 PRISM STEAM GENERATOR INTERNALS



NOTES

QF = HEAT FLUX (ID) (RIGHT HAND SCALE)
SHELL T = SODIUM TEMPERATURE
TWO = OUTER DUPLEX TUBE (OD) TEMPERATURE
TWI = INNER DUPLEX TUBE (ID) TEMPERATURE
H_2O T = WATER/STEAM TEMPERATURE

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Figure 5.6—4 PRISM STEAM GENERATOR TEMPERATURE AND HEAT FLUX PROFILE

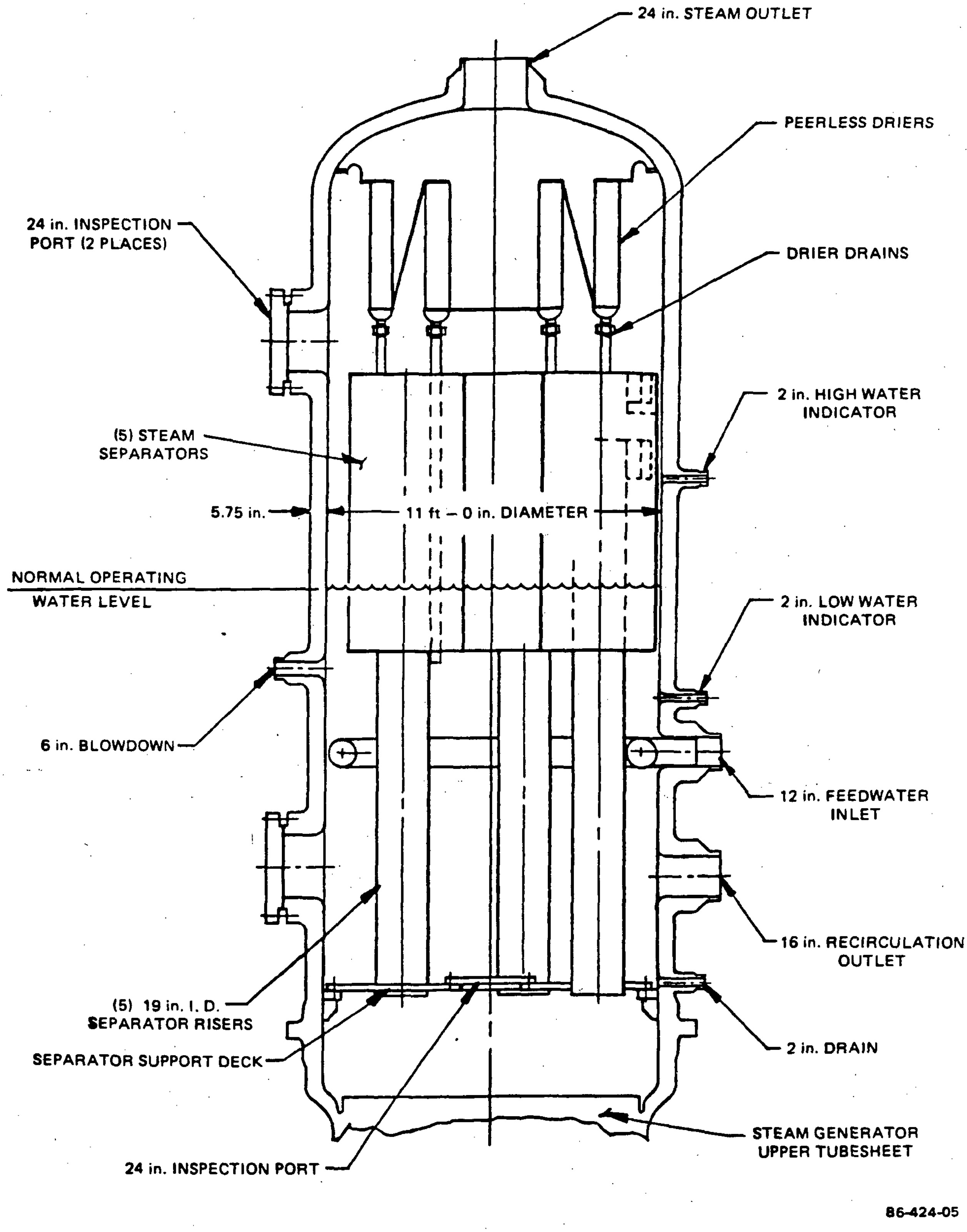


Figure 5.6–5 PRISM STEAM DRUM INTERNALS

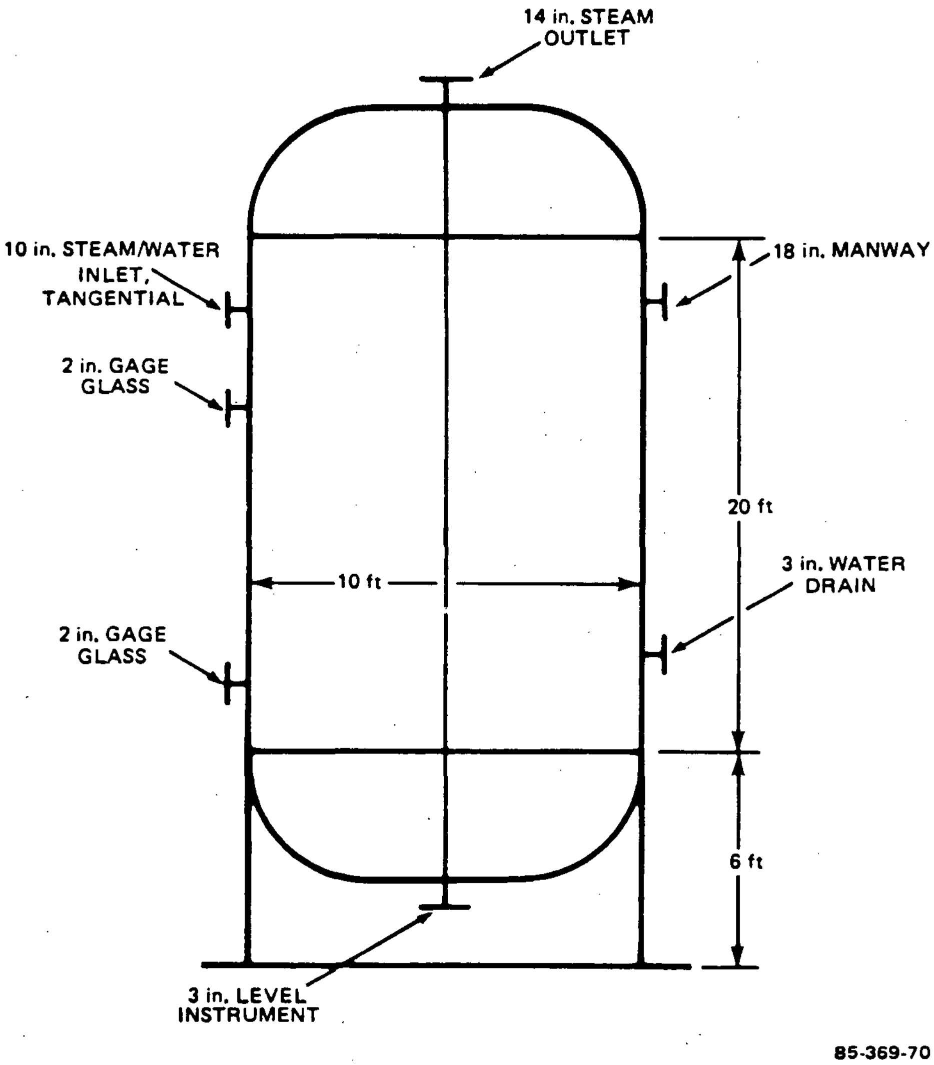
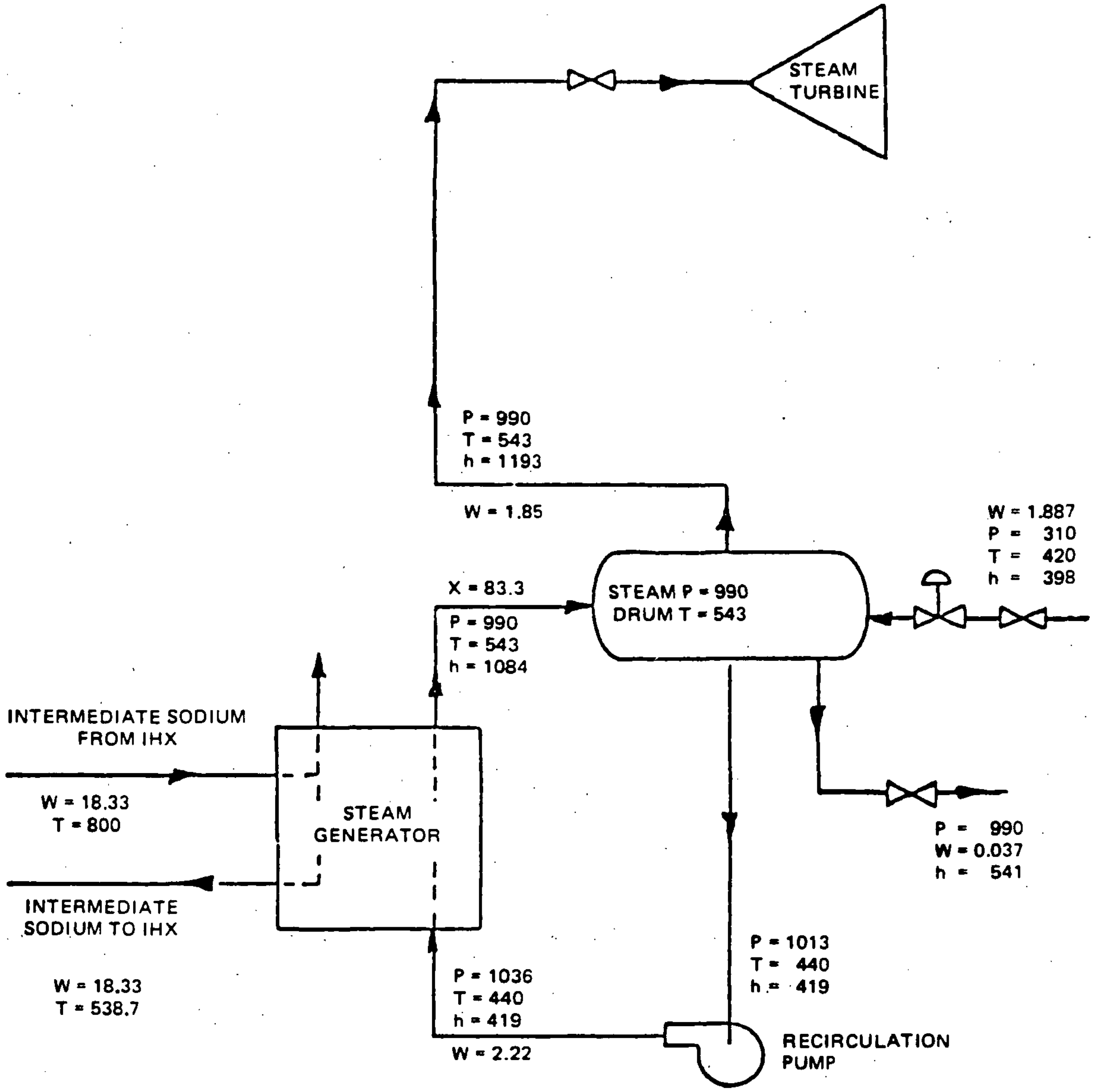


Figure 5.6–6 PRISM STEAM GENERATOR WATER DUMP TANK



W = lb m/hr x 10^{-6} T = °F	h = Btu/lbm P = psia	X = STEAM QUALITY

ASSUMPTIONS:

TUBE SIDE SURFACE	FOULED
HEAT TRANSFER COEFFICIENT	LOWER LIMIT
PLUGGING ALLOWANCE	3%

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Figure 5.6–7 STEAM GENERATOR SYSTEM THERMAL-HYDRAULIC DESIGN CONDITIONS

5.7 Shutdown Heat Removal Systems

5.7.1 Design Bases

The function of the PRISM shutdown heat removal systems are:

1. The reactor vessel auxiliary cooling system (RVACS) removes reactor decay and sensible heat following reactor shutdown at all times and is the final shutdown system when the normal heat rejection path through the Intermediate heat transport system and steam generator system are unavailable.

2. The auxiliary cooling system (ACS) provides additional (non-safety-related) removal of reactor decay and sensible heat following reactor shutdown when the normal heat removal path through the intermediate heat transport system is available but the steam/water system is unavailable.

5.7.2 Design Description

The shutdown heat removal systems provided in the PRISM plant are illustrated in Figure 5.7-1. The three methods involved use of condenser cooling, the ACS which removes heat from the the steam generators, and the RVACS which removes heat directly from the reactor.

Looking at Figure 5.7-1, the order of their use is essentially from right to left. Normally shutdown heat removal is by condenser cooling. Failing that, shutdown heat is removed from the steam generators by the ACS augmented by some steam venting initially when water is still available. If the water has been lost from the steam generator, ACS will work in conjunction with RVACS to reduce reactor and heat transport component temperatures and cool down the plant. If the IHTS is not available, RVACS will remove heat directly from the reactor vessel by natural circulation air flow. The estimated usage frequencies of the ACS and RVACS are ten times and less than one time per module life time, respectively. Condenser cooling system and ACS are constructed to high quality industrial standards whereas the RVACS is constructed as a safety-related system.

5.7.2.1 Reactor Vessel Auxiliary Cooling System

A unique and significant feature of PRISM is its reactor vessel auxiliary cooling system (RVACS). This system can dissipate all of the reactor's decay heat through the reactor vessel and containment vessel walls by radiation and convection to naturally circulating air outside the containment vessel without exceeding acceptable structural temperature limits. A description of the system and its performance is presented in the subsequent subsections.

<u>General System Description</u>

The RVACS operates continuously, but functions at its intended high heat removal rate only when all other reactor heat removal systems are inoperative. That circumstance has a probability for occurrence of less than once in sixty years. Should it occur, however, RVACS performs its function without any human or mechanical action. Primary sodium flow through the reactor core is maintained by natural circulation. The decay heat generated in the core is removed by the primary sodium and transferred to the reactor vessel. The heat is in turn transferred to the containment vessel, and then to air which naturally circulates as it is heated. The flow paths of the primary sodium and the air are illustrated in Figure 5.7-2. The primary sodium flow path is the same as the normal flow path initially but is slightly different at later times when the sodium has expanded and raised its surface level above the top of the reactor vessel liner so that the sodium flows behind the liner and down along the reactor vessel wall (instead of down through the intermediate heat exchangers as for normal operation). The rest of the sodium flow path is the same as for normal operation. The air flow path shown in Figure 5.7-2 is described below.

Air Flow Paths - Figure 5.7-3 is an isometric cut-away of the reactor silo and head access area (HAA) which includes the air passageways constituting the air flow path. Arrows show the flow directions.

Figure 5.7-4 presents plan views of the reactor silo and HAA to show the air inlet and outlet plena.

Atmospheric air enters the RVACS through four horizontal inlet openings in the concrete structure above grade. It turns downward and flows into the lower of two horizontal plena (the inlet plenum on Figure 5.7-4) under the HAA floor slab. Within the inlet plenum, the air flow turns horizontally in a radial direction toward the reactor module and then turns again, this time vertically down into the outer annuli around the module. The turns are required to provide radiation shielding. Near the bottom of the module, the air flow turns 180° and enters the air riser annulus, where it cools the containment vessel and is heated. The heating provides the natural draft head required to maintain air flow through the system.

Near the top of the module, the hot air turns to flow radially outward in the higher of the two horizontal plena. From the outlet plenum (Figure 5.7-3) the hot air enters four vertical ducts, each of which is centered in the vertical inlet air passageways near the inlet openings. The vertical ducts carry the hot air through the roof of the inlet air passages and exhaust it straight up into the atmosphere from four stacks several feet above the inlet openings.

Structures - Most of the structures which form the RVACS flow passages are made from concrete and structural steel because they also function as radiation and missile shielding. The collector cylinder which separates the annuli in the silo, the plate work which separates the horizontal plena, the vertical exhaust ducts, and the grill work (not shown) at the face of each air inlet and exhaust are made from structural steel only.

The collector cylinder is made from steel and is thermally insulated on its outside surface as is shown on Figure 5.7-5. This thermal insulation is required to minimize regenerative heating of the cold air in the downcomer annulus and to protect the concrete silo wall from excessive temperatures. The plena separator and the exhaust ducts are structural grade carbon steel, and the grills at the openings to the outside air are also carbon steel. The thick structures of the inlet air shafts are made from prefabricated cellular carbon steel assemblies into which concrete is poured.

Design Data - The air flow path of the RVACS has flow areas varying from 38 square feet to over 200 square feet with the minimum area occurring in the riser annulus next to the containment vessel. Air flow rates, velocities, and other thermal-hydraulic characteristics are given in Table 5.7-1. The steel structure temperatures for normal and RVACS operating conditions are given in Table 5.7-2. For concrete structures which could be subject to temperatures higher than 150°F, a refractory admixture is added that will allow a maximum operating temperature of about 250°F.

Table 5.7-3 shows the codes that are used for design, construction, and inspection of the RVACS structures. Construction of the collector cylinder is not covered by the ASME Code and will be governed by special design specifications yet to be developed.

5.7.2.2 Auxiliary Cooling System (ACS)

The ACS consists of a thermally insulated steam generator shroud, as shown in Figure 5.7-6, to form an annulus on the outside of the steam generator for natural air circulation. The shroud is made up of several sections that can be bolted together for installation and removal for steam generator in-service inspection. Steam generator building air enters the shroud at the base and exits near the top to the atmosphere through a stack. Louvers are provided at the inlets and outlet for flow control.

During normal plant operation, the louvers are closed to minimize steam generator heat loss. For reactor cooling, the louvers are opened to allow air circulation by natural convection which removes heat directly from the steam generator surface. Sodium flow in the IHTS is maintained by the intermediate pump pony motor while primary sodium flow through the core is maintained by natural circulation.

The ACS is not a safety-related system, but rather is used to increase plant availability by significantly reducing the reactor cool down time after water has been lost from the SG system.

5.7.3 Design Evaluation

5.7.3.1 RVACS Performance Summary

The thermal hydraulic analysis results presented in this section are for the design basis accident event referred to as an RVACS event. This event is an emergency or level C service event and consists of a normal scram with primary pump coast down and simultaneous loss of the intermediate heat transport system (IHTS). Thus, heat removal by the normal heat removal path through the steam condenser and through the ACS is precluded. The cause of the IHTS sodium loss can be either: (1) steam generator pressure relief rupture disc failure, (2) sodium dump as a result of a major sodium/water reaction in the steam generator, and (3) a major IHTS pipe break. The expected frequency of this event is only one (1) per module lifetime. It is only for these circumstances that the safety-grade RVACS is solely depended upon to remove shutdown decay heat from the reactor.

Heat removal during shutdown conditions by RVACS involves two main heat transport mechanisms or processes as indicated in Figure 5.7-2. First, there is the removal of heat from the core by natural convection of sodium and subsequent transport of the heat to the reactor vessel. The second process involves transport of the heat from the reactor vessel to the containment vessel and the collector cylinder primarily by thermal radiation and the subsequent heat removal from these structures by convection to a naturally circulating air stream. Both of these processes have been analyzed using a COMMIX model and the TSAP lumped parameter model. Performance summaries of the short-term transient from the COMMIX analysis and for the long-term transient using TSAP are presented in the following. Further details of these analyses are provided in Reference 5.7-1.

5.7.3.1.1 Short-Term Transient Results

Results for the transient calculation over its first 20 minutes are presented. The principal information of interest to this evaluation is to verify that the transition from forced flow to natural convection flow is smooth and regular and occurs without any major fuel temperature peaks. Therefore, the presentation will concentrate on these parameters. The temperature responses at other points in the reactor of interest to structural evaluations are presented in Reference 5.7-1.

The normalized driver node (I=1) flow velocity and heat source are shown in Figure 5.7-7. The plotted value for the heat source is the rate at which heat is transferred from the fuel to the coolant. Also, the velocity is used to represent mass flow. From 0-5 seconds the power-to-flow ratio is larger than 1 and the driver outlet temperature increases to a local peak as indicated in Figure 5.7-8. Power-to-flow ratio is less than 1 and the driver outlet temperature will decrease until system heatup causes the inlet temperature to increase. A second power-to-flow crossover occurs near 700 seconds as seen in Figure 5.7-7. The short-term driver temperature peak at 5 seconds is seen to be approximately 950°F followed by a rapid decrease of approximately 250°F over approximately 25 seconds yielding a transient cooldown rate on the order of 10°F/s. The driver outlet temperature decrease is halted at approximately 50 seconds by the increase in the reactor inlet temperature. The behavior of the driver outlet temperature at longer times is characterized by a steadily increasing trend as indicated in Figure 5.7-9.

The corresponding radial blanket flow and temperature responses are similar to the drive response except that the temperature peak is only ~835°F at about 5 seconds. Also, the temperature decreased from its peak by approximately 140°F over a time of about 100 seconds which is a much lower temperature transient than the driver received. The results also showed that the flow in the blanket assembly decreases to near zero at about 700 seconds which is also the time of minimum flow of about 1.5 percent of fuel power flow in the driver. This condition is being investigated further.

In summary, the COMMIX transient analyses show a smooth and regular transition to natural convection without excessive fuel temperature peaks. The driver flow rate reaches a minimum of about 1.5 percent of fuel flow near 700 seconds into the transient followed by a gradual increase.

5.7.3.1.2 Long-Term Transient Results

The characteristic of the long-term temperature transient experienced by the PRISM during RVACS event is illustrated in Figure 5.7-10 which gives the average core outlet temperature. There is a relatively rapid heat-up during the early stages of the transient followed by a relatively slow long-term cooldown transient. Maximum core sodium outlet temperatures of 1133°F and 1182°F are reached at 30 hours for both the expected and conservative cases analyzed.

The maximum sodium temperature peaks occur near the point where the RVACS heat rejection rate equals the decay heat generation as illustrated in Figure 5.7-11 for the conservative case. The RVACS heat rejection at peak sodium temperature is about 2.5 MWt as compared to about 0.9 MWt at normal reactor operating conditions.

The acceptable performance of RVACS is illustrated in Figure 5.7-10 since the average core sodium outlet temperature is considerably less than the 1200°F defined as the emergency or level C service limit.

The cooldown time to reach the hot standby condition with absence of any other cooling was projected to be 80 days for the conservative case. It was this long cooldown time and corresponding reduction in plant availability that led to introduction of an ACS (described in Section 5.7.2.2) employing air cooling of the SG shell.

Additional sodium and steel temperature transients are given in Figure 5.7-12 for the heat-up and peaking time period. The core sodium outlet temperatures represent the average for the entire core.

The sodium hot pool temperatures as seen from Figure 5.7-12 are approximately 30°F lower than the core outlet sodium temperatures and the average reactor vessel temperature is about 40°F lower at the top than the hot pool. The cold pool sodium temperature is approximately 115°F less than the hot pool.

The reactor vessel inside temperature is seen to be maintained slightly less than the hot sodium plenum at the top. The containment vessel temperature transients as seen from Figure 5.7-12 are considerably colder than the reactor vessel.

Overall, RVACS has sufficient heat removal capacity to meet all shutdown heat removal needs for the reactor. The system maintains primary sodium temperatures below the sodium 1200°F limit established for this condition. The temperature reached during this event is also low enough to prevent fuel cladding damage. No operator action is required to start RVACS. The system functions continuously with significant heat removal occurring when vessel temperatures go above the normal operating temperatures. RVACS performance can be readily monitored and easily maintained.

5.7.3.2 ACS Performance Summary

The RVACS event (described in Section 5.7.3.1) analysis was expanded to include the heat removal provided by the ACS. Results of this study are summarized in Figure 5.7-13 which gives core outlet temperature transients for two conditions RVACS cooling and RVACS cooling with ACS operation. The transient results show that the cool down time is reduced from 80 days to about 7 days. Also peak core sodium temperatures reached during the transient are reduced from less than 1200°F (RVACS cooling) to less than 900°F (RVACS cooling, plus ACS operation).

The above analysis illustrates the plant availability gain resulting from the use of the ACS. Slow plant cool downs, which would occur if only RVACS is used, are avoided.

5.7.3.3 RVACS Operation with Reactor Vessel Leak

A leak in the reactor vessel would cause the sodium level in the outlet plenum to fall to about one foot above the IHX inlet, and the annulus between the reactor vessel and containment vessel would fill with sodium to approximately the same level. With regard to shutdown heat removal, the effective length of the RVACS would be reduced by the amount of loss of height of sodium in the outlet plenum and the natural convection flow path for the primary sodium would be restricted to the path through the IHX, shield, EM pump and then back to the core inlet. Sodium flow in the annulus between the liner and reactor vessel would be by natural convection only because the sodium level would have fallen well below the overflow level. On the positive side, the RVACS would become more effective since the inert gas in the annulus between the reactor vessel and containment vessel would be replaced by sodium which is many times more thermally conductive.

To provide a better understanding of this case, the PRISM thermal hydraulic model discussed in Section 5.7.3.1 was modified to represent the reactor vessel leak condition and a computer run was executed for an RVACS only cooldown with nominal core decay power. Figure 5.7-14 gives a comparison of the predicted average core outlet temperature during normal RVACS operation with the temperature for the reactor vessel leak condition. The maximum temperature reached during RVACS only operation without a leak is about 1133°F at 30 hours after start of cooldown whereas the maximum temperature reached during the vessel leak condition is 1083°F at about 24 hours after start of cooldown. Hence, the reactor vessel leak condition does not pose a threat to the shutdown heat removal function of RVACS.

5.7.3.4 RVACS Tolerance Against Fouling and Flow Blockage

The RVACS is not subject to the types of failure modes associated with conventional active cooling systems which employ pipes, valves, pumps, and dampers. Although no credible failure modes have been identified, two postulated challenges to heat removal have been analyzed. These two challenges can be categorized as: (1) blockage of the air flow passages, and (2) sodium aerosol fouling caused by the ingestion of NaOx, the reaction products produced by a sodium fire in the IHTS.

With respect to flow blockages, the most likely blockage location is at the bottom of the silo where the air stream turns upward. Any rubble, debris, sand, or liquids would tend to collect at the bottom of the channel. This could eventually prevent the air from turning and flowing under the lower edge of the collector cylinder. However, most blockage phenomena would not lead to complete blockage of the air passageway due to the nature of the event and the permeability of the rubble. In addition, ISI and on-line performance monitoring during normal operation can easily detect incipient blockage via air exit temperature and flow rate measurements.

To evaluate these effects, several flow blockages were analyzed. The results show that a 30 percent flow rate would be required to limit the maximum core outlet sodium temperatures to less than the faulted (Service Level D) limit of 1300°F. The probability of such severe air flow blockage occurring is extremely small. The analysis also shows that heat rejection into the surrounding concrete and earth structures alone is not sufficient to prevent sodium boiling.

A thermal-hydraulic model for RVACS was used to perform a number of parametric evaluations of the RVACS thermal performance under fouled and/or degraded emissivity conditions. Results show that the RVACS is very tolerant to unusual and unexpected operating conditions. For example, deposition of a 0.25-inch thick layer of NaOx on the RVACS heat transfer surfaces (more than 6000 lb of plateout) as a result of a major IHTS sodium spill and fire increased the expected peak sodium temperature during a long-term RVACS transient by only 48°F. This preliminary analysis indicates that credible design basis IHTS sodium fires will not be capable of preventing RVACS from protecting the health and safety of the public.

5.7.4 Tests and Inspections

The RVACS is a nuclear safety-related system which is essential to safe reactor cooldown following a loss of flow accident. RVACS operates at all times and is a completely passive system. The RVACS system will be continuously monitored by measuring air flow rate and exit air temperature, and by monitoring for water intrusion, radiation and fire/smoke. Periodic remote visual examination of the RVACS flow passage will be performed to check for blockage and to inspect the collector cylinder integrity. The inspection will be done by inserting a remote miniature TV camera through the RVACS inlet vents or outlet stacks. Testing of the reactor vessel emissivity coating for degradation also will take place by installing material coupons in the reactor vessel/containment vessel annuli and periodically removing a coupon for examination.

References - Section 5.7

5.7-1 Hunsbedt, A. (Editor), "Thermal Hydraulic Analysis," General Electric Company Report GEFR-00782, August 1986.

TABLE 5.7-1

RVACS AIR FLOW CONDITIONS

AIR FLOW IN ANNULI AROUND REACTOR MODULE

	NORMAL REACTOR OPERATION		RVACS SHUTDOWN HEAT REMOVAL OPERATION	
	IN	OUT	IN	OUT
TEMPERATURE (°F)	100*	178	100*	305
RATE (LBM/SEC)	33	33	49	49
MAX. VELOCITY (FT/SEC)		8.2		13.8
HEAT REMOVED (MWt)		0.65		2.5
TOTAL SYSTEM ΔP (LBF/FT^2)		0.52		1.14

* DESIGN AMBIENT TEMPERATURE

TABLE 5.7-2

STRUCTURAL TEMPERATURES
NORMAL AND RVACS OPERATING CONDITIONS

STRUCTURE (1)	NORMAL REACTOR OPERATION		RVACS SHUTDOWN HEAT REMOVAL OPERATION	
	TOP (2)	BOTTOM (2)	TOP	BOTTOM
REACTOR VESSEL	760	600	1102	1003
CONTAINMENT VESSEL	441	340	778	654
COLLECTOR CYLINDER	263	202	556	413

(1) STRUCTURE TEMPERATURES (IN °F) ARE AVERAGES ACROSS WALL THICKNESS

(2) ELEVATIONS COINCIDING WITH TOP AND BOTTOM OF COLLECTOR CYLINDER

TABLE 5.7-3

RVACS STRUCTURES CODES & STANDARDS

CONSTRUCTION	
o REACTOR VESSEL	ASME SECTION III, CL. 1 & CC N-47
o CONTAINMENT VESSEL	ASME SECTION III, CL. 1 & CC N-47
o CONCRETE STRUCTURES	ACI349-80 SEISMIC CAT. 1
IN-SERVICE INSPECTION	
o REACTOR VESSEL	ASME SECTION XI, DIV. 3, SUBSECTION IBM
o CONTAINMENT VESSEL	ASME SECTION XI, DIV. 3, SUBSECTION IVC

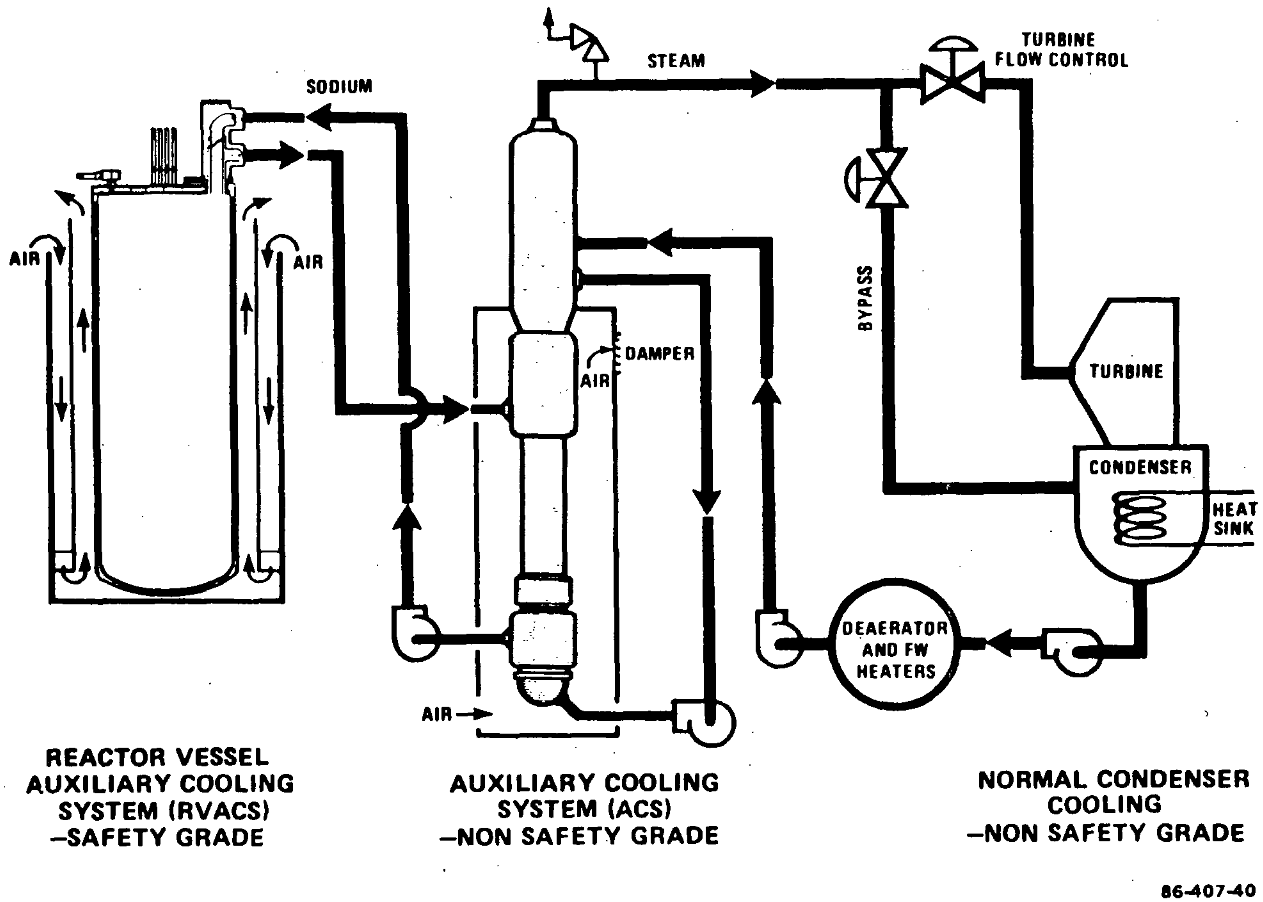


Figure 5.7–1 PRISM SHUTDOWN HEAT REMOVAL SYSTEM

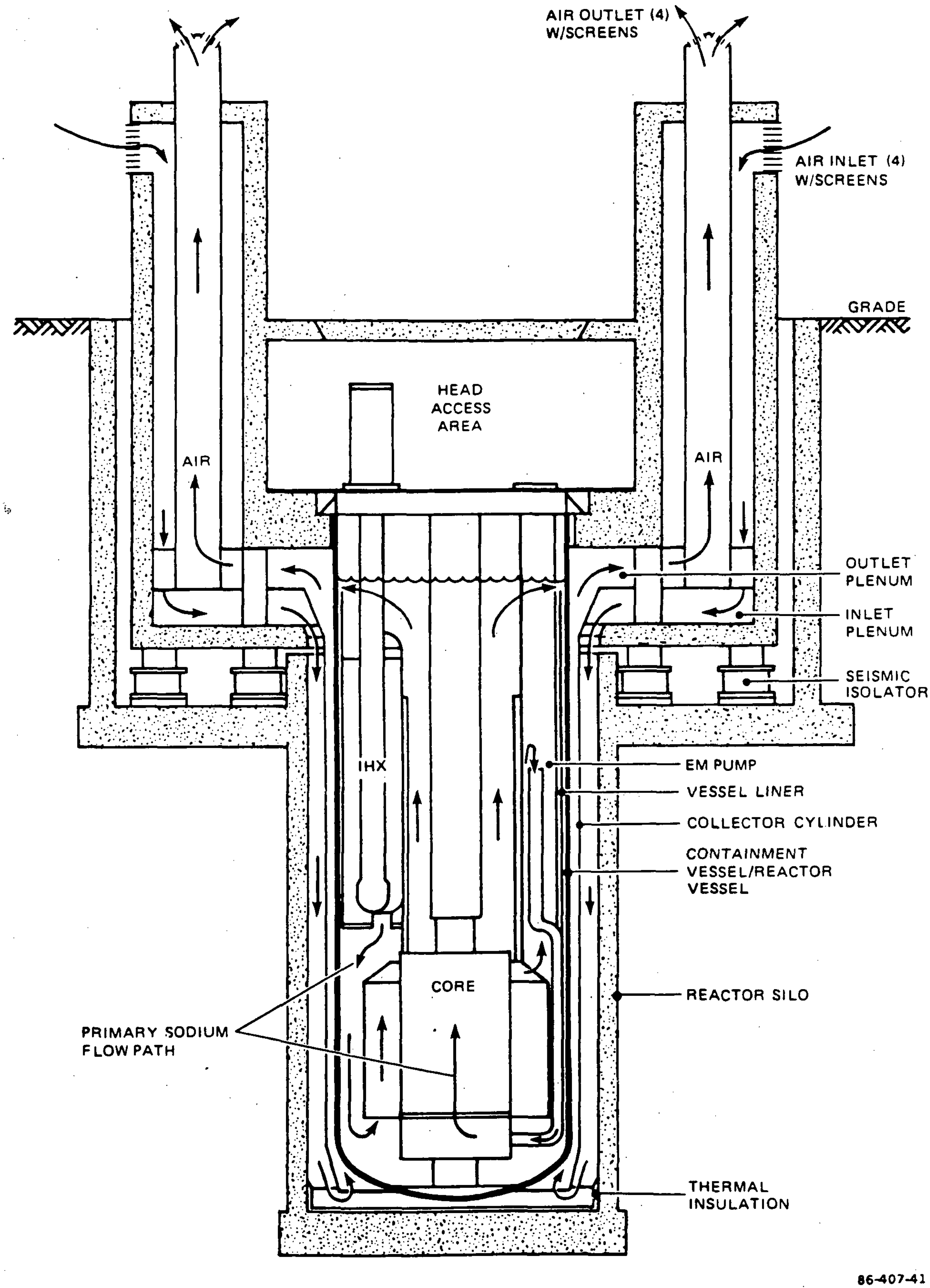


Figure 5.7–2 PRIMARY SODIUM AND RVACS AIR FLOW CIRCUIT DURING RVACS OPERATION FOR HEAT REMOVAL

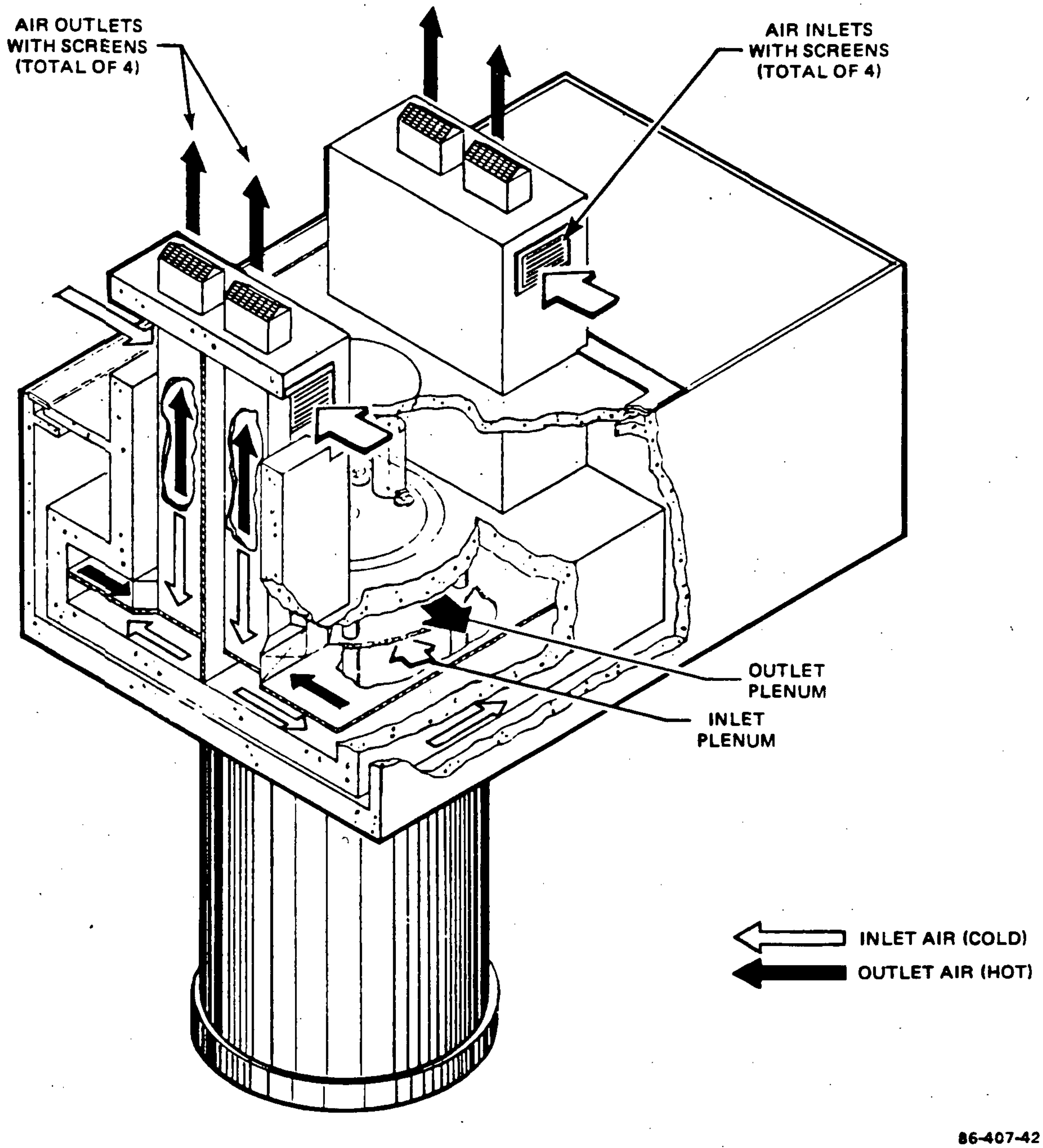
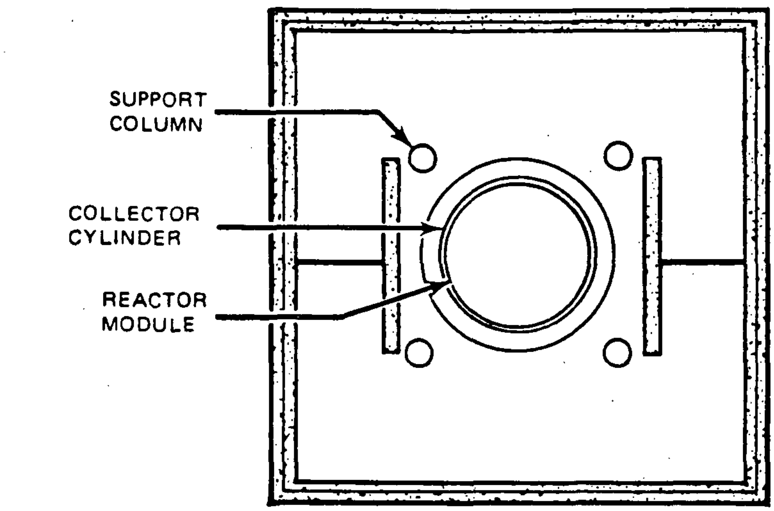
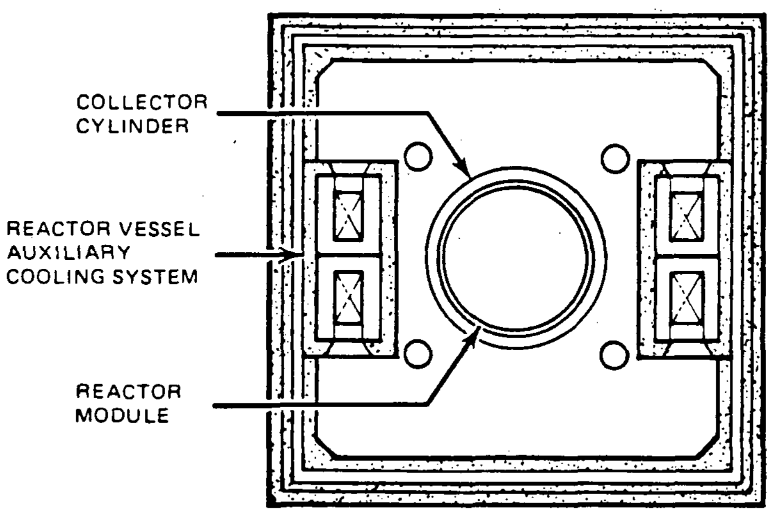


Figure 5.7–3 REACTOR SILO AND HEAD ACCESS AREA SHOWING RVACS AIR PASSAGEWAYS



INLET PLENUM



OUTLET PLENUM

86-407-43

Figure 5.7—4 PLANS OF INLET AND OUTLET PLENA

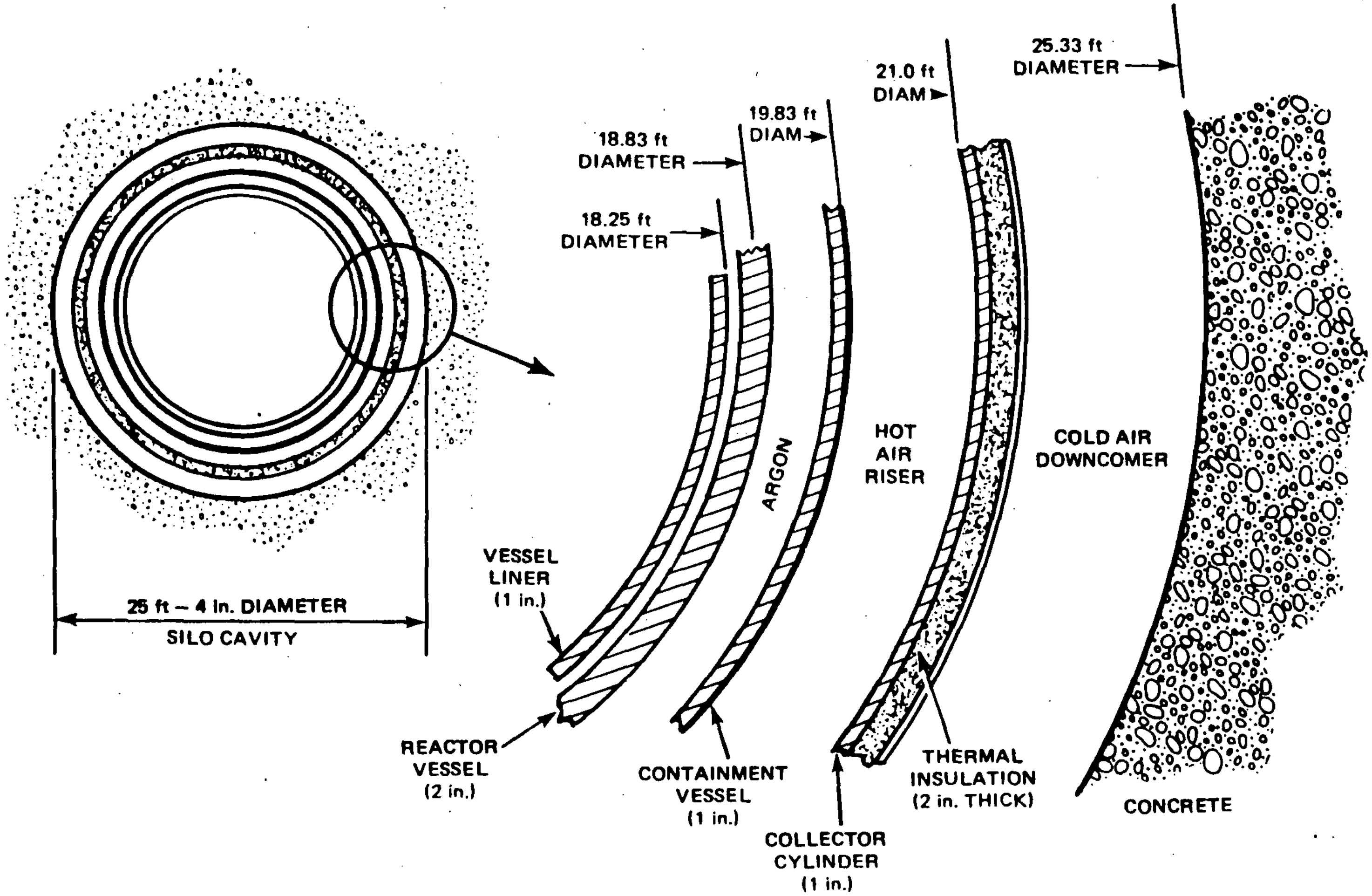


Figure 5.7–5 CROSS-SECTION OF RVACS IN REACTOR SILO REGION

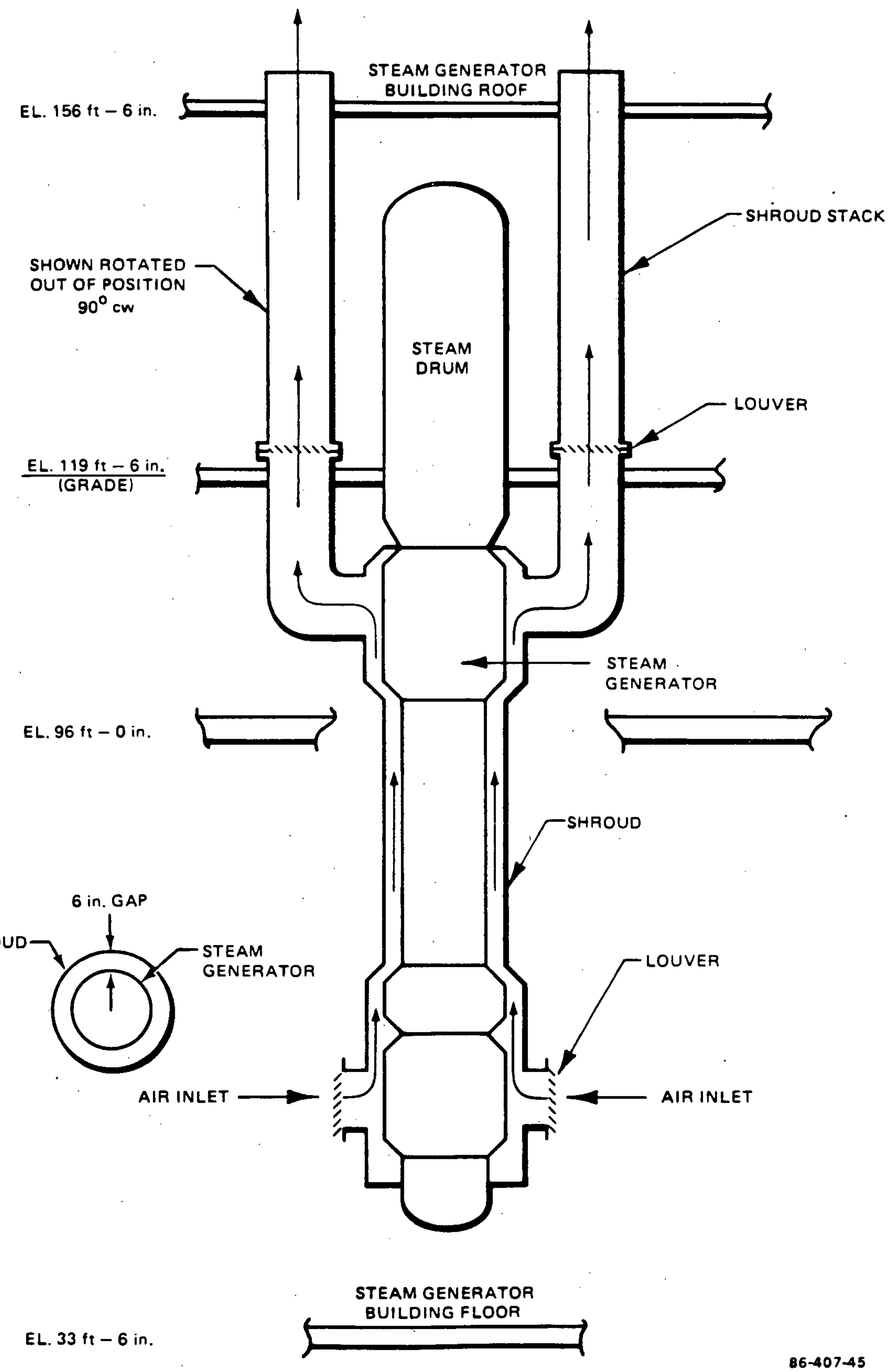


Figure 5.7–6 PRISM AUXILIARY COOLING SYSTEM (SCHEMATIC)

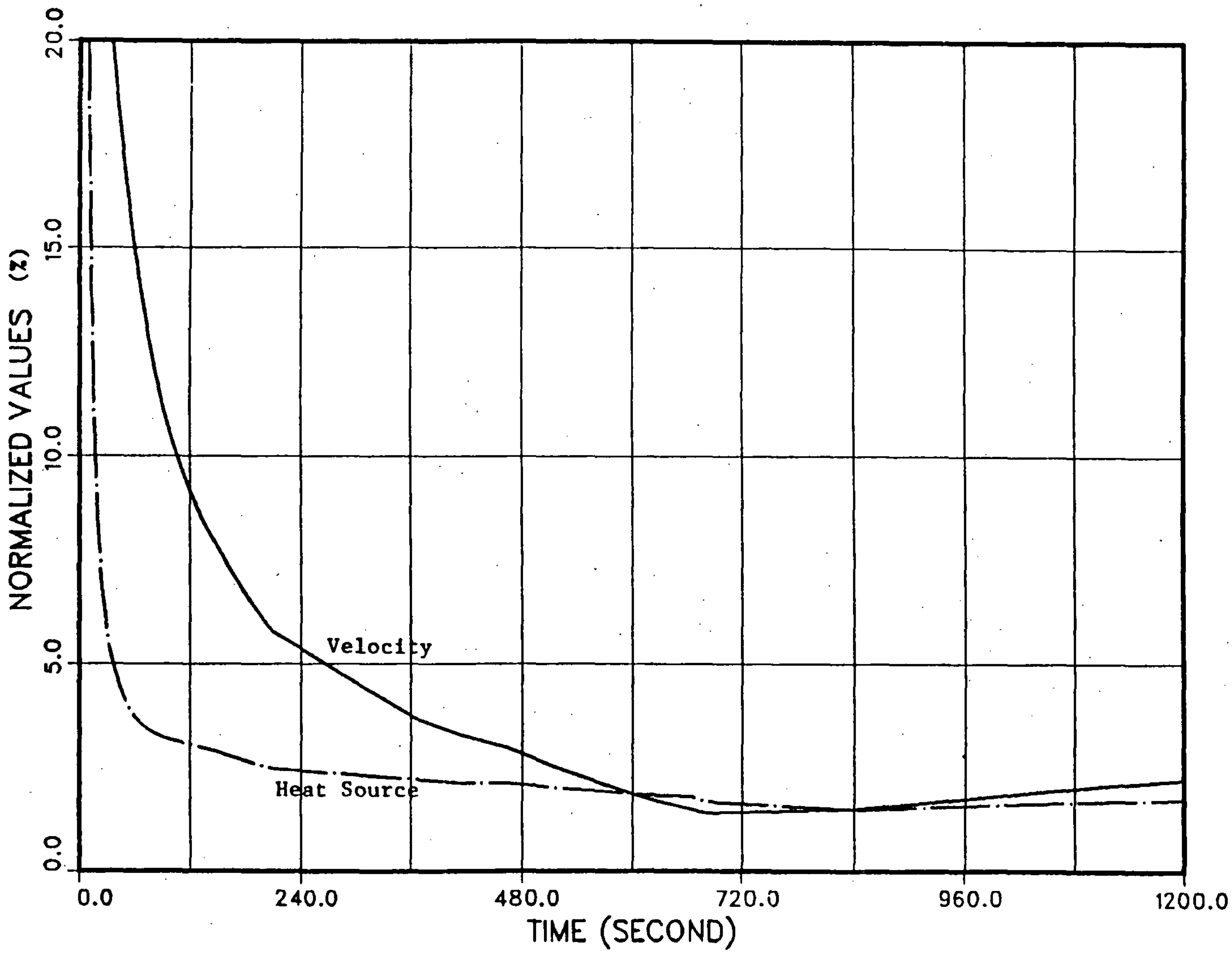


Figure 5.7–7 DRIVER FLOW VELOCITY AND HEAT SOURCE DURING RVACS TRANSIENT

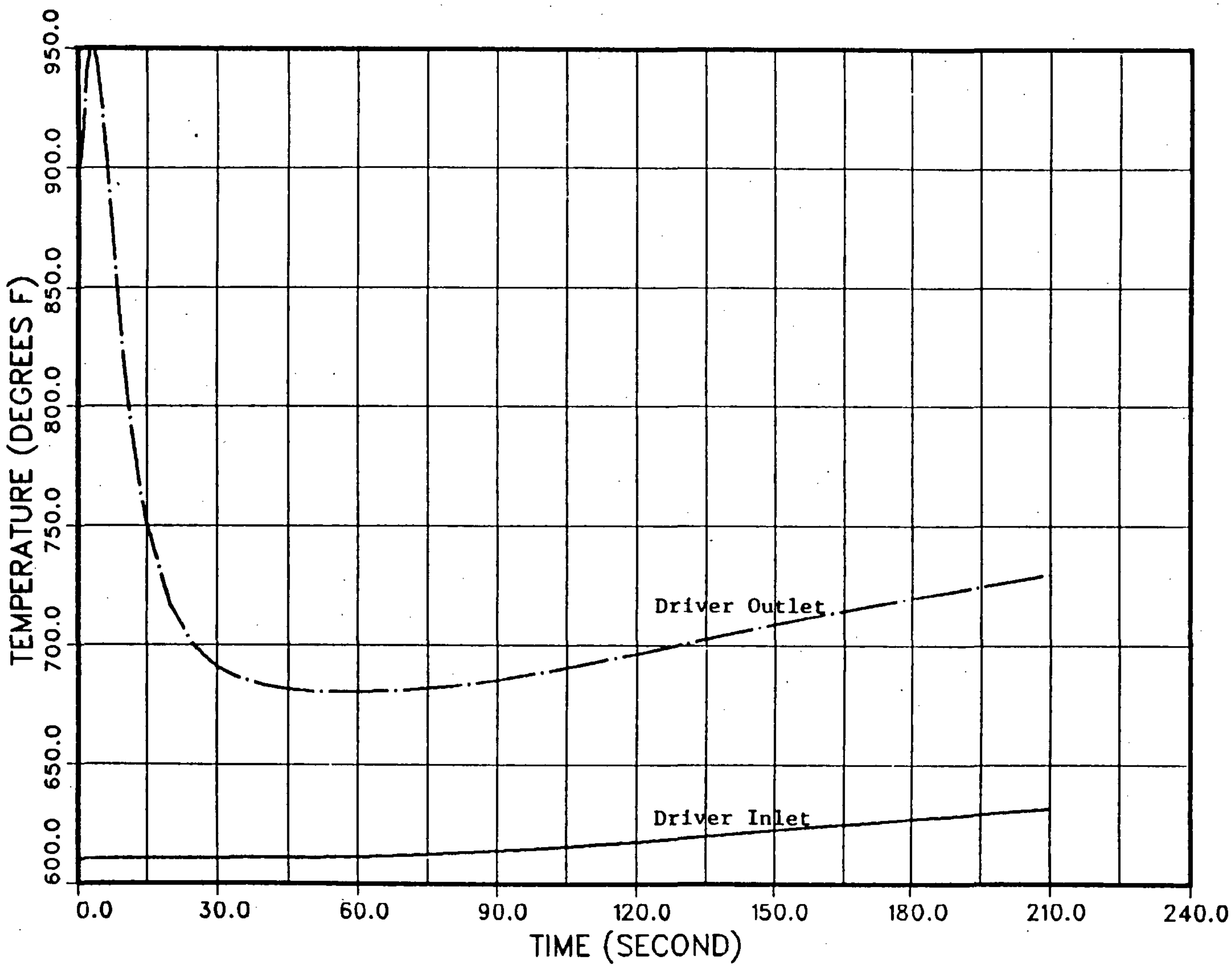


Figure 5.7–8 DRIVER INLET AND OUTLET TEMPERATURES DURING RVACS TRANSIENT

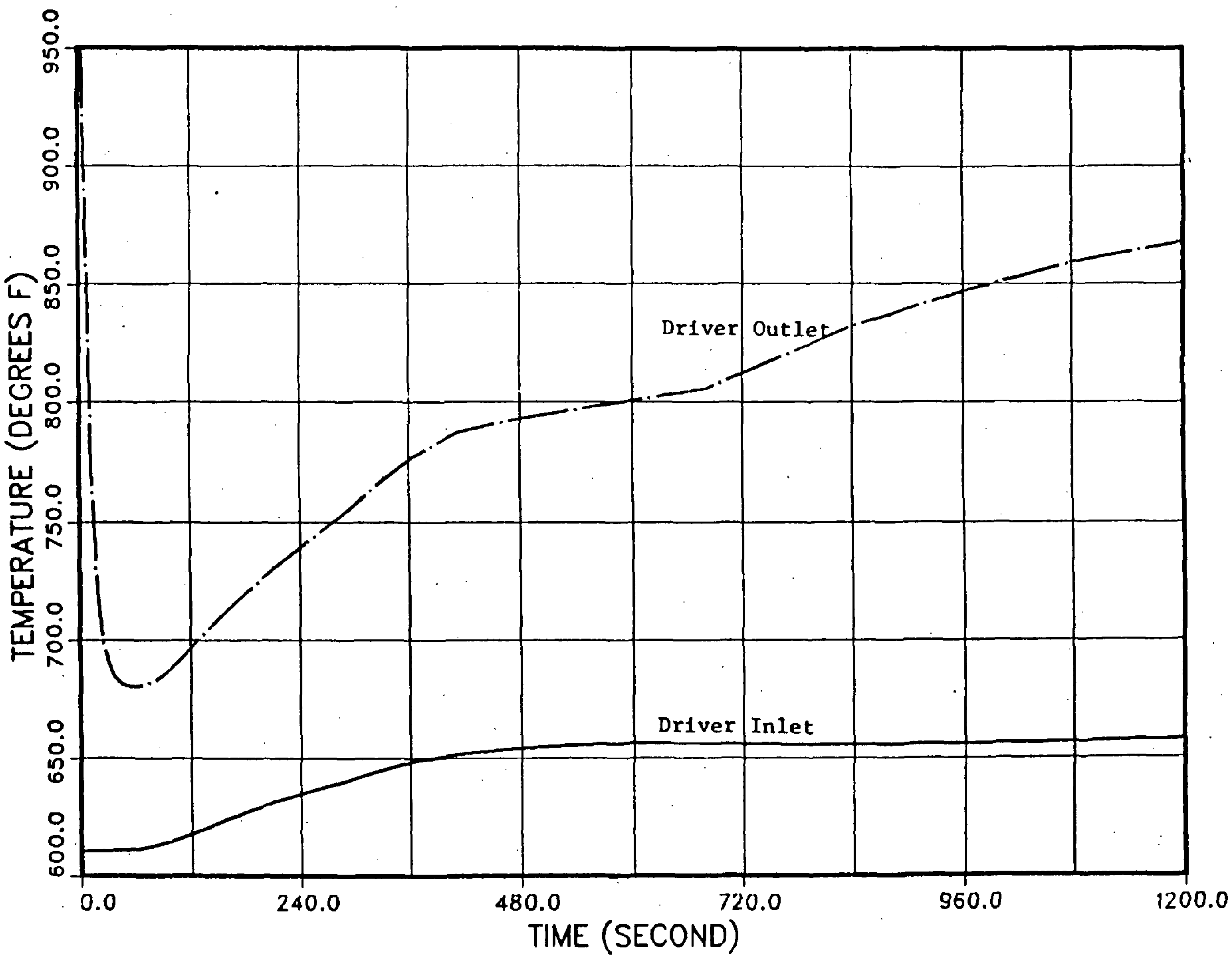


Figure 5.7–9 DRIVER INLET AND OUTLET TEMPERATURES DURING RVACS TRANSIENT

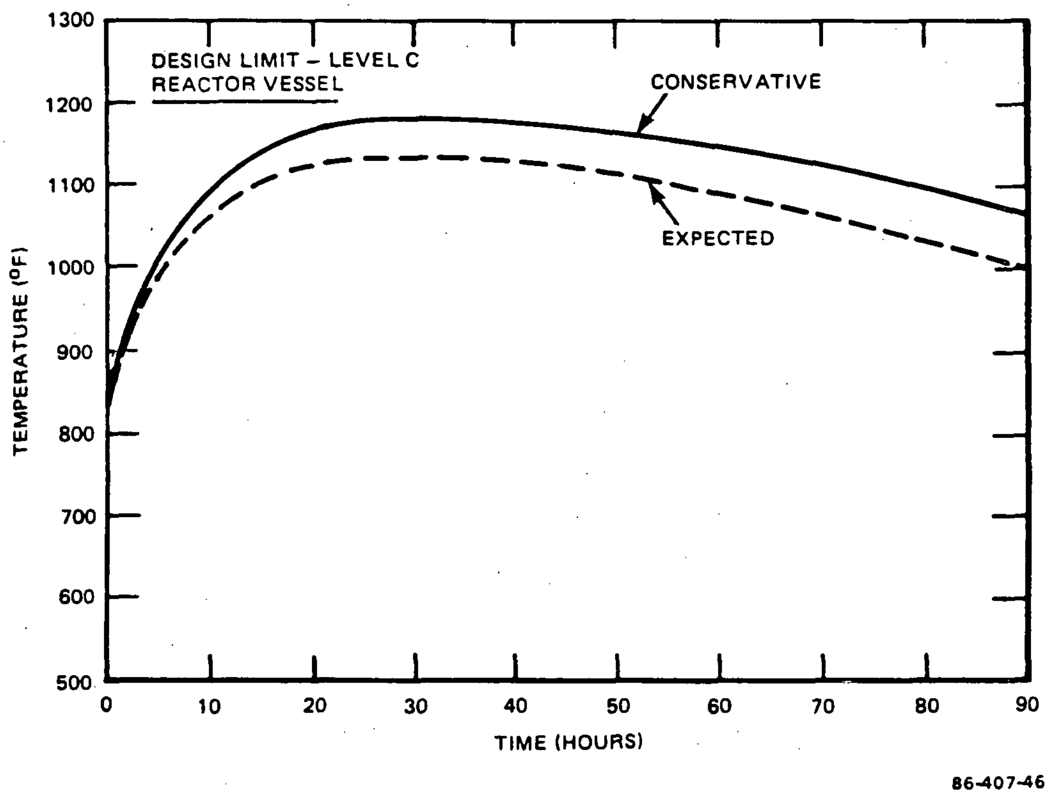


Figure 5.7–10 RVACS PERFORMANCE – AVERAGE CORE OUTLET TEMPERATURE AS A FUNCTION OF TIME

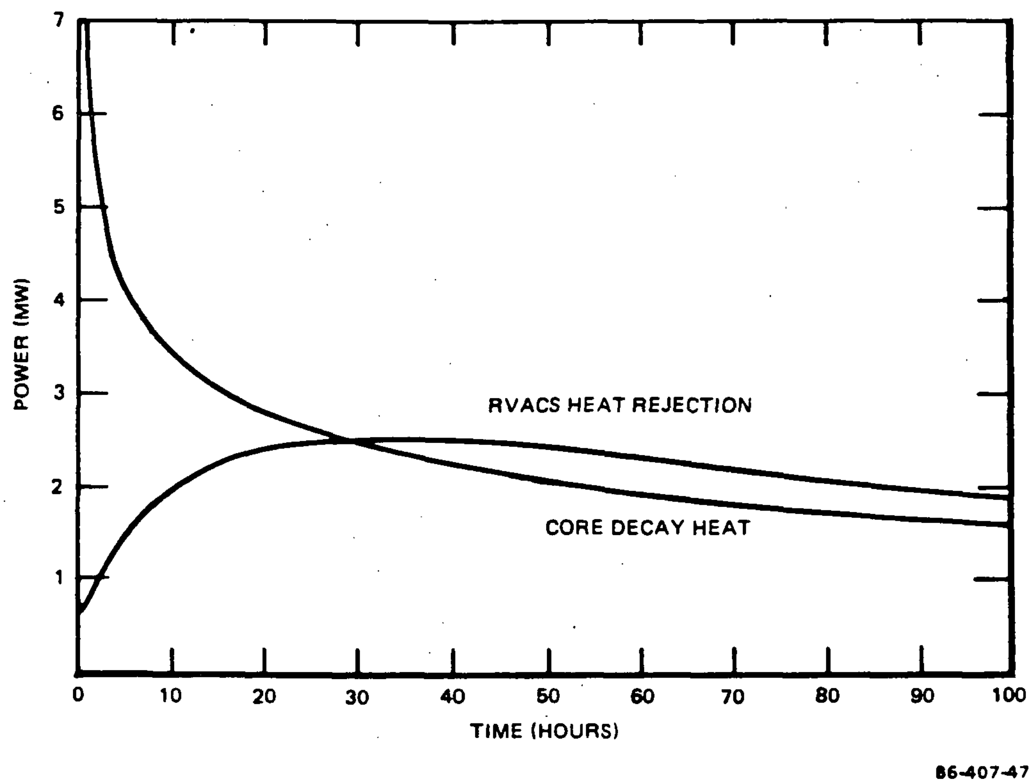


Figure 5.7–11 COMPARISON OF CORE DECAY HEAT TO RVACS COOLING

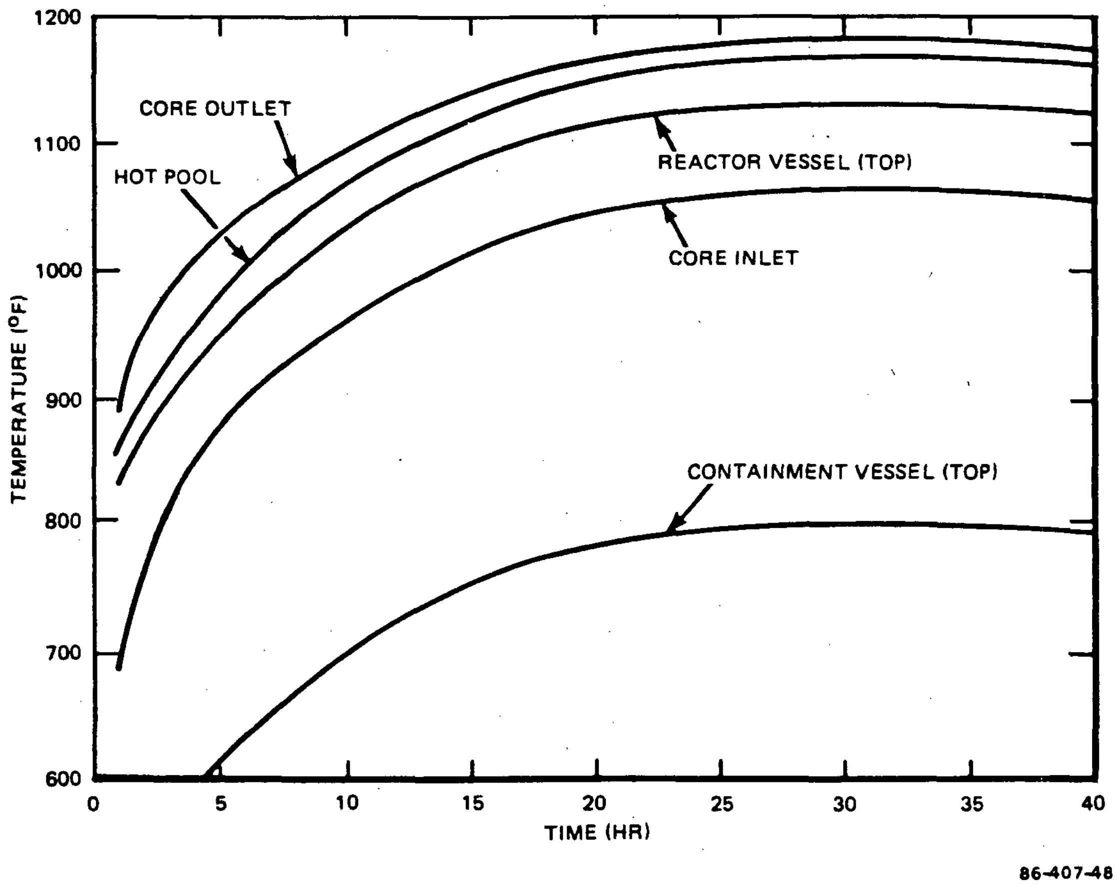


Figure 5.7—12 SODIUM AND VESSEL TEMPERATURE FOR AN RVACS EVENT — CONSERVATIVE CASE

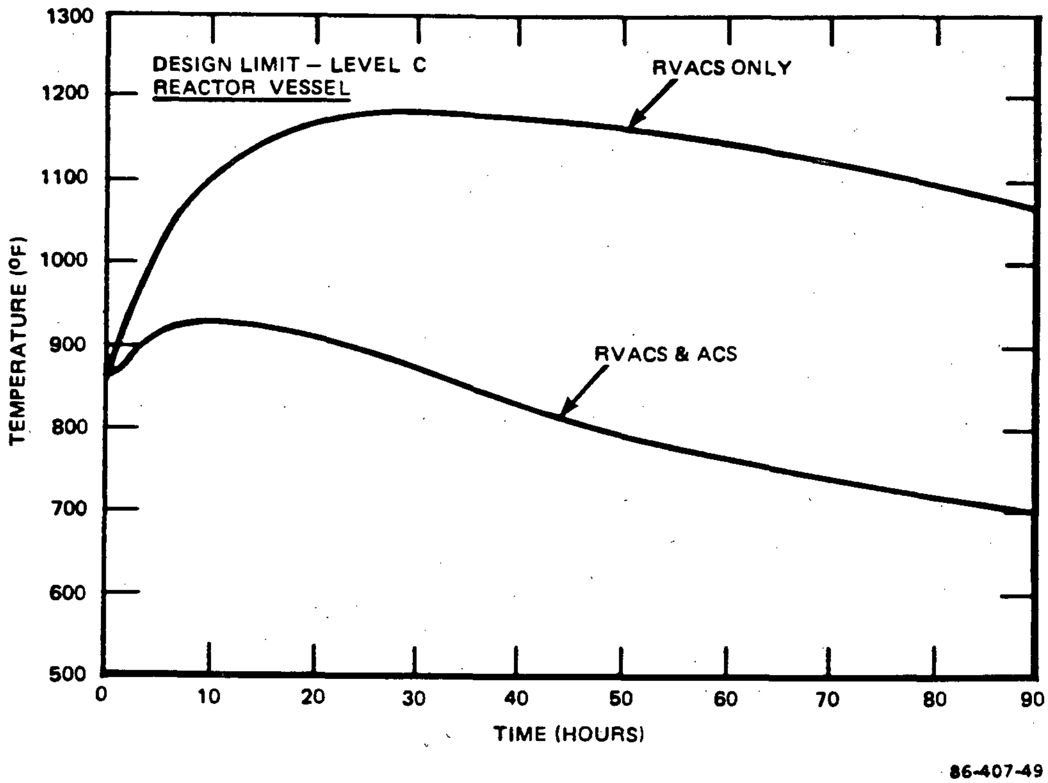


Figure 5.7–13 AVERAGE CORE OUTLET TEMPERATURE AS A FUNCTION OF TIME FOR RVACS ONLY AND RVACS PLUS ACS

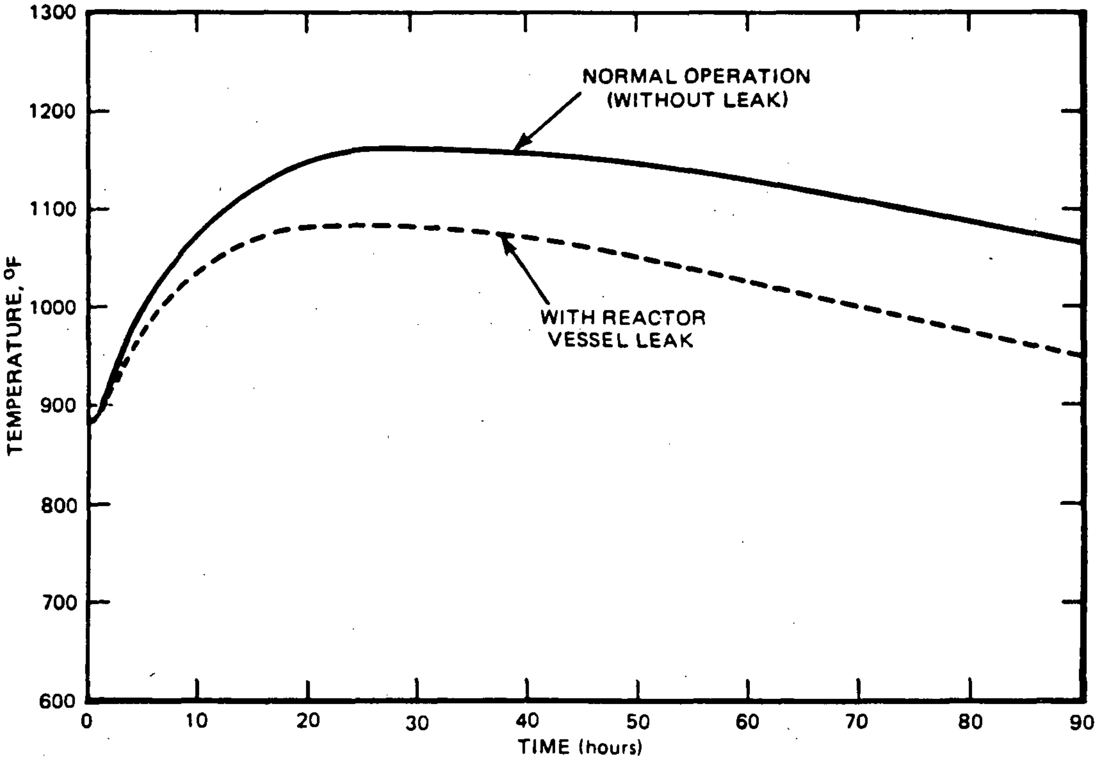


Figure 5.7–14 RVACS OPERATION WITH REACTOR VESSEL LEAK NOMINAL DECAY POWER

5.8 In-Service Inspection of Components

An in-service inspection assessment has been developed for the reactor module based on the requirements of Section XI, Division 3, of the ASME Boiler and Pressure Vessel Code (B&PVC), "Rules for In-Service Inspection and Testing of Components of Liquid Metal Cooled Plants." This section summarizes the assessment of the PRISM inspectability.

Section XI of the ASME B&PVC specifies the general type and extent of the In-Service Inspection (ISI) required for Class 1, 2 and 3 nuclear power plant systems or components. Division 3 of Section XI applies specifically to liquid metal cooled plants. The intent of the ASME ISI requirements is to maintain the nuclear power plant and to return the plant to service, following plant outages, in a safe and expeditious manner. The rules require a mandatory program of examinations, testing, and inspections to evidence adequate safety.

Only those systems and components classified as nuclear safety class systems are required to conform to the ASME requirements. The following are classified as nuclear class systems or components:

- the reactor internals;
- the reactor vessel and closure;
- the containment vessel;
- the reactor vessel auxiliary cooling system (RVACS).

The other systems, including the intermediate heat transport system, are classified as high quality industrial standard systems which are non-nuclear safety class. The non-nuclear class systems are not required for safe shutdown of the plant and the ASME Code ISI requirements do not apply to these systems. However, the PRISM ISI plan includes the non-nuclear class systems and components. The ISI plan for the non-nuclear class systems was developed with the intent to ensure safe and economic operation.

The Code describes a number of available ISI techniques which fall into the following general categories:

- o Visual examinations
- o Surface examinations
- o Volumetric examinations
- o Continuous monitoring
- o Alternative examinations

For the pool type PRISM reactor and the intermediate heat transport system, visual examinations and continuous monitoring are the primary inspection techniques. Visual examinations detect the appearance of a component and can be direct, remote, or can use less conventional equipment such as dimensional gaging and under-sodium scanning. Continuous monitoring detects liquid metal and inert gas leaks.

A preservice inspection must be performed before the plant is placed in commercial service. Following the start of commercial operation, the ASME Code allows two alternate inspection program schedules. These schedules are presented in Tables 5.8-1 and 5.8-2. An ASME code case will be required to extend these tables from the current 40-year plant life to 60 years required by PRISM. During the first three years of commercial operation, a plant may choose to switch inspection schedules. After that time, the plant must adhere to the chosen schedule. Each inspection interval may be extended or decreased (but not cumulatively) by as much as 1 year. In addition, if the plant experiences an extended outage, six months or more, the inspection interval may be extended for a period of time equivalent to the outage time.

5.8.1 Nuclear Class Components

ASME Section XI, Division 3 ISI requirements were developed specifically for liquid metal cooled reactors; therefore, the requirements account for the differences between liquid metal cooled reactors and water or gas cooled reactors. Besides the obvious characteristic of liquid metal opacity, liquid metal cooled reactors are often fabricated from austenitic stainless steels. Stainless steel exhibits good ductility and toughness at typical LMR operating conditions justifying a leak-before-break strategy for piping and vessels. These same austenitic stainless steel properties indicate that measurable deflections or dislocations will occur before structural failure of the supports. Thus, it is not necessary to examine all areas of each component or system. Rather, the ISI program is aimed at determining the general condition and operability of the reactor and detecting problems before they become significant.

5.8.1.1 Reactor Vessel and Closure

5.8.1.1.1 Reactor Vessel

The reactor vessel is a Class 1 component protected by the containment vessel. The ISI requirements for the reactor vessel are given in examination category B-A of Table IMB-2500-1 of the Code, "liquid metal retaining welds in vessels protected by guard vessels". A combination of visual (VTM-2) examination and continuous monitoring is specified by the Code for the reactor vessel, but an alternative examination method is planned using a combination of continuous monitoring methods with remote visual examination used only for follow-up examination.

The Code VTM-2 examination is conducted on the outside surface of the component. The chosen ISI method must be able to discern "accumulations of liquids, liquid streams, liquid drops, and smoke." The Code allows for either direct or remote visual examination using aids such as periscopes and TV cameras. VTM-2 examination does not require the removal of external coverings such as insulation.

The reactor vessel is enclosed in the containment vessel. As an inherently safe design feature, the annulus size between the reactor and containment vessels is a nominal five inches. The small annulus size ensures core coolability in the event of a major reactor vessel leak, as the IHX primary system inlet will remain covered with coolant even if the annulus fills to the height of the pool sodium. Although this feature greatly enhances the inherent safety of the PRISM reactor, the small annulus size does not permit the use of a remote viewing device which can be maneuvered laterally to examine all the reactor vessel welds. Therefore, for the PRISM reactor an alternate examination method, a combination of continuous monitoring techniques with remote visual examination used for follow-up examination, will be used.

The annulus will be continuously monitored for sodium leaks using sodium ionization detectors (SID) and continuity detectors or sparkplugs. Filter analysis devices and gas analysis will be used on a periodic basis. The filters and a gas sample also will be analyzed if one of the continuous monitoring devices indicates a leak to verify the leakage indications. A pressure transducer will also be installed in the annulus. Several ISI access ports are provided around the annulus for insertion/removal of the continuous monitoring instrumentation and remote viewing equipment. During testing and operation at test facilities and experimental reactors, the continuous monitoring devices have proven to be very sensitive to sodium leaks. Sodium level transducers and cover gas pressure transducers located inside the reactor also will detect any large leak through the reactor vessel.

If a leak indication is given by two or more monitoring devices, either continuous monitoring devices or the filter and gas analyses, remote viewing using a small TV camera will be used to verify the existence of and to try to identify the size and location of the leak. Miniature 2-inch diameter cameras are available for in-reactor inspection of nuclear power stations. A TV camera has an advantage over a boroscope or periscope in that it can be maneuvered through the curve on the bottom head of the annulus and gain visual access to the area just below the core.

5.8.1.1.2 Reactor Closure

The reactor closure is covered by two examination categories in Table IMB-2500-1 of the Code; B-C 'cover gas retaining components' and B-G 'bolting'. Both categories require continuous monitoring as the ISI method.

Continuous sodium aerosol and radiation monitoring shall be conducted on the exterior of the liquid metal or radioactive gas containing components in the head access area (HAA). These systems will be able to detect a failure in the liquid metal or gas boundaries from the resultant presence of sodium aerosol, sodium vapor or radiation.

5.8.1.2 Reactor Internals

The ISI requirements for the reactor internals are specified in Category B-N of Table IMB-2500-1 in the Code. The inspection category for reactor internals is visual (VTM-3). The objective of the VTM-3 examination is to "determine the general mechanical and structural conditions" of the reactor internals.

The opacity of liquid sodium increases the difficulty of visually inspecting the reactor internals. The Code therefore allows for the use of various techniques to 'see' inside the pool. The following methods can be used for a VTM-3 inspection: remote visual examination, under-sodium scanning, or remote dimensional gaging. As the "presence of loose parts, debris, and loss of integrity at bolted or welded connections to components shall be detectable" during a VTM-3 inspection, alternate inspection techniques such as acoustic emission monitoring, accelerometers and strain gages may also be used.

The in-service inspection of the reactor internals will be carried out using a number of examination methods. The primary examination methods will be continuous monitoring of pressures, temperatures, and dimensional gaging. Under-sodium scanning will be used for additional inspection if there is an indication of a failure or if adequate information cannot be gathered using continuous monitoring and dimensional gaging. In-service inspection above the sodium level will be done with a miniature remote TV camera.

In hard-to-access areas of the reactor pool, including the area below the horizontal redan, continuous monitoring may give the most reliable and accurate information about the condition of the internal structures and components. Sensors will be installed in strategic locations inside the reactor to provide information about particular internal components and structures. For instance, temperature sensors installed at the core outlet will give information on the continued coolant flow through the core. Continuous monitoring will be the primary source of information on the operation of the EM pumps and the IHX. Sensors which can perform these functions in the reactor environment are currently available.

The reactor design has no nuclear class piping outside the reactor pool. Within the pool, the core plenum inlet piping is nuclear class, but because it is immersed in sodium, it follows the Code requirement for reactor internals rather than liquid metal retaining piping. Continuous monitoring of pool pressures and temperatures will determine if leakage is occurring in the pipe.

Dimensional gaging using the in-vessel transfer machine (IVTM) will be used to provide additional information about the integrity of pool components. A great deal of information concerning the integrity of the core will be obtained during normal refueling operations when the position of roughly one third of the fuel pins are indexed as they are replaced. A failure of the core support structure may be indicated by inconsistency in fuel element height, an increase in the force required to remove or insert fuel elements, or the change in fuel element position. Also during refueling, the integrity of the fuel storage cylinder can be asserted. The IVTM will indicate any irregularities in height as stored fuel is removed and hot fuel is placed into storage.

The range of the IVTM is 24 feet from the top of the core to the top of the vessel just below the closure head. The IVTM grapple has a 6 ft radius circle of travel. The IVTM is mounted on the rotating plug and so its range is extended beyond the grapple radius.

If a failure indication is detected by continuous monitoring or dimensional gaging, under-sodium viewing or ranging may yield additional information without requiring drainage of the primary sodium. The under-sodium scanning device will be inserted through an ISI port on the rotatable plug to provide the maximum scanning range. If scanning of the area below the top of the core is necessary, the area can be made accessible by removing some of the core assemblies or by removing an IHX and lowering inspection equipment through the redan.

5.8.1.3 Containment Vessel

The containment for the reactor is provided by the reactor closure and the containment vessel which surrounds the reactor vessel. The containment vessel also functions as a guard vessel for the reactor. ISI for the guard vessel is governed by examination category C-B 'welds in guard tanks and guard pipes' of Table IMC-2500-1 in the Code. For welds in low alloy steel guard vessels, VTM-3 visual inspection is required.

The Code was further examined to determine if additional ISI requirements should apply to the vessel owing to its function as the containment vessel. None of the requirements set forth in Division 3, "Rules for Inspection and Testing of Components of Liquid-Metal Cooled Plants," specifically apply to containment vessels. Therefore, Division 1, which applies to light-water plants, was examined. The requirements of Section IWE for pressure retaining welds in vessels requires a VT-3 visual examination for shell welds. The VT-3 examination has the same intent and requirements as a VTM-3 examination. Therefore, the VTM-3 examination specified for guard tank welds in Section 3 of the Code will govern ISI of the containment vessel. The goal of the ISI will be to ensure that the containment vessel is structurally sound so that it can contain any sodium leaks from the reactor vessel.

VTM-3 will detect the loss of integrity at welded connections, but will not detect the loss of inert gas that may result from a small leak. For this reason, continuous monitoring of the pressure in the RV/CV annulus will be provided. A loss of pressure in the annulus (normally pressurized to 29.4 psia during operation) will be the first indication of a leak. VTM-3 inspection, by inserting a remote viewing device into the RV/CV annulus and into the CV/collector cylinder annulus, will be used as follow-up detection if a loss of pressure indication is given by two or more monitoring devices. Additional backup inspection will be done by pulling and analyzing a gas sample from the RV/CV annulus. If a leak has occurred, air intrusion into the annulus will occur.

5.8.1.4 Vessel Supports

The reactor vessel and containment vessel are supported by modular support brackets at the top of the reactor. The ISI for these supports is covered by category F-A of Table IMF-2500-1 of the Code. This category for 'plate and shell' type supports specifies VTM-3 visual examination for mechanical connections to liquid metal and cover gas retaining components.

The reactor vessel is supported by the reactor closure and containment vessel, which in turn are supported by support brackets bolted to the floor of the head access area. The support brackets are directly accessible from the HAA and can be visually inspected during reactor shut- down by removing 1-2 ft of shielding and insulation from above the supports. Although the supports are directly accessible, the shear lug upon which the containment vessel sits is not visible without removing the support. To avoid removal, ultrasonic inspection of the shear lugs will take place to provide 100% inspection of the supports. Follow-up inspection in the case of an anomaly will involve removing the bracket for direct visual inspection.

5.8.1.5 Seismic Support System

The seismic isolation support system functions to provide support for the silo and reactor module at all times. Only in the case of a seismic event is the isolation against horizontal loads required. It is essential to plant safety to ensure that the system will function as designed during a seismic event. The isolation support system consists of a number of isolators mounted on concrete support columns. The seismic isolators are fabricated from steel and rubber. The ISI for this system is visual inspection. The columns and isolators are located in a chamber under the RVACS inlet plenum. The chamber is accessible and direct visual examination of the isolators is possible. In addition, capability for removing and testing any single isolator is also provided. Mockups of the isolators will be located adjacent to the functioning supports and will be removed for testing and detailed examination if the visual examination indicates damage. The qualification program for the isolators is shown in Table 1.5-1.

5.8.1.6 RVACS

The RVACS is a nuclear safety related system which is essential to safe reactor cooldown following a loss of flow accident. RVACS operates at all times and is a completely passive system. The RVACS system will be continuously monitored by measuring air flow rate and exit air temperature, and by monitoring for water intrusion, radiation and fire/smoke. Periodic remote visual examination of the RVACS flow passage will be done to check for blockage and to inspect the collector cylinder integrity. The inspection will be done by inserting a remote miniature TV camera through the RVACS inlet vents or outlet stacks. Testing of the reactor vessel emissivity coating for degradation also will take place by installing material coupons in the RV/CV annuli and periodically removing a coupon for examination.

5.8.2 Non-Nuclear Class Components

5.8.2.1 Intermediate Heat Transport System

In-service inspection will be performed on the IHTS sodium components in compliance with the ASME Boiler and Pressure Vessel Code, Section XI, Division 3, Class 3, "Rules for In-Service Inspection of Nuclear Power Plant Components." The in-service inspection for the IHTS is described in Section 5.5.4.1.

5.8.2.2 Steam Generator System

In-service inspection will be performed on the SGS components in compliance with the ASME Boiler and Pressure Vessel Code, Section XI. In-service inspection of steam/water components will be performed in compliance with Division 1, Class 3, Rules for Light Water Cooled Plants and in-service inspection of sodium components will be performed in compliance with Division 3, Class 3, Rules for Sodium Cooled Plants. The in-service inspection for the SGS is described in Section 5.6.4.1.

TABLE 5.8-1

INSPECTION PROGRAM A

Inspection Interval	Inspection Period, Calendar Years of Plant Service	Minimum Examinations Completed, %	Maximum Examinations Credited, %
1st	3	100	100
2nd	7	33	67
	10	100	100
3rd	13	16	34
	17	40	50
	20	66	75
	23	100	100
4th	27	8	16
	30	25	34
	33	50	67
	37	75	100
	40	100	...

TABLE 5.8-2

INSPECTION PROGRAM B

Inspection Interval	Inspection Period, Calendar Years of Plant Service	Minimum Examinations Completed, %	Maximum Examinations Credited, %
1st	3	16	34
	7	50	67
	10	100	100
2nd	13	16	34
	17	50	67
	20	100	100
3rd	23	16	34
	27	50	67
	30	100	100
4th	33	16	34
	37	50	100
	40	100	...

CHAPTER 6
ENGINEERED SAFETY FEATURES

CHAPTER 6

ENGINEERED SAFETY FEATURES

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Chapter 6 ENGINEERED SAFETY FEATURES

6.1 General

Engineered Safety Features are specific provisions of the design that prevent the occurrence of, or mitigate the effects of, serious accidents. The PRISM design incorporates three such features:

1. Containment Systems,
2. Reactor Vessel Auxiliary Cooling System (RVACS), and
3. Head Access Area (HAA) guard pipes.

Descriptions of the containment systems are provided in Section 6.2 of this chapter.

The RVACS system provides a highly reliable, entirely passive means for residual heat removal in case of failure of other heat transport paths. Description of RVACS and evaluation of the design are in Chapter 5.7.

Intermediate sodium piping in the HAA is surrounded by guard pipes that protect the containment boundary against sodium spills. Description of this Engineered Safety Feature is in Chapter 5.1.3.

6.2

6.2 Containment Systems

6.2.1 Functional Design

6.2.1.1 Design Bases

The containment system and associated auxiliary system shall provide a leak tight boundary that will contain the accidental release of core fission products and primary coolant.

The containment shall be designed to withstand the static and dynamic loads produced by postulated accidents.

Mechanical loading on the containment boundary is caused by annulus and cover gas pressure increases due to accidents that cause temperature increase. On the containment vessel alone, sodium leakage from the reactor vessel would also produce hydrostatic pressure not normally present. The sodium coolant is at low pressure and is considerably sub-cooled even during severe transients. Therefore, the coolant is not a significant pressure loading source. The cover gas region is also a low pressure region with no identifiable source of significant pressurization other than system temperature and level variations caused by differential expansion between the sodium and the reactor vessel. The cover gas pressure is approximately atmospheric during normal operation and increases to about 5 psig during loss-of-cooling events.

Due to the inherent negative feedback mechanisms within the PRISM core, even beyond design basis events do not challenge the containment integrity. The reactor is designed to accommodate beyond design basis events without control rod insertion with minor mechanical loading of the containment boundary. Local faults are also limited in their extent of damage because of features that prevent inlet blockage and the inherent ability of sodium cooling of stochastic occurrences of local fuel pin damage. Small amounts of fuel debris that may be postulated to enter the coolant from local damage will not threaten the reactor vessel since there is considerable margin for cooling small particulate in the sodium.

Thus, it is judged inappropriate to arbitrarily postulate containment damage and radiological source terms associated with core melting. The Site Suitability Source Term evaluated in Section 6.2.3 is sufficiently large to actually provide margin against postulated severe core damage as long as such postulated severe damage is not able to challenge the reactor and containment vessel integrity.

The containment boundary extends inward from the reactor closure to the intermediate heat exchanger. This boundary is required to withstand the full steam pressure loading (1000 psia), so that the containment boundary will not be compromised in the unlikely event of a sodium-water reaction and SWRPRS failure.

The overall safety approach for PRISM includes demonstrating its inherent shutdown and shutdown heat removal mechanisms thus providing a convincing demonstration of the inherent behavior of the reactor module. The ability to gain such evidence is dependent on retaining the close coupling of the reactor heat source with natural air cooling of the containment vessel. The approach emphasizes safety design features that can be fully demonstrated even for beyond design basis challenges.

6.2.1.2 System Design

Figure 6.2-1 shows the PRISM reactor module. The reactor closure and containment vessel make up the containment boundary. These features and the potential leaks path are illustrated in Figure 6.2-2.

The reactor vessel supports the core which is immersed in a pool of liquid sodium. A containment vessel surrounds the reactor vessel. The reactor closure provides the top cover for the reactor vessel and containment vessel. A head access area located above the reactor closure houses various reactor systems. If the reactor vessel should leak below the sodium level, the containment vessel will retain the sodium, the core will still be immersed, and the sodium flow circuit will be maintained. On loss of the intermediate heat transport system, decay heat removal is provided by natural air circulation on the exterior side of the containment vessel through the reactor vessel auxiliary cooling system (RVACS).

The space above the liquid sodium in the sealed reactor vessel is filled with an inert cover gas (helium) which reaches atmospheric pressure at full power. Because the cover gas is sealed within the primary system, its pressure drops as the system temperature decreases, reaching approximately 6 psi at 400°F. During operation of the reactor, noble gases and other fission products can be released from failed fuel and can migrate to the reactor cover gas space. However, most solid fission products will be retained within the fuel matrix. With cladding failure, a fraction of the halogens, cesium, and other fission products in the fuel pin to cladding gaps will be released (see Section 6.2.3.3, "Source Term Analytic Factors"). Iodine and other halogens will be chemically bonded to the sodium with only small fractions entering the cover gas space. The two potential pathways for cover gas leakage are:

1. Through the reactor closure (L_1) to the head access area and then directly to the environs. The head access area is a closed volume but is cooled by ventilation air flowing through the space.

2. Through the reactor vessel (L_2) and then through the containment vessel (L_3) to the RVACS.

The following volumes represent the current PRISM reactor configuration:

o Cover gas space:	1770 cu ft
o Reactor vessel/containment vessel annulus:	1000 cu ft
o Head access area:	20000 cu ft

The reactor closure and the containment vessel comprise the major structural elements of the containment, and are classed as Seismic Category I structures. They will be designed in accordance with applicable sections of ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, Division 1.

6.2.1.3 Design Evaluation

Almost all the containment gas leakage is expected to be associated with penetrations in the reactor closure (pathway L_1, see Figure 6.2-2). A second path involves leakage from the cover gas to the containment vessel annulus and then to the RVACS annulus and the environs (pathway L_2 and L_3). However, this leakage is expected to be negligible except when postulated reactor vessel failures are included in the accident scenario. The magnitude of cover gas leakage through the containment boundary is defined by the containment leak rate (CLR) expressed in percent of contained cover gas volume per day.

The reactor closure and its penetrations are designed to assure that total cover gas leakage through pathway L_1 will be less than 0.1% of the contained cover gas volume per day. Each penetration listed in Table 5.2-7 and depicted in Figure 5.2-2 will be sealed by high quality elastomeric and/or metallic seals and backed up by seal welded covers. For example, the rotatable plug shown in Figure 5.2-3 is sealed to the closure by a metal-to-metal pressure surface (i.e., a ledge seal) a buffered double elastomeric seal, and a welded canopy seal.

Containment leakage from a low pressure liquid metal reactor will tend to diminish with time due to sodium aerosol plugging. The aerosol agglomerates into larger particulate masses which then collect on surfaces. In the case of containment leakage on the order of 0.1% volume per day, the leakage path will be so small (fractions of a millimeter), that the aerosols will rapidly plug the hole and reduce leakage or, at least, retain larger particulates. In the site suitability analysis (see Section 6.2.3) no credit was taken for this phenomenon.

It is expected that leakage through the reactor or containment vessel walls will be functionally non-existent. The vessels will be fabricated, welded, and tested for leak tightness in the factory and before operation. During reactor operation the containment vessel annulus is filled with argon gas and the volume is continuously monitored to detect sodium or helium cover gas leakage from the reactor vessel and argon leakage out of the containment (or air in), thus providing an implicit on-line vessel test. The leak tightness of the vessel is measured on-line by measuring the conditions of pressurized inert gas in the annulus between the two vessels. Neither the annulus or cover gas are actively serviced during plant operation and, therefore, loss of pressure or vacuum or a change in composition would indicate a leak.

6.2.1.4 Testing and Inspection

The two parameters measured during containment leak rate testing are temperature and pressure. The change in gas pressure related to time is translated to leak rates in percent volume per day using the ideal gas equation of state.

Temperatures are measured by RTDs located in the reactor cover gas space and the annulus between the reactor vessel and the guard vessel. Pressures are measured using high precision quartz gauges.

Leakage through the reactor closure (L_1) is determined by continuously measuring the sodium level and cover gas pressure within the reactor vessel, as well as monitoring the head access area for gaseous fission products. The annulus between the reactor and containment vessels is maintained at a positive pressure to assure that leakage through the reactor vessel (L_2) is zero. Through vessel leakage (L_3) can be detected by a loss of the positive pressure in this sealed space and measured by maintaining the cover gas space at the same pressure while monitoring the pressure decay rate.

Based on leak testing experience of light water reactor containments, it is expected that leakages as low as 0.03% volume per day could be measured. Therefore, the reference design case of 0.1% per day represents a credible case from the leak testing standpoint.

In-service Inspection requirements for the reactor closure are described in Section 5.8.1.1.2.

In-service Inspection requirements for the containment vessel are described in Section 5.8.1.3.

6.2.1.5 Instrumentation Requirements

Instrumentation provided for containment leak testing (see Section 6.2.1.4) will be used for continuous measurement of parameters that indicate potential challenges to the containment boundary. Cover gas temperature and pressure will be continuously monitored by RTD's and quartz gauges to detect challenges to the reactor closure.

Sodium leakage from the reactor vessel which could challenge the containment vessel will be detected by monitoring pressure within the sealed space between the two vessels.

6.2.1.6 Materials

Materials to be used in components of the containment boundary are as follows:

Containment Boundary Component	Material
Reactor closure and components	304SS
IHX	304SS
Containment vessel	2-1/4 Cr-1Mo

6.2.2 Containment Isolation System

There are no penetrations of the Containment Vessel. Penetrations of the Reactor Closure that require isolation are limited to five-3" Sodium Processing lines and one-1 1/2" Cover Gas Processing line. During Reactor Operation, these lines are closed with redundant isolation valves. Both valves are located as close as practical to the reactor closure head and are within the reactor Head Access Area (HAA). The HAA provides environmental protection from tornado generated missiles as well as weather protection. Interlocks controlled by the Reactor Protection System prevent opening of these valves unless the reactor is in a shutdown mode.

During shutdown operations the coolant and covergas cleanup lines are automatically closed on any leak indication, such as a high radiation alarm or sodium aerosol within the HAA.

Containment penetrations which are unique to PRISM include the adapters, gate valves, and inerted transfer casks which are used during refueling and primary component replacement operation, when the reactor is shut down. Dual gate valves and inert gas purging of the adapters and shielded transfer casks are used to assure that air does not leak into the primary system and that the containment barrier always includes at least one passive containment barrier during those maintenance activities. These attachments are designed as extensions of containment when they are in use.

Other Reactor Closure penetrations include:

1. Rotatable plugs,
2. Head-mounted instruments,
3. Equipment openings (EM, pumps, IHX, IVTM),
4. Fuel transfer port,
5. ISI ports, and
6. Control Rod Drive (CRD) lines.

As discussed in Section 6.2.1.3, Design Evaluation, all reactor closure penetrations are redundantly sealed by elastomeric and/or metallic seals backed up by seal-welded covers. Beneath the Reactor Closure, a bellows design separates CRDs from the cover gas space. Above the Reactor Closure, each CRD is contained in a separate enclosure. All reactor closure penetrations are redundantly sealed, as described in Section 6.2.1.3, Design Evaluation.

Provision is made for isolation of the Head Access Area (HAA) by means of electrically-operated dampers in HVAC ducts. This is a fire-protection feature that is not considered part of containment. The dampers are actuated through the Reactor Protection System as part of the fire control actions. They are used to limit air ingress in the event of an accident that exposes sodium to air in the HAA.

6.2.3 Site Suitability Analysis

6.2.3.1 General

This section describes calculated offsite doses from the Site Suitability Source Term (SSST). Results of the analysis show that the PRISM design will meet 10 CFR 100 requirements.

The rest of this section describes: the isotopic inventory for PRISM (Section 6.2.3.2), the radiological source terms (6.2.3.3), Atmospheric Dispersion Factors (6.2.3.4), Dose Conversion Factors (6.2.3.5), dose limits (6.2.3.6), and comparison of calculated offsite doses to dose limits (6.2.3.7).

6.2.3.2 Core Inventory

The isotopic inventory is based on partial core data specified in Table 6.2-1. The isotopic activity of a full core was obtained by multiplying the data from Table 6.2-1 by the ratio of power levels (425 MWt/36.25 MWt). Five percent was added to account for a 105 percent power level.

6.2.3.3 Source Term and Analytic Factors

For the reference design case the following cover gas source terms are chosen for the site suitability evaluation:

Source Composition	
o Noble gases (xenon and krypton):	100%
o Halogens (iodine and bromine):	0.1%
o Particulates (cesium and rubidium):	0.1%
o Transuranics (plutonium):	0.01%
Containment Leak Rate	0.1%/Day
Meteorology	R.G. 1.4
Core Failure	Whole Core
Applicable Limits	10CFR100

These choices bound all consequences of design basis accidents because the PRISM reactor by its design is inherently protected from major fuel failure. The core will always be submerged in a pool of sodium. Natural cooling of the sodium pool (and fuel) during reactor shutdown by the reactor vessel auxiliary cooling system (RVACS) will prevent overheating and avoid major core damage. The conservative approach that has been used in establishing SSST was used to select the release fractions.

The core transient analysis for both design basis events and selected beyond design basis events shows that because of inherent shutdown characteristics and assured RVACS cooling, fuel temperatures stay well below boiling and temperatures that could lead to cladding failure. The non-mechanistic selection of the above release fractions can easily be seen to be conservative for all design basis accidents.

Noble gases released from failed fuel are assumed to bubble through the liquid sodium to the cover gas space without holdup. Although 100% of the core noble gas inventory was selected for the SSST evaluations, it is unlikely that more than 5 to 10% of the fission gas inventory could be released to the cover gas space by fuel pin failures that might be postulated for even beyond design basis accidents.

It is believed that most of the transuranics, primarily plutonium, will be held in the fuel matrix. A small fraction may be released if the fuel pin cladding is damaged. Most of the transuranics which are released into the sodium will be retained by the liquid sodium or plated out on internal reactor surfaces. Thus, only very small fractions are expected to collect in the cover gas space.

There will be significant fractions of halogens, cesium and other fission products, up to 10% of the core inventory, in the fuel pin to cladding gaps. Iodine is usually chemically bound to the cesium in these gaps. With cladding failure some of the activity is released from the cladding but most of it would be retained by the sodium. In the environment of a pool reactor the iodine and other halogens not held in the fuel pin gaps will be chemically bound to sodium and any cesium present in the pool. This cesium will behave like sodium, (i.e., stay contained in the pool) with only a small fraction becoming airborne in the cover gas space.

No credit has been taken in either case for attenuation in the HAA, even though it is likely that it will be isolated under the postulated accident conditions. This is consistent with the approach to SSST evaluations since these features are not designed as containment systems, but represent additional conservatism in emergency planning evaluations.

6.2.3.4 Atmospheric Dispersion Factors

Atmospheric dispersion factors (X/Q) were defined for the two boundaries used:

1. Exclusion Area Boundary (EAB) at 0.5 miles
2. Low Population Zone (LPZ) Boundary at 2.0 miles

The X/Q factors were selected from Regulatory Guide 1.4 (95 percentile). These X/Q factors are as follows:

- o 0-2 Hr EAB: 1.00×10^{-3} sec/m^3
- o 0-8 Hr EAB: N/A
- o 0-24 Hr EAB: N/A
- o 1-4 Day EAB: N/A
- o 4-30 Day EAB: N/A
- o 0-8 Hr LPZ: 1.30×10^{-4} sec/m^3
- o 8-24 Hr LPZ: 2.50×10^{-5} sec/m^3
- o 1-4 Day LPZ: 8.00×10^{-6} sec/m^3
- o 4-30 Day LPZ: 1.75×10^{-6} sec/m^3

6.2.3.5 Dose Conversion Factors

Doses were calculated for each individual organ and summed with an appropriate ICRP-26 weighting factor in order to calculate a whole body-risk equivalent (WB-RE) dose. Thyroid and whole body doses were calculated using dose conversion factors from Regulatory Guide 1.109. In addition to the standard thyroid and whole body doses, organ doses were calculated for lung, bone, bone surface, red bone marrow, and liver using the dose conversion factors of NUREG/CR-0150 (Ref. 6.2-1). The ICRP-26 weighting factors are:

- o Thyroid 0.03
- o Lung 0.12
- o Bone Surface 0.03
- o Red Bone Marrow 0.12
- o Liver 0.06
- o Whole Body 1.00

The bone dose is not included in calculating the WB-RE dose.

6.2.3.6 Dose Limits

The defined dose limits are based on:

1. The guideline values of the Code of Federal Regulations 10CFR100 adjusted for construction permit (CP) doses per Regulatory Guide 1.4.

2. The dose criteria used in the CRBRP review for the lung, liver, bone, bone surface, and red bone marrow for both cases.

The dose limits are:

o Thyroid	150.0 rem
o Lung	37.5 rem
o Bone	150.0 rem
o Bone Surface	150.0 rem
o Red Bone Marrow	37.5 rem
o Liver	75.0 rem
o Whole Body	20.0 rem
o WB-RE	24.5 rem

6.2.3.7 Site Suitability Analysis Results

The results of the analyses performed are given in Table 6.2-2. The calculated doses are compared to 10CFR100 guidelines. The doses calculated for the EAB and LPZ are below the guideline values.

The most limiting organ is the bone surface for which the doses are dominated by plutonium (transuranic) sources.

References - Section 6.2

6.2-1 "Estimate of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occurring in Routine Releases from Nuclear Facilities, Vol. III," by D. E. Dunning, Jr., et al., NUREG/CR-0150, October 1981.

TABLE 6.2-1

PARTIAL CORE INVENTORY

Isotope	Activity (Ci)	Isotope	Activity (Ci)
U-237	2.54400+05	MO-99	7.00500+06
U-239	3.55400+07	TC-99M	6.13300+06
NP-239	3.55200+07	ZR-100	5.52800+06
PU-238	7.54100+04	NB-100	3.71700+06
PU-239	1.82800+04	NB-101	6.35900+06
PU-241	4.07800+06	MO-101	7.60200+06
AM-241	1.41300+04	TC-101	7.60600+06
CM-242	5.36000+05	MO-102	7.62100+06
BR-85	6.76300+05	TC-102	7.63600+06
KR-85	1.95200+04	MO-103	7.58300+06
KR-85M	6.85500+05	TC-103	7.75200+06
BR-87	7.85600+05	RU-103	8.02300+06
KR-87	1.13500+06	RH-103M	7.23000+06
BR-88	6.72500+05	MO-104	6.85100+06
KR-88	1.56800+06	TC-104	7.52300+06
RB-88	1.63900+06	RH-104	8.74800+05
KR-89	1.67600+06	MO-105	4.51000+06
RB-89	2.10000+06	TC-105	5.88200+06
SR-89	2.12300+06	RU-105	6.12900+06
KR-90	1.59000+06	RH-105	6.10800+06
RB-90	1.82400+06	RH-105M	1.71600+06
SR-90	1.21600+05	TC-106	4.92900+06
Y-90	1.29000+05	RU-106	3.67200+06
KR-91	1.08200+06	RH-106	3.67500+06
RB-91	2.49100+06	TC-107	2.90500+06
SR-91	2.90700+06	RU-107	3.89700+06
Y-91	2.90000+06	RH-107	3.91300+06
Y-91M	1.68900+06	TC-108	1.38500+06
RB-92	2.21600+06	RU-108	2.74200+06
SR-92	3.47800+06	RU-109	1.94300+06
Y-92	3.51700+06	RH-109	2.01800+06
RB-93	1.64000+06	PD-109	2.10000+06
SR-93	6.22200+06	PD-109M	1.01200+06
Y-93	4.42700+06	AG-109M	2.09900+06
RB-94	8.53900+05	SB-125	8.74200+04
SR-94	4.12100+06	TE-125M	1.79400+04
Y-94	4.78200+06	TE-127	6.33300+05
SR-95	3.69000+06	TE-127M	8.30700+04
Y-95	5.33300+06	SN-128	7.73200+05
ZR-95	5.46100+06	SB-128M	8.63700+05
NB-95	5.32800+06	SB-129	1.61000+06
ZR-97	6.03200+06	TE-129	1.59900+06

TABLE 6.2-1 (Cont'd)

PARTIAL CORE INVENTORY

Isotope	Activity (Ci)	Isotope	Activity (Ci)
NB-97	6.08300+06	SN-130	1.55500+06
NB-97M	5.73000+06	SN-131	1.27900+06
ZR-98	6.30200+06	SB-131	3.68400+06
NB-98	6.42200+06	TE-131	4.01500+06
Y-99	1.55900+06	TE-131M	6.90300+05
ZR-99	6.13100+06	I-131	4.54200+06
NB-99	6.50100+06	TE-132	6.02800+06
I-132	6.16000+06	CE-147	2.40600+06
SB-133	1.80400+06	PR-147	2.48800+06
TE-133	4.70500+06	ND-147	2.55600+06
TE-133M	2.63800+06	PM-147	8.13200+05
I-133	7.79500+06	CE-148	1.82100+06
XE-133	7.60500+06	PR-148	2.05700+06
TE-134	6.02300+06	PM-148M	3.38300+05
I-134	8.38600+06	PR-149	1.55200+06
CS-134	1.87100+05	ND-149	1.64600+06
TE-135	3.49000+06	PM-149	1.64600+06
I-135	7.57900+06	ND-151	1.01200+06
XE-135	8.51800+06	PM-151	1.01800+06
XE-135M	1.68700+06	SM-151	1.35500+04
I-137	3.03900+06	EU-154	1.25000+04
XE-137	6.79000+06	EU-155	5.54500+04
CS-137	3.52500+05	CS-142	2.41000+06
BA-137M	3.34500+05	BA-142	5.45600+06
I-138	1.40700+05	LA-142	5.63100+06
XE-138	5.56700+05	CS-143	1.04300+06
CS-138	7.20800+06	BA-143	4.58600+06
XE-139	4.86200+06	LA-143	5.13200+06
CS-139	6.61700+06	CE-143	5.17400+06
BA-139	6.78300+06	PR-143	5.01600+06
XE-140	2.74100+06	BA-144	3.14900+06
CS-140	5.68100+06	LA-144	6.29300+06
BA-140	6.30300+06	CE-144	3.28100+06
LA-140	6.55500+06	PR-144	3.28900+06
XE-141	1.13900+06	PR-144M	3.94600+04
CS-141	4.47500+06	CE-145	3.61300+06
BA-141	6.32700+06	PR-145	3.61500+06
LA-141	6.36500+06	CE-146	3.00700+06
CE-141	6.43800+06	PR-146	3.02300+06

NOTE: This isotopic inventory is based on partial core data. To obtain the inventory of the full core, a multiplication factor of 425 MWt/36.25 MWt should be applied.

TABLE 6.2-2

RADIOLOGICAL ASSESSMENT RESULTS

Dose Summary at the EAB (0.5 Miles)

Organ	2-Hour Dose	10CFR100	% of Limit
Thyroid	3.29	150.0	2.19%
Lung	5.81	37.5	15.48%
Bone	19.77	150.0	13.18%
Bone Surface	63.54	150.0	42.36%
Red Bone Marrow	8.02	37.5	21.38%
Liver	24.88	75.0	33.17%
Whole Body	1.56	20.0	7.80%
Whole Body Equivalent	6.72	24.5	27.41%

Dose Summary at the LPZ (2 Miles)

Organ	30-Day Dose	10CFR100	% of Limit
Thyroid	2.62	150.0	1.74%
Lung	6.38	37.5	17.02%
Bone	23.13	150.0	15.42%
Bone Surface	74.30	150.0	49.53%
Red Bone Marrow	9.30	37.5	24.79%
Liver	2.94	75.0	3.92%
Whole Body	0.53	20.0	2.67%
Whole Body Equivalent	4.90	24.5	20.00%

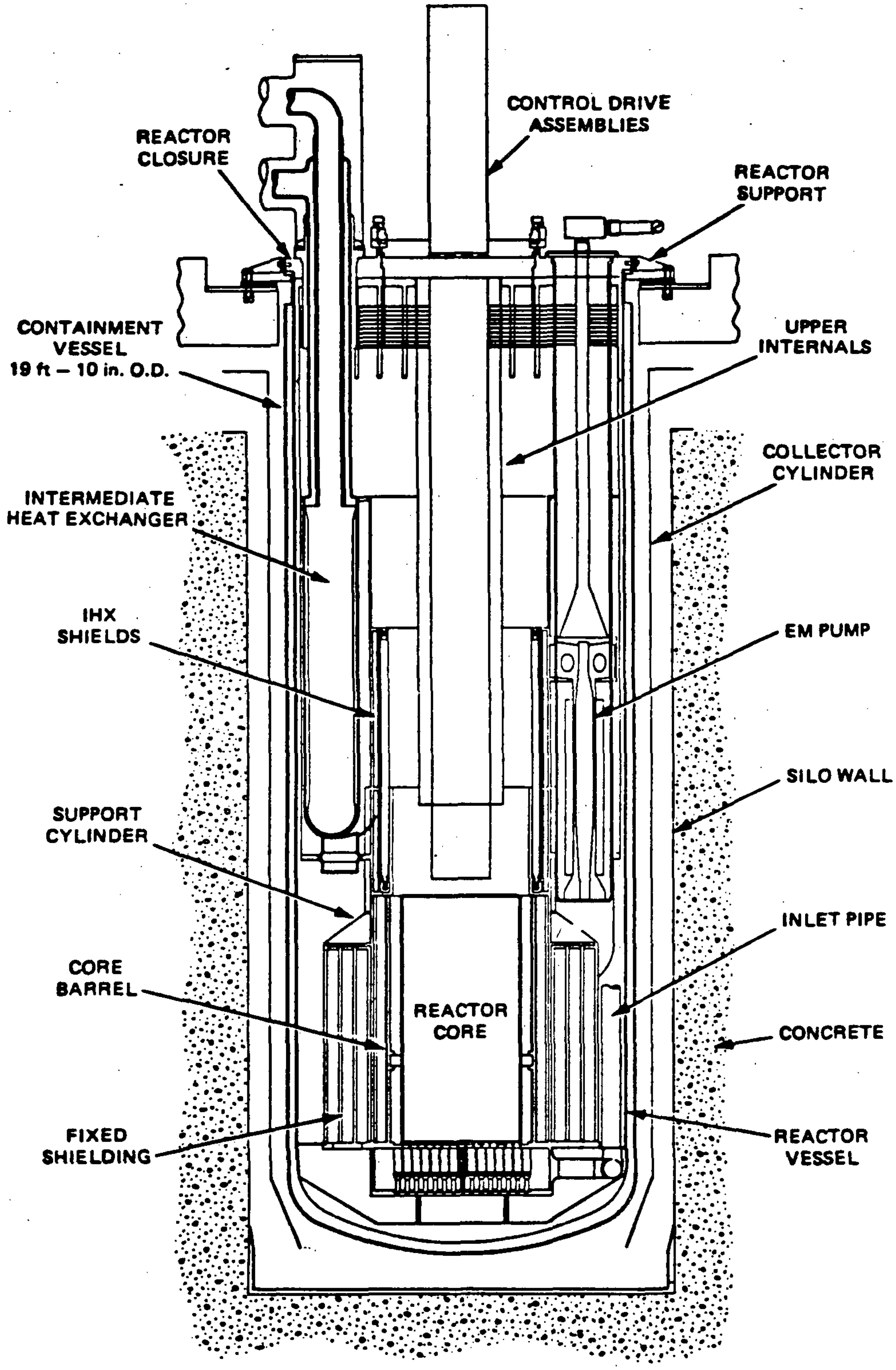

Figure 5.1–1 REACTOR MODULE

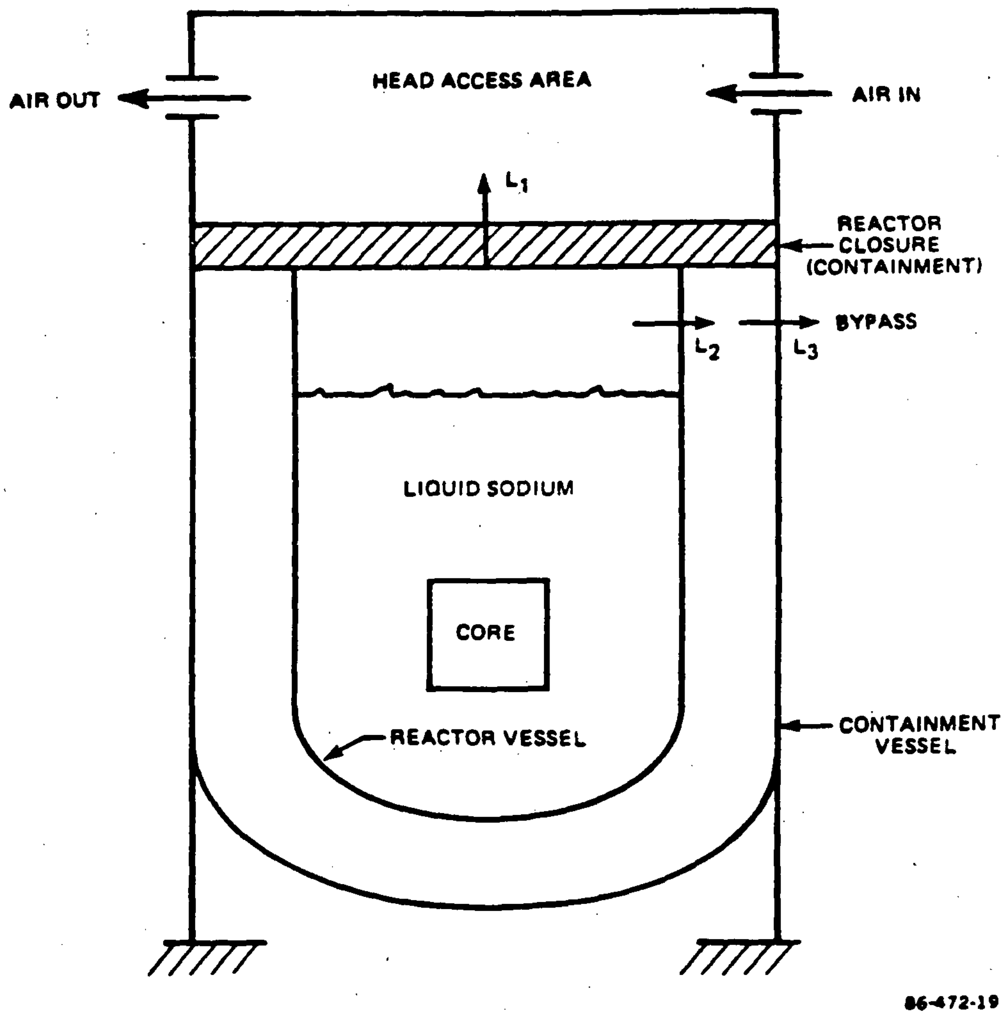

Figure 6.2–2 PRISM CONTAINMENT ILLUSTRATION

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CHAPTER 7

INSTRUMENTATION AND CONTROLS

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Chapter 7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

The plant control system (PCS) provides the hardware, and software for plant control, protection of the plant investment, and data handling and transmission. The PCS functions will utilize highly reliable redundant digital equipment and uninterruptible power supplies.

The nine nuclear steam supplies, three turbine generators, and the associated balance of plant equipment in the standard PRISM plant are controlled from a single control center staffed by three reactor operators and supervised by a shift engineer.

A high level of plant automation and an efficient man-machine interface (consisting of touch screens and touch panels mounted on sit-down consoles) enable the operators to direct all PRISM control operations. Plant overview displays, display system and process mimics, and diagnostic programs enable the operators to quickly determine the plant's state and provide an appropriate response for those infrequent occurrences when operator action is required. The plant investment protection function of the PCS will automatically runback or shutdown the reactor or major components as required to minimize component damage.

Multiplexing of plant signals and the use of fiber optic data highways permit all necessary control and data signals to be transmitted between local systems and controllers to the control center over a small number of redundant, physically separated busses. This contributes to economic, reliable control of the distributed plant systems.

The Reactor Protection System (RPS) provides digital electronics to initiate reactor module safety-related trips for the protection of plant personnel and the safety of the surrounding public in the event of a plant accident. There are nine local and independent automatic Reactor Protection Systems. Each local RPS consists of four separate identical sensor and electronic logic divisions, each located immediately adjacent to the reactor in equipment vaults. The RPS provides independent 1E conditioning and monitoring of sensors to determine plant status during and following a plant accident. All safety-related data handling and information transmission is provided locally by the individual module related RPS.

7.1.1 Identification of Safety-related Systems

The following systems contain safety-related instrumentation and control equipment.

The building electrical power system provides safety-related batteries and battery chargers for the control rod latch coils and control rod drive in motors of the reactor system.

The reactor system includes a safety-related drive-in motor for each control rod and a safety-related synchronous machines to supply coastdown energy to each primary sodium EM pump following reactor trip.

The inert gas receiving & processing system provides a safety-related pressure boundary at those portions of the helium gas distribution subsystem that connect to the reactor system.

The reactor instrumentation system provides the safety-related instrumentation in the reactor vessel and head access area. These sensors provide inputs to the reactor protection system supporting its safety-related monitoring and control functions.

The primary sodium service system contains safety-related valves at the reactor closure assembly to assure isolation of the containment boundary.

7.1.2 Identification of Safety Criteria

The Reactor Protection System will meet the PRISM principal design criteria in Section 1.2.1.2 and the intent of the following NRC Regulatory

Guides and NRC Branch Technical Positions as interpreted for the PRISM liquid metal fast breeder reactor concept.

RG 1.11	Instrument Lines Penetrating Primary Reactor Containment
RG 1.17	Protection of Nuclear Power Plants Against Industrial Sabotage
RG 1.22	Periodic Testing of Protection System Actuation Functions
RG 1.28	QA Program Requirements (Design & Construction)
RG 1.30	QA Requirements (Installation, Inspection and Test of Instrumentation and Electrical Equipment
RG 1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants
RG 1.41	Preoperational Testing of Redundant On-Site Electrical Power Systems to Verify Proper Load Group Assignments.
RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems.
RG 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems.
RG 1.62	Manual Initiation and Protection Actions.
RG 1.64	QA Program Requirements for the Design of Nuclear Power Plants
RG 1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants.

RG 1.73	Qualification Tests of Electrical Valve Operators Installed Inside the Containment of Nuclear Power Plants.
RG 1.75	Physical Independence of Electric Systems.
RG 1.89	Qualification of Class 1E Equipment for Nuclear Power Plants.
RG 1.97	Instrumentation for Light-water Cooled Nuclear Power Plants to Access Plant Conditions During and Following an Accident.
RG 1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants.
RG 1.105	Instrument Spans and Setpoints
RG 1.118	Periodic Testing of Electric Power and Protection System.
RG1.120	Fire Protection Guidelines for Nuclear Power Plants.
RG 1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants.
RG 1.152	Criteria for Programmable Digital Computer System Software in Safety Related Systems of Nuclear Power Plants
RG 1.153	Criteria for Power, Instrumentation, and Control Portions of Safety Systems
BTP EICSB 21	Guidance for Application of Regulatory Guide 1.47.

BTP EICSB 22 — Guidance for Reactor Protection System Anticipatory Trips.

The design of the Plant Protection System will comply with the intent of the following industrial standards:

IEEE Std. 308 — Criteria for Class 1E Power Systems for Nuclear ~~Power~~ Generating Stations.

IEEE Std. 317 — Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations.

IEEE Std. 323 — Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

IEEE Std. 334 — Type Tests of Continuous Duty Class 1E Motor for Nuclear Power Generating Stations.

IEEE Std. 336 — Installation, Inspection, and Testing Requirements for Class 1E Instrumentation and Electric Equipment at Nuclear Power Generating Stations.

IEEE Std. 338 — Criteria for Periodic Testing of Nuclear Power Generating Station Safety Systems.

IEEE Std. 344 — Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.

IEEE Std. 352 — Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems.

IEEE Std. 379 — Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems.

IEEE Std. 381 Criteria for Type Tests of Class 1E Modules Used in Nuclear Power Generating Stations.

IEEE Std. 382 Standard for Qualification of Safety-Related Valve Actuators.

IEEE Std. 383 Standard for Type Test of Class 1E Electric ~~Std.~~ Cables, Field Splices, and Connections for Nuclear Power Generating Stations.

IEEE Std. 384 Criteria for Independence of Class 1E Equipment and Circuits.

IEEE Std. 422 Guide for the Design and Installation of Cable Systems in Power Generating Stations.

IEEE Std. 467 Standard Quality Assurance Program Requirements for the Design and Manufacture of Class 1E Instrumentation and Electric Equipment for Nuclear Power Generating Stations.

IEEE Std. 494 Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Stations.

IEEE Std. 497 Criteria for Post Accident Monitoring Instrumentation for Nuclear Power Generating Stations.

IEEE Std. 518 Guide for the Installation of Electrical Equipment to Minimize Noise Inputs to Controllers for External Sources.

IEEE Std. 566 Recommended Practice for the Design of Display and Control Facilities for Central Control Rooms of Nuclear Power Generating Stations.

IEEE Std. 577	Requirements for Reliability Analysis in the Design and Operation of Safety Systems for Nuclear Power Generating Stations.
IEEE Std. 603*	Criteria for Safety Systems for Nuclear Power Generating Stations.
IEEE Std. 627	Design Qualification of Safety Systems Equipment Used in Nuclear Power Generating Stations.
IEEE Std. 765	Preferred Power Supply for Nuclear Power Generating Stations
ANSI/IEEE/ ANS-7432	Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations
IEEE Std. 730	Standard for Software Quality Assurance lans
ANSI/ANS-4.1	Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations
ANSI/ISA-S67-01	Transducer and Transmitter Installation for Nuclear Safety Applications
ISA-S67.06	"Response Time Testing of Nuclear Safety-Related Instrument Channels in Nuclear Power Plants"

* IEEE Std. 603 is intended to replace IEEE Std. 279 required in 10CFR50.55a, (h). If this has not occurred by the time of application for certification then IEEE Std. 279 will be applied.

7.2

7.2 Reactor Protection System

7.2.1 Description

The reactor protection system (RPS) functions to limit the consequences of initiating faults in a module of a PRISM plant so that radioactive release guidelines are maintained and the public is protected.

The RPS is classified safety-related and provides for the protection of the health and safety of the public. Each RPS is located at the reactor module, the principal safety-related element of the plant.

The design basis events (DBE) which require Reactor Protection System action are identified in Table 7.2-1. To select trip parameters and set the trip levels, two qualitative features of each DBE are necessary. These are an estimate of the magnitude of the critical transient parameter and definition of acceptable limits. Computer modeled transients will be run in order to determine the required trip levels to prevent exceeding the acceptable limits.

Trip parameters and design basis events are given in Table 7.2-2 for the reactor shutdown system. A trip parameter is defined as a parameter, such as flux, temperature, flow, etc., which responds to a given DBE to trip (scram) the reactor.

The RPS relies on the reactor parameters (sodium temperature, pressure, flow, and flux). All safety-related RPS trip related instrumentation is hardened into the module area. Hardened safety-related instrumentation is confined to the reactor silo area. Neutron flux gives a direct measure of reactor thermal power. Primary sodium flow rate is determined from ~~the~~ measure of the EM pump discharge pressure. The RPS responds to a deviation to these parameters and causes a reactor trip (scram) and signal to the plant control system to adjust the balance of the plant.

An RPS reactor trip consists of the following actions:

1. release all control rod absorber bundles for gravitational insertion,
2. activate the post trip control rod drive-in motors to assure full control rod insertion under power,
3. initiate EM pump coastdown, and
4. provide a trip signal to the plant control system for related investment protection action in the balance of plant (BOP).

Specific trip parameters which protect the reactor against each DBE are shown in Table 7.2-2.

The RPS is made up of safety-related equipment from the sensor through and including the isolation device communicating with the PCS via the data handling and transmission system (DHTS). There are no electronic components located within the reactor or in the HAA, only sensors and actuators. All electronic components are located in vaults arranged adjacent to the HAA, Figure 7.2-1. These instrumentation vaults are accessible under accident conditions.

The electronics for the RPS is divisionalized into four divisions. Each division sensor, cabling, and electronics is electrically and physically isolated. There are four physically separate isolated instrument vaults, one for each division, separated from the HAA and separated one from another. All signal conditioning electronics and the RPS divisional logic are contained within these vaults. The RPS electronics uses a quad redundant scheme, Figures 7.2-1a and 7.2-1b, that provides each sensor or actuation signal with its own input or output electronics. The RPS logic is duplicated in each of four identical sections following the same quad redundant scheme. Input or output to the PCS is through a multiplexer/coupling device to the plant fiber optic DHTS. Local safety-related readout for the RPS is provided within the vaults. Manual trip is possible from each panel within these vaults, the control center (CC) and the remote shutdown facility (RSF). (The information used by a roving operator to initiate a manual trip from an RPS instrument vault is direction to do so received from a block licensed reactor operator or the shift supervisor.) Communication with with the CC is via a quad redundant fiber optic ring network. This network (DHTS) and the CC are all constructed to non safety-related standards. RPS instrumentation readout, recording, alarm and trip approach monitoring capabilities are provided at the CCR operator console displays.

The RPS is implemented with a quad redundant system in order to give the RPS very high reliability. This quad redundant system has provisions to avoid system failure after faults have caused error within the system. Fault tolerance is the survival attribute of a system that allows it to preserve its expected behavior after faults manifest themselves with the system. This quad redundant system employs fault tolerant features such as protective redundancy.

The functions of protective redundancy, namely fault masking, fault detection and system recovery, are predominantly discussed in association with fault tolerance; i.e., a system which is fault tolerant must be able to detect faults (that is detect erroneous states of a system), it must be able to mask those faults (that is confine them to a fault zone) and, isolate them in such a way that their effects cannot propagate to other parts of the system. It must be able to recover from a fault by reconfiguration, or retry or other means which would allow the system to continue its proper function without fault in its total mission.

In addition, the system is required to have fault containment zones which are designed in such a way as to be replaceable, or renewable or repairable. In the case of fault tolerant equipment, replacement is simplified to the point where the parts are easily inserted and removed from the system, then repaired "off-line" in the plant's instrument maintenance shop.

Another important feature of the quad redundant system is its self-diagnostics capability. The diagnostic system is extensive enough to identify faults into fault containment zones which are replaceable or repairable. That means that diagnostics are able to indicate whether a fault is in a sensor, the signal conditioning or other circuits along the process. If this identification cannot be properly made, then the replacement or repair of systems becomes cumbersome and potentially damaging. For example, the diagnostics extend all the way to the switches sending electric current to the actuators. The diagnostics must also be able to detect and isolate faults in the inter-vault communications system so that these systems can be replaced or repaired while total mission is being uninterrupted.

A feature of the RPS is that each division of the RPS is located within its own individual vault. This provides spatial redundancy and separation. A failure within one division cannot physically propagate to another. In order to accomplish the features of fault tolerance with separation superimposed, an exchange of data must be made between vaults. This is done in the RPS with fiber optic cables. These fiber optic cables provide the requisite communications and electrical isolation between vaults to accomplish electrical isolation and independency.

The reactor protection system is quad redundant from the sensor input to the actuator switches. There is an optical communications system which communicates with the plant control system. The I/O communications between the vaults are a highly restricted communications system that is dedicated to the task of inter-divisional RPS communications and provides the necessary diagnostics, reconfiguration, etc., that are needed to perform fault tolerant Reactor Protection System functions.

7.2.2 Analysis

7.2.2.1 RPS Block Diagram

The following discussion focuses on the conceptual RPS configuration. In Figure 7.2-1a, there are four blocks of equipment, one in each vault (one vault is illustrated). They are interconnected by an optically isolated intercommunications bus.

Equipment in each of the RPS instrument vaults (Figure 7.2-1a) is electrically Class 1E, identical and consists of the following:

1. Inputs
   - o Sensor selection
   - o Analog signal conditioning
   - o Sensor verification
   - o Analog-to-digital signal voltage conversion

2. Buffer memory for data word storage and distribution to other divisional RPS logic systems via the data exchange and interfaces.

3. CPU, the RPS central processing unit logic that utilizes a two-out-of-three software logic scheme and performs the checks against safety setpoints.

4. Output trip drivers, the interface electronics between the RPS logic processor and the trip breaker hardware.

5. Trip breakers, the two-out-of-four hard wired logic elements interfacing directly with the scram hardware (latch coils, control rod drive-in motors, etc.)

6. RPS manual trip, buttons in each instrument vault that bypass all RPS logic for reactor scram. An input to the RPS logic initiates the automatic logic trip sequence as a backup to the manual function.

Figure 7.2-1b illustrates the trip logic of a PRISM RPS and presents a truth table to indicate how a trip decision is made. The following discussion follows the example indicated by the shaded area in Figure 7.2-1b.

1. The information from each divisional sensor is sampled and stored within its own division.

2. Next, the four divisions exchange their sensor readings with the other divisions.

3. All four divisions now have the observed information from all four sensors.

   NOTE: That in our example (shaded area of table from Figure 7.2-1b) two sensors are indicating that a trip is necessary (Sensor B and Sensor D).

4. All divisions process data.

   Division A processes the data from Divisions B, C, and D.
   A = Spare
   Division B processes the data from Divisions A, C, and D.
   B = Spare
   Division C processes the data from Divisions A, B, and D.
   C = Spare
   Division D processes the data from Divisions A, B, and C.
   D = Spare

5. Division A processes the data from Sensors B, C, and D. (Division A's data is the spare.) Using the two-out-of-three logic, Division A will call for a trip based upon Sensors B and D indicating the requirement for a trip.

6. The two sets of contacts (#2) in the two-out-of-four hard wired logic will open.

7. Division B uses the data from Sensors A, C, and D. Only Sensor D is calling for a trip. Thus, Division B will not issue a trip.

8. Division C uses the data from Sensors A, B, and D. Using the two-out-of-three logic, Division C will call for a trip based upon Sensors B and D indicating the requirements for a trip.

9. The two sets of contacts (#3) in the two-out-of-four hard wired logic will open. These two contacts open in conjunction with the opening of the #2 contacts (see 6 above), will result in an interruption of the electrical current to the latch coils. The

absorber bundle (control rod) will be dropped - scramming the reactor.

10. Division D uses the data from Sensors A, B, and C. Only Sensor B is calling for a trip. Thus, Division D will not issue a trip.

When one division of the RPS fails, the remaining three divisions revert to a mode where the three good divisions exchange data. The two-out-of-three logic is still used to sense a trip. Here, it is assumed that a loss of the electronics for one RPS division means the data from that division's sensor is unavailable.

7.2.2.2 Actuator Switching Breakers

Should the readings or calculations processed by the protection software exceed a given RPS trip setpoint, each division outputs a trip signal. The signal is a voted (two-out-of-three with a dynamic spare) output within each division.

The trip breakers (illustrated in Figures 7.2-1b and 7.2-2) form a two-out-of-four voting logic unit. Each breaker is an optically coupled unit in which the breaker contacts are held closed by the application of light. Thus, de-energizing or failure of a light emitting diode will result in the opening of the breaker contacts to form a fail-safe design. With the two-out-of-four voting logic arrangement, any one division may be tested through to trip or taken out of service at any time without resulting in an inadvertent reactor trip. The high availability of the reactor is maintained while continuing to provide the flexibility for test, calibration and/or service without the need for or risk of shutting the reactor down.

In the PRISM RPS design, each division of the RPS is physically in a separate instrument vault. The two breakers for Division A are in the Division A vault. Likewise, Divisions B, C, and D contain their own breakers. Any two of the divisional logic units in agreement that a reactor trip is necessary, will result in a release of the latch coil holding current.

In the PRISM design, there are six control rods. There is one set of eight optically coupled trip breakers for each control rod. Division A RPS CPU will drive a total of twelve light emitting diodes with the twelve latch (A) trip breakers located within a breaker panel in RPS instrument Vault A. Other RPS trip outputs are handled in a similar fashion. There are similar breaker arrangements for the control rod carriage drive-in motors and for the EM pump breakers.

If the electrical current for two or more sets of divisional breaker trip coils is interrupted (two-out-of-four logic), the breaker contacts interrupt the electrical current to the control rod latch coils to release all absorber bundles.

Figure 7.2-2 illustrates the quad output circuit to energize the latch coils to hold a control rod. Each division of the RPS is provided with a circuit breaker assembly with three sets of contacts. One set is used for the latch coil circuit and the additional contacts are involved in feedback for the automated diagnostic testing of the trip function. The four circuit breakers are wired with their contacts such that successful operation of any two divisions can perform the trip function. The circuit breakers are designed such that in their non-energized state their operating contacts are an open circuit. Thus, for normal operation the circuit breaker must be continually energized. The circuit breakers are designed to fail open. This arrangement can accommodate a circuit breaker failure either <u>open</u> or <u>closed</u> and still perform its trip function. The circuit breaker for any one division may be removed at any time for maintenance activities without causing the rod drop or impacting the reactor module's availability. There are two power supplies provided in a parallel arrangement to supply the holding current for the control rod release mechanism. Each power supply is supplied with isolation devices such that one power supply may fail without affecting the second. One power supply may be removed from the circuit at anytime without causing a rod release. Maintenance activities on the supply may proceed without impacting the reactor module's availability.

Figure 7.2-3 illustrates the quad output circuit for a control rod drive-in motor. The circuit breakers for the automatic actuation of the motor are normally closed contacts that must be held open by the electronics. Any two divisions outputting a trip signal will initiate the drive-in action. Any two manual contact closures will also initiate drive-in. Figure 7.2-3 shows that the manual trip also starts the automatic trip function as a backup action to assure that a trip sequence is complete. When a control rod mechanism reaches the fully inserted position, the end of stroke limit switches will open to stop the drive-in motor. The circuit remains active until trip recovery actions are completed such that any attempt to withdraw a control rod will result in reactivation of the control rod drive-in motor.

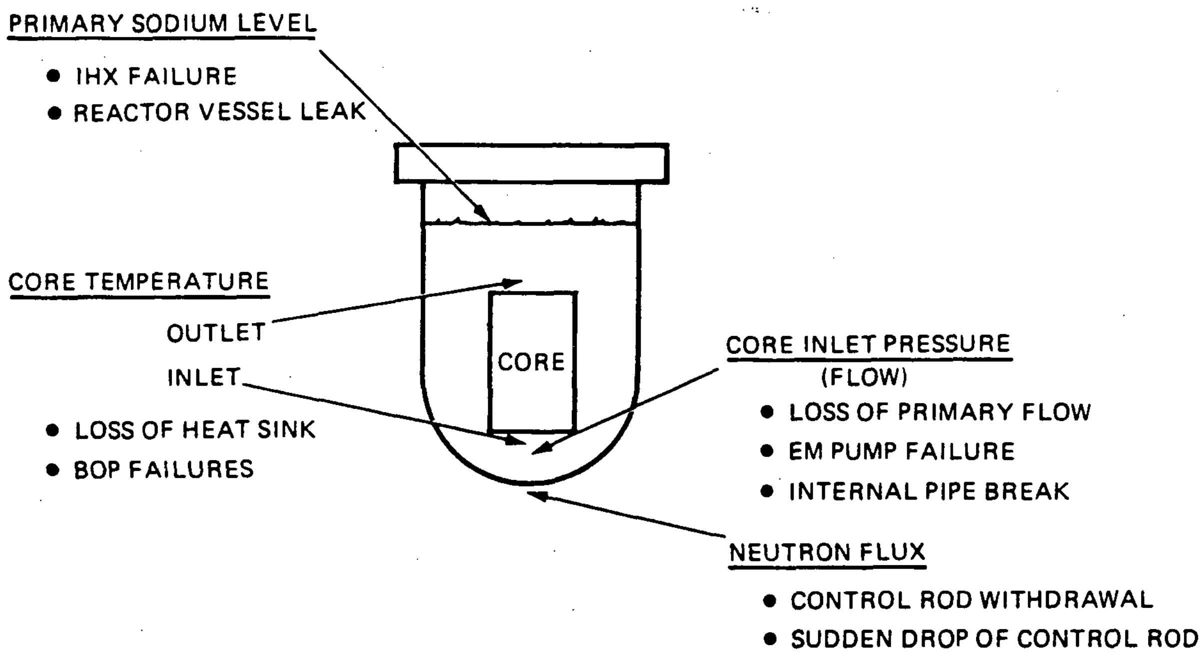
TABLE 7.2-1

DESIGN BASIS EVENTS FOR THE RPS

DBE No.	Description
1	Control assembly withdrawal at startup
2	Control assembly withdrawal at full power
3	Small reactivity insertions
4	Drop of single control rod at full power
5	Reactor vessel breach
6	Sudden core radial movement
7	Misoperation of reactor plant controller
8	Cold sodium insertion
9	Large gas bubble through reactor
10	Control assembly withdrawal at startup max. mechanical speed
11	Control assembly withdrawal at full power max. mechanical speed
12	Loss of off-site electrical power
13	Spurious primary E.M. Pump trip
14	Spurious intermediate pump trip
15	Loss of backup heat sink
16	Loss of turbine condenser
17	Loss of normal feedwater
18	Excessive primary pump flow
19	Intermediate pump overspeed
20	Loss of RPS/AM Instrumentation System
21	Single intermediate pump seizure
22	PHTS design basis pipe leak
23	IHTS design basis pipe leak
24	Steam line pipe break
25	Na-water reaction
26	Primary pump electrical failure
27	Safe shutdown earthquake

Table 7.2–2

● DESIGN BASIS EVENTS vs. TRIP PARAMETERS



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RPS COMPONENT LOCATION

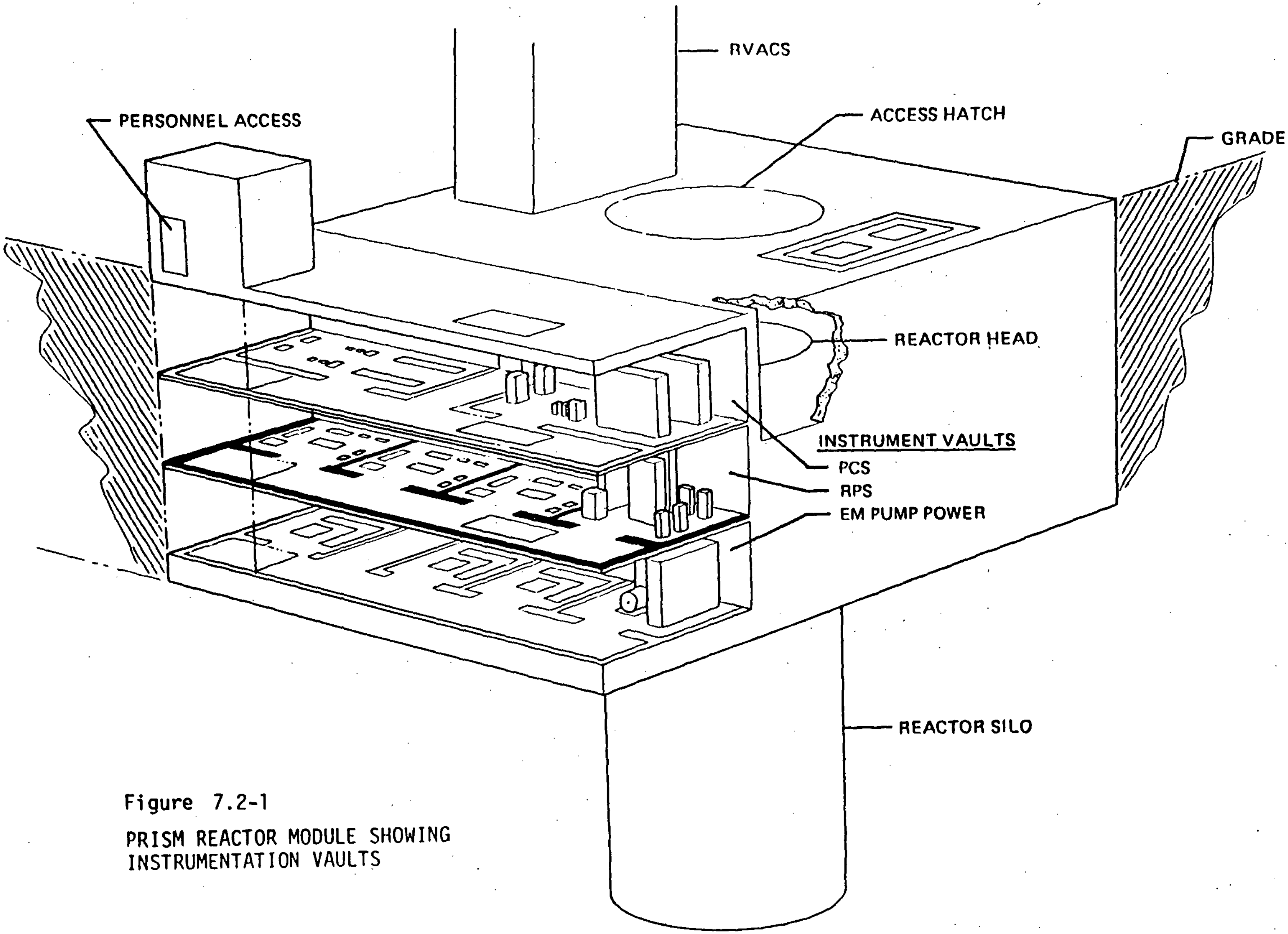
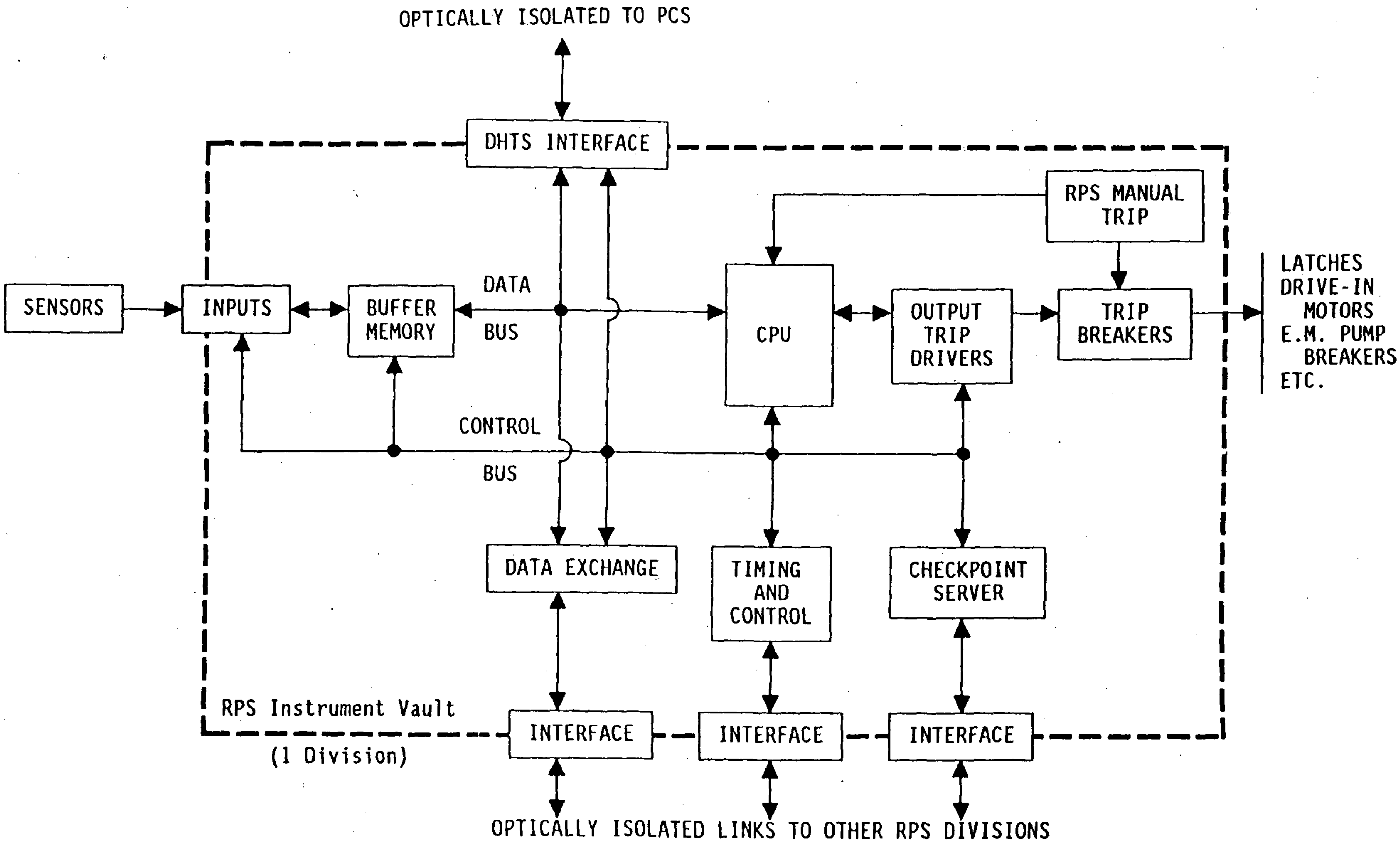


Figure 7.2-1

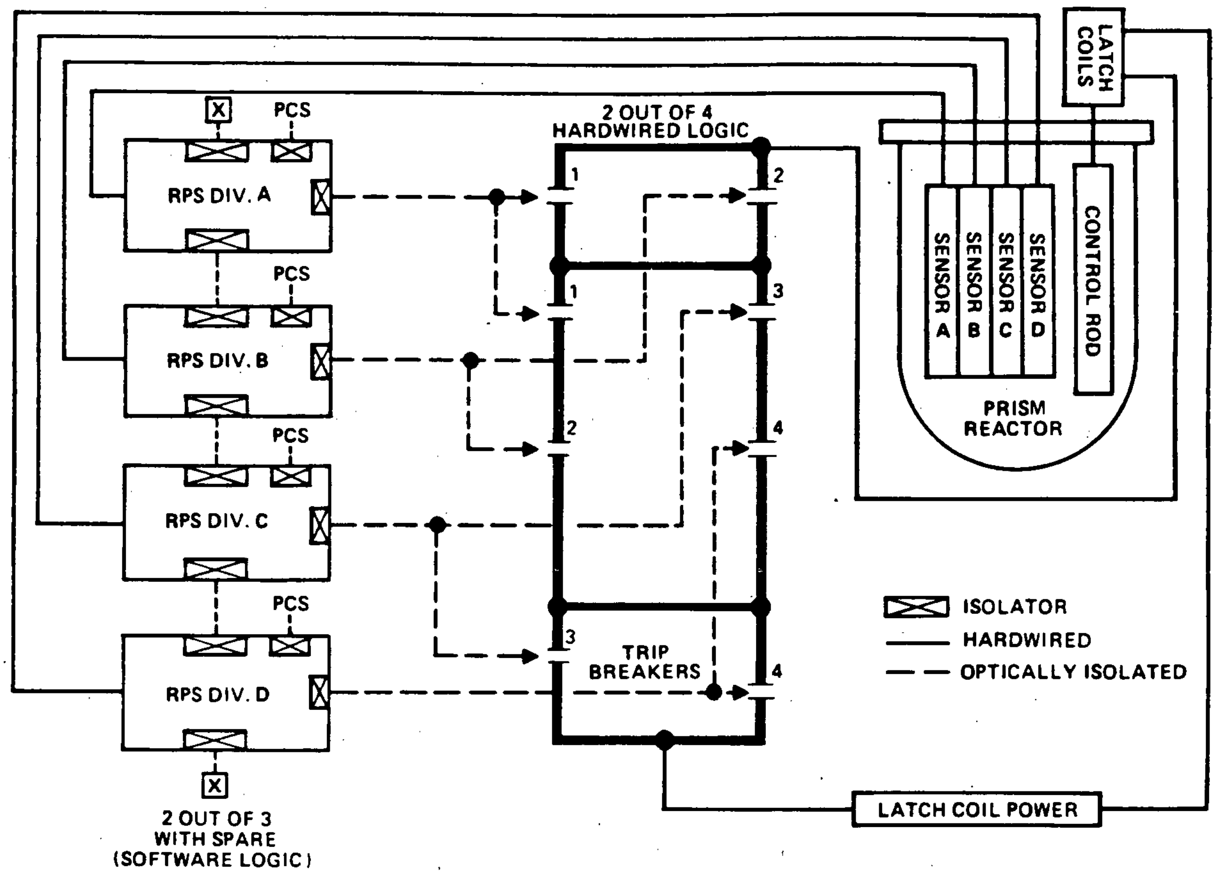
PRISM REACTOR MODULE SHOWING INSTRUMENTATION VAULTS



All equipment
indicated is safety-
related

(Typical of 4 Divisions)

FIGURE 7.2-1a RPS SYSTEM BLOCK DIAGRAM



SENSOR CONDITION ALL SENSORS + ELECTRONICS OK

	SENSORS				ELECTRONICS 2/3 Logic				2/4
	A	B	C	D	A	B	C	D	TRIP?
ALL SENSORS OK	-	-	-	-	-	-	-	-	NO
	-	-	-	T	-	-	-	-	NO
	-	-	T	-	-	-	-	-	NO
	-	T	-	-	-	-	-	-	NO
	T	-	-	-	-	-	-	-	NO
	-	-	T	T	T	T	-	-	YES
	-	T	T	-	T	-	-	T	YES
	T	T	-	-	-	-	T	T	YES
EXAMPLE →	-	T	-	T	T	-	T	-	YES
	T	-	T	-	-	T	-	T	YES
	T	-	-	T	-	T	T	-	YES
	-	T	T	T	T	T	T	T	YES
	T	-	T	T	T	T	T	T	YES
	T	T	-	T	T	T	T	T	YES
	T	T	T	-	T	T	T	T	YES
	T	T	T	T	T	T	T	T	YES

LEGEND
T = TRIP SIGNAL
(SAFETY SETPOINT EXCEEDED)

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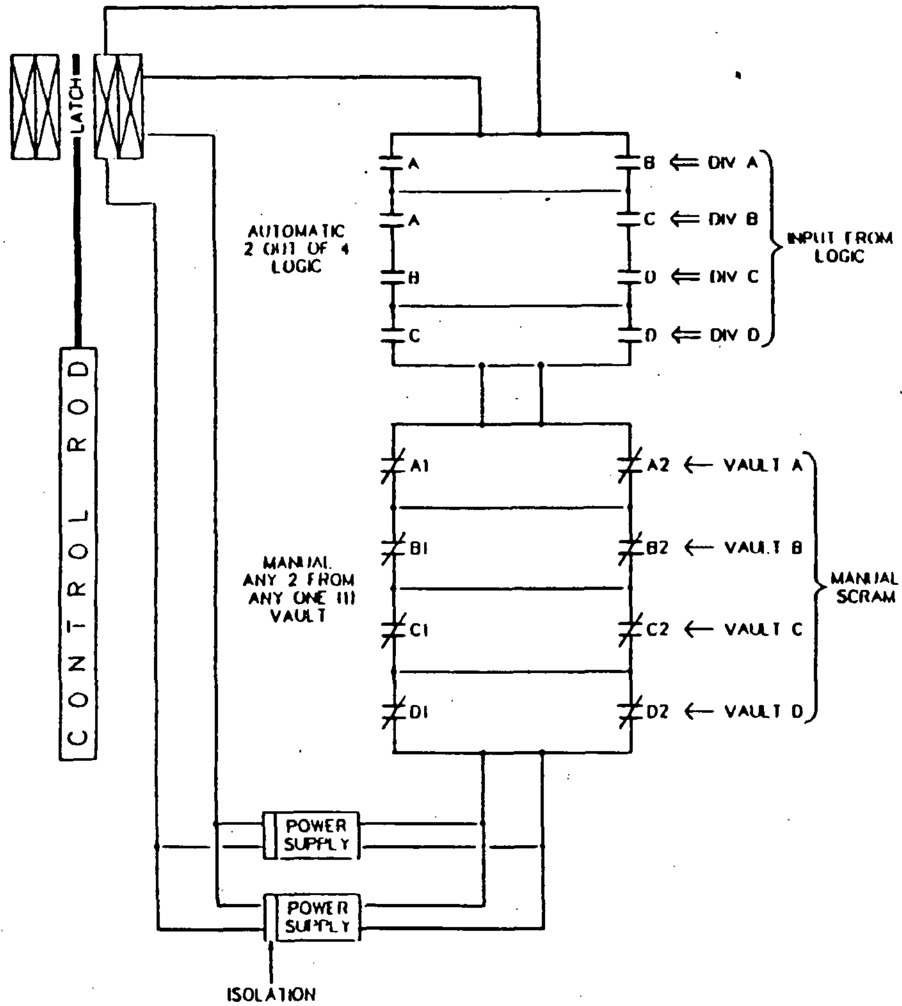
Figure 7.2–1b RPS TRIP LOGIC AND TRUTH TABLE

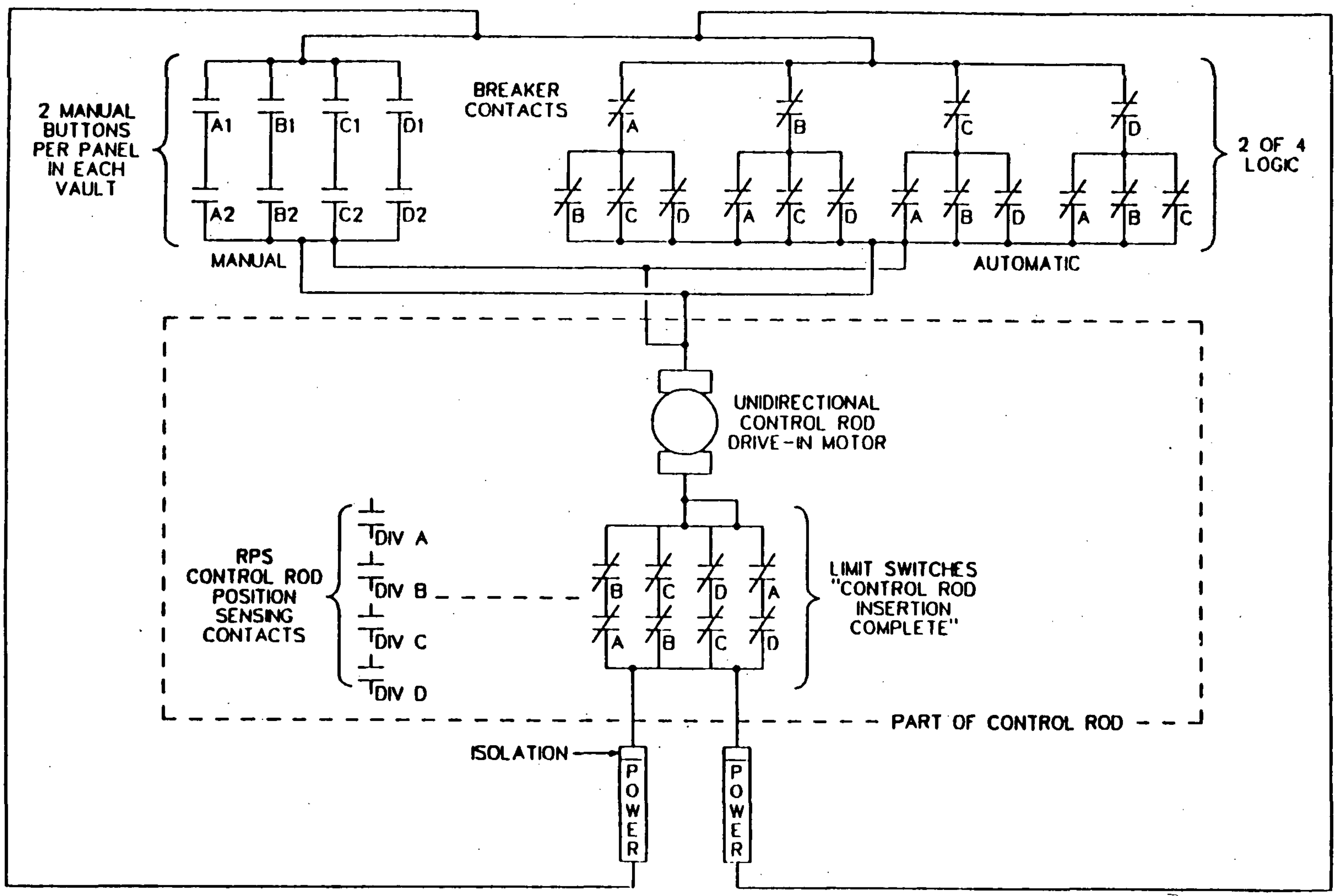


SCRAM LATCH
RELEASE LOGIC

87-293-03

Note: ALL COMPONENTS SHOWN ARE SAFETY-RELATED

Figure 7.2–2 CONTROL ROD SCRAM LATCH RELEASE SWITCHING LOGIC



CONTROL ROD SCRAM
MOTOR DRIVE IN CIRCUIT

Note: ALL COMPONENTS SHOWN ARE SAFETY RELATED

87-293-02

Figure 7.2–3 CONTROL ROD DRIVE-IN SWITCHING CIRCUIT

7.3

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7.4 Systems Required for Safe Shutdown

Because of the inherent safety characteristics of the reactor module, no plant control system (PCS), or individual module reactor protection system (RPS), are required to assure public safety. In events that include postulated failure of active heat removal systems and successful actuation of the shutdown system, the passive RVACS will continue to cool the reactor indefinitely and no active control is required. The PRISM inherency features, acting without either PCS or RPS functions, will bring the reactor to a hot standby condition. It is then necessary to insert the control rods to bring the reactor to a refueling temperature for service. The control rods are used by the RPS for all automatic trips and by the PCS for reactor control and fast runback/shutdown following design basis events (including safe shutdown earthquakes or the detection of failed fuel elements).

A reactor scram includes insertion of the control rods and a coastdown of the EM pumps. The primary design basis for initiating an EM pump coastdown as part of a reactor trip is to cover the loss of electrical power design basis event. EM pumps do not provide the inertia to sustain flow following a loss of electrical power as do mechanical pumps. The flow in an EM pump stops as rapidly as does the decay of the magnetic flux following an interruption of the electrical power. The sudden cessation of flow through the reactor core could lead to local overheating, cladding failure and possibly limited boiling within the inner fuel assemblies. Consequently, the primary flow is lowered at a rate to match approximately the inherency features response. This is referred to as the primary flow "coastdown." With EM pumps, coastdown is provided by the inertia stored in a synchronous machine. With the interruption of electrical power to the EM pumps and their synchronous machines, the machines begin to convert stored energy to electrical power to provide the requisite primary flow "coastdown."

7.5 Safety Related Instrumentation

An overview of the reactor instrumentation system is given by Figure 7.5-1. Three main sub-groups are identified:

1. The Reactor Protection System (RPS)
2. The Plant Control System (PCS)
3. Sensors within the reactor module which are defined by systems interfacing with the reactor.

The first group consists of instrumentation which provide signals for initiating reactor trips, and instrumentation which will provide the operator with knowledge on the state of the reactor module in the event of an accident. The sensors, signal conditioning equipment, and logic system are Class 1E equipment. The second and third groups are not 1E systems. The second group contains all of those sensors used for plant control and operator knowledge on the state of the plant during normal operations. The final group consists of those monitors whose function and requirements are defined as part of the specification of a component or subsystem.

This section covers the instrumentation in the first group, which provides protection and accident monitoring. The control system instrumentation is described in Section 7.7. Monitoring instrumentation is defined in Section 7.6 of this chapter. Instrument systems in the third group are covered in the descriptions of the appropriate interfacing reactor system.

7.5.1 Reactor Protection System Instrumentation

The Reactor Protection System is activated when a threshold signal is generated by any one of the safety function parameters:

1. Reactor core neutron flux
2. Reactor core outlet temperature
3. Cold pool temperature
4. Pump Discharge inlet pressure
5. Primary sodium level

7.5.1.1 Flux Monitoring System

Neutron flux leaking from the core provides a measure of core flux, and is monitored to detect rapid changes in flux. After calibration the flux monitors also indicate the thermal power produced in the core. Flux monitors provide the most rapid indication of any potential overheating condition and are used to protect the integrity of the fuel pins by initiating a reactor trip or control shutdown.

PRISM requires the neutron flux monitors cover a range of 1×10^{-6} % to 130% full operating power for control and accident monitoring.

The safety-related power range flux monitor will provide a signal to indicate excessive reactivity insertion and prevent overpower in the reactor core. The flux monitor provides signals proportional to the neutron flux leaking from the core. The neutron flux is directly proportional to the thermal power being generated. Exceeding a specified flux amplitude will initiate a trip.

The flux monitors are located in drywells in the concrete silo. These drywells will be surrounded by neutron thermalizing blocks and gamma shielding to provide a suitable environment for the detectors. The signal conditioning equipment complementing the flux monitors is located in the instrumentation vaults. Connection cables between the flux monitors and the instrumentation vaults will run in a protected conduit. The expected temperature in the drywell during normal operating conditions is not expected to exceed 500°F.

Low-level range flux detectors are provided in drywells near the outer radius of the UIS and are located six inches above the top of the reactor core to measure core fission power and subcriticality during shutdown and refueling. This flux detector set is also used along with special core assemblies containing neutron sources during the initial core loading. These detectors are used during refueling and startup operations. They are not safety related instruments.

7.5.1.2 Core Outlet Temperature Monitoring

The primary goal of the reactor protection system is to prevent or limit release of radioactive material beyond the site boundaries. The fuel pin clad is the first barrier, containing radioactive gases released during fission. The temperature of the clad must be maintained below failure thresholds. The outlet temperature of sodium leaving the reactor core, (an indication of clad temperature) is measured. If the core outlet temperature exceeds a threshold, the reactor is tripped.

A temperature monitoring sensor provides a signal to indicate excessive sodium temperatures in the hot pool. Its function is to prevent overheating of the load bearing structures, especially the reactor vessel; and prevent the fuel pin clad/fuel interface reaching a temperature at which an eutectic alloy is formed, which could cause mechanical damage to the clad.

The thermowell will be attached to the upper internal structure (UIS). The sodium flowing from the core is adequately mixed, providing a mixed mean outlet temperature, at a location just below the stored fuel assemblies. These thermowells will penetrate the head and be attached to the UIS. Each thermowell will be capable of holding four sensor elements, of which one will be the primary sensor with the other three elements spare or providing calibration functions. Four thermowells will be provided to allow for redundant and diverse measurement of the core outlet sodium temperature.

With the PRISM inherently safe design, natural circulation cooling and core reactivity limitations assure that boiling temperatures are not reached. The RPS safety setpoints are at temperatures well below boiling.

7.5.1.3 Loss of Sodium Flow Through the Reactor Core

Loss of sodium flow can also cause high fuel pin temperatures. A measure of the primary flow rate will be provided.

When the reactor is generating power, a low sodium flowrate signal will initiate a reactor trip. No direct measurement of the total core flowrate is feasible. The loss-of-flow measurement will be indirect, using the EM pump discharge pressure as an indicator of primary flow.

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7.5.1.4 EM Pump Discharge Pressure

Sodium pressure is measured at the EM pump discharge. The sensors consist of NaK-coupled bellows with strain gage instrumented diaphragm pressure transducers as shown in Figure 7.5-2. Four instrument pipes connect the pump discharge with wet wells under the closure. One bellows assembly is located at the measurement site (in the wet well) and isolates the primary coolant from the NaK while transferring the pressure to the NaK. A tube connects that bellows with a bellows above the closure. The upper bellows separates the NaK from an oil-filled chamber. The pressure transducer attaches to the chamber and the oil couples the pressure signal from the NaK to the transducer diaphragm. The diaphragm isolates the oil from the surrounding air and carries a strain gage to provide a signal proportional to coolant pressure. The NaK-coupled bellows are provided in wet wells through the closure and are replaceable. They use simple in-reactor mechanical components with high reliabilities that are anticipated to last the life of the plant. The strain gage assemblies, being less reliable, are external to the module in the HAA and are replaceable. The sensors are normally referenced to ambient air in the HAA. For calibration, a known pressure source is supplied to the air vent, resulting in a simple, fast, reverse calibration.

7.5.2 Accident Monitoring System Description

The PRISM reactor protection system also incorporates several monitoring systems that provide important information, but do not have safety trip functions. These are the reactor vessel heat removal system effectiveness and emergency power supply condition.

Those sensors that provide information required by the operator to assess the state of the reactor system and its associated heat removal systems are part of the accident monitoring system and are Class 1E. The accident monitoring function sensor requirements are indicated in Figure 7.5-3. Further requirements for each of the subsystems shown in this figure are discussed below.

7.5.2.1 Trip Actuation and Function Completion Sensing

Sensors are provided to indicate that the trip actions are completed, and the operator is informed that the required actions have been taken. Any device that performs a trip function will be monitored to ensure the action is completed.

Each control rod consists of the absorber with a knob at the top of the rod-like structure. The knob is held in a collet arrangement such that a downward motion of the collet out of a restraining cylinder, allows the absorber to drop. The collet is connected by a long rod to the top of the control rod assembly (above the reactor head) where a continuously energized electromagnet holds the rod. An interruption of the electrical current to the electromagnet (a trip) allows the rod to release the absorber to drop into the core under the force of gravity. A limit switch is opened when the control rod has completed its full insertion travel. The state of this limit switch is monitored.

Each control rod can be moved using small motors for trimming the power level during operation. A more powerful motor has sufficient capability to override the trimming motor and drive the rods into the core. This motor is actuated following a trip, and remains activated until the power is switched off by deliberate action of the operator. The state of these two motors is sensed.

The closure of the valves on all head penetrations is confirmed. The state of the active portion of the valve is monitored. A limit switch indicates complete closure of these head isolation valves.

An auxiliary synchronous machine supplies coastdown power to the EM pump when the normal supply is de-energized, Figure 7.5-4. This machine is connected in parallel with the pump. Upon de-energization of the EM pump, the stored kinetic energy is utilized for coastdown. The state of this synchronous machine will be monitored during the run-down to confirm expected performance.

7.5.2.2 Heat Generation Source Monitoring

Three primary heat generating sources exist within the reactor vessel: 1) reactor core 2) electromagnetic pumps, and 3) stored fuel. Each of these sources is considered in detail below.

Measurement of quantity of heat generated in the core is provided by the mixed mean outlet temperature, the mixed mean inlet temperature, and the flow rate of sodium through the core. The first and last parameters were discussed earlier in Section 7.5.1. The remaining parameter is discussed in this section of the report. The temperatures measured at the outlet of the four EM pumps will be used to estimate the core inlet temperature. Each pump will have a single drywell containing four temperature measuring devices.

A total of 19 spent fuel elements can be stored in the reactor with a maximum heat generation rate of approximately 0.2 MWt. This could be up to about 20% of the decay heat generated by the core, but more typically will be less than 10%. The temperature drop across the bundles is measured, and it will be assumed the total quantity of sodium flowing through the core also flows over the fuel bundles stored in the vessel. The inlet temperature to the fuel storage section is measured. The mixed mean outlet temperature will be measured using drywells at each IHX, each drywell containing temperature sensors.

The electrical input to the pump is assumed to be totally converted to heat energy. No significant heat losses except to the sodium pool have been identified. Therefore, the heat input from the pumps will be obtained from measuring electrical power input (wattmeter).

7.5.2.3 Shutdown Heat Removal Systems

The quantity of heat transferred from the reactor vessel via the intermediate heat exchanger (IHX) requires information with respect to the primary sodium flow through each unit. It is impractical to measure this information directly. An indirect measure of the heat transfer will be obtained by measuring the increased enthalpy on the secondary side of the IHX. The temperatures at the inlet and outlet of the IHX units will be monitored.

The quantity of heat transferred in the RVAC system is estimated from the enthalpy change in the air flowing though the duct system. The change in enthalpy will be calculated from the mass flowrate of air, its humidity, and the temperature differential between inlet and outlet flows.

Air mass flow is a function of air density (temperature), air velocity, and duct cross-sectional area (fixed by the physical design of the system). The present concept for monitoring the air mass flowrate through the RVACS is based upon the use of a pitot tube dynamic and static air pressure measurement system with two sensors located in the exit chimney of each of the four RVACS stacks. The air flowrate in each stack is determined from the differential pressure measurement and the air density obtained from air temperature measurement.

A thermistor type device is used to measure the air temperature. The air temperature can be measured to +/-2% of full scale and the air velocity can be resolved to +/-2% of full scale with a +/-0.2% repeatability. There are numerous types of meters that could work reliably in such an environment.

7.5.2.4 Emergency Power Supply Monitoring

Power to the critical instrumentation systems will be provided by local Class 1E battery-backed supplies. The status of the 1E power supply will be monitored.

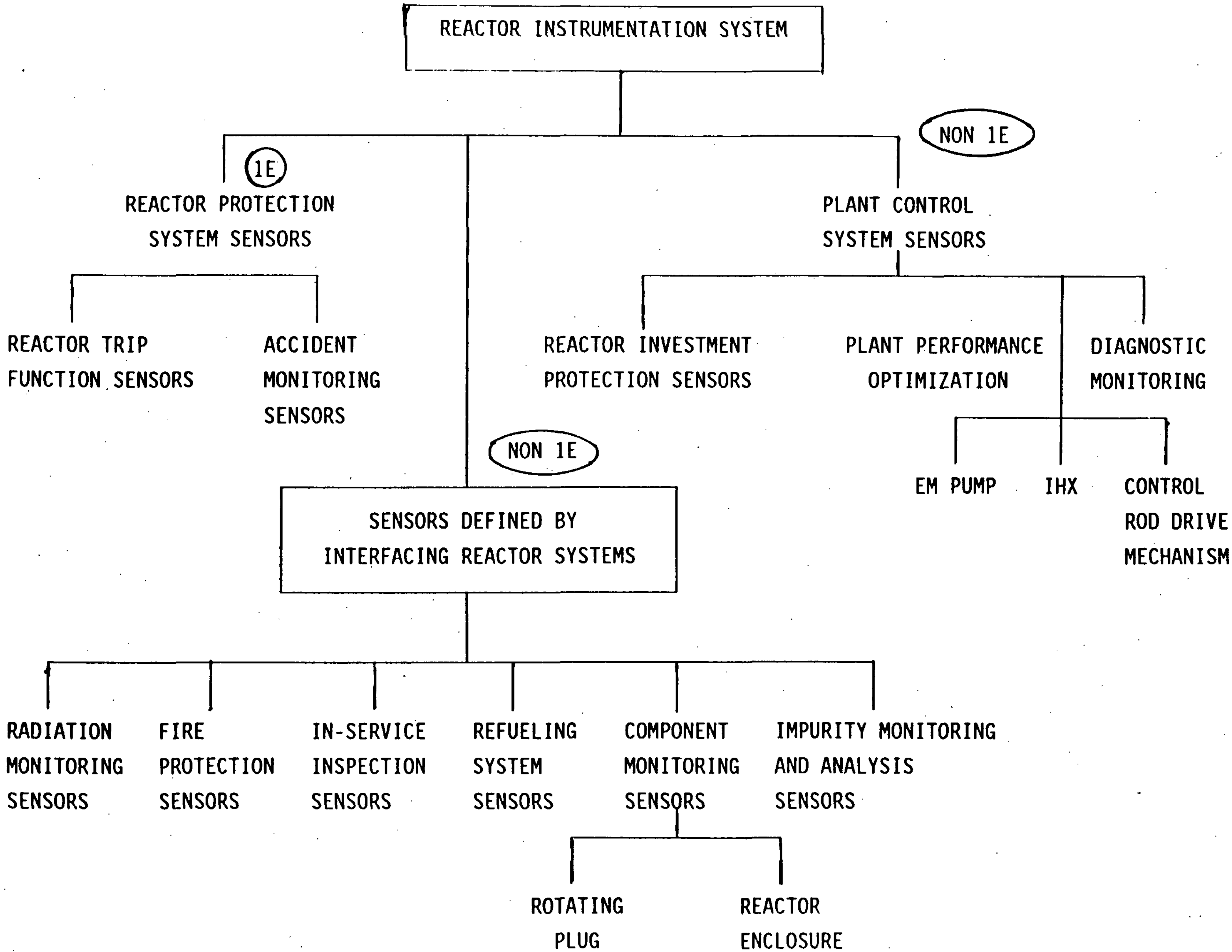


Figure 7.5-1 - OVERV[illegible] REACTOR INSTRUMENTATION

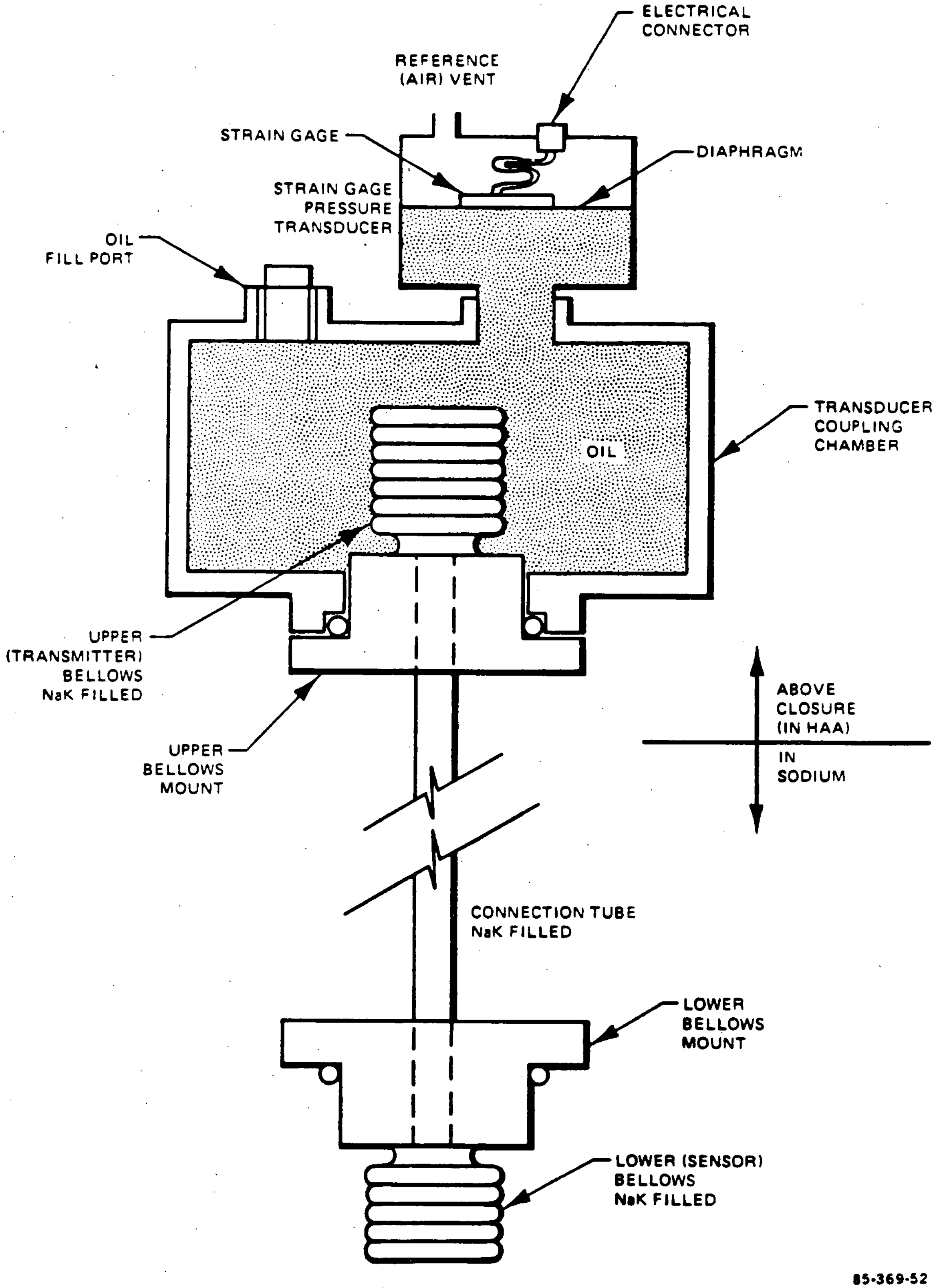
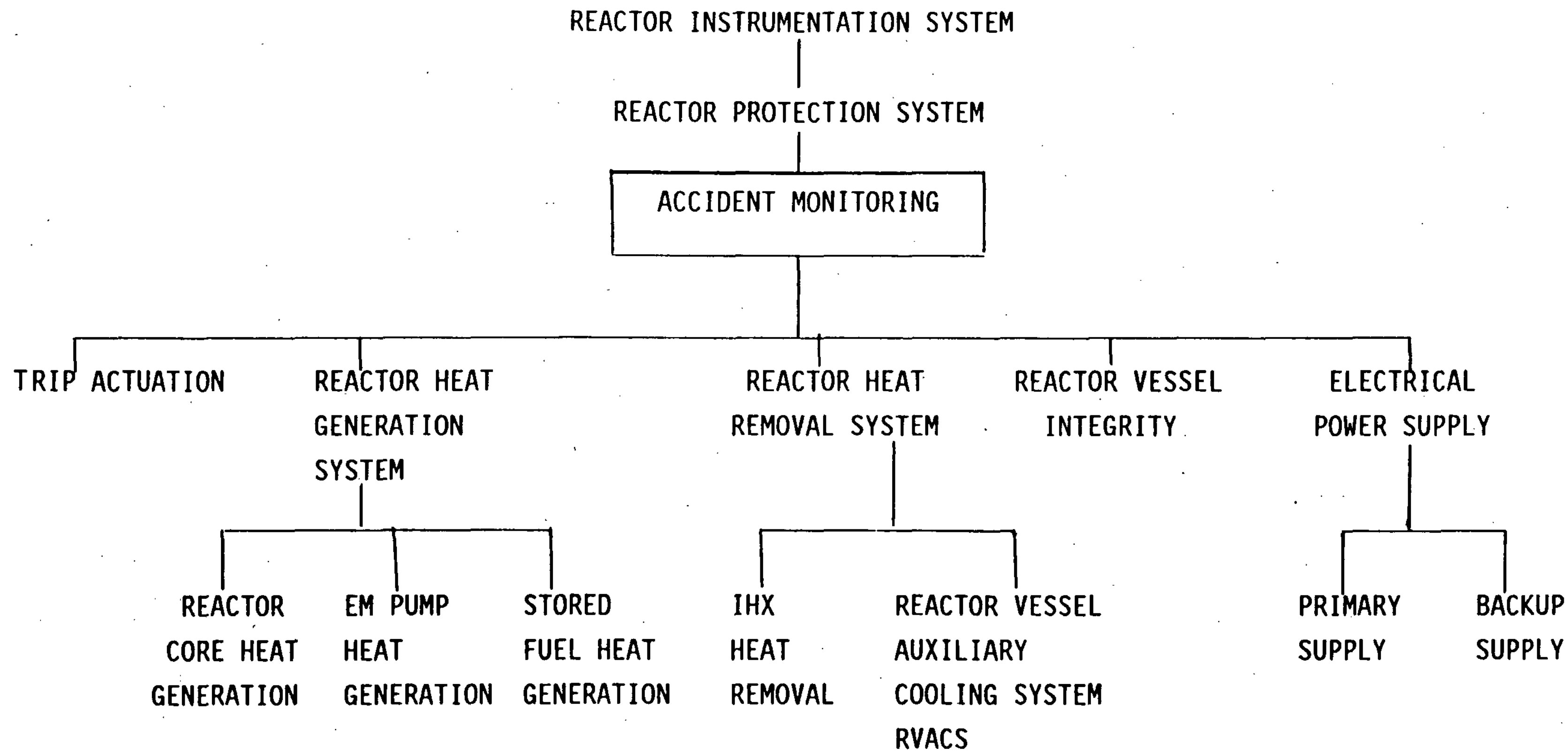


Figure 7.5–2 SODIUM PRESSURE SENSOR



<u>PURPOSE</u>:	CONFIRM TRIP FUNCTIONS ARE COMPLETED (PROVIDE STATE OF TRIP FUNCTIONS)	QUANTIFY HEAT GENERATION RATES AND TEMPERATURE PROFILE IN THE REACTOR VESSEL	QUANTIFY HEAT REMOVAL RATES FROM THE REACTOR VESSEL	CONFIRM STATE OF REACTOR AND CONTAINMENT VESSEL INTEGRITY	QUANTITY THE AVAILABILITY OF NORMAL AND EMERGENCY POWER SUPPLIES

Figure 7.5-3 - [illegible]NT MONITORING SYSTEM

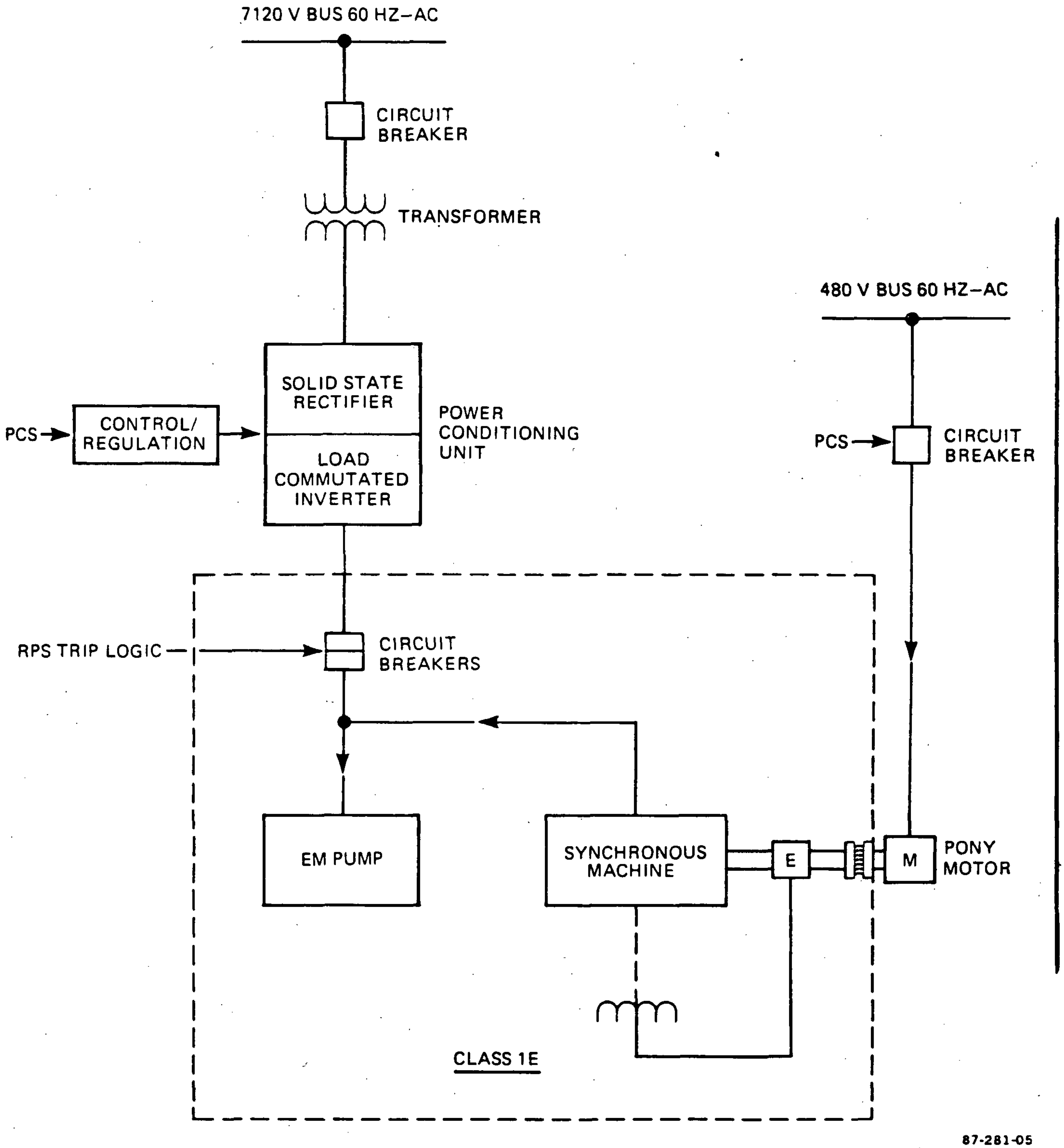


Figure 7.5–4 E. M. PUMP POWER SUPPLY

7.6 Instrumentation and Monitoring Systems

An overview of the instrumentation systems defined by various reactor subsystems, and of monitoring systems performing specialized diagnostic functions is given in Figure 7.5-1. Each of these systems are covered in more detail below.

7.6.1 Radiation Monitoring System

The primary protection against radiation release is provided by several mechanisms.

1. The inherent safety features of the reactor system.
2. The reactor protection system which functions to limit the damage to the reactor due to accidents.
3. The accident monitoring system which provides the operators with quantitative information on the state of the plant, allows the possible initiating events to be recognized, and provides information to evaluate the incident and assure protective actions are effective and complete.

The radiation monitoring system will measure radiation levels during all plant operating, shutdown, abnormal, and accident conditions. Monitors will be positioned in the areas surrounding the module, in the head access area, the safety qualified equipment vaults, and at the site boundaries.

7.6.2 Fire Protection Monitoring

Fire protection monitoring will be provided in accordance with 10CFR50.48 and 10CFR50, Appendix R.

7.6.3 In-Service Inspection

In-service inspection requirements and approach are discussed in Chapter 5.0.

7.6.4 Impurity Monitoring

The PRISM reactor is designed to operate as a hermetically sealed system. It is only opened during annual refueling periods. The reference design for primary system processing and impurity monitoring uses a permanent cold trap. The system is used intermittently to purify sodium from one of three adjacent PRISM reactor modules (a power block). The purification system and associated impurity monitors are contained in a hardened building. The temperature distribution in the cold trap crystallizer and economizer is monitored with thermocouples. The pressure drop across the cold trap is monitored during operation to assess the degree of crystallizer plugging. A plugging temperature indicator is used to measure sodium impurity concentration. Since the monitoring equipment is an integral part of the purification subsystem the primary sodium is not monitored during most of the reactor power operation. Sodium samples can be extracted using the sodium primary pump, with a separate return line to the vessel. The sodium multipurpose sampler is used for:

1. obtaining sodium samples
2. equilibration of metal specimens for analysis of active oxygen, carbon, or hydrogen in the sodium
3. sampling for particulates in the sodium

The multi-purpose sampler is located in an inerted cell and remotely operated with master-slave manipulators. A flow-through sampler is used to obtain a sample of sodium from the flowing sodium stream. It is located in a shielded, inerted glove box in the head access area (HAA). Sodium monitoring and sampling will be done only during reactor shutdown.

To measure for gross in-leakage of air into the reactor cover gas space, provision is made to obtain a grab sample of cover gas. The sample is obtained using a grab sampler connected to a one-inch nozzle on the reactor vessel top head. A 100 cc sample of gas is obtained. After the sample is in the collector it is allowed to decay for two weeks and cool to the ambient HAA temperature. The sample is taken to the Laboratory for analysis.

7.6.5 Refueling Neutron Flux Monitor

Low-level range flux detectors are provided in drywells near the outer radius of the UIS and are located six inches above the top of the reactor core to measure core fission power and subcriticality during shutdown and refueling. This flux detector set is also used along with special core assemblies containing neutron sources during the initial core loading. These detectors are withdrawn from the core area during normal power operation and are only used during refueling.

Neutron flux in the reactor core changes by a factor of 10^{11} or more from initial shutdown condition to operation at full power. When the core is initially installed into the reactor the neutron flux in the core is extremely low, and may be below the reliable operation range of a monitor.

7.6.6 Diagnostic Monitoring

An overview of diagnostic monitoring systems is given in Figure 7.6-1.

7.6.6.1 Fuel Element Detection and Location (FEDAL) System

The FEDAL system for the PRISM reactor will employ delayed neutron monitoring, and fission gas monitoring for breached pin detection with fuel pin gas tagging for assembly location. System configuration and functions are as follows:

Subsystem	Key Functions
Delayed Neutron Monitoring	- Monitor Primary Sodium for Fuel-Sodium Exposure
Fission Gas Monitoring	- Monitor Reactor Cover Gas for Fission Gases - Indicate Number of Breached Pins in the Core
Pin Gas Tagging/Tag Recovery and Analysis	- Locate Assembly with Breached Pin

Two delayed neutron monitoring stations, a fission gas monitor, and fuel pin gas tagging are employed in each reactor. A single tag gas recovery and analysis system services all nine plant reactors. Given that the reactor technical specification allows operation with up to two fuel pin breaches in the core for extended periods, a FEDAL systems will be required to indicate to the operator that these limits are being maintained. Thus, each reactor is equipped with FEDAL instrumentation that indicates the status of the core in terms of the number of breaches, their fission gas, and DN activity.

A breach is considered any break in the cladding allowing material to pass into the coolant or into the pin. Breached pin detection will require cover gas monitoring to indicate fission gas releases by breached fuel and primary sodium monitoring to indicate fuel-sodium chemical reactions. A breach may release fission gases only or progress to fuel-sodium exposure and release both fission gases and solid or liquid fission products.

The use of fission gas monitoring of the cover gas verifies fission gas inventories are maintained well below source term design basis. The reactor containment boundary consists of the reactor head and containment vessel. A conservative fission gas inventory and containment leakage design rate is assumed. It can be postulated that defective fuel with initial weld defects could allow significant levels of fission gases to accumulate in the reactor. The use of the fission gas monitor in each module will detect these birth defects and the reactor will be shut down before inventory levels of concern are approached.

It should be noted run beyond clad breach (RBCB) operation with a small number of breached fuel element clad is not a safety concern and the FEDAL system is not safety related. Testing has shown that breaches behave in a very benign manner. A breach, once it develops, progresses very slowly and in a benign manner. Another important consideration is that clad breaches are random occurrences and do not propagate from pin to pin.

7.6.6.2 Fission Gas Monitor

The function of the fission gas monitor is to detect pin (fuel element clad) breaches, count the number of pin breaches in the core, and transmit the information for operator display. The fission gas monitor samples the cover gas and through analysis of the gamma ray spectra, determines the concentrations of selected fission gases. Data from the fission gas monitor is processed to inform the operator on the state of the core in the reactor.

A passive-diffusion fission gas monitor (illustrated in Figure 7.6-2) is employed for continuous cover gas monitoring. Fission gases in the cover gas, released from breached fuel pins, are transported from the cover gas to the detector by diffusion. A key feature of the design is the diffusion tube which extends from the cover gas (just above the sodium level) to the collimator within the reactor head. Gas transport time is such to allow the high Ne-23 levels decay to near background yet permit the fission gas isotopes to reach the gamma detector field of view. A heater, near the top of the tube, maintains the tube at hot pool temperatures to prevent sodium condensation and maintain a static temperature inverted gas column.

Fission gases are detected and identified by a gamma detector located above the head at the end of the collimator. Detection is with a germanium (Ge) crystal which generates an electrical pulse whose voltage amplitude is proportional to the energy of the gamma rays emitted by the fission gases. The crystal is maintained at a very low temperature by housing it in a liquid nitrogen filled cryostat.

7.6.6.3 Delayed Neutron Monitoring

The function of the delayed neutron monitor is to detect fuel exposed to primary sodium. The delayed neutron (DN) monitor continuously monitors the primary sodium for the presence of sodium borne fission products that decay by neutron emission. These products (mainly bromine and iodine) are released by the fuel into the sodium. As they decay over time, they release neutrons which are detected by the DN detectors located within the main heat exchangers.

Delayed neutron monitoring (illustrated in Figure 7.6-3) is with two in-reactor detector stations. Each station has four DN detectors with one being an in-place spare. The DN detectors are located in an IHX drywell. Signals from the tri-detectors are processed to provide a continuous reading of several diagnostic parameters that are used to indicate the amount of fuel to sodium exposure. These parameters include 1) the sodium transport time for the DN emitters traveling from the breached assembly to the detectors, and 2) the isotopic holdup time which is a measure of the effective aging of the DN precursor between their birth in the fuel and their release to the sodium. These are combined with measured count rate information to compute the equivalent recoil area for any exposed fuel in the core.

7.6.6.4 Fuel Pin Gas Tagging

Gas tagging will be employed to locate the assembly containing the breached pin. Gas tagging consists of the addition to the fuel pins of small amounts of gas having a unique isotopic composition for each assembly. When the pin cladding fails, the tag gas is released which makes it possible to locate the failed assembly by mass spectrometric analysis of the reactor cover gas. Lower cost Ar-Ne tag gas systems are usable with helium cover gas. A total of 150 tags (one for each fuel and blanket assembly) are used, with a common tag for all the pins of each fuel assembly. The tag gas is loaded into each pin by an evacuate-and-backfill technique. Candidate tag isotopes are Ne-20, Ne-21, Ne-22, Ar-36, Ar-38, and Ar-40. These are blended in various amounts to make up the 150 unique tags.

7.6.6.5 Tag Gas Recovery and Analysis

An off-line, plant-wide approach is employed for tag gas recovery and analysis (Figure 7.6-4). The tag gas recovery and analysis equipment is located in the fuel cycle facility and the cover gas is (with the tag) transported from the modules to the radwaste building. The gas is transported in a cover gas vehicle which is part of the cover gas cleanup system. For tag gas recovery and analysis, the reactor is shut down and the cover gas vehicle is connected to the reactor. The cover gas (with the tag) is transferred to the vehicle storage tank, and the reactor is charged with fresh helium. The vehicle is disconnected and moved to the radwaste building where it is connected to the tag recovery and analysis equipment.

At the radwaste building, the helium cover gas is passed through charcoal beds to first remove any fission gases. The tag gases are recovered from the cover gas by passing the helium through cryogenic temperature charcoal beds. The tag gases are then further concentrated and passed through a mass spectrometer for identification.

Once the tag species present in the cover gas are known, assembly location in the core is determined.

7.6.6.6 Analysis of FEDAL Systems

Operation with failed oxide fuel testing program conducted in EBR-II showed a cladding breach allows only very minute quantities of fuel release into the sodium, the main releases being fission product gases. When a breach occurs, sodium comes in contact with the fuel inside the pin, the sodium reacts with the fuel and the resulting reaction products cover up the breach area. This in a sense seals over the breach and essentially prevents fuel from being transported into the primary sodium. The main fission product releases from a breach are the fission gases. These bubble up through the sodium and the long life gases accumulate in the cover gas. Thus from a contamination standpoint, operation with two breaches in the core is not a limiting condition. The behavior of failed metal fuel is expected to be even more benign.

Based on fuel reliability, breached assembly recovery time and reactor availability considerations, the PRISM reactor will operate with failed fuel to the next refueling shutdown with up to two breaches residing in the core or until reaching FG or DN limits. Studies indicate that for a wide range of fuel reliability values, most of the availability gain in achieved by allowing up to two breaches in the core with residence times up to 180 days. The PRISM reactor operates as a sealed system which requires about two weeks to replace an assembly. This makes it advantageous to allow the breach to reside in the core to the end of the normal refueling outage.

The fuel for PRISM is expected to be very reliable with a nominal failure rate on the order of 0.2 breaches/reactor/year. Converting this into operation conditions, approximately 85% of the time the reactor will operate without any breaches in the core, 13% of the time with one breach and about 2% with up to two breaches in the core. This means that PRISM will operate with failed fuel very infrequently.

7.6.7 Loose Parts Monitoring (LPM)

It is a design objective in PRISM to use a commercially available LPM system that meets the intent of Regulatory Guide 1.133, "Loose Part Detection Program for the Primary System of Light-Water Cooled Reactors".

The regulatory guide requires the detection of loose parts in light-water-cooled reactors during normal operation. The primary purpose of the loose-part detection program is the early detection of loose metallic parts in the primary system. Early detection can provide the time required to take appropriate actions to avoid or mitigate safety-related damage to or malfunctions of primary system components.

7.6.8 Safeguard and Security Systems

Safeguard and security information is discussed in Appendix C.

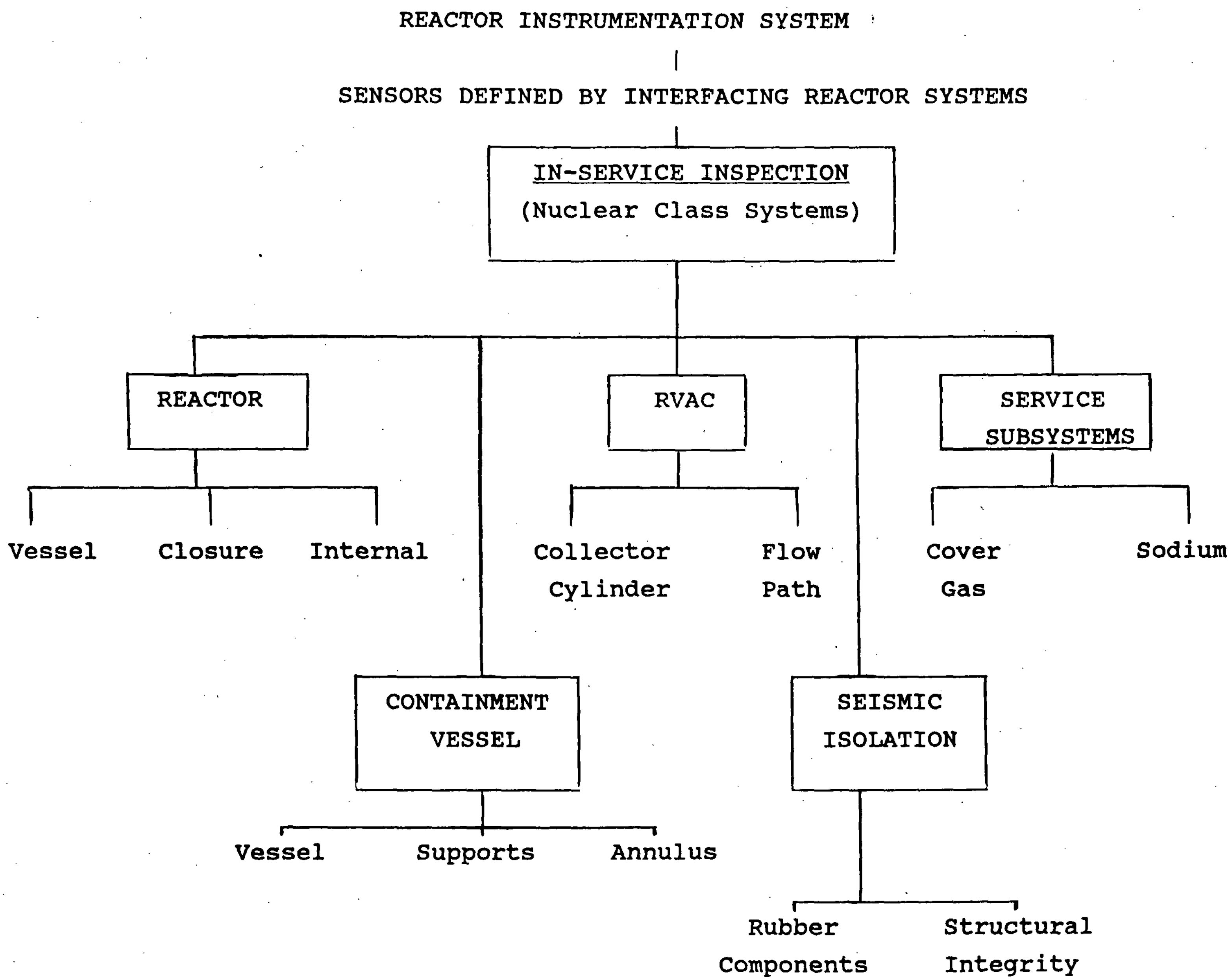


Figure 7.6–1 OVERVIEW OF ISI MONITORING SYSTEMS

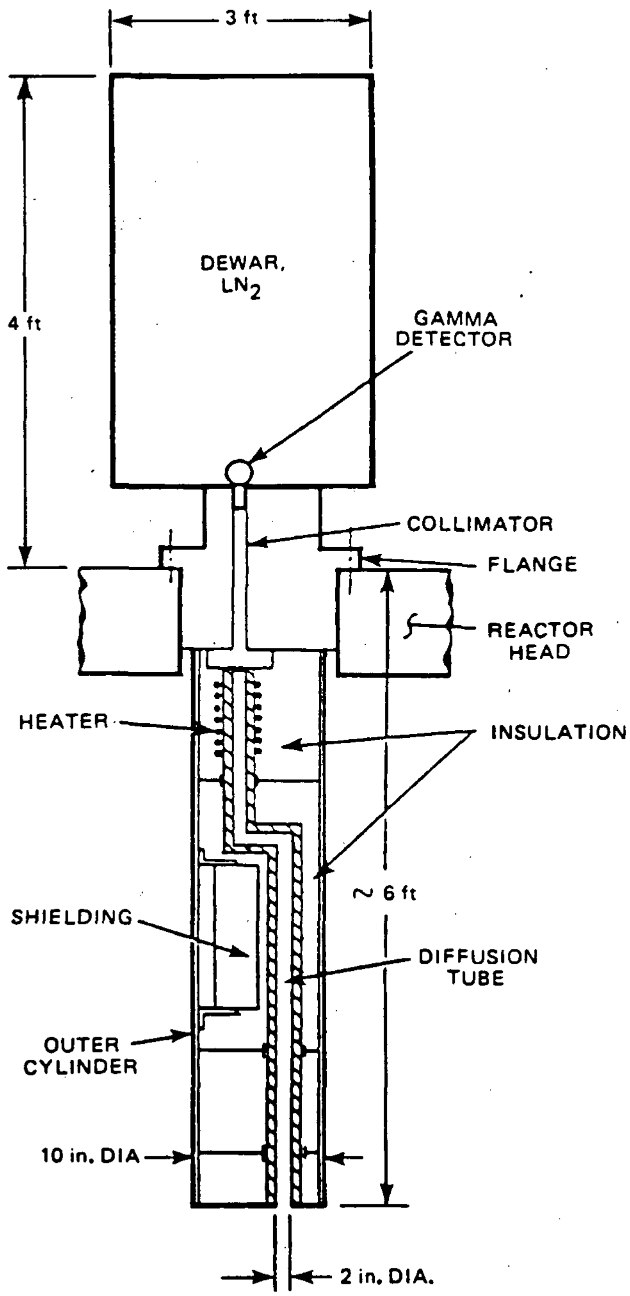


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Figure 7.6–2 PASSIVE DIFFUSION FISSION GAS MONITOR

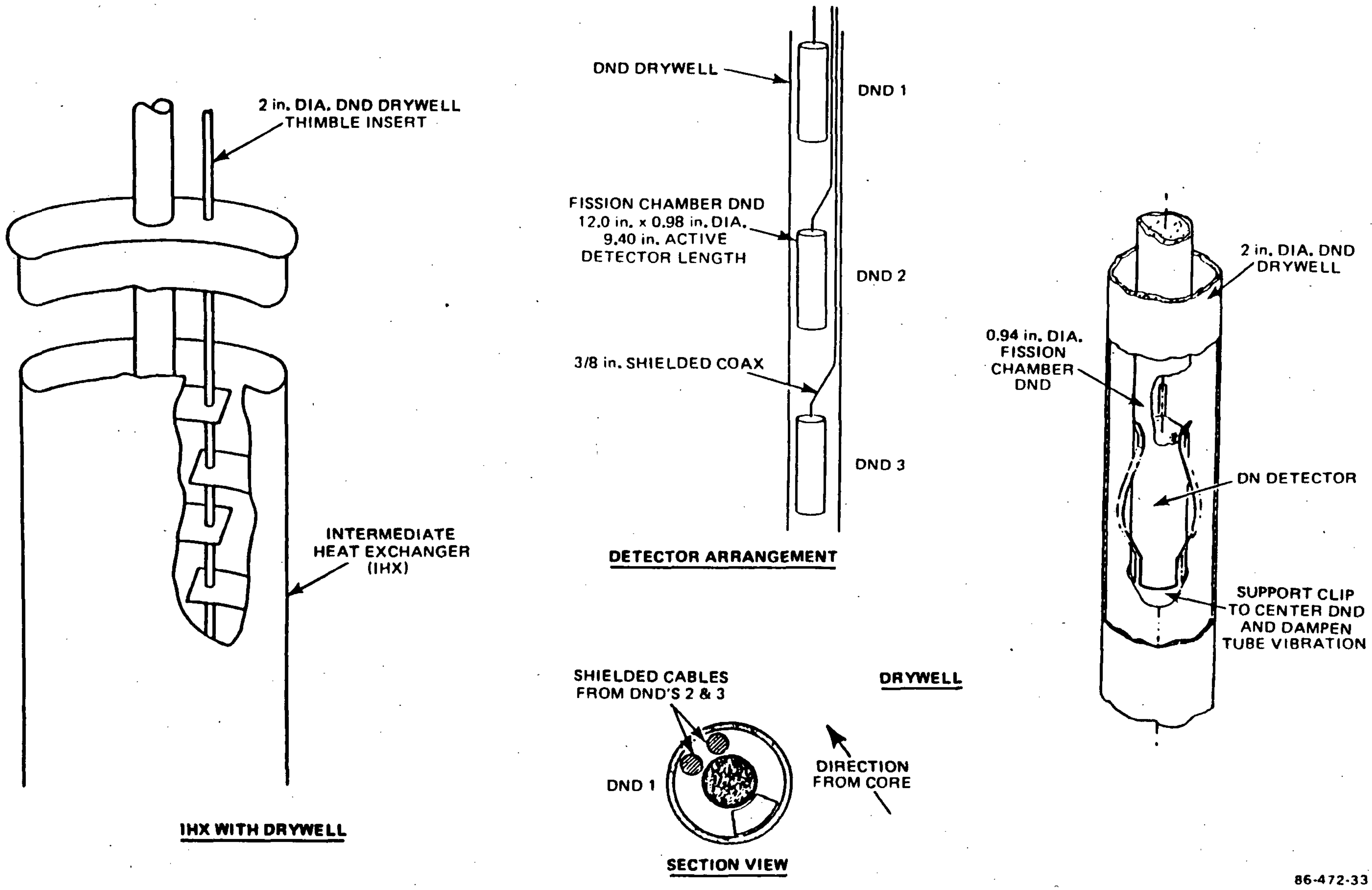

Figure 7.6–3 DELAYED NEUTRONS DETECTOR DESIGN

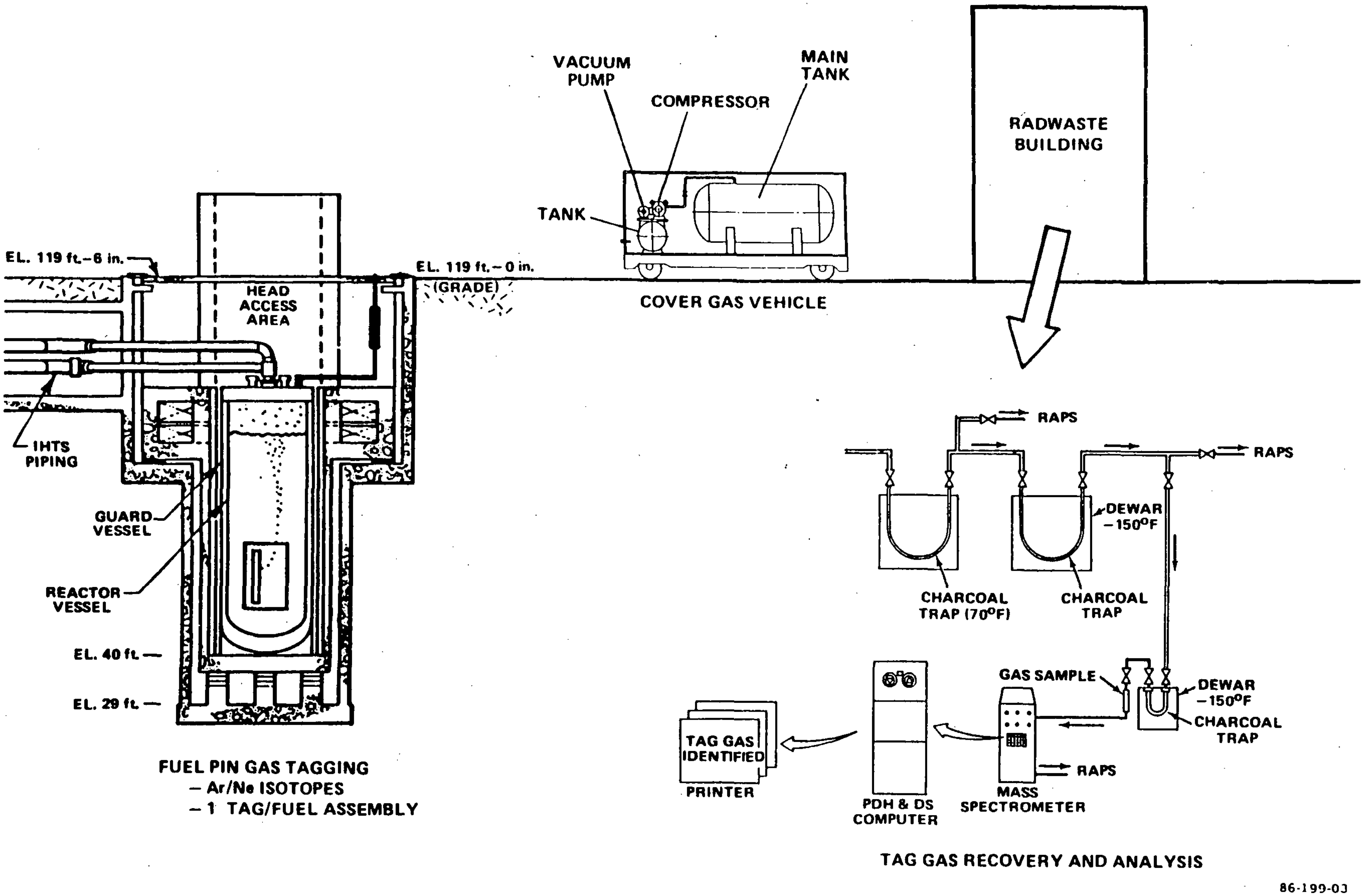

Figure 7.6–4 TAG GAS RECOVERY AND ANALYSIS

7.7 Control Systems Not Required For Safety

Digital computer-based systems are used for highly automated, reliable and efficient multi-module plant operation. A plant-level overview of the hierarchical distributed digital PCS and its relations to the RPS are illustrated in Figure 7.7-1. There are four major levels of control in the hierarchy:

- o Plant level
- o Block level
- o System level
- o Local (subsystem) level

The plant and block level controllers are located in the control building in the Information Management Center (IMC). System and subsystem controllers are distributed through the plant, to provide coordinated control of the multi-modular PRISM plant. The plant level control system coordinates operation of the three power block control systems, and each block controller coordinates its three NSS and one BOP control system. Each NSS and BOP controller in turn coordinate their respective subsystems and local controllers. This distributed hierarchy is further illustrated in Figures 7.7-2 and 7.7-3. As shown in Figure 7.7-2, the man-machine-interface is arranged such that each operator can observe and confirm the plant operation for which he has been assigned. Since all plant data is available to each operator interface, an operator is capable of assisting a fellow operator in monitoring and control during a period of high work load or an unplanned event.

The plant control system uses fault tolerant controllers and data highways with multiple redundancy at all levels of control to achieve high plant availability and reliability. In case of a controller failure, the automated test and diagnostic features of the system will immediately detect the bad channel and the system will automatically reconfigure around the faulted component. The diagnostic features provided identify failures to the lowest replaceable module level and this allows the failed channel to be repaired/replaced in a much shorter time than a second failure is expected to occur. This configuration results in a highly reliable control system. The plant control system will be in automatic mode most of the time, freeing operators for more appropriate judgmental tasks. Through appropriate controller input, an operator may exercise semi-automatic or manual control if required without disabling major automatic protective features, such as electrical power equipment protection (short circuit or overload), overpressure devices (rupture discs, relief valves) motor operated valve actuator protection (torque switches), etc.

The three power blocks, consisting of nine reactor modules and three steam turbine generators are automatically controlled and automated to a level such that each power block can be controlled by a single operator.

A high level of plant automation is used such that under normal circumstances when everything is working properly virtually no operator control action is required.

The plant level controller (PLC) automatically supervises and coordinates the balancing of load between the power blocks, and what contribution they make to the total power generation demanded by the grid controller.

The block level controller (BLC) automatically supervises and coordinates each nuclear steam supply system within the power block, and what its contribution to total turbine steam flow will be to meet the demands of the PLC.

The PLC also monitors the performance, response, and limitations on each NSSS and turbine generator unit, as coordinated by each BLC.

Most sequential plant operations are automated. Operator sequence start permissives and hold releases (allowing continuation to the next sequence step) are provided in the 0-25% power range. Continuous coordinated multi-module operation is provided over the 25-100% power range. Figure 7.7-3 shows the control structure used during continuous operation. The plant coordination function is a part of the plant level control function (shown in Figure 7.7-2) and is a part of the distributed plant control. This coordinates operation of the three power blocks to achieve plant level goals. The power block feed forward demand shown in

Figure 7.7-3 improves system response. On step changes it provides a setpoint equal to the subsystems estimated final steady state condition. The optimal power block coordination function, also shown in Figure 7.7-3 along with the feed forward, is part of the power block control system. These controllers provide optimal direction to the distributed local control subsystems, as shown in Figure 7.7-3. Hierarchically below the power block controllers, the local process sub- system controllers manipulate control rod drives, pumps, valves, the main turbine, electrical power systems and auxiliary systems. Should communica- tions be lost with the power block controllers, these local control systems would continue to operate the plant at the fixed load setpoint or initiate plant shutdown if needed.

Manual operator override of the automatic control mode results in removal of the appropriate portion of the system from coordinated automatic control. This is recognized by the remaining portions of the system which are on automatic control. The operator initiated manually generated commands are transmitted over the plant data bus to the appropriate local controller.

At the local controller level, the plant control system is partitioned into three module nuclear steam supply system controllers and the associ- ated turbine-generator/BOP controller for each power block. This is shown in Figures 7.7-2 and 7.7-3. The major subsystems which comprise the NSSS control system and turbine generator and BOP control system have multiple redundancy and are shown in Figures 7.7-3 and 7.7-4.

Distributed local control, as shown in these figures, enables the nine module NSSS control systems, the three turbine generators and BOP control systems to operate simultaneously. The distributed processing also reduces the information transmission required for subsystem coordination and display.

Fault diagnosis and appropriate plant investment protection actions are also performed in parallel at the local controller level throughout the plant. This permits rapid response to correct, limit or protect against possible component damaging plant events.

Figure 7.7-3 shows the concept of utilizing methods of modern optimal control. Mathematical models of the portion of the plant being controlled are incorporated into each subsystem controller. The model provides an estimate of variables useful for control which are not directly measured, and the accuracy of this estimating is assured by on-line updates of the model with real plant data. Feedback gains from the measured and calculated states are continuously adjusted to provide optimal control of each subsystem. Subsystem information is passed upward to the system and power block controllers where optimal coordination of the power block subsystems is performed and control directives are passed down from these higher level controllers to the subsystem controller.

The reactor control subsystem (RCS) provides local control over the reactor module operating power level and core sodium exit temperatures by automatically positioning the control rods to meet module power demand. To protect plant investment, the RCS can also initiate a fast runback of reactor power. Incorporation of the fast runback feature significantly reduces the number of reactor protection system (RPS) directed reactor module scrams. Display of rod positions and core power and temperature distribution calculations assist the operator in monitoring the performance of the RCS.

Figure 7.7-3 shows the remainder of the NSSS control system contains subsystem controllers for the primary and intermediate sodium heat transport systems, the steam generator and recirculation loop, and individual steam generator feedwater flow control. The primary sodium pumps can be energized by the PCS. These pumps are controlled by the PCS during startup and fast runback operations. The intermediate sodium pumps (main drive and pony motors) are constant speed, and are energized and tripped by the PCS. The steam generator system drum level, drum drain flow, recirculation flow and individual steam generator feedwater flow are all controlled by the PCS through the local NSSS subsystem controllers.

The turbine generator and other BOP subsystem controllers coordinate feedwater pump operation, turbine inlet steam pressure, turbine steam and bypass flows, condensate-feedwater train flows, and other support systems, such as gland steam seal, vacuum control, main generator excitation and voltage control, etc.

The control system, whose hierarchy and physical distribution has just been described, performs many functions. The aggregate of these functions is referred to as a "control engine". These functions and key interconnecting information flowpaths are shown in Figure 7.7-5. A description of these functions using the numbering convention in Figure 7.7-5 follows.

1. Data Acquisition and Validation

   The PCS is required to observe (monitor) the status of plant components and processes and determine the operational and control states of the plant. The Data Acquisition function gathers data from the subordinate PCS subsystems (i.e., NSS, BOP, DHTS, etc.) validates the source, updates the archive and distributes the data to other internal functions. The information and data is aggregated (grouped logically and meaningfully) by the Decision Support function and made available for selection as required.

   The data/information sampling rate is individually set for each input and may be changed as the plant state dynamics or data communication requirements dictate. Data are validated by comparing the sampled data against archived records of data acquired from previous sampling. If discrepancies are observed, the data packet is flagged for subsequent analyses. The digital data packet contains source identification and time tagging information.

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2. State Estimator

The State Estimator provides the capability to observe plant status when direct sensing of process variables or component status is not available. Data are composed from related measured data available from the Data Acquisition function, archives or by estimating variables. The State Estimator validates incoming data and, if unable to validate, identifies the reading and substitutes an estimated value based upon historical and current related data. This may be necessary if the Data Acquisition function has flagged a data packet as having come from a non-validated signal source such as a defective component.

3. Performance Analyzer

The Performance Analyzer monitors the performance of plant startup, shutdown and energy production maneuvers, analyzes the economics of reactor modules and turbines operating at different power levels, and evaluates the states of the plant. The Performance Analyzer uses validated, composed and estimated data from the State Estimator. The Performance Analyzer assesses overall performance and determines the states of systems. It anticipates future states based upon observed trends and selected strategies, then routes the outcome of the analyses to other internal functions.

4. Diagnostician

The Diagnostician observes plant status and supports the function of making plant control decisions. The Diagnostician performs internal system diagnostics and sequence-of-events monitoring. The Diagnostician provides for timely fault identification and anomaly diagnosis so that the Control Strategist can select alternate strategies. The Diagnostician has a key role in meeting the requirements to sustain plant operation during off-normal conditions.

Invalid data, discrepancies between expected and predicted performance, and anomalies or failures detected by self-tests are verified, analyzed and evaluated. The Diagnostician determines if the anomaly detected is an incipient failure, a faulted component, a false alarm, is a process upset, an inaccuracy in the model parameters, or is a perturbation caused by machine drift and/or plant aging. When the cause or condition is identified, the Diagnostician routes the information to the Configuration Manager and, ultimately, to the Control Strategist for a control decision.

5. Control Strategist

The Control Strategist has a major role in coordinating plant control, making plant control decisions, effecting plant control, and accepting plant direction. The Control Strategist contains the control logic used to meet requirements for:

a. operation of modules and turbines at different power levels,
b. sustaining operation through plant maneuvers under loss of a major component or system,
c. responding to off-normal service limit events,
d. automatically accepting load demands from the grid dispatcher,
e. determining plant state maneuvers, and
f. allocating demands for, and rates of change in, energy production and conversion.

The Control Strategist makes decisions regarding the importance of the control objectives and goals based upon inputs from the operator, Configuration Manager and Performance Analyzer. It resolves conflicting goals and objectives and selects the corresponding control strategy to meet the goals and objectives that are given precedence. The Configuration Manager assists the conflict resolution by identifying acceptable plant component and process configurations.

The Control Strategist receives input from the Performance Analyzer about the current plant state and receives new goals and target states from the operator. The Control Strategist first identifies which goals are currently achievable based upon the current configuration of the plant components and processes. A database maintained by the Configuration Manager is available to the Control Strategist listing the current operational capabilities and limitations of equipment and systems. The process of selecting a control strategy for each achievable goal is executed according to a prioritized scheduling scheme.

6. Configuration Manager

The Configuration Manager uses observed plant status and component and subsystem availability data to determine acceptable energy production and conversion configurations. The configurations are those configurations to be used in order to meet applicable plant operational requirements such as:

a. sustain operation under loss of a major component or system,
b. respond to trips, loss of feedwater, overcooling, depressurization, etc.,
c. meet reliability goals,
d. minimize forced outages.

The Configuration Manager serves as a continuous on-line advisor to the Control Strategist and responds to requests for reconfiguration of plant components and processes. Using inputs from the Maintenance Manager the Diagnostician, the Configuration Manager maintains an operability database (i.e., capabilities, limitations, availability, etc.) for key components and systems. The Configuration Manager determines the operability of the components and subsystems. The operability is assessed for each operating mode. Requests for operability status can come from the Control Strategist, the Diagnostician, the Maintenance Manager or the operator.

7. Maintenance

Maintenance requirements are met by a set of internal functions referred to as the Maintenance Analyzer and Maintenance Planner. These maintenance functions observe plant status, make decisions, and report plant information during operation and refueling. The maintenance function schedules maintenance activities. It receives requests for component maintenance, assigns priority based upon availability and current plant goals and objectives, and schedules preventive and corrective maintenance in conjunction with the Configuration Manager. The maintenance function responds to inputs from the Diagnostician and analyzes the maintenance requirements.

The maintainers supervise maintenance activities and they are assisted by the automated Maintenance Analyzer and Planner. The maintainers make decisions concerning the maintenance required (adjust, recalibrate, replace-on-line, remove-and-repair, repair-in-place, modify, etc.). They observe and make decisions as necessary for prioritizing maintenance activities and determining personnel assignments.

8. Control Validator

The Control Validator provides a check-before-execute capability to validate all control commands input by the operator or associated with the selected control strategy. The Control Validator compares an expected control state with the current and target control states. This validation process is one that provides an accurate indication of whether or not the operational objectives can be met (within established operating rules and requirements) by the selected strategy and configuration. Making this indication available to operators and other elements of the control system is an example of the reporting function of the Control Validator.

If a control command is invalid, the source/originator of the command is alerted to cancel the command or provide an alternate command for subsequent validation. This iterative validation process continues until the command is validated by the Control Validator or until the operator is delegated validation responsibility. Upon validation, the Control Validator routes the request to the Command Generator.

9. Command Generator

The Command Generator formulates the specific detailed set of control instructions to controlled subsystems and components based upon the validated input received from the Control Validator. The Command Generator retrieves a set of predefined, subordinate instructions and corresponding system component or device destination. Each set of command instructions selected is compared against the originating request prior to issue. If a discrepancy occurs, a discrepancy message is routed to the Control Validator, indicating that subordinate instructions cannot be validated for that command.

Command instructions include identification and order-of-execution information. The Command Generator is used to effect plant control by issuing digital control commands (instructions) to the PCS subsystems or non-PCS (interfacing) systems (such as a trip request to the RPS).

10. Decision Support

The plant operator has a role in making plant level decisions and information must be reported to the operator for this purpose. The Decision Support function provides the capability to assist that decision making responsibility allocated to the operator. The Decision Support function serves as an advisor to the operator. The Decision Support continually receives real time information from the Performance Analyzer, the Diagnostician, the Maintenance Manager, the Configuration Manager and the Control Validator.

The Decision Support function provides acquisition and abstraction of data and it processes the data into information that is intelligible and coherent for the operator. The Decision Support emphasizes circumstances which should be called to the operator's attention.

Included in this category is the disturbance analysis and safety parameter display function which involves the processing of plant data and the presentation of it to the operator in a way that assists him in the performance of his monitoring and control duties during both normal and abnormal modes of plant operation. The proposed data is that necessary to:

1. properly interpret and validate key process measurements;

2. monitor plant system configurations and conditions that provide timely indication of the approach to technical specification limits;

3. determine the precise mode of plant operation and operating trends;

4. aid operators in determining if safety requirements are being challenged and what the priority of system functions is during a plant disturbance.

Implementation of the control engine requires operating system software which has real-time, multi-task, interrupt driven features. These features are especially important for the block level controllers which have a direct interface with the highly interactive operator consoles.

The block level controllers (or computer systems) also require a software data base management system to provide control, storage, and access of all data and programs necessary to support the on-line and off-line processing requirements of the plant technical and support staffs.

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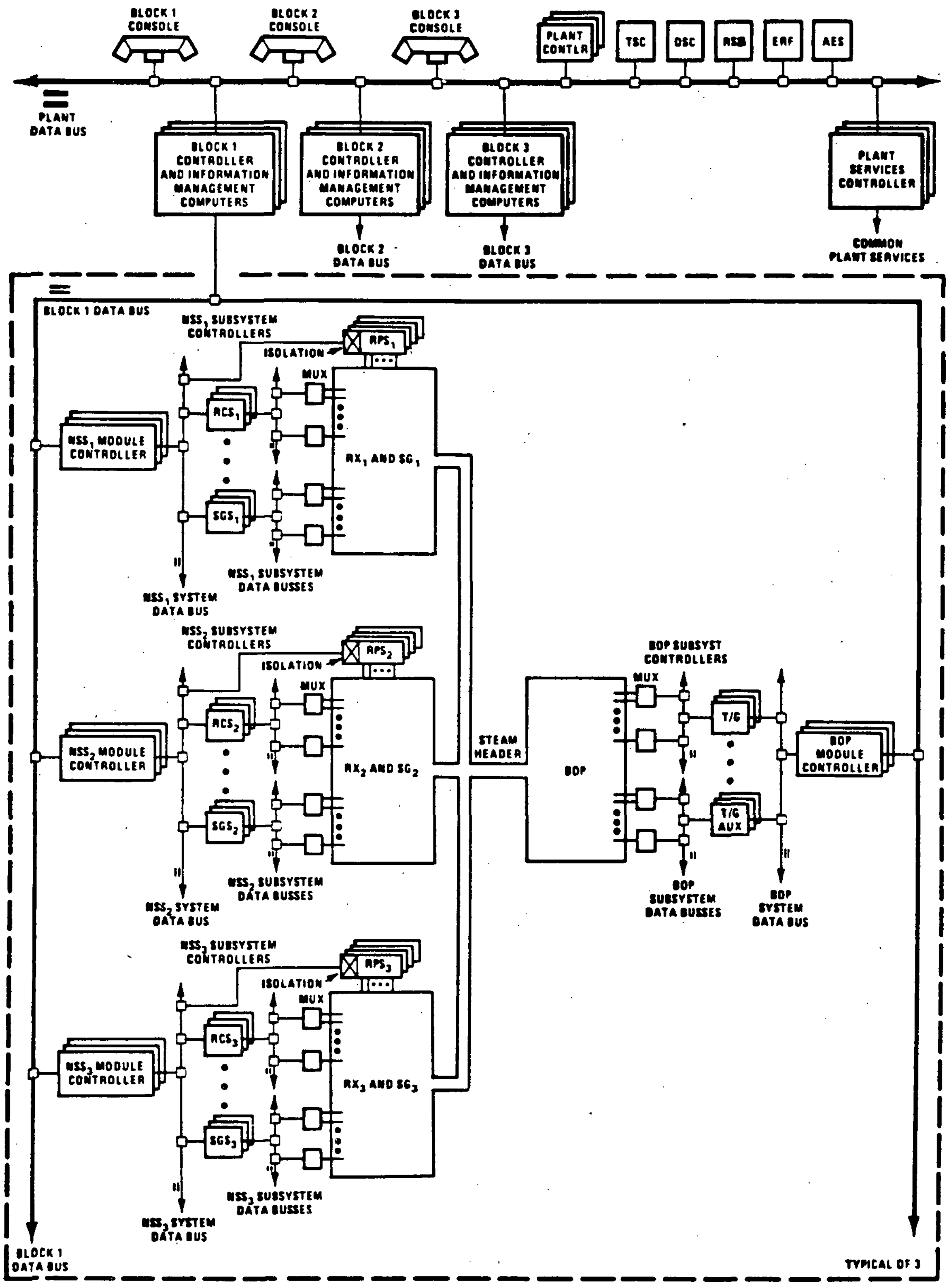

Figure 7.7–1 PRISM PLANT CONTROL AND PROTECTION SYSTEM CONCEPT

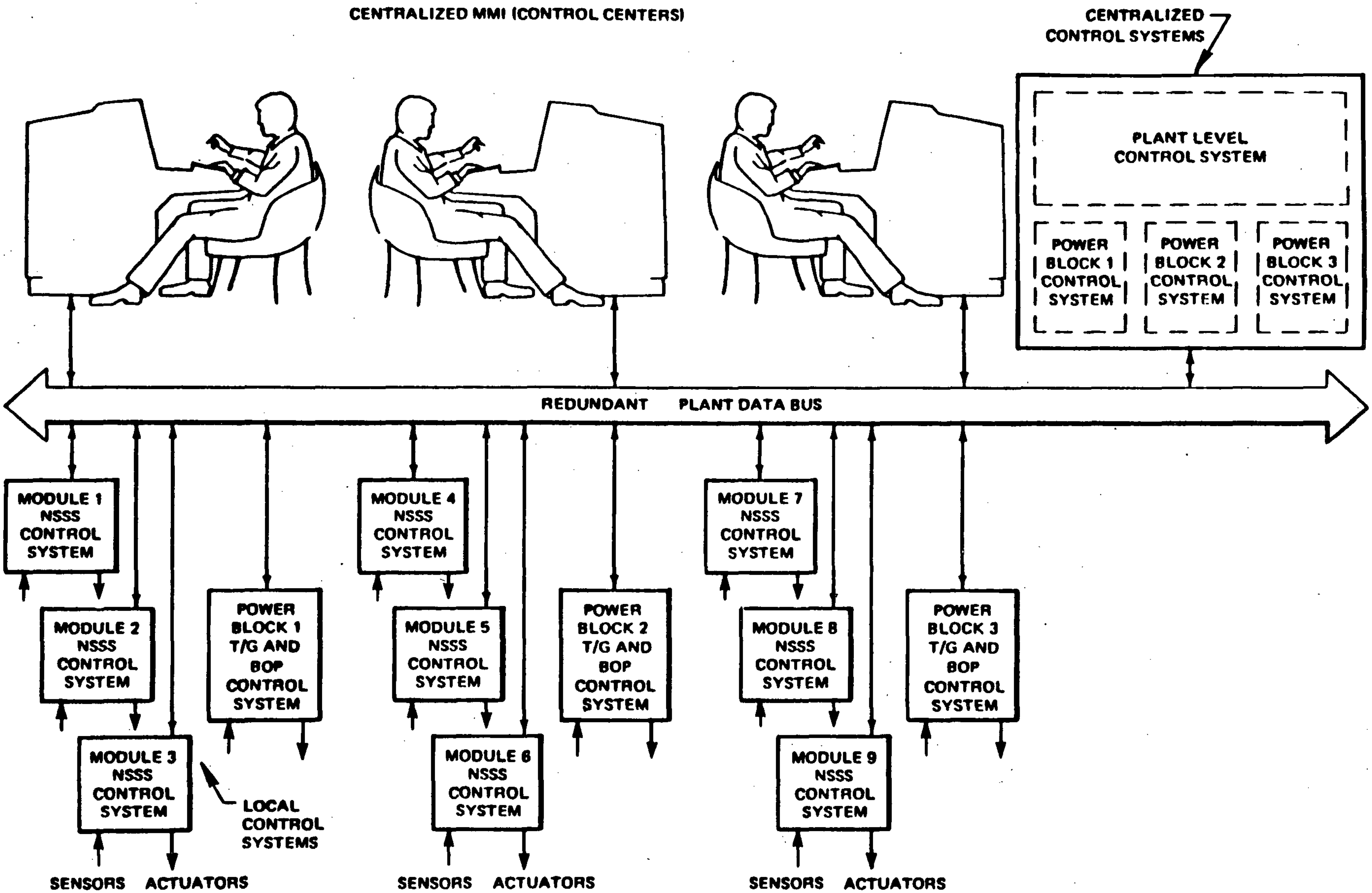

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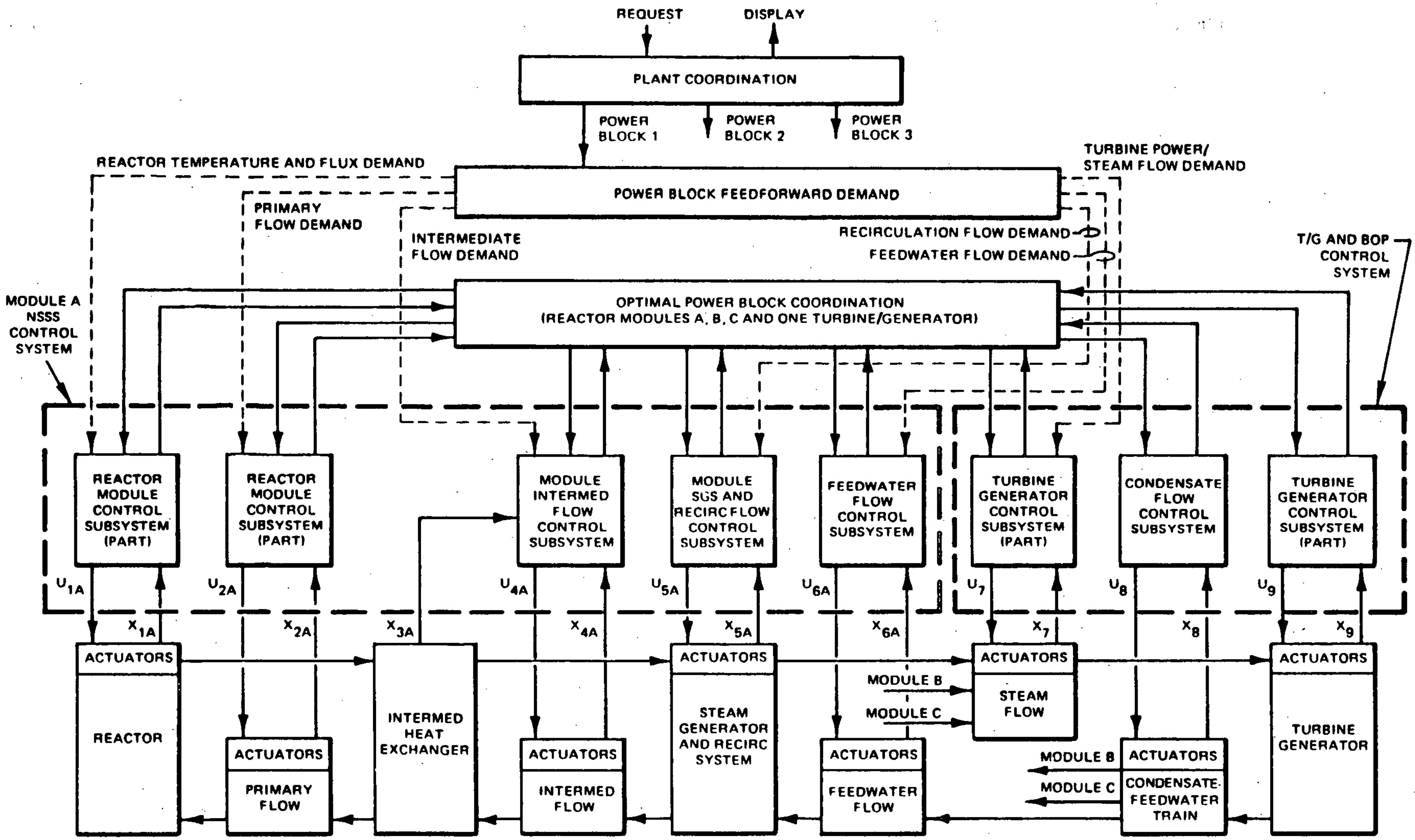
Figure 7.7–2 PRISM DISTRIBUTED PLANT CONTROL



NOTE: SUBSYSTEM CONTROLLERS CONTAIN SIMPLIFIED SUBSYSTEM (INCLUDING ACTUATOR) MODELS

U'S ARE CONTROL SIGNALS; X'S ARE SUBSYSTEM STATE SIGNALS

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Figure 7.7-3 PRISM DISTRIBUTED OPTIMAL CONTROL STRUCTURE

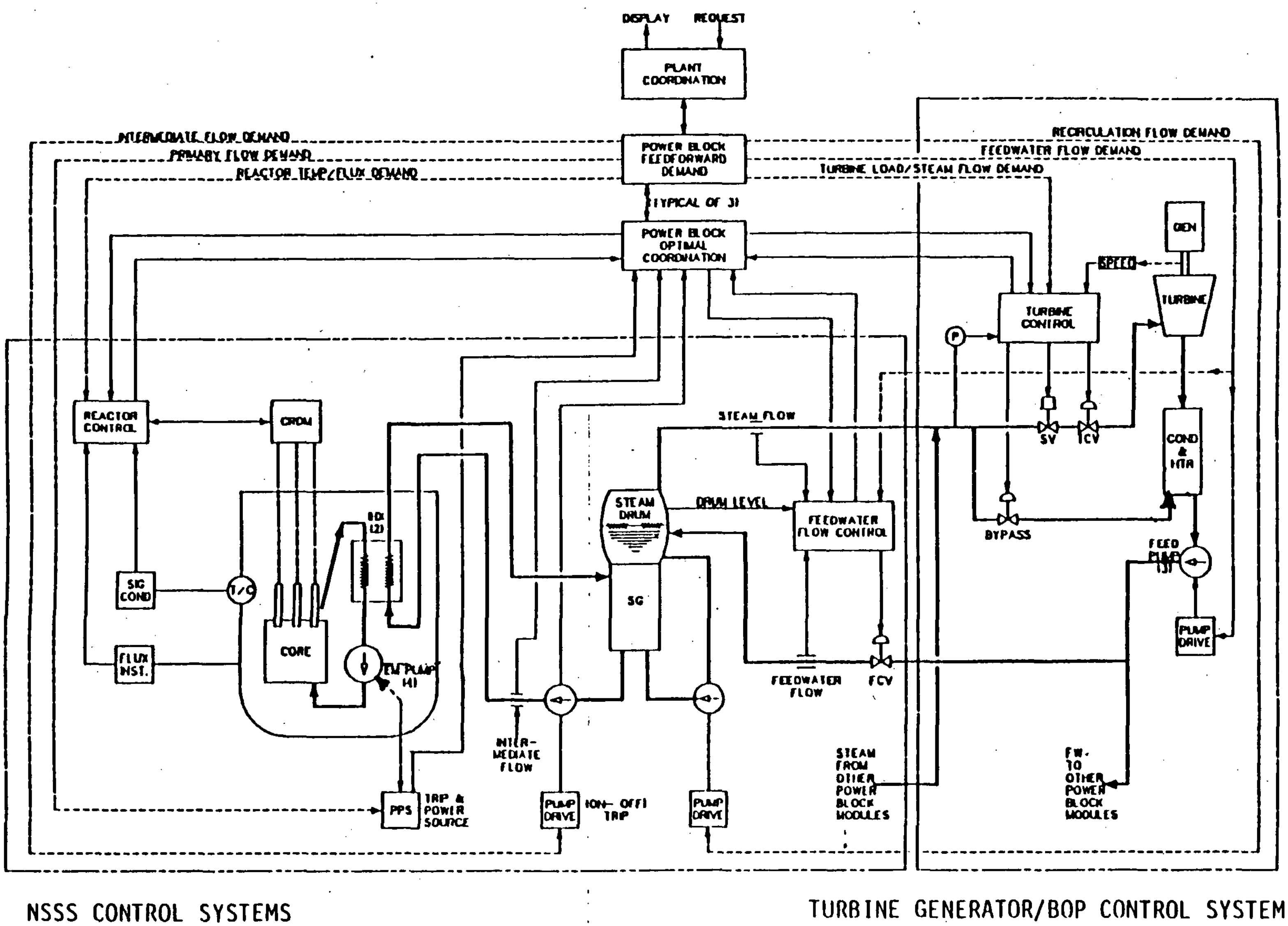

Figure 7.7–4 MAJOR PLANT CONTROL SYSTEM PHYSICAL INTERFACES

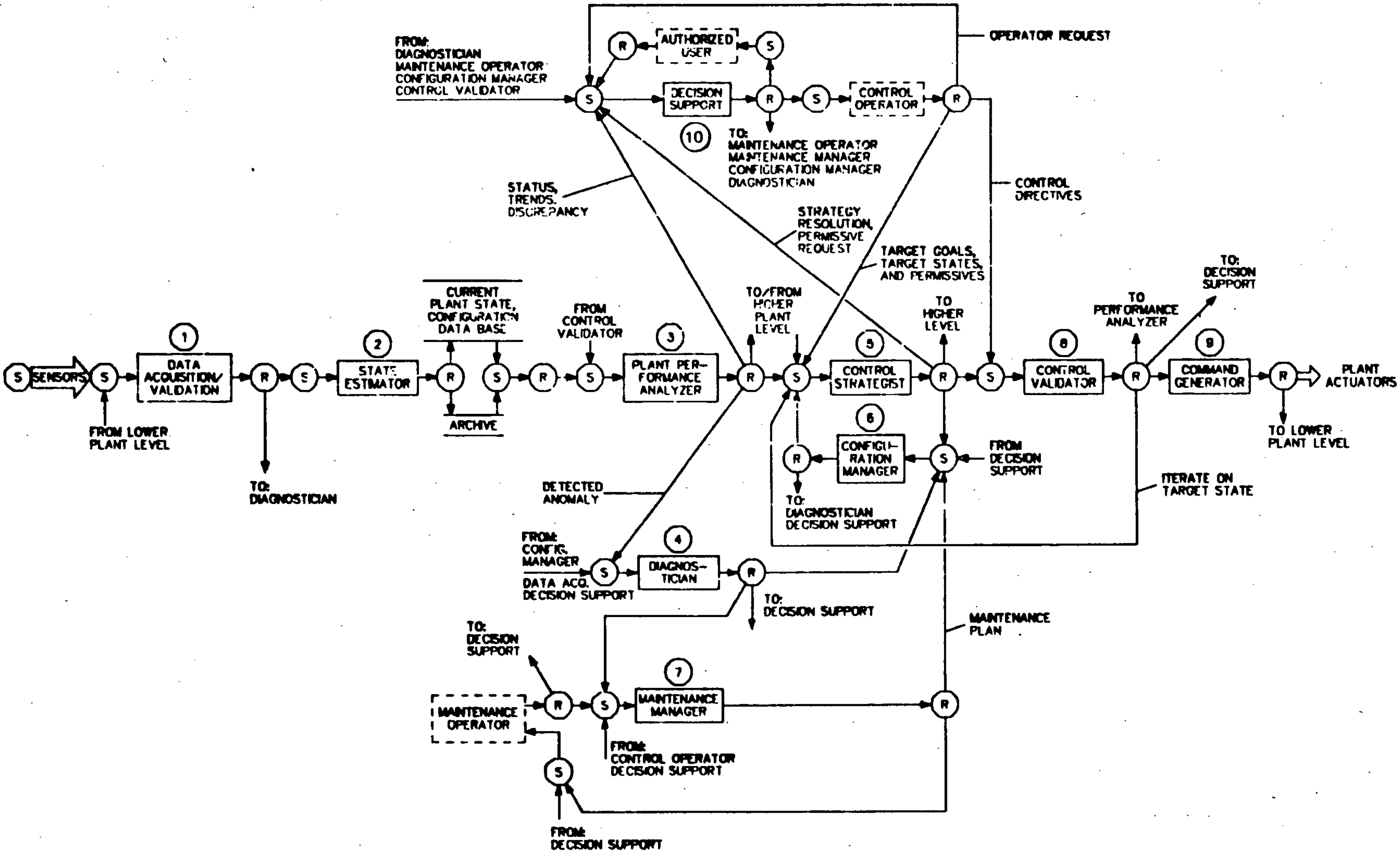

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Figure 7.7–5 CONCEPTUAL GENERIC CONTROL ENGINE MODEL

7.8 Data Handling and Transmission System (DHTS)

The DHTS consists of multiple sets of optical communication cables, multiplexers, data highway buses, and interfaces to the various controllers. The principal components of the DHTS are shown schematically in Figure 7.7-1. The DHTS performs two principal functions, data handling and signal transmission, respectively.

The key functions of DHTS are the following:

- o Accepts directions from the controller on where to send and receive communications and the purpose of the communications (e.g., control command, request, monitor, print, display, file, annunciate, etc.). This activity occurs at the local, block, and plant controller levels.
- o Multiplexes non-safety classified data for plant control and monitoring.
- o Provides the sole means of physical interconnection of plant, block, system and subsystem controllers when electrical signals are exchanged between them using non-hard wired circuits.
- o Receives data from the plant control and reactor protection systems for monitoring the multi-module plant.

The major portions of the DHTS signal transmission network consists of a hierarchical set of fiber optic data busses (each with multiple redundancy) designed to meet the signal transmission requirements between devices that comprise the centralized plant controls, local control systems, instrumentation, and actuators. The DHTS also receives and transmits optically buffered RPS signals for display monitoring purposes. Physical location of device interfaces to the data busses is not limited by the network design. The network can interface with a wide variety of electronic equipment and devices (e.g., digital or analog controllers, computers or conventional, non-intelligent devices such as relays, converters and solid state control and logic systems).

The control and management of the system data transmission input/output is achieved by servicing priorities and interrupts associated with the multiplexers linked to the local controller data busses. Communications are validated for transmission errors, stored in memory and compared to static or dynamic fault limits and corrected as appropriate. As faults are detected, they are logged and flagged for distribution to the appropriate display subsystems. Continuous diagnostic testing is performed on the plant and local level data busses. Should a failure be detected on a data bus, the system automatically routes signals to another data bus, validates its operation and notifies the appropriate technical personnel of the failure.

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The data bus interface provides a standard electrical and mechanical interface to the data highway. The data bus interface consists of a multiplexer and transreceiver. Multiplexers receive inputs in the form of data communication (bit packets) from the data bus via the transreceiver or preprocessed discrete and analog instrumentation signals from plant systems. The multiplexers provide outputs in the form of digital and analog instrumentation signals to plant systems and data communication to the data highway via the transreceiver. The transreceiver electronically implements the access (transmission) and extraction (reception) of data on and from the data highway.

The DHTS and the centralized plant controls are physically interfaced by the transreceiver. The transreceiver consists of an electrical interface or tap to the plant data highways, I/O controllers containing serial/parallel communications formatters and direct or shared memory access hardware and dedicated data busses. These devices provide the signal transition and linkage between DHTS and the control complex display, control and processing equipment. Each I/O controller performs high speed data transfer with the control complex processors.

The allocation of demultiplexing functions between DHTS and the control complex is determined on the basis of the type, functions and signal transmission requirements. It is planned that all signals originating within the control center for transmission to remote locations will be multiplexed by the DHTS over digital data busses. Any exceptions to this design standard will be determined on a case by case basis by evaluating the requirements for unique interfaces between the centralized plant controls and the other systems.

Multiplexing will enable control and monitoring signals to be transmitted over a small number of cables. The signals share the transmission medium once they are processed for digital communication. Multiplexing results in a significantly cost-effective reduction in plant instrumentation, control cable quantities, cable spreading areas, and critical path plant construction schedule. Multiplexers are strategically located throughout the plant to optimize cable savings. These multiplexers, along with the communication devices and plant-wide fiber optic data busses, constitute the signal transmission portion of DHTS.

7.9 Plant Control Complex

The plant control complex includes the man-machine interface (MMI) displays and command interfaces for six functional areas: 1) central control room (CCR), 2) technical support center (TSC), 3) operations support center (OSC), 4) auxiliary shutdown console in the Reactor Service Building (RSB), 5) information management center (IMC), and 6) administrative and engineering support (AES). Each of these areas provides unique support to the operators who direct the control management, engineering, and maintenance of the plant. These functions, usually separate, are unified in PRISM for optimal interface and operator control of the multi- module plant. Figure 7.7-1 shows how the functional areas interface with the DHTS.

Central Control Room (CCR)

The following criteria have been established for design guidance and meeting the operability goal in that the PRISM plant (3 reactor modules and 1 turbine generator per power block, providing a total plant consisting of 3 power blocks) can be operated by 3 senior operators supervised intermittently by the shift engineer.

1. Balance the operator's workload between his active involvement in supervising control of plant processes and his full awareness of the status of plant behavior. Allocate to computerization the following functions:

    a. Plant-wide energy management.
    b. Plant operating configuration management.
    c. On-line performance analysis.
    d. Control strategy validation.
    e. Sensor/command/actuator validation.
    f. Plant-wide diagnostics and decision aids.
    g. Maintenance planning and status.

2. Provide support to the operator by supplying real-time information on the state and condition of the plant so that he adequately supervises the automatic operations of the plant.

3. Automate system and subsystem operation to the maximum extent, consistent with high plant availability, efficient and safe operation.

4. Following failures permit the operator to continue serving supervisory functions rather than being required to provide manual control of the affected system until the system has been repaired and restored to service. A list of equipment failures and the resulting PCS/operator action is shown in Table 7.9-1.

5. Allocate control center workspace only to essential human-machine interfaces reflecting automated, distributed control.

6. Provide diagnostic displays, backed up with decision aids. Intelligent digital signal processing and control systems provided for PRISM will be capable of predetermining (by trend and/or pattern analysis) that an off-normal event is imminent. The traditional alarm system (with a large window box, flashing lights, and klaxon style horn) which is found in the control rooms of today's operating power plants, will not exist in the PRISM advanced control complex.

7. The interactive displays, by which the operator monitors plant and system performance, contain color, shape, and other coding mechanisms to identify plant and system status for the operator. Should the situation warrant notification to the operator, such notification will take place through the interactive displays.

The control center for the PRISM Plant is a unified design incorporating human factors engineering. The successful implementation of complex man-machine interfaces in control systems is realized through today's electronic technology. Advanced engineered control room layouts, with their greater capability and smaller size through use of digital components, results in a more efficient control center design. The PRISM control center is located on the ground level floor of the control building and is shown in Figure 7.9-1.

Preliminary analysis has indicated that through the use of extensive automation, the primary man-machine interface within the control center can be served by three consoles arranged as shown in Figure 7.9-2. Video display units (VDU) replace hard wired indicators and recorders. Selected plant data is also continuously stored to provide permanent records of plant operations. The VDU's have flexible data formats, so that upon loss of a VDU, the same information may be brought up on other VDU display units. Touch sensors on the VDU's and touch data entry keyboards associated with the VDU's replace hard wired switches and controllers.

Digital controller electronic racks and processing equipment are separately located remote from the main central control room operator consoles. Transmission of signals between the various levels of controllers and the CCR is performed over plant data highways as shown in Figure 7.7-1. For improved plant security, entrance to the main control room is electronic key card controlled to allow only the CC staff and shift supervisor entry. Work permit issuance will not require face to face operational and maintenance supervisory personnel contact. Kitchen, lounge and bathroom facilities are provided adjacent to the control center also for improved operator comfort and overall control center security.

Technical Support Center (TSC)

The TSC, shown in Figure 7.9-1, provides technical support to plant operating personnel during emergency conditions, relieves operators of peripheral duties and communications, and limits personnel traffic in the main control room.

All the displayed and printed TSC information is available for the shift technical advisor during normal operating conditions or for the technical support personnel (up to 25) during emergency conditions. Requirements for total floor space, space for private NRC conferences, plant records, and telephone communications are also met. The close proximity permits needed face to face communication and avoids major security barriers between TSC and CCR personnel.

Auxiliary Shutdown Console in the Reactor Service Building (RSB)

The auxiliary shutdown console is a backup control console to manually initiate a remote shutdown of any of the nine reactor modules and monitor their safe shutdown in the event the control room becomes uninhabitable. The console is located in the RSB which is within the nuclear island security fence. The RSB is constructed to protect the operator against seismic and fire events, and special air conditioning is provided to protect the operator from smoke and noxious airbone contamination. Control functions from this console are limited to reactor shutdown and monitoring to assure that the plant is in a safe shutdown condition. For consistency in man-machine interfaces, the display and control surfaces associated with the auxiliary console are identical to those used in the control room.

Operations Support Center (OSC)

The OSC is located in the control building and provides for the integration of group technical tasks. These include required maintenance, surveillance, testing and calibration tasks that must go on continuously during plant operation. There are also a considerable number of local control system functions that are necessary in the course of starting up or shutting down the plant. Rather than allowing the control center to be crowded by personnel involved with these functions (primarily technicians), these personnel are instead assembled and supervised at a central location, the OSC. The OSC always functions under the authority of control center technical management.

This facility, along with the maintenance offices VDU displays with access to the plant data base maintenance information, combine to provide the OSC support cited in NUREG-0696.

Information Management Center (IMC)

An information management center is provided in the control building (Figure 7.9-1) to house computing equipment to provide a plant wide data base, large scale plant performance calculations, engineering analysis and report files. This center will include a secure area and will permit maintenance and computer operational activities of the plant and block level controllers to be performed without interfering with other opera tions.

Administrative and Engineering Support (AES)

Administrative, engineering, and maintenance supervisors will be provided with work stations as necessary to perform their defined duties. The work stations will allow them to access the DHTS to read real plant data for purposes such as:

- Utility billing
- Fuel management and fuel inventory
- Daily plant operations log
- Abnormal occurrence reporting
- Plant performance, heat balance, efficiency calculations
- Component operating status, performance analysis, maintenance status, etc.
- Sodium, water, gas turbine fuel inventory, etc.

Operator Training Simulator

The current design philosophy has dictated that the operator training simulator is located on-site but outside the main control building. The reason being to reduce the flow of traffic and possibly risk of unauthorized personnel gaining access to the main control room. The simulator will be able to monitor the plant data bus to permit on-line observation of actual plant operation. Consoles with VDU groupings essentially the same as those in the control center will be used. Computer simulation of plant dynamics will be used in the initial phase of the project for both safety analysis and operating control systems design. The control center will then be designed according to test results obtained from dynamic mockups of the facility. These mockups will be electronically active and driven by a real-time PRISM simulator. An operator training simulator will be developed in the last phase of work, combining the final control center design with a real-time PRISM plant. The operator training simulator will be simulator will be utilized to train the plants main control center operating crews. The simulator will provide plant specific simulator training. Use of an on-site simulator permits flexible scheduling of operator training, minimizing any interference with the plant operating schedule.

Table 7.9-1 - EQUIPMENT FAILURES AND PCS/OPERATOR ACTION

Equipment Failure	PCS and Operator Action
1. Plant major component failure - (failure within scope of PCS automation)	o PCS reconfigures components whenever possible to maintain power generation of the power block or reduces load (rapid runback to match plant available capacity. o PCS apportions load among the remaining fully operational modules to maximize power generation capability of remaining equipment. o Operator alerted of failure. o Operator alerts roving operator and maintenance superintendents.
2. Plant major component failure - (failure outside scope of PCS)	o PCS senses inability to meet demanded load on affected module and reduces power to a safe level. o PCS apportions load among the remaining fully operational modules to maximize power generation capability of remaining equipment. o Operator alerted to failure. o Operator takes manual control action to reconfigure failed components; and later increase power level if safe to do so. o Operator isolates failed components from the operating system. o Operator alerts roving operator and maintenance supervisors.
3. PCS controller failure - (single channel)	o PCS is fault tolerant and redundant. Controller self-isolates failed component by self-reconfiguration. o PCS/controller remains operational. o Roving operator and maintenance supervisor alerted. o On-line repair initiated. o Controller fully operational (3 channels within 4 hours).

Table 7.9-1 - EQUIPMENT FAILURES AND PCS/OPERATOR ACTION
(Continued)

Equipment Failure	PCS and Operator Action
4. PCS controller total failure	o Action is dependent on the level of control in the hierarchy.
	o If upper level controller (ULC) fails or all communications with ULC lost, then "hold" command initiated within the lower level controller (LLC). Roving operator and maintenance superintendent alerted to investigate.
	o If LLC fails, or communications lost, roving operator and maintenance superintendent alerted.
	o Local coordinated action may be taken with telephone CCR communication to shutdown affected module (or block if BOP controllers failed), by coordinated action between the roving operator at the local station and CCR operator.
	o PCS remains operational for remaining module controllers.
	o If power block is running at reduced load, shutdown of a reactor or steam generator module due to local subsystem controller failure may be compensated for by PCS automated operation of remaining reactor/steam generator and BOP module controllers by automatic load reallocation.

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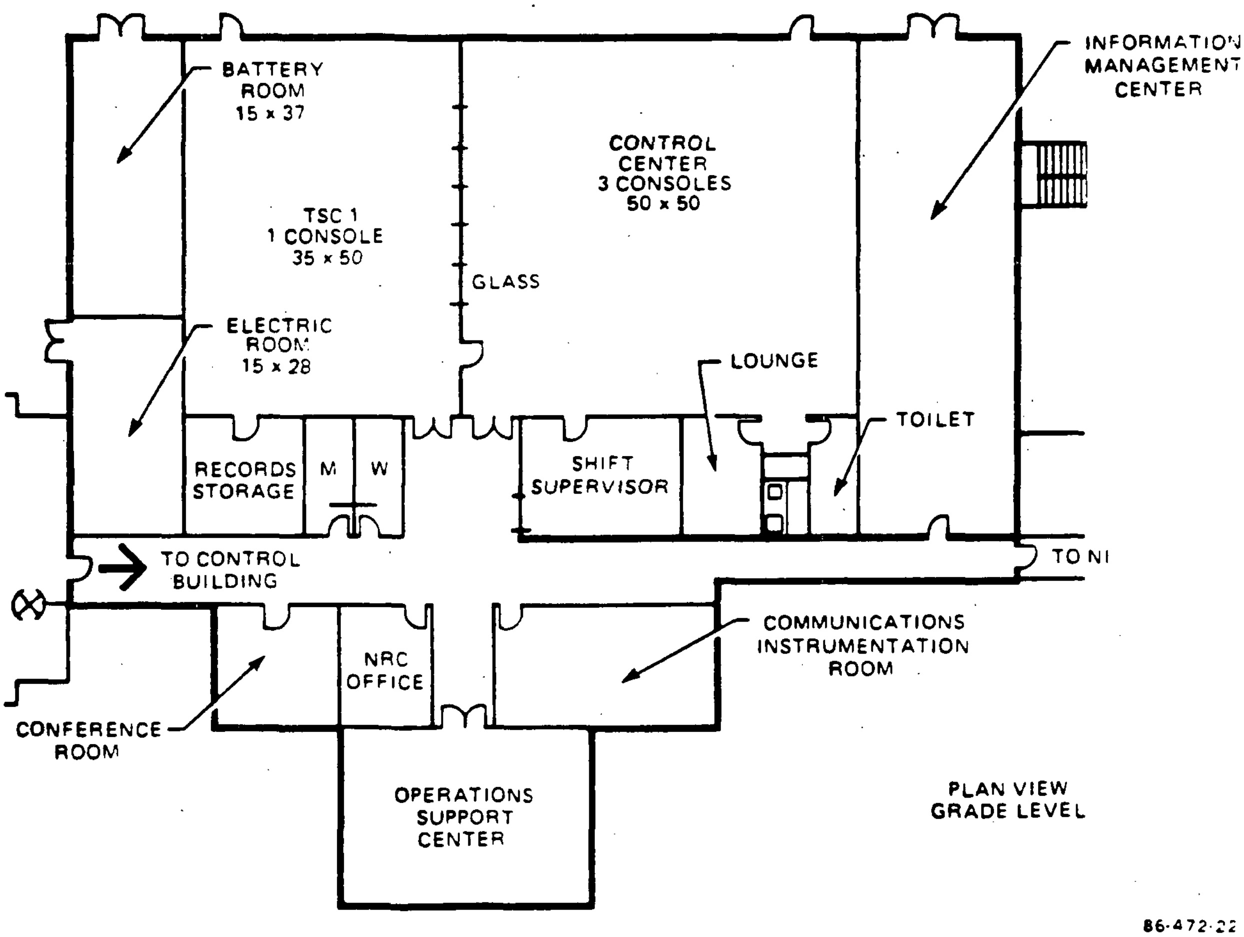

Figure 7.9–1 CONTROL BUILDING AND CONTROL CENTER LAYOUT

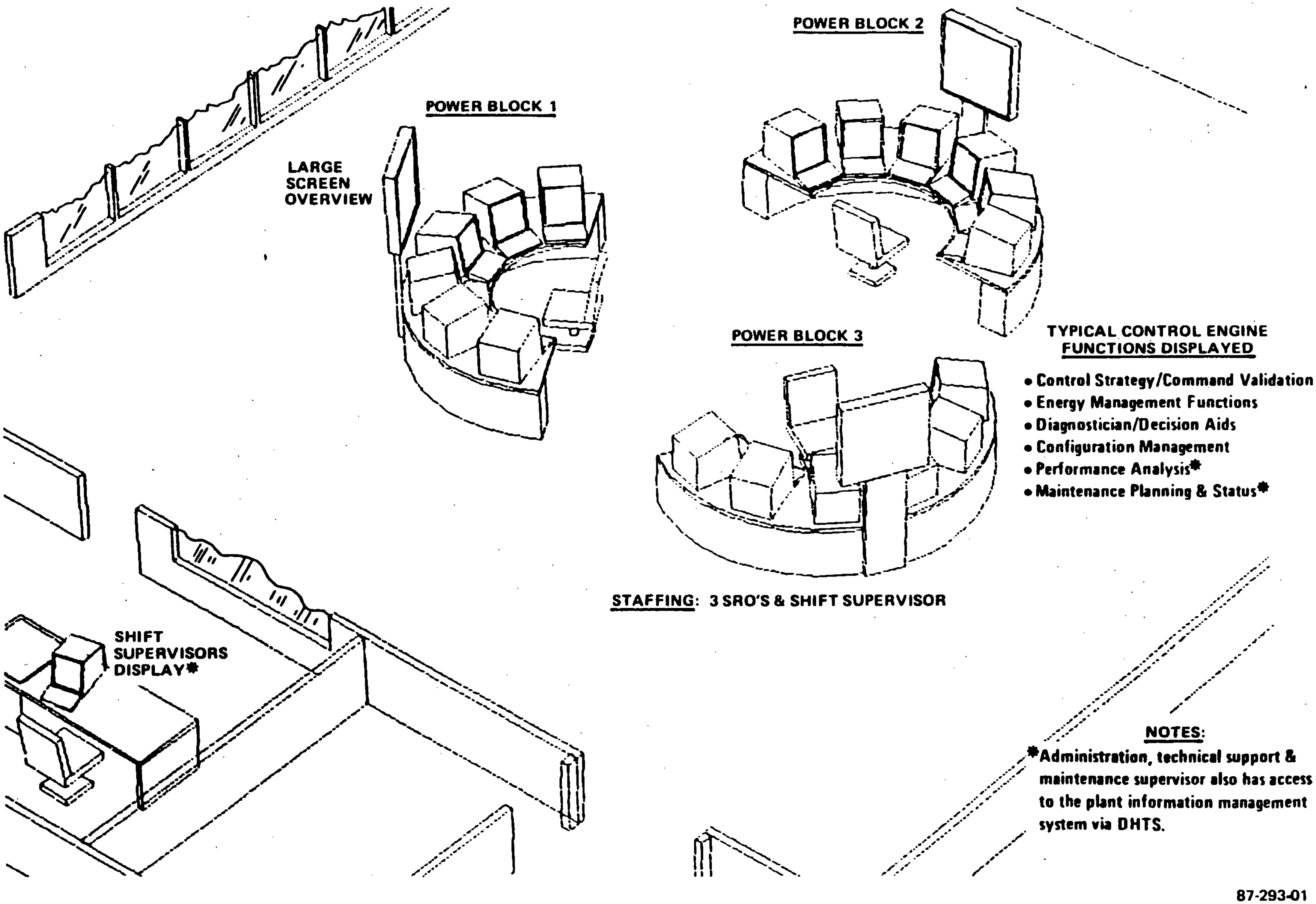

Figure 1.2–17 CONTROL CENTER OPERATOR INTERFACES

CHAPTER 8

CHAPTER 8

ELECTRIC POWER

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Chapter 8 <u>ELECTRIC POWER</u>

8.1 Introduction

The function of the power transmission system is to export all available power generated by the plant and supply adequate auxiliary power to operate the electrical loads required during all modes of operation. The transmission system includes the generator circuit breakers, main step up transformer, unit auxiliary transformers, common station service transformers and associated bus and cables to form a complete power distribution network.

Each power block operates independently, furnishing power to the utility grid through the generator step up transformer and high voltage switchyard breaker. The common station service system provides a secondary source of offsite power to each power block and supplies power to common services of the facility.

The utility distribution system provides preferred offsite power sources for each power block and a physically independent, secondary offsite power supply for the common station service system. The system buses and cables are designed to withstand the maximum short circuit and full load currents. Transformers provide the necessary power distribution requirements, limit short circuit currents to acceptable values and ensure adequate voltage levels during all modes of unit operation. Protective systems, operational displays, automated switching, automated system diagnostics loading analysis, and automated reconfiguration control will be provided by the plant wide control system.

8.2 Offsite Power System

8.2.1 Description

The offsite power system consists of preferred and secondary power sources from the utility distribution network.

Power from the preferred offsite sources is delivered to each power block through high voltage switchyard breakers arranged in a highly reliable ring bus. The ring bus will accommodate multiple ties with the utility transmission network.

Power from a secondary offsite source is delivered to the station service power system through a separate high voltage ring bus.

The preferred and secondary offsite sources are connected to the utility grid by separate, physically independent transmission lines.

8.2.2 Analysis

The preferred and secondary offsite power systems are designed with sufficient separation and independence to assure reliable operation. Protective relays and high voltage breakers provide system fault protection.

8.3 Onsite Power Systems

8.3.1 AC Power System

The onsite ac power system consists of 7.2 kV, 480/277V, and 208/120 V systems for each power block and for the common station service system.

Each power block 7.2 kV system is a parallel radial system with ties to the common station service system to provide high reliability. The onsite power system is shown in the main single line diagrams Figure 8.3-1.

Power is normally furnished to the 7.2 kV systems from the preferred offsite power source through the unit auxiliary transformers. In the event of a transformer failure or loss of the preferred offsite source, power will be available from the secondary offsite source through the common station service system. The station service transformers are sized to continuously carry the common station service loads in addition to the full auxiliary load from one power block. Two standby non-safety related gas turbine generators are provided on the station service 7.2 kV system. In the event that secondary offsite power is also lost, the generators will furnish power to common equipment loads essential to maintaining plant operation and preventing major equipment damage.

Power from the 7.2 kV system is stepped down to 480 V at load center substations for operation of large 480 V loads and distribution to 480 V motor control centers. The load centers and motor control centers are distributed throughout each power block and the common facilities.

The turbine building and common station service 480 V power distribution networks are arranged in secondary selective systems using double ended load center substations with tie breakers. Each substation is designed to handle the complete load center power requirement with the tie breaker closed.

The reactor building 480 V power distribution networks are arranged in radial systems providing separate buses for the four reactor protection channels.

Power is distributed throughout the facility for general plant 208/120 V lighting and power requirements. Power for 208/120 V panels is provided from the 480 V system through small power distribution transformers.

Instrument ac 120 V power is provided in the common facilities and in each power block for instrumentation and control circuits requiring a regulated voltage supply.

Essential 120 V ac power is provided from battery backed uninterruptable power supply (UPS) systems in each turbine building and in the common facilities for control and instrumentation functions requiring uninterruptable power. Separate vital 120 V ac UPS systems are provided in each reactor building to provide uninterruptable power for the reactor protection system. Separate UPS power supplies are provided for each of the four reactor protection system (RPS) and the four plant control system (PCS) channels. There are no non-Class 1E loads connected to Class 1E buses.

8.3.2 DC Power Systems

Power for the 125 V dc system is provided by batteries and battery chargers. Separate systems are furnished for the common facilities and for each turbine and reactor system. The 125 V dc system provides power for emergency oil pumps, dc control and instrumentation circuits and the essential and vital ac UPS systems. The UPS system for the reactor protection system is Class 1E and is shown on Figure 8.3-2.

Power for the 48 V dc systems is furnished from batteries and chargers located in the reactor buildings. The 48 V dc systems provide power for the control rod drive-in motors and latch coils and are Class 1E safety-related. Separate batteries are provided for each of the four channels for each drive-in motor and latch coil system. The 48 V dc systems are shown on Figure 8.3-3 and 8.3-4.

8.3.3 Electromagnetic Pumps Power Supply

The primary sodium electromagnetic (EM) pumps are normally supplied from the ac distribution system. On loss of this system, power is required for a controlled coastdown of pumps for a period of about two to three minutes to prevent core temperatures from exceeding acceptable limits. This power is provided by a synchronous machine which in normal operation just motors on the 1110 V EM pump power source. The diagram for the EM pump power supply is shown on Figure 8.3-5 and power cable separation in Figure 8.3-6. Each EM pump is provided with a separate power supply and coastdown system. The coastdown characteristics required are shown in Section 5.4.3.3.

Power to the EM pump is normally supplied from the 7.2 kV, 3 phase, 60 Hz ac distribution system through a dedicated input transformer and solid state power conditioning unit. This unit is also utilized to supply power to the EM pump during startup and normal shutdown operation.

An auxiliary synchronous machine supplies coastdown power to the EM pump when the normal power supply through the power conditioning unit is lost. This machine is connected in parallel with the EM pump. Normally it is running unloaded, in an overexcited mode of operation, supplying the reactive power requirements of the EM pump and power conditioning unit. Upon loss of the normal power supply, the stored kinetic energy in the synchronous machine is utilized for coastdown of the EM pump.

The power conditioning unit includes a solid state rectifier section, a dc reactor link and a load commutated inverter (LCI). The LCI system is highly reliable and is designed with flexible operating characteristics, which allows variation of both frequency and voltage over a wide range. The power conditioning unit is rated at 1500 kVA with an output of 1110 V, 3 phase, 20_Hz. The LCI unit is provided with the required control, protective, monitoring and alarm circuits. A built-in diagnostic feature facilitates trouble shooting and reduces repair time significantly.

The synchronous machine is rated at 2000 kVA, 1110 V, 3 phase, 20 Hz. It is provided with a fast acting excitation system, which is capable of controlling excitation and voltage of the machine over a wide range. The inertia of the machine rotating parts is capable of providing the required coastdown energy for the EM pump.

The electrical system includes all required power and control devices. In order to ensure a very reliable coastdown operation, the synchronous machine is designed and qualified as Class 1E. A circuit breaker is provided for isolation of the Class 1E portion of the system from the non-Class 1E portion.

8.3.4 Fire Protection for Cable Systems

Cables for the power distribution system are furnished with highly flame retardant insulation and jacketing materials. Fire stops are provided in cable trays and at all room penetrations. Ionization and heat detectors which are part of the plant fire detection and alarm system provide early warning of a fire. Appropriate fire suppression will be provided as described in Section 9.7.1.

8.3.5 Analysis

The onsite electrical power systems, including the ac and dc power systems are designed with sufficient independence, redundancy and protection so that postulated single failures affect only a single load group. The remaining redundant loads provide for safe plant control and shutdown. Plant control and building electrical power is shown in Figure 8.3-7.

The power distribution system utilizes isolated phase buses, non-segregated phase buses and insulated cables in conduits and cable tray to provide a highly reliable power system. Cables associated with four channel RPS, PCS and control rod drive systems are physically separated to maintain independence of redundant circuits. The power system is designed to limit or withstand the maximum calculated fault currents and to regulate system voltage within acceptable limits. Electrical protective devices and alarms are provided to ensure safe and reliable operation.

The power systems are designed in accordance with applicable regulatory guides, IEEE standards, state and local codes.

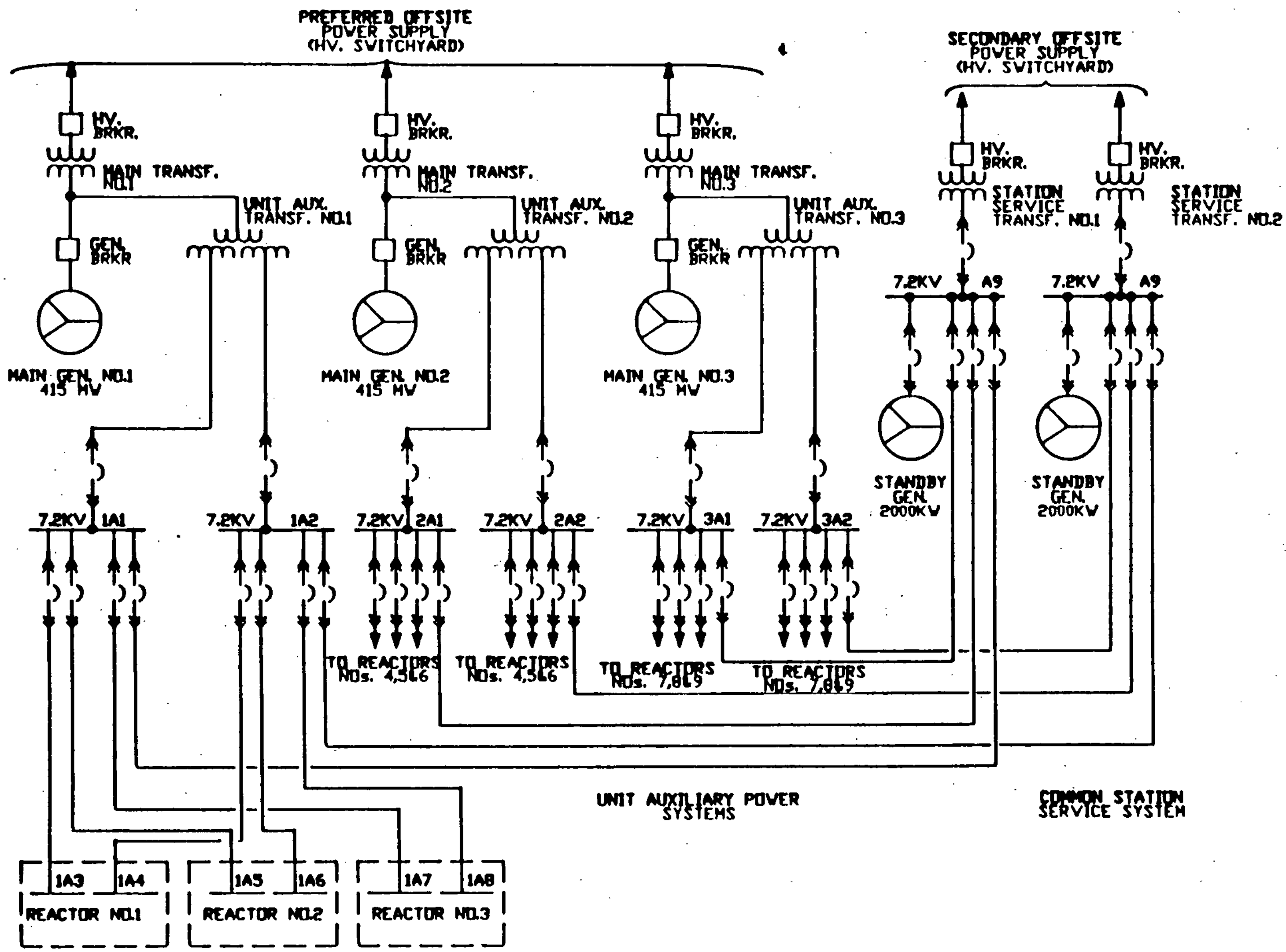


Figure 8.3-1. MAIN SINGLE LINE DIAGRAM
SH. 1

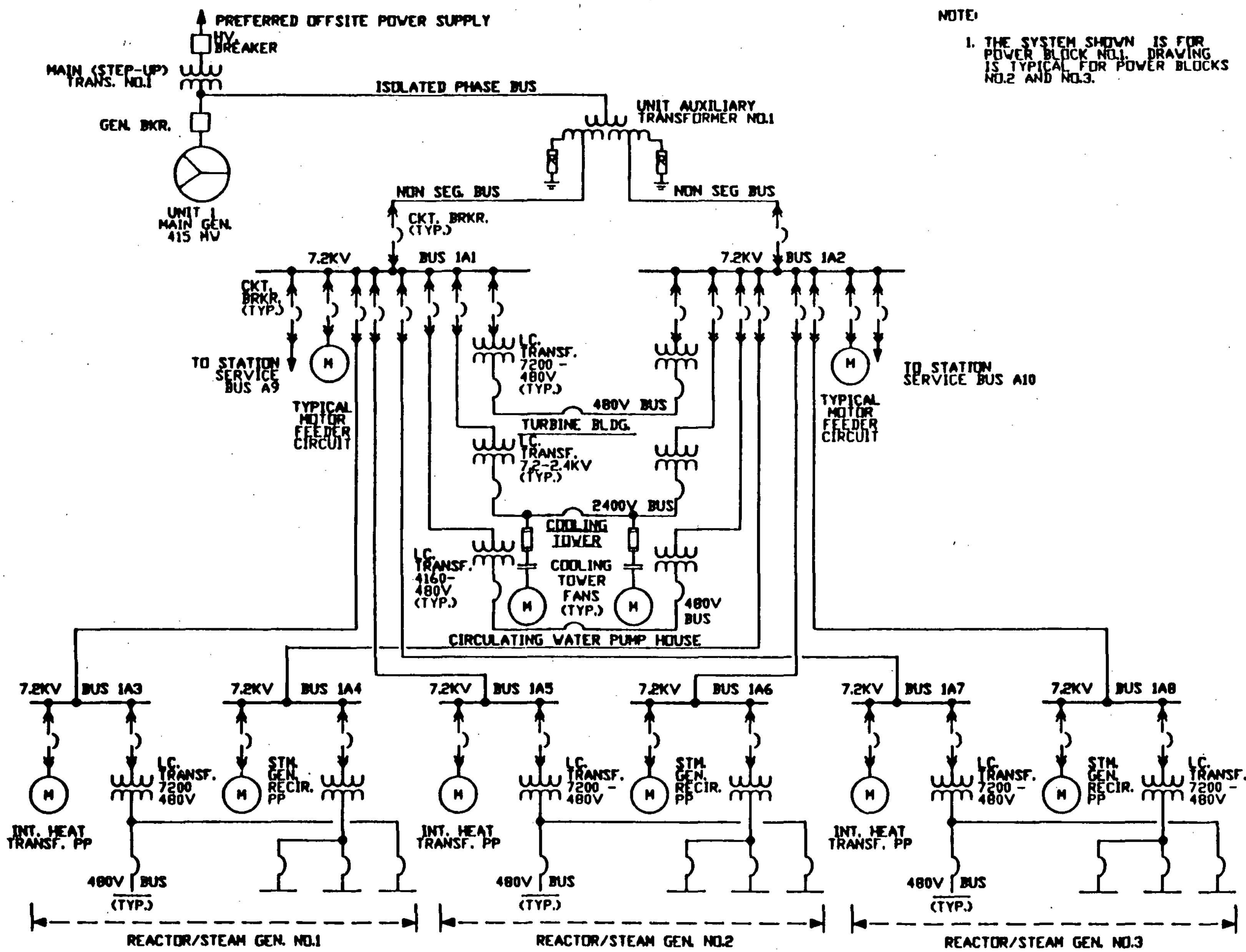
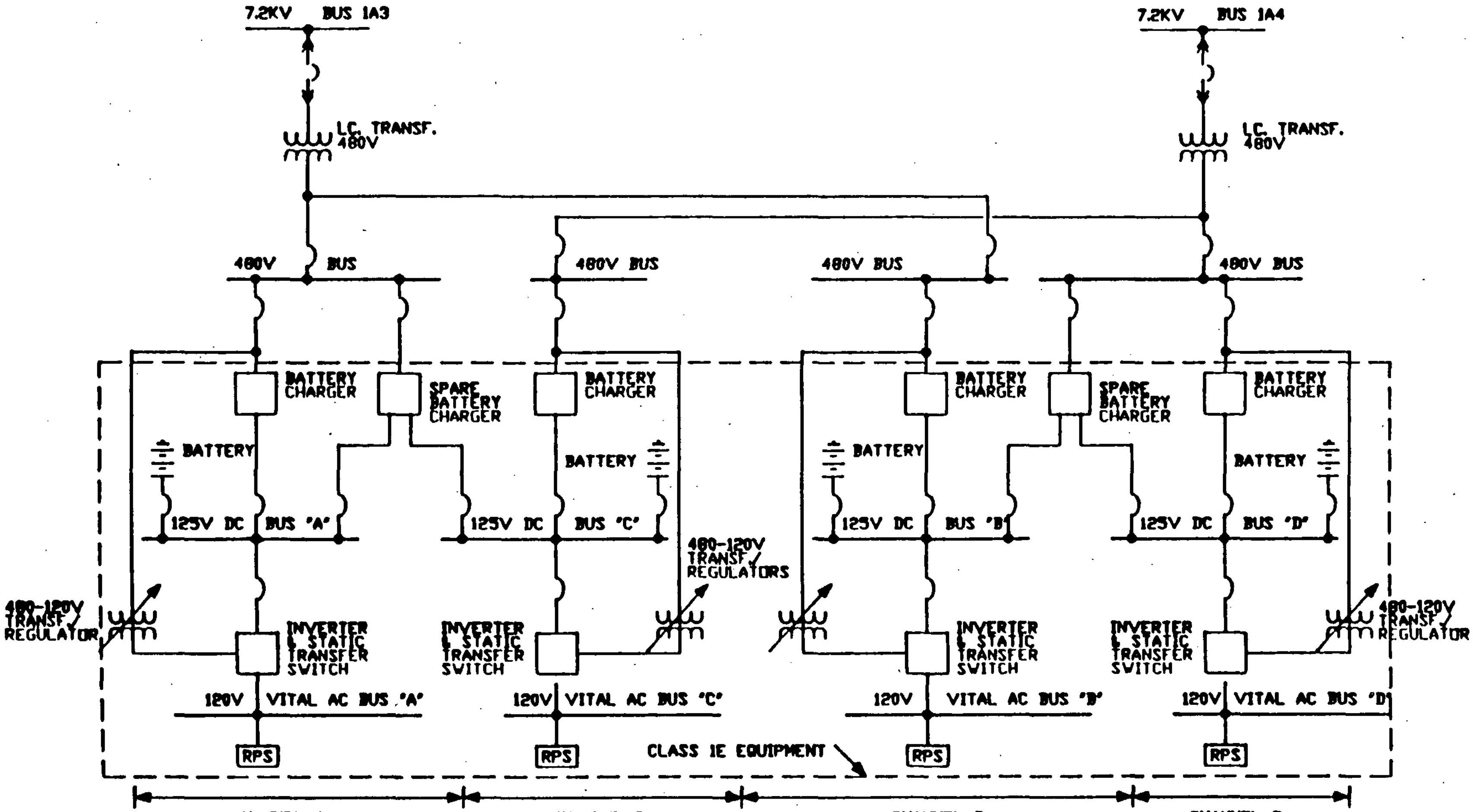


Figure 8.3-1. MAIN SINGLE LINE DIAGRAM
SH. 2



NOTE:
1. THE SYSTEM SHOWN IS FOR REACTOR NO.1. DRAWING IS TYPICAL FOR REACTORS NOS. 2 THRU 9.

Figure 8.3-2 CLASS 1E DC AND VITAL AC SINGLE LINE DIAGRAM

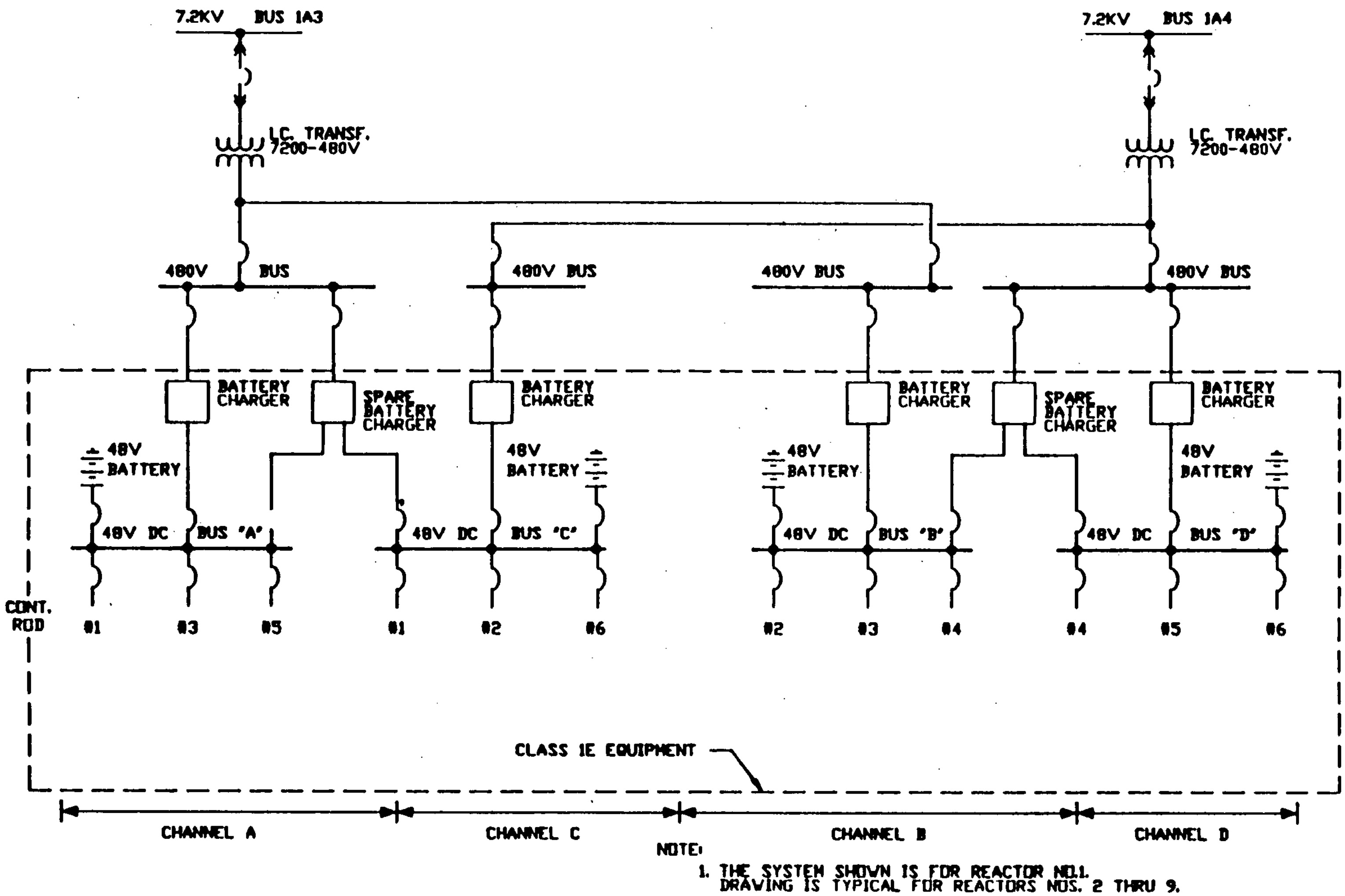


Figure 8.3-3. CONTROL ROD LATCH COIL POWER SUPPLY

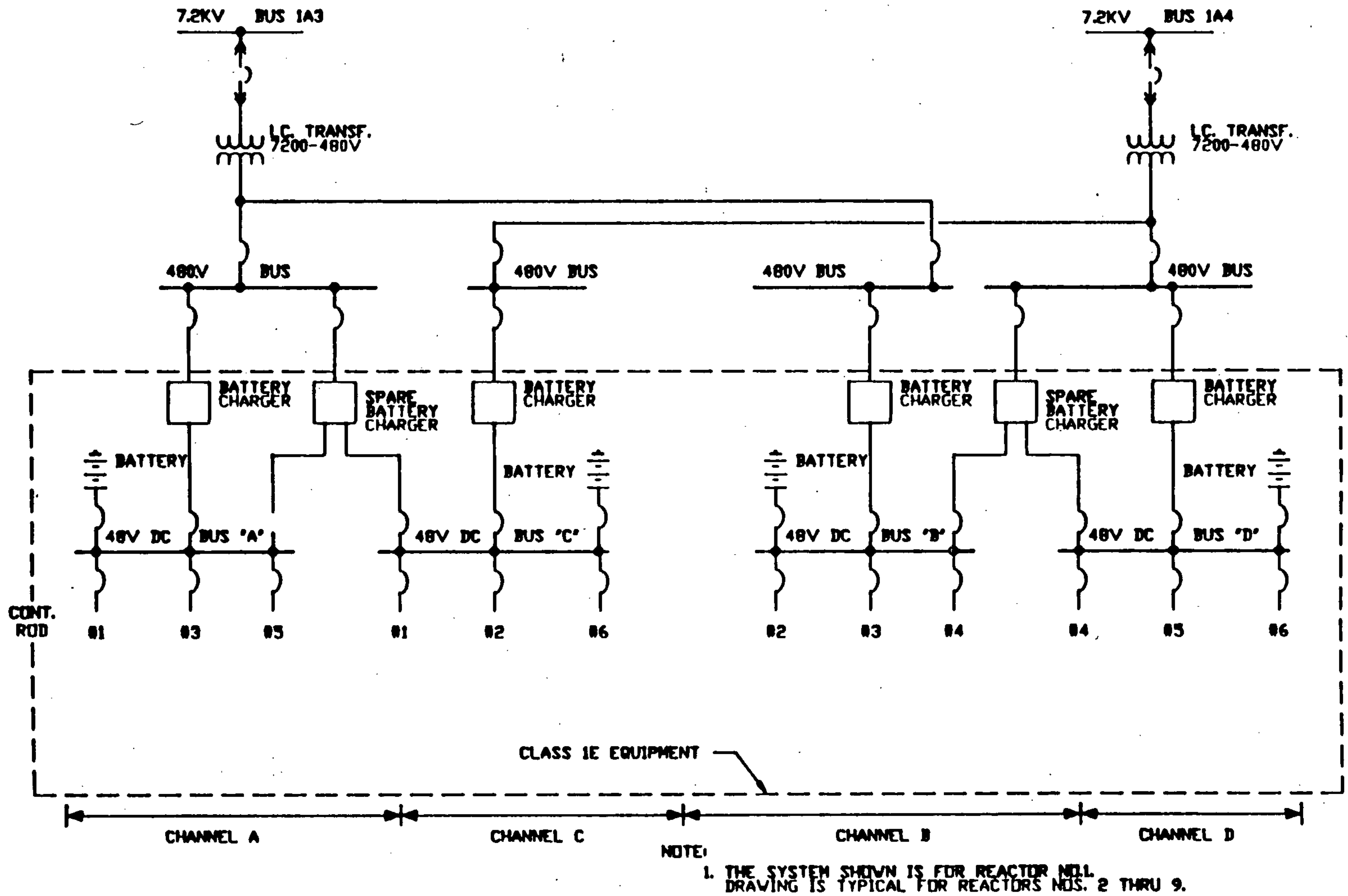


Figure 8.3-4. CONTROL ROD DRIVE IN MOTOR POWER SUPPLY

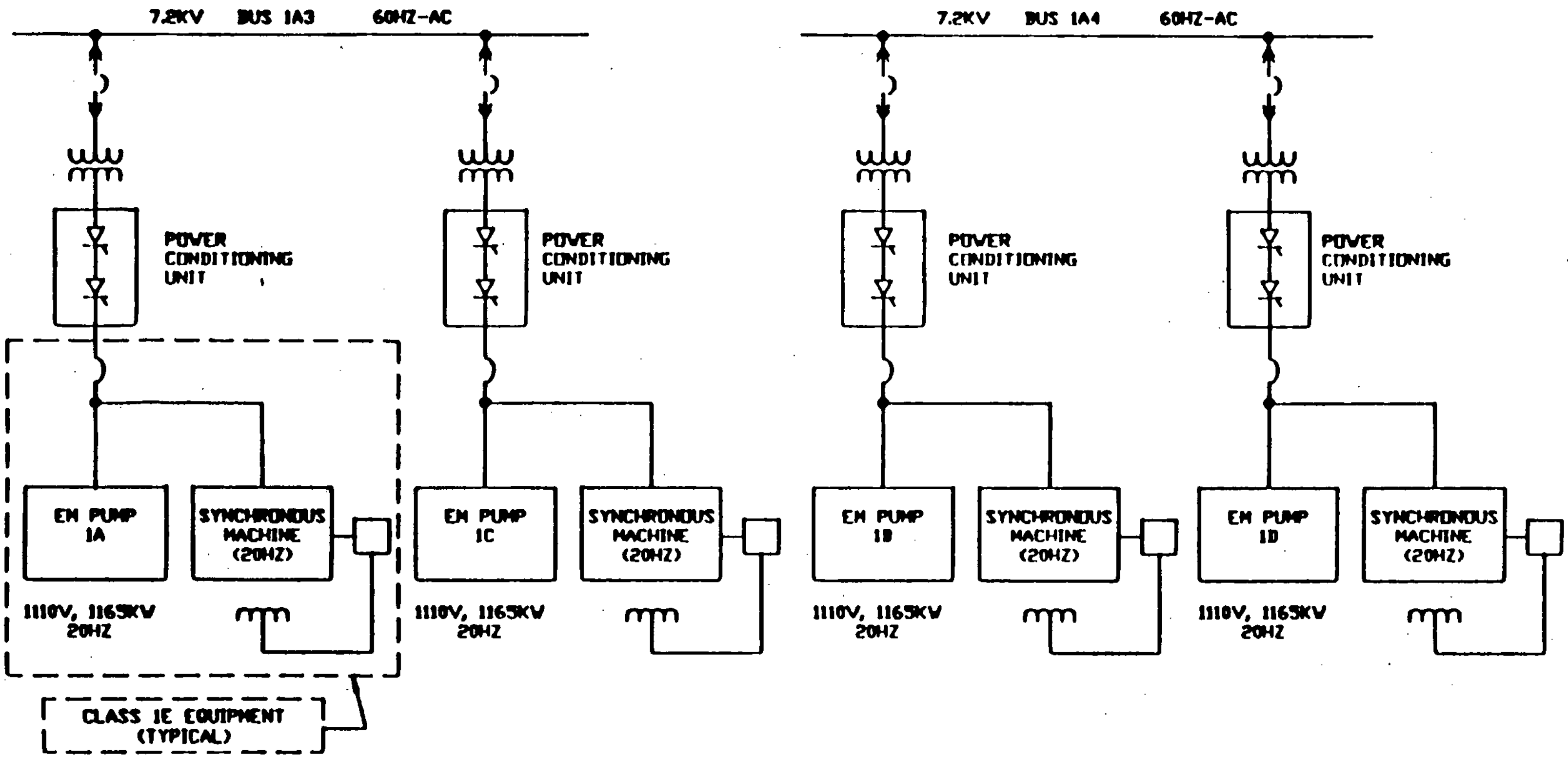


Figure 8.3-5. EM PUMP POWER SUPPLY

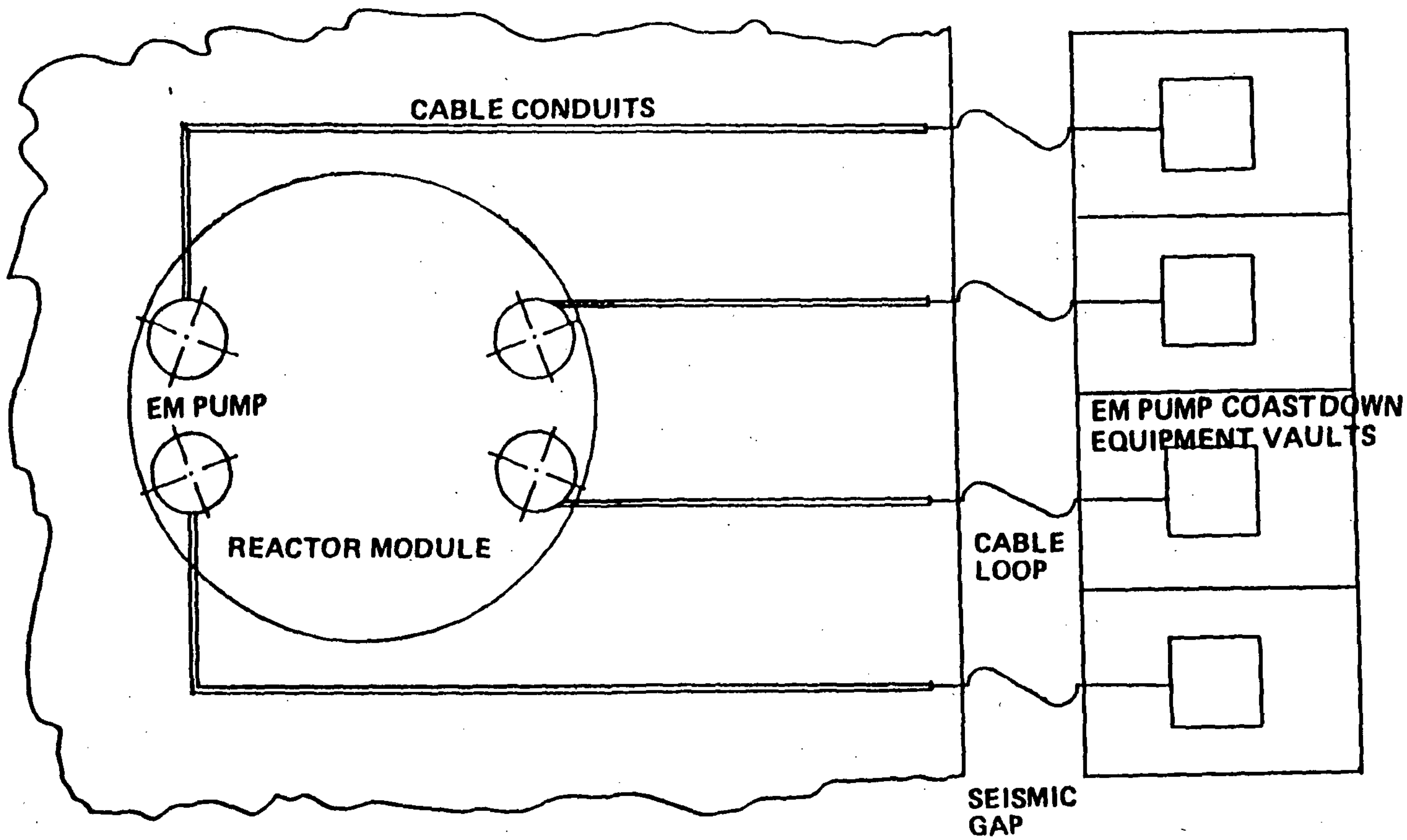


Figure 8.3-6 **EM PUMP COASTDOWN POWER CABLE SEPARATION**

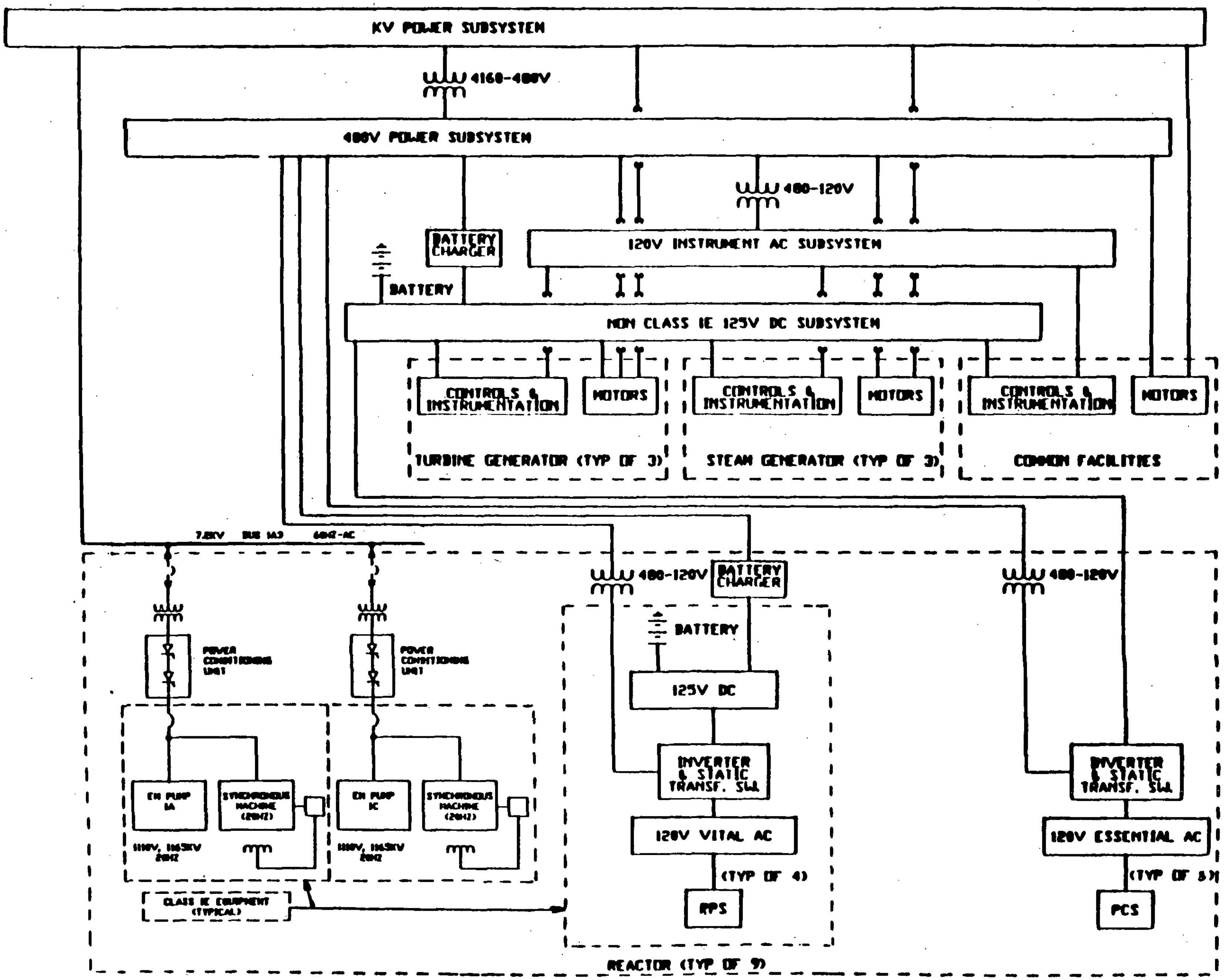


Figure 8.3-7 - PLANT CONTROL AND BUILDING ELECTRICAL POWER