In addition, the containment structure will withstand the following Tornado generated missiles (only one missile was considered acting at any time simultaneously with the 360 mph wind load):

Horizontal Missiles

1) 4" x 12' wood plank at 300 mph

2) 4000 lb auto at 50 mph less than 25' above the ground (25 ft2 contact area).

Vertical Missiles

1) 4" x 12' x 12' wood plank at 90 mph

2) 4000 lb auto at 17 mph less than 25' above the ground (25 ft2 contact area).

Specific structural effects as the result of missile impact are: 1) missile penetration and 2) structural response to dynamic impact. In addition to the overall structural effects such as overturning moment and base shear, the local structural effects must be considered in the design for tornado wind and generated missile loads. For missile loads, limited local plasticity, structural dynamic response ductility and redistribution of stresses in redundant structures due to plastic action was permitted.

Consideration of tornado loads was not a factor in the design of the Containment structure. The 3 psig negative pressure is approximately 4% of the maximum internal pressure load (1.5P=70.5 psig) thus stresses introduced into the rebar from this load are very small.

5.1.4 Penetrations

5.1.4.1 General

In general, a penetration consists of a sleeve embedded in the concrete wall and welded to the containment liner. The weld to the liner is shrouded by a continuously pressurized channel which is used to demonstrate the integrity of the penetration-to-liner weld joint. The pipe, electrical conductor cartridge, duct or equipment access hatch passed through the embedded sleeve and the ends of the resulting annulus were closed off, either by welded end plates, bolted flanges or a combination of these.

Differential expansion between a sleeve and one or more hot pipes passing through it was accommodated by using a bellows type expansion joint between the outer end of the sleeve and the outer end plate, as shown on Figure 5.1-12.

The components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

Pressurizing connections were provided to continuously demonstrate the integrity of the penetration assemblies.

5.1.4.2 Types

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Electrical Penetrations

"Cartridge" type penetrations are used for all electrical conductors passing through the Containment. The penetrations are provided with a pressure connection to allow continuous pressurization. Insulating bushings or fused glass seals are used to provide a pressure barrier for the conductor.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

Figure 5.1-13 shows a design of typical electrical penetrations. There are approximately 60 electrical penetrations.

Piping Penetrations

Double barrier piping penetrations are provided for all piping passing through the Containment. The pipe is centered in the embedded sleeve which is welded to the liner. End plates are welded to the pipe at both ends of the sleeve. Several pipes may pass through the same embedded sleeve to minimize the number of penetrations required. In this case, each pipe is welded to both end plates. A connection to the penetration sleeve is provided to allow continuous pressurization of the compartment formed between the piping and the embedded sleeve. In the case of piping carrying hot fluid, the pipe is insulated and cooling is provided to maintain the concrete temperature adjoining the embedded sleeve at or below 150 F.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

Cooling is provided for most hot penetrations through the use of air-to-air heat exchangers. These are made in accordance with the ASME UPV Code, Section VIII, by welding together two embossed sheets of 10 gage carbon steel material, the embossments forming coolant passages. The unit is rolled into the form of a cylinder with an outside diameter slightly smaller than the respective inside diameter of the penetration sleeve. The exchanger is placed inside the sleeve and outside the pipe insulation, with the inlet and outlet coolant connections penetrating the sleeve between the outside concrete wall surface and the bellows expansion joint. The coolant to be used is ambient air fed by a centrifugal blower which is backed up with a full sized spare. The isolation features and criteria for piping penetrations are given in Chapter 6. Figure 5.1-12 shows typical hot and cold pipe penetrations.

Loss of cooling for the sleeve is highly improbable. The heat shield has no moving parts, and the cooling air is at low pressure. There are redundant blowers to assure that cooling air is not lost for a significant time. The blowers operate off a diesel bus and can be manually started following a blackout. The thermal insulation on the pipe wall reduces heat flow to the liner sleeve. Operation of the cooling unit can be ascertained by opening the "flow through" connection of the penetration pressurization system on the penetration sleeve and observing the temperature of the cooling air emerging.

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In order to lose significant structural properties, concrete must be held continuously at 500 to 600 F. The hottest penetrations are the main steam lines, which normally operate at a temperature of 507 F. The results of a two dimensional transient heat transfer analysis indicated that in the improbable case that all cooling air would be lost to the main steam penetrations, the surrounding concrete would reach a maximum temperature of 200 F in approximately 100 hours and 280 F in approximately 1000 hours. It is highly improbable that cooling air would be lost a very long period of time since the failure of any of the air blower drive motors is alarmed in the control room. Even if the adjoining concrete did reach these temperatures (200 – 300 F), the strength of the structure would not be impaired for two reasons:

1) No credit was taken for the tensile strength of the concrete.

2) These temperatures have substantially no effect on the strength of the penetration sleeve or the reinforcing bar in the area of the penetration.

A total of approximately 80 pipes pass through approximately 50 penetration sleeves, 23 of which are considered thermally hot. In addition, several spare sleeves (capped and pressurized) are provided for the possible future addition of piping.

Equipment and Personnel Access Hatches

An Equipment Hatch was provided. It was fabricated from welded steel and furnished with a double-gasketed flange and bolted dished door. The hatch barrel is embedded in the containment wall and welded to the liner. Provision was made to continuously pressurize the space between the double gaskets of the door flanges and the weld seam channels at the liner joint, hatch flanges and dished door. Pressure is relieved from the double gasket spaces prior to opening the joints. The Personnel Hatch is a double door, mechanically-latched, welded steel assembly. A quick acting type, equalizing valve connects the Personnel Hatch with the interior of the containment vessel for the purposes of equalizing pressure in the two systems when entering or leaving the containment. The Personnel Hatch doors are interlocked to prevent both being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened.

Remote indicating lights and annunciator situated in the Control Room indicate the door position status. An emergency lighting and communication system operating from an external emergency supply is provided in the lock interior. Emergency access to either the inner door from the containment interior or to the outer door from outside, is possible by the use of special door unlatching tools. The design was in accordance with Section VIII of the ASME Code.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

Outage Equipment Hatch (OEH)

Outage Equipment Hatch can be used in place of the Equipment Hatch at Elevation 95'-0" in the Containment Building during outages. The OEH will be attached to the Containment Building using the same attachments for the Equipment Hatch. The OEH door can be closed and sealed in less than 30 minutes and is designed to withstand the radiation release from a fuel handling accident involving recently-irradiated fuel (i.e, fuel subcritical for less than 84 hours).

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The OEH is provided with sealed service penetrations for the passage of service lines (i.e., compressed air, electricity, fluid carrying hoses, instrumentation, fiber optic cables, etc.).

Special Penetrations

1) Fuel Transfer Penetration

A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the Reactor Containment and the spent fuel pit. The penetration consists of a 20-inch stainless pipe installed inside a 24-inch pipe. The inner pipe acts as the transfer tube. The transfer tube is fitted with a pressurized double gasketed blind flange on the refueling canal end to seal the reactor containment. The terminus of the tube outside the containment is closed by a standard gate valve. The outer pipe is welded to the containment liner and provision is made by use of a special seal ring for pressurizing all welds essential to the integrity of the penetration during plant operations. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures. Figure 5.1-14 shows a sketch of the fuel transfer tube.

2) Containment Supply and Exhaust Purge Ducts

The ventilation system purge ducts are each equipped with two quick-acting tight-sealing valves (one inside and one outside of the containment) to be used for isolation purposes. The valves are manually opened for containment purging, but are automatically closed upon a signal of high containment pressure or high containment radiation level. The space between the valves is pressurized above calculated peak accident response pressure, while the valves are normally closed during plant operation. See Section 5-3, Containment Ventilation System, and Section 6.4, Containment Air Recirculation Cooling and Filtration System.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

Two solenoid controlled, pneumatically operated butterfly valves are provided for each purge penetration, one on each side of the containment building wall. Two penetrations, one supply and one exhaust, are required. Valves are spring-loaded to fail closed.

The space between the valves is pressurized from the pressurization system through an electrically operated three-way solenoid valve. The pressure is maintained only when valves are closed and must be relieved before butterfly valves can be opened. Failure to release this pressure will prevent valves from opening.

Failure of any of the valves to open will prevent the fans from running. Tripping or either of the purge fans will automatically close the butterfly valves and pressurize the space between the valves. Failure of any of the valves to close will prevent the adjacent space from being pressurized, and sound the loss-of-pressurization alarm.

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Loss of pressure for either zone will be displayed by individual indicating lights at the Main Control Board.

The valve control solenoids and pressurization solenoids are controlled from a single control switch on the fan room control panel. The cycle is initiated by setting the control switch to "open" position. This will energize the pressurization alarm.

When the pressure between the valves has been relieved, the valves control solenoids are energized and the valves opened. If for any reason, any of the four valves fail to open within a given time after the cycle is initiated, all four valves will close and pressure will be restored. The circuit is interlocked to prevent inadvertent opening of the valves during S.I. condition.

Once all four valves have been opened, the operator has a pre-determined time (approximately one minute) to start the purge supply fan. Failure to do so will cause all four valves to close.

Position indicating lights for each of the four valves are provided on the Fan Room Control Panel and Main Control Board.

3) Sump Penetrations

The piping penetration in the containment sump area is not of the typical sleeve-to-liner design. In this case, the pipe is welded directly to the base liner. The weld to the liner is shrouded by a test channel which is used to demonstrate the integrity of the liner.

5.1.4.3 Design of Penetrations

Criteria

The liner is basically not a load-carrying member because it is subjected to strains imposed by the reinforced concrete; nevertheless, the liner was reinforced at each penetration in accordance with the ASME Code Section VII. The weldments of liner to penetration sleeve are of sufficient strength to accommodate stress concentrations and adhered strictly to ASME Code Section VIII requirements for both type and strength.

Liner stress is imposed on the cylindrical penetration as a circular uniform load acting around the circumference of the penetration. The penetration thicknesses were chosen to accommodate this load without causing severe distress at the opening.

The penetration sleeves and plates were designed to accommodate all loads imposed on them under operating conditions (thermal effects and internal penetrations and test pressures) and accident conditions (loads resulting from all strains, internal pressures, and seismic movements).

In the design of the piping penetration sleeves and the piping going through them, maximum total stress in all cases was limited to a value below the yield stress of the material involved; therefore, no plastic design criteria were employed. In particular, piping whose failure would result in a Loss-of-Coolant Accident and the main steam and feedwater pipe penetrations and pipe supports in the Containment Building were designed to prevent the formation of a plastic

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"hinge" in the pipe should any of these pipes rupture. This was accomplished by effectively anchoring these pipes at 90° elbows connected to all these pipes adjacent to the penetration both inside and outside the building, and by restraining these pipes along their run inside the building and outside the building to the first stop valve. The anchors and restraints were designed to prevent a breach of containment at the piping penetrations should any of these pipes rupture inside, immediately outside, or within the penetration itself. The penetrations were designed to the strength of the pipe and no further considerations are necessary.

To insure that a Loss-of-Coolant Accident acting simultaneously with an earthquake would not result in a breach of containment by causing a failure of one or more pipe penetrations through the Containment Building wall, the following methods were used:

All auxiliary piping attached to the Reactor Coolant System which passes through penetrations in the Containment Building wall must also pass through the circular secondary shield wall approximately fifteen feet inside the building as illustrated in Plant Drawing 9321-F-25012 [Formerly Figure 5.1-2]. The total number of pipes in this category is very limited. They were examined individually and suitable restraints or anchors were used either at or within the secondary shield wall to prevent a Loss-of-Coolant Accident or a failure of one of these pipes within the secondary shield wall from causing the failure of the building penetrations through which the pipes pass. In some cases, it was physically impossible for any conceivable movement of the end of those pipes attached to the Primary Coolant System to be reflected at the building penetration and impose other than ordinary operating loads at these points. In other cases, it was necessary to design restraints for the pipes at the secondary shield wall to withstand the failure of the pipe within the wall in tension. Some auxiliary pipes attached to the Reactor Coolant System are attached at points which will not move; for instance, the reactor coolant pump seal water injection pipes and the steam generator blowdown pipes. In general, these have restraints at the secondary shield wall designed for normal loads plus the reaction forces resulting from the double ended rupture of these pipes within the shield wall.

All Containment Building piping penetrations except main steam and feedwater were designed as anchors for the pipes passing through them and transmit piping loads to the reinforced concrete wall. The anchorage strength exceeds the maximum combined forces imposed by the effects on the piping penetration of dead loads, loads induced from a Loss-of-Coolant Accident, thermal expansion of the pipe, penetration air pressure, and earthquake loads.

The piping penetrations were designed to transmit the above combined loadings to the concrete structure without exceeding the yield strength of the penetration steel. Typical penetration details are shown in Figure 5.1-12. Load transfer from the pipe to penetration anchorage is limited to the actual loads induced or to the ultimate strength capacity of the pipe in bending, shear, axial, or torsional loadings.

All piping penetrating the Containment meet the requirements of the USAS B31.1.0 Power Piping Code. In the case of the main steam and feedwater lines, the supports, inside and outside the Containment Buildings to the second isolation valve, were designed so that a failure of any one of these pipes does not result in breach of containment of the failure of any other main steam or feedwater pipe between the steam generator and the second isolation valve.

The design of all containment building piping penetration sleeves and end plates except the new Steam Generator Blowdown Penetrations (AA, BB, CC, and DD) and Service Water Penetration (SS) was in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII. The Steam Generator Blowdown Penetration Sleeves and end plates and the Service Water

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Penetration (SS) end plates were designed in accordance with the requirements of the 1986 edition of the ASME Boiler and Pressure Vessel Code, Section III subsection NC.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the AMSE Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

Pipes which penetrate the containment building wall and which are subject to machinery originated vibratory loadings, such as the Reactor Coolant Pumps, had their supports spaced in such a manner that the natural frequency of the piping system immediately adjacent to the penetrations is greater than the dominant frequencies of the pump. Pipe line vibration was checked during preliminary plant operation; and where necessary, vibration dampers were fitted. This checking and fitting effectively eliminates vibrating loads as a design consideration.

Materials

The material for penetrations including the Personnel and Equipment Access Hatches, together with the mechanical and electrical penetrations is carbon steel, conforming with the requirements of the ASME Pressure Vessels Code Section VIII, and exhibiting ductility and welding characteristics compatible with the main liner material. The Equipment Hatch, penetration sleeves and Personnel Lock meet the Charpy V-notch impact values for a minimum of 15 ft-lbs at -50°F.

The stainless steel expansion joints (bellows) of the hot penetration expansion joints were protected from damage in transit and during construction by sheet metal covers fastened in place at the fabricator's shop. These were left in place permanently if there was no interface with nearby piping or equipment.

Due to cracking in the bellows of the Main Steam and Boiler Feedwater penetrations, replacement bellows were installed. The replacement bellows are constructed of improved materials.

The materials making up the penetrations conform to the following specifications:

	Item	Specification	Minimum Yield Strength (PSI)	Minimum Tensile Strength (PSI)	Elongation
1.	Mech. Penetration Sleeve – 12" Dia. & under**	ASTM A333, Gr. 1	30,000	55,000	35% in 2"
2.	Mech. – Over 12" Dia.**	ASTM A201 Gr. B to A300	32,000	60,000	22% in 8"
3.	Rolled Shapes+	ASTM A36, ASTM A131 Gr. C	36,000 32,000	58,000 58,000	20% in 8" 21% in 8"

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4. End Plates	a)	ASTM A300 C1.1 Fire Box A201, Gr B(1)	32,000	60,000	22% in 8"
	b)	ASTM A240 Type 304L+	25,000	70,000	40% in 2"
***	c)	ASTM A516, Gr. 60	32,000	60,000	21% in 8"
5. Fuel Transfer Tube+		ASTM A240 Type 304L	25,000	70,000	40% in 2"
6. Bellows+	a)	ASTM A312 Type 304L	25,000	70,000	35% in 2"
	b)	ASME SB168 Inconel 600++	35,000	80,000	30% in 2"
7. Elec. Penetrations**		ASTM A333 Gr. 1	30,000	55,000	35% in 2"
8. Equip. Hatch Insert**		ASTM A300 C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"

** The Equipment Hatch, penetration sleeves and Personnel Lock were Charpy tested to a minimum of 15 ft-lbs at -50°F.

\+ No specific NDTT requirements

++ Main Stream and Main Feedwater penetrations

*** Service Water Penetration SS end plates were Charpy V-notch tested to a minimum of 20 ft-lbs (1of 3 test only) at 0°F or lower with a minimum average of three tests of 25 lbs

Item	Specification	Minimum Yield Strength (PSI)	Minimum Tensile Strength (PSI)	Elongation
9. Equip. Hatch Flanges**	ASTM A300, C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
10. Equip. Hatch Head**	ASTM A300 Firebox A201, Gr. B	32,000	60,000	22% in 8"

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11. Personnel Hatch**	ASTM A300, C1.1. Firebox A201, Gr. B	32,000	60,000	22% in 8"
12. Piping Penetration Reinf.*	ASTM A442, Gr.60	32,000	60,000	22% in 8"

13. Outage Equipment Hatch is designed and built in accordance with ASME Section VIII, 1989 and made of ASTM A516 Grade 60 or higher Grade material. The structural steel members are made of SA 36; Pipe penetrations are made of ASTM A 106 Grade B.

NOTE:

* The liner plates for the shell, bottom and dome were impact tested on a longitudinal section at 15 ft-lbs at a temperature 30 degrees below the service temperature of -50°F.

** The Equipment Hatch, penetration sleeves and Personnel Lock were Charpy tested to a minimum of 15 ft-lbs at -50°F.

Consideration of Jet Loads, Missile Impact and Tornado Loads for Openings

The 3'-0" thick crane wall, the 4'-0" and 6'-0" thick Refueling Canal and the 2'-0" thick operating floor are capable of resisting jet force loads and missiles from primary coolant piping. Thus, jet force loads and missiles from the potential failure of the Primary Coolant System are contained within the reactor coolant compartment shield walls and cannot impinge on the containment structure walls; consequently, these loads were not considered in design of large openings. All other missiles terminate inside these concrete shield walls and consequently were not factored into the large opening design. Large openings are shielded or are far enough away to preclude impingement from main steam and feedwater pipe break loads.

Tornado loads are small compared to the seismic loadings. The tornado shear loads from torsion and translational wind force and the overturning moments caused by wind load have a minimum factor of safety of approximately 2.5 when compared with earthquake shears and moments which were used to size the seismic reinforcing bars. The tornado moment and shears are in fact smaller than the minimum earthquake moments and shears considered in design. On this basis, the seismic bars provide more than an adequate mechanism for resisting tornado loads. In addition, tornado loads act independently of other severe loads; therefore, the Equipment Hatch and Personnel Lock reinforced concrete bosses were designed for simultaneous design basis accident and earthquake loads, which were larger than tornado loads, are of more than adequate strength to resist tornado loads.

The containment structure will not be penetrated by the tornado-generated missiles. The concrete sections around large openings are thicker than the 4'-6" Containment wall and so no further consideration of tornado missiles at the large openings was necessary. The large openings have shielded walls of sufficient thickness to protect against tornado missiles.

Consideration of Curvature of the Wall in the Finite Element Analysis

Curvature of the containment cylinder wall was included in the finite element analysis for large openings by assigning three coordinates to each node point in the model. This in effect idealizes the structure as a series of chords of a circle with radii equal to the containment cylinder reference surface. Since the widest element in the fine model at the Equipment Hatch

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opening is 50", or approximately 1% of the total circumference, the chords adequately represent the curvature of the containment surface. Since the shape and stiffness of the structure was accurately represented in the model, all forces and effects were included in the computer output.

The procedures used to design for the six stresses and the justification for all structural elements (rebar) provided to resist the forces or stress resultants outputted by the computer are discussed in detail in Appendix 5A. All concrete in tension was considered cracked in the finite element analysis.

5.1.4.4 Leak Testing of Penetration Assemblies

A proof test was supplied to each penetration by pressurizing the necessary areas to 54 psig. This pressure was maintained for a sufficient time to allow soap bubble and Freon sniff tests of all welds and mating surfaces. Any leaks found were repaired and retested; this procedure was repeated until no leak existed.

5.1.4.5 Construction

The qualification of welding procedures and welders was in accordance with Section IX, “Welding Qualifications” of the ASME Boiler and Pressure Vessel Code. The repair of defective welds was in accordance with paragraph UW-38 of Section VIII “Unfired Pressure Vessels.”

For penetrations between 9" and 18", all the reinforcing bars including primary and secondary vertical bars and diagonal bars are grouped around the penetrations. Due to the continuity of the bars and the relatively small opening size, no special provisions were needed to resist normal, shear and bending stresses. The penetrations are keyed into the concrete, thus creating an edge loading which induces torsion into the walls. The loads are small and the rebar feels little effect from this torsional loading.

For penetrations greater than 18" to 4'-0" the bars are continuous. Since reinforcing is continuous around penetrations, steps were taken to insure that no local crushing of concrete occurred.

From an article, "Detailing and Placing Reinforcing Bars" by Paul F. Rice from Concrete Construction, January 1965, it was determined that in order to prevent local crushing of the concrete a minimum bend diameter of 31 times the bar diameter is required when the reinforcing is stressed to yield. The angle of bend in the rebar determines the force which is transmitted to the concrete in the event the bar tries to straighten out due to tension. For this reason, most bars were bent at 10 degrees except at large penetrations, including the Equipment Hatch, Personnel Lock, main steam and feedwater, and air purge penetrations, where the deviation of the bar from its centerline is too large to permit a 10o bend. In these cases, the bars were bent at 30 degrees but a tie back system was used which prevents a build-up of forces. To further prevent this buildup (in all cases except the equipment hatch penetration) the line of force makes an angle of one-half of the angle of bend, from a horizontal line for the vertical bars and from a vertical line for the horizontal bars and is tangent to the outside of the penetration.

Details of the Personnel and Equipment Hatch design are presented in Section 3.4 of Appendix 5A.

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Concrete was poured in nominal 5' lifts, 360 degrees with no stagger. Approximately one week was allowed to elapse between pours and the surface was left rough, thoroughly cleaned by air blowdown, and all laitance removed. Joints were thoroughly wetted and slushed with a coat of neat cement grout immediately before placing of new concrete except for the exterior of the containment where surfaces were thoroughly wetted but not grouted.

5.1.4.6 Testability of Penetrations and Weld Seams

All penetrations, the Personnel Air Lock and the Equipment Hatch were designed with double seals which are normally pressurized at a minimum pressure greater than the calculated peak accident pressure. Individual testing at 115% containment design pressure is also possible.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

The containment ventilation purge ducts are equipped with double isolation valves and the space between the valves is permanently piped up to the penetration pressurization system. The space can be pressurized to 115% of design pressure when the isolation valves are closed. The purge valves fail in the closed position upon loss of power (electric or air).

All welded joints in the liner have steel channels welded over them on the inside of the vessel. During construction, the channel welds were tested by means of pressurizing sections with Freon gas and checking for leaks by means of a Freon sniffer. Most welds are continuously pressurized during power operation at a minimum pressure greater than the calculated peak accident pressure. Liner welds that are not pressurized during power operation are those welds associated with disconnected sections of the Weld Channel Pressurization System. The integrity of the welds associated with any disconnected sections of the Weld Channel Pressurization System is verified by integrated leak rate testing.

Test connections are provided on the Penetration and Weld Channel Pressurization System lines to the Equipment Hatch and Personnel Airlock to allow for leak testing of the PWCP connections.

The use of the weld channel pressurization system may necessitate periodic relief of pressure buildup within the containment, should the system leak into the containment structure.

When pressure relief of the Containment is required during normal operation, it is accomplished using the containment pressure relief line and not the containment purge lines. However, the pressure relief exhaust is routed through charcoal filters which have an iodine removal efficiency of 90.0%. Prior to pressure relief operations, the Containment Auxiliary Charcoal Filter System (see Section 5.3) may be operated to reduce the activity in the containment atmosphere. Assuming 1% fuel defects and 50 lbs/day leakage of reactor coolant into the Containment, the containment atmosphere activity has the maximum value of 20.4 x MPC for iodines and 135.5 x MPC for noble gases after approximately 16 hours of operation of the containment auxiliary charcoal filter system whose efficiency for iodine removal is 90%.
The activity released to the environment as a result of depressurizing the Containment from 1.0 psig to 0 psig at 1500 CFM for 2 hours based on the above abnormal conditions is:

a) For iodines: 8.26×10^{-15} curies expressed as equivalent I-131

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b) For noble gases: 6.68 curies expressed as equivalent Xe-133

The maximum expected operating conditions considered as normal are taken as 0.2% fuel defects and 14.4 gpd leakage of reactor coolant into the Containment. For these conditions, the containment atmosphere activity is 8.16 x MPC for iodines and 65.0 x MPC for noble gases after approximately 16 hours operation of the containment charcoal filter system whose efficiency for iodine removal is 99%. The activity released to the environment as a result of depressurizing the containment from 1.0 psig to 0 psig at 1500 cfm for 2 hours through the purge line carbon filters (iodine removal efficiency of 99.0%) based on these maximum operating conditions is:

a) For iodines: 2.49×10^{-6} curies expressed as equivalent I-131

b) For noble gases: 3.21 curies expressed as equivalent Xe-133

5.1.4.7 Accessibility Criteria

The Containment is completely closed whenever the core is critical or whenever the primary system temperature is above 200 F, except as required for brief periods necessary to relieve the Containment to keep the pressure below a reasonable level (1-2 psig) or to purge the Containment in preparation for Containment entry.

Limited access to the Containment through personnel air locks is possible with the reactor at power or with the primary system at hot shutdown for special maintenance or periodic inspections. Access at power would normally be restricted to the areas external to the reactor equipment compartment primarily for inspection and maintenance of the air recirculation equipment, incore instrumentation chamber drives, and instrument calibration.

After shutdown, the Containment vessel is purged to reduce the concentration of radioactive gases and airborne particulates. This purge system was designed to reduce the radioactivity level to doses defined by 10 CFR 20 for a 40-hour occupational work week, within 2-6 hours after plant shutdown. Since negligible fuel defects are expected for this reactor, much less than the 1% fuel rod defects used for design, purging of the Containment is normally accomplished in less than 2 hours. To assure removal of particulate matter the purge air will be passed through a high efficiency filter before being released to the atmosphere through the purge vent.

The primary reactor shield was designed so that access to the primary equipment is limited by the activity of the primary system equipment and not the reactor.

5.1.5 System Design Evaluation

5.1.5.1 Reliance on Interconnected Systems

The containment leakage limiting boundary is provided in the form of a single, carbon steel liner on the vessel having double barrier weld channels and penetrations. Each system whose piping penetrates this boundary was designed to maintain isolation of the Containment from the outside environment. Provisions are made to continuously pressurize penetrations and most weld channels and to monitor leakage from this pressurization.

5.1.5.2 System Integrity and Safety Factors

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Pipe Rupture – Penetration Integrity

The penetrations for the main steam, feedwater, blowdown and sample lines were designed so that the penetration is stronger than the piping system and that the vapor barrier will not be breached due to a hypothesized pipe rupture.

Major Component Support Structures

The support structures for the major components were designed to resist all thrust forces, moments and torques associated with either a Reactor Coolant System or main steam pipe break. All primary structural steel elements were designed for stresses not exceeding yield stress due to these forces.

5.1.5.3 Containment Structure Components Analyses

The details of radial, longitudinal and horizontal shear analyses for the containment reinforced concrete are given in Section 5.1.3.

5.1.5.4. Performance Capability Margin

The containment structure was designed based upon limiting load factors which were used as the ratio by which accident and earthquake loads were multiplied for design purposes to ensure that the load/'deformation behavior of the structure is one of elastic, low strain behavior. This approach places minimum emphasis on fixed gravity loads and maximum emphasis on accident and earthquake loads. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors primarily provide for a safety margin on the load assumptions. Tabulations of load combinations and load factors utilized in the design which provide an estimate of the margin with respect to all loads are referenced in Section 5.1.2.

5.1.6 Minimum Operating Conditions

The minimum operating conditions which are applicable to the Containment System are given in the Technical Specifications.

5.1.7 Containment System Structure-Inspection and Testing

Initial Containment Leakage Rate Testing

Criterion: Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result form the design basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance. (GDC 54 of 7/11/67)

After completion of the containment structure and installation of all penetrations and weld channels, integrated leakage rate tests were performed prior to initial plant operations to establish the respective measured leakage rates and to verify that the leakage rate at the peak accident conditions is no greater than 0.075 percent by weight per day of the containment stream-air atmosphere at the calculated peak accident conditions. The leakage rate tests were performed using the absolute method. The duration of each test was not less than 24 hours.

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Periodic Containment Leakage Rate Testing

Criterion: The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime. (GDC 55 of 7/11/67)

The peak accident pressure integrated leakage rate test is conducted at periodic intervals during the life of the plant, and also as appropriate in the event major maintenance or major plant modifications are made.

A leak rate test at the peak accident pressure using the same test method as the initial leak rate can be performed at any time during the operational life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

Provisions for Testing of Penetrations

Criterion: Provisions shall be made to the extent practical for periodically testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident. (GDC 56 of 7/11/67)

Penetrations were designed with double seals which are continuously pressurized above accident pressure. The large access openings such as the Equipment Hatch and Personnel Air Lock are equipped with double gasketed doors and flanges with the space between the gaskets connected to the pressurization system. The system utilizes a supply of clean, dry, compressed air which places the penetrations under an internal pressure above the peak calculated accident pressure.

A permanently piped monitoring system is provided to continuously measure leakage from all penetrations.

Leakage from the monitoring system is checked by continuous measurement of the integrated makeup air flow. In the event excessive leakage is discovered, each penetration can then be checked separately at any time.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

Provisions for Testing of Isolation Valves

Criterion: Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits. (GDC 57 of 7/11/67)

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

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Initiation of containment isolation employs coincidence circuits which allow checking of the operability and calibration of one channel at a time. Removal or bypass of one signal channel places that circuit in the half-tripped mode.

Local leak rate testing of containment isolation valves is performed in accordance with Technical Specification 5.5.15. The Containment Leakage Rate Program is in accordance with the guidance contained on Regulatory Guide 1.163, except as noted in the Technical Specification.

Field and operational inspection and testing were divided into three phases:

1) those taking place during erection of the Containment Building liner; construction tests

2) those taking place after the containment structure was erected and all penetrations were complete and installed; pre-operational tests

3) monitoring during reactor operation; post-operational tests

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

5.1.7.1 Construction Tests

During erection of the liner, the following inspection and tests were performed:

Bottom Liner Plates

All liner plate welds were tested for leak tightness by vacuum box. The box was evacuated to at least a 5 psi pressure differential with the atmospheric pressure.

After completion of a successful leak test, the welds were covered by channels. A strength test was performed by applying a 54 psig air pressure to the channels in the zone for a period of 15 minutes.

The zone of channel-covered welds was pressurized to 47 psig with a 20% by weight of Freon-air mixture. The entire run of the channel to plate welds was then traversed with a halogen leak detector.

The sensitivity of the leak detector is 1 x 10-9 standard CC per second. The sniffer was held approximately ½ inch from the weld and traversed at a rate of about ½-inch per second. The detection of any amount of halogen, indicating a leak, required weld repairs and retesting.
After the halogen test was completed all liner welds not accessible for radiography were pressurized with air to 47 psig and soap-tested. Any leaks indicated by bubbles were repaired and retested. Where leaks occurred, welds were removed by arc gouging, grinding, chipping and/or machining, before rewelding. In addition, the zone of channels was held at the 47 psig air pressure for a period of at least two hours. The drop in pressure was not to exceed the equivalent of a leakage of 0.05% of the containment building volume per day. Compensation for change in ambient air temperature was made if necessary.

Vertical Cylindrical Walls and Dome

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For the liner, a complete radiograph was made of the first 10 feet of full penetration weld made by each welder or welding operation. A minimum of a 12" film "spot" radiograph was made every 50 feet of weld thereafter on the side walls and dome, except where back-up plates are used. The radiograph films were given to United Engineers and Constructors for their review.

When a spot radiograph showed defects that required repair, two adjacent spots were radiographed. If defects requiring repair were shown in either of these, all of the welding performed by the responsible operator or welder was 100% radiographed to determine the end of defect.

The performance and acceptance standards for all radiography is ASME Section VIII, Paragraph UW51.

The liner plate to plate welds were tested for leak tightness by vacuum box techniques. After successful completion of the spot radiography and vacuum box tests and subsequent repair of all defects, the channels were welded in place over all seam welds in a pre-determined zone. A strength test was performed on the liner plate weld and the channel weld by pressurizing the channel with air at 54 psig for 15 minutes. In addition, each zone of channel covered weld was leak tested under the Freon-air mixture at 47 psig.

In location where radiography was not possible, such as the lower courses of shell plates where back-up plates were used, and where liner bottom welds and floor plates were made to angles and tees, the liner fabricator welded on a 2" long overrun coupon. The overrun coupon was chipped off, marked for location and given to United Engineers and Constructors for testing. These welds are also vacuum box tested.

Welded studs were visually inspected, and at least one at the beginning of each day's work and another at approximately mid-day were bend-tested to 45 degrees for each welder. Studs failing visual or bend-testing were removed.

While the liner is not a pressure vessel, industry experience has shown that leaks in pressure vessels normally occur at joints. For this reason and following current liner fabrication practice, there was no radiographic or other non-destructive examination of liner plate.

<u>Liner Erection Tolerance</u>

Deviations from the allowable erection tolerance standards were located, documented and, in most cases, eliminated during the normal erection of the liner. This was accomplished by jacking against the polar crane wall, utilizing tubular beams, capped by beams of sufficient cross-sectional area to insure against localized buckling of the liner plate. For areas above the concrete polar crane wall, the required tolerances were met and maintained by circular plate wind girders. For the isolated cases where the liner could not be jacked into tolerance, a Non-Conformance Report (NCR) was written and forwarded to the architect-engineer with a complete survey of the area for an engineering evaluation together with a waiver request. This documentation is maintained by the Authority. Only minor deviations were experienced.

<u>Concrete Compression and Slump Testing</u>

The compression test samples consisted of six 6" x 12" cylinders for each 100 cu. yd. or portion thereof, per class, per day. A minimum of one set of six cylinders was made for pours of less than 100 cu. yds. Three cylinders were broken at 7 days and 3 cylinders at 28 days. The basis

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for rejection was failure to develop a minimum compressive strength at 28 days of 15% above the nominal design strength as proven on an average of the three cylinders.

A slump test was performed on each truckload of concrete used in the first four lifts (20 feet) for the containment exterior wall and was recorded for each sample from which compression test cylinders were made. For all other concrete, a slump test was made and recorded for three truckloads of concrete from each class of concrete per 100 cubic yards (or portion thereof) placed per day. A Quality Control inspector was present during the pour and visually checked the concrete from each truck. Any concrete which appeared to be near or over the limit was slump tested. Wet loads were rejected. The maximum slump for all pours was 5 inches except for special pours when specific approval was received from the Architect-Engineer. In no case was the slump permitted to exceed 7 inches.

The statistical results of compression testing for the 28 day breaks were:

a) 100% of the cylinder break tests exceed the minimum requirement

b) 75% of the cylinder break tests exceeded the minimum requirement by at least 1000 psi

c) 50% of the cylinder break tests exceeded the minimum requirement by at least 1250 psi

d) 25% of the cylinder break tests exceeded the minimum requirement by at least 1750 psi

e) 10% of the cylinder break tests exceeded the minimum requirement by at least 2250 psi

The samples for compression and a slump testing of concrete were taken from the point of discharge from the truck. There was no occurrence of pour removal or concrete rejected from these test results.

<u>Cadweld Splice Test Program</u>

In the Cadweld Test Program, tests were performed on production Cadwelds which had been removed (specifically for testing) from the Containment Building after placement. Of the first 141 production Cadwelds tested in this program, all test results were in excess of the minimum specified strengths.

The following test results were obtained from the actual Cadweld test reports submitted to WEDCO from Consolidated Testing Laboratory. Of the Cadwelds tested:

100% had ultimate strengths of at least 79,000 pis

75% had ultimate strengths of at least 95,100 psi

50% had ultimate strengths of at least 97,600 psi

25% had ultimate strengths of at least 102,600 psi

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10% had ultimate strengths of at least 105,100 psi

A statistical analysis of these results was performed using the methods outlined in Appendix 5A, Section 5.2.1.

The mean value of the ultimate strength of the splices was 99,580 psi with a standard deviation of 9.960 psi and a total range of 32,750 psi. Of the total at least 99% had an ultimate strength of 76,373 psi. No Cadwelds were rejected on the basis of test results from the Cadweld Test Program.

Penetrations

Strength and leak tests of individual penetration internals and closures and sleeve weld channels were performed in a similar manner to the above and all leaks repaired and the penetration or weld channel retested until no further leaks were found.

5.1.7.2 Pre-Operational Tests

All penetrations, and the welds joining these penetrations to the containment liner and the liner seam welds, were designed to provide a double barrier which can be continuously pressurized at a pressure higher than the calculated peak accident response pressure of the containment. This blocks potential sources of leakage with a pressurized zone and at the same time provides a means of monitoring the leakage status of the containment which is more sensitive to changes in the leakage characteristics of these potential leakage sources. Certain liner welds are no longer continuously pressurized. Therefore, the leakage status of these welds is no longer continuously monitored. The integrity of these welds is verified by integrated leak rate testing.

After the Containment Building was complete with liner, concrete structures, and all electrical and piping penetrations, Equipment Hatch and Personnel Lock in place, the following tests were performed:

1) Strength Test:

   A pressure test was made on the completed building using air at 54 psig. This pressure was maintained on the building for a period of at least one hour. During this test, measurements and observations were made to verify the adequacy of the structural design. For a description of observations, cracks, strain gauges, etc., refer to the Containment Report, Appendix 5A.

2) Integrated Leakage Rate Tests:

   Integrated leakage rate tests were performed on the completed building using the absolute method. These leakage tests were performed with the double penetration and weld channel zones open to the containment atmosphere.

3) Sensitive Leak Rate Test:

   After it had been assured that there were no defects remaining from construction, a sensitive leak rate test was conducted. The sensitive leak rate test included only the volume of the weld channels and double penetrations. This test is considered more sensitive than the integrated leakage rate test, as the

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instrumentation used permits a direct measurement of leakage from the pressurized zones. The sensitive leak rate test was conducted with the penetrations and weld channels at a minimum pressure greater than the calculated peak accident pressure and with the Containment Building at atmospheric pressure. The leak rate for the double penetrations and weld channel zones was equal to or less than 0.2% of the containment free volume per day.

In order to verify that the structural response of the Containment to pressure loads is in accordance with design assumptions and to provide assurance that the structure was constructed in accordance with the design to resist pressure loads, a Structural Integrity Test (SIT) was performed.

Readings and measurements were taken at 0 psig, 12 psig, 21 psig, 41 psig and 54 psig (the latter is 115% of the design pressure of 47 psig) during pressurization, and at 41, 18, 21, 41, and 0 psig during depressurization.

The following gross deformation measurements were taken during the SIT using invar wire extensometers. This provided a means for taking all measurements inside the containment structure thus eliminating effects of weather and temperature. All results were remotely recorded during the test and data was quickly reduced.

a) Radial deformation of the containment wall was measured at 15 locations in the thickened Equipment Hatch boss and the transition area from the thickened boss to the 4'-6" cylinder wall.

b) Diameter change in the containment structure was measured at 10 locations spaced at approximately 10'-0" between elevations 101'-0" and 191'-0".

c) Radial deflection of the containment cylinder wall was measured at elevation 91'-0".

d) Vertical deflection of the Containment was measured at elevations 95'-0", 143'-0" and 191'-0" and at the apex of the dome. Redundancy was provided for the measurement at the apex of the dome.

Detailed crack measurements were made prior to the test, at peak test pressure of 54 psig, and following depressurization at five areas of the exterior shell, each of at least forty square feet in area. The areas of detailed measurement were: a quadrant of the personnel lock concrete boss, and ten foot wide strips spanning elevations 43'-0" to 48'-0", 115'-0" to 120'-0", and 188'-0" to 193'-0".

In addition, the exposed surface of the containment shell was visually inspected prior to the test, at 41 psig during the ILRT, and following depressurization. These inspections were for purposes of monitoring the general crack pattern and for specifically following the behavior of the most significant crack.

5.1.7.3 Acceptability of Testing Program

AEC Safety Guide No. 18 "Structural Acceptance Test for Concrete Primary Reactor Containments" was followed for testing except in the following areas:

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1) The pattern of measurement points around the largest opening (equipment hatch) were not as shown in Figure C of Safety Guide 18 which indicated 12 points symmetrically located to measure radial and tangential deflections. The Indian Point 3 Structural Integrity Test required taking of radial measurements at 15 locations around the equipment hatch.

   Due to access restrictions, no deflection readings were taken on the lower vertical axis of this opening; the 15 measurement locations were symmetrically positioned in the remaining accessible area around this opening. Tangential deflections were not taken, as they were insignificant compared to the radial deflections. The second largest opening (personnel hatch) was structurally loaded in a manner similar to the equipment hatch; no deflection measurements were taken for the personnel hatch opening. This program of radial deflection measurements provided the necessary data to verify that anticipated deformations were taken into account and were within acceptable limits.

2) The structural integrity of the OEH was tested in the Vendor's shop to 7.5 psig for 10 minutes, then the pressure was dropped to 6 psig and the air supply was closed. All tests were performed in accordance with the requirements of ASME B&VP code Section VIII, 1989 Code Part UG-99 or UG-100 for the fabricated Carbon Steel.

5.1.7.4 Post-Operational Tests

The double penetrations and most weld seam channels which were installed on the inside of the liner in the Containment are continuously pressurized to provide a continuous, sensitive and accurate means of monitoring their status with respect to leakage. Certain liner welds are no longer continuously pressurized. Therefore, the leakage status of these welds is no longer continuously monitored. The integrity of these welds is verified by integrated leak rate testing.

No periodic structural integrity tests of the Containment are planned. Periodic peak pressure containment integrated leakage rate test (ILRTs) are performed in accordance with the Technical Specifications. Peak pressure tests are to be conducted as appropriate in the event major maintenance or major plant modifications are made. As a prerequisite to the ILRT, a detailed visual examination of the accessible interior and exterior surfaces of the containment structure and its components is required to uncover any evidence of deterioration which may affect the containment integrity. However, no degradation of structural integrity is expected. The Authority does not consider periodic structural integrity tests as warranted either separately or in conjunction with other tests.

The Containment Leakage Rate Testing Program details requirements for inspection of the accessible interior and exterior surfaces of the containment structure and its components. This periodic surveillance of the Containment and associated structures is visual and includes critical areas as well as a general examination of the accessible surfaces for deterioration. The inspection is also performed prior to any integrated leak test. The insulation attached to the steel liner is designed so that sections can be removed to facilitate inspection of the liner.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

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Provisions have been made for access to the upper external parts of the containment structure. These provisions consider the use of movable scaffolding while performing periodic inspection and testing during the service life of the facility.

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3. United States Atomic Energy Commission – Nuclear Reactors and Earthquakes, TID-7024, 1963.

4. Blume, J., N. Newark, L. Corning – Design of Multistory Reinforced Concrete Building for Earthquake Motions – Portland Cement Association.

5. Timoshenko, S., and S. Woinowsky-Kreiger, Theory of Plates and Shells, Second Edition, McGraw-Hill, 1954.

6. American Concrete Institute, Code for Reinforced Concrete Chimney Design, ACI-505.

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TABLE 5.1-1

FLOODED WEIGHTS – CONTAINMENT BUILDING

Item	Flooded/Equipment Weight, lb
Pressurizer –1	346,000
Steam Generators – 4	3,816,400
Reactor – 1	
(a) Vessel	868,000
(b) Internals	420,000
(c) Piping	1,000,000
Reactor Pumps – 4	824,000
Accumulator Tanks – 4	529,000
175 Ton Polar Crane – 1	650,000
Ventilation Fans – 5	656,000
Reactor Coolant Drain Tank – 1	20,000
Pressure Relief Tank – 1	129,000
Other Miscellaneous Equipment	100,000
TOTAL	9,358,400

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5.2 CONTAINMENT ISOLATION SYSTEM

5.2.1 Design Basis

Each system whose piping penetrates the Containment's leakage limiting boundary was designed to establish or maintain isolation of the Containment from the outside environment under the following postulated conditions:

a) Any accident for which isolation is required (severely faulted conditions) coincident with

b) An independent single failure or malfunction (expected faulted condition) occurring in any active system component within the isolated bounds.

Piping penetrating the Containment was designed for pressures at least equal to the containment design pressure. Containment isolation valves were provided, as necessary, in lines penetrating the Containment to assure that no unrestricted release of radioactivity can occur. Such releases might be due to rupture of a line within the Containment concurrent with a Loss-of-Coolant Accident, or due to rupture of a line outside the Containment, which connects to a source of radioactive fluid within the Containment.

In general, isolation of a line outside the Containment protects against releases due to rupture of the line inside concurrent with a Loss-of-Coolant Accident, and closes off a line which communicates with the containment atmosphere in the event of a Loss-of-Coolant Accident.

Isolation of a line inside the Containment prevents flow from the Reactor Coolant System or any other large source of radioactive fluid in the event that a piping rupture outside the Containment occurs. A piping rupture outside the Containment at the same time as a Loss-of-Coolant Accident is not considered credible, as the penetrating lines are of seismic Class I design up to and including the second isolation barrier and are assumed to be an extension of the Containment.

Normally lines located inside the Containment building that are required to function after an accident are located outside the missile barrier. An exception to this is a portion of the closed loop Component Cooling Water system which is located along the inside of the Crane Wall by the 31 Steam Generator. This is acceptable based on the "Modification of the General Design Criteria 4 requirements for protection against Dynamic effects of postulated pipe ruptures." This takes into account the Leak Before Break methodology, which relaxes the pipe rupture requirements for the Reactor Coolant Loop. The Component Cooling Water piping is also protected by concrete walls from the Pressurizer Surge line located on the other side of the Containment therefore the piping meets the intent of the original design criteria of being protected from credible missiles.

The isolation valve arrangement provides two barriers between the Reactor Coolant System or containment atmosphere, and the environment.

System design is such that failure of one valve to close will not prevent isolation.

The containment isolation valves were examined to assure that they are capable of withstanding the maximum potential seismic loads.

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To assure their adequacy in this respect:

a) Valves were located in such a manner as to reduce the accelerations on the valves. Valves suspended on piping spans were reviewed for adequacy for the loads to which the span would be subjected. Valves were mounted in the position recommended by the manufacturer.

b) Valve yokes were reviewed for adequacy, and strengthened as required for the response of the valve operator to seismic loads.

c) Where valves are required to operate during seismic loading, the operator forces were reviewed to assure that system function is preserved. Seismic forces on the operating parts of the valve are small compared to the other forces present.

d) Control wires and piping to the valve operators were designed and installed to assure that the flexure of the line does not endanger the control system. Appendages to the valve, such as position indicators and operators, were checked for structural adequacy.

e) The design of control systems for automatic containment isolation valves is such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves requires deliberate operator action.

Containment Isolation Valves Criteria

Isolation valves were provided as necessary for all fluid system lines penetrating the Containment to assure at least two barriers for redundancy against leakage of radioactive fluids to the environment in the event of a Loss-of-Coolant Accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving was designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safeguards systems.

Valves utilized in systems for containment isolation service were selected based on tight shutoff requirements, speed of operations, and materials suitable for service in a particular environment relative to temperature, pressure and radiation activity.

The criteria for level of reliability for control valves listed were based on satisfactory operation of the containment isolation valves for the operating life of the plant with the required leak tightness assured by testing and corrective maintenance as required.

The criteria of reliability for swing stop, check, and gate valves listed were based on documented material, quality assurance, compliance for inspection, welding qualification, seismic criteria, testing, and technical specifications for required leak tightness complying with valve manufacturer's standard practice.

Table 5.2.1 provides a summary of containment isolation valve type, actuator, and closure time established for systems penetrating containment.

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With respect to numbers and locations of isolation valves, the criteria applied were generally those outlined by the seven classes described in Section 5.2.2. Specific containment isolation valves are listed in FSAR Table 5.2-3.

5.2.2 System Design

The seven classes listed below are general categories into which line penetrating containment may be classified. The seal water referred to in the listing of categories is provided by the Isolation Valve Seal Water System described in Section 6.5. The following notes apply to these classifications:

1) The "not missile protected" designation refers to lines that are not protected throughout their length inside containment against missiles generated as the result of a Loss-of-Coolant-Accident. These lines, therefore, are not assumed invulnerable to rupture as a result of a Loss-of-Coolant Accident.

2) In order to qualify for containment isolation, valves inside the Containment must be located behind the missile barrier for protection against loss of function following an accident.

3) Manual isolation valves that are locked closed or otherwise closed and under administrative control during power operation qualify as automatic trip valves.

4) A check valve qualifies as an automatic trip valve in certain incoming lines not requiring seal water injection.

5) The double disk type of gate valve was used to isolate certain lines. When sealed by water or gas injection, this valve provides two barriers against leakage of radioactive liquids or containment atmosphere. In certain cases, a double disc valve was used in place of two valves in series having seal water or gas injection between them.

6) In lines isolated by globe valves in series (inboard and outboard) outside containment and provided with seal water injection, the following applies:

   a) On process lines ingressing containment (incoming lines) IVSWS will be required to wet the stem packings on both the inboard and outboard valve. IVSW wets the valve plug as well as the stem packing of the RCP seal water injection line containment isolation valves (CH-MOV-250A through D),

   b) On process lines egressing containment (outgoing lines) IVSWS will be required to wet only the stem packing on the inboard valve. One exception would be the Steam Generator Blowdown CIVs where both the inboard and outboard valves stem packings are wetted by IVSWS.

7) Excessive loss of seal water through an isolation valve that fails to close on signal is prevented by the high resistance of the seal water injection line. A water seal at the failed valve was assured by proper slope of the protected line, or a loop seal, or by additional valves on the side of the isolation valves away from the Containment.

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8) Lines penetrating containment were designed to the same seismic criteria as the containment vessel up to and including the second isolation barrier. These portions of the penetrating lines are therefore to be considered extensions of the containment.

A review of the Containment Isolation System (NUREG-0578) indicated that there were a number of valves, which automatically reset to the previous position upon reset of containment Phase A isolation. These valves were under operator control via operating procedures to be placed in the closed position prior to resetting of Phase A. Circuits for these valves have been modified to preclude automatic opening on reset. The modification to the valve circuits entailed the installation of pushbuttons that work in conjunction with the containment isolation reset switches so that each valve control circuit has to be reset or the valve will be inhibited from opening.

Class 1 (Outgoing Lines, Reactor Coolant System)

Outgoing lines connected to the Reactor Coolant System which are normally or intermittently open during reactor operation were provided with at least two automatic trip valves in series located outside the Containment. Automatic seal water injection was provided for line in this classification.

Class 2 (Outgoing Lines)

Outgoing lines not connected to the Reactor Coolant System which are normally or intermittently open during reactor operation, and not missile protected or which can otherwise communicate with the containment atmosphere following an accident, were provided, as a minimum, with two automatic trip valves in series outside containment. Automatic seal water injection was provided for lines in this classification with the exception of the reactor coolant pump seal water return line, which was provided with manual seal water injection. Most of these lines are not vital to plant operation following an accident.

Class 3 (Incoming Lines)

Incoming lines connected to open systems outside containment, and not missile protected or which can otherwise communicate with the containment atmosphere following an accident were provided with one of the following arrangements outside containment:

1) Two automatic trip valves in series, with automatic seal water injection. This arrangement was provided for lines, which are not necessary to plant operation after an accident.

2) Two manual isolation valves in series, with manual seal water injection. This arrangement was provided for lines, which remain in service for a time, or are used periodically, subsequent to an accident.

Incoming lines connected to closed systems outside containment, and not missile protected or which can otherwise communicate with the containment atmosphere following an accident were provided either with two isolation valves in series outside containment with seal water injection between them or, at a minimum with one check valve or normally closed isolation valve located either inside or outside containment.

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The closed piping system outside containment provides the necessary isolation redundancy for lines, which contain only one isolation valve.

Exceptions are the containment spray headers and the safety injection header associated with the boron injection tank, which was valved in accordance with safeguards requirements. The containment spray headers have locked-open double disk gate valves while the safety injection header has either single normally-open double disk gate valves or two normally open gate valves arranged in series.

Class 4 (Missile Protected)

Incoming and outgoing lines which penetrate the Containment and which are normally or intermittently open during reactor operation and are connected to closed systems inside the Containment and protected for missiles throughout their length were provided with at least one isolation valve located outside the Containment. Seal water injection was provided for certain lines in this classification.

Class 5 (Normally Closed Lines Penetrating the Containment)

Lines which penetrate the Containment and which can be opened to the containment atmosphere but which are normally closed during reactor operation were provided with two isolation valves in series or one isolation valve and one blind flange.

Class 6 (Special Service)

There are a number of special groups of penetrating lines and containment access openings. Some of these are discussed below.

Each ventilation purge duct penetration was provided with two tight-closing butterfly valves, which are closed during reactor power operation and are actuated to the closed position automatically upon a containment isolation or a containment high radiation signal.

One valve is located inside and one valve is located outside the Containment at each penetration. The space between valves is pressurized by air from the Penetration and Weld Channel Pressurization System, whenever they are closed.

The containment pressure relief line is similarly protected. However, since the line can be opened during reactor power operation, three tight closing butterfly valves in series are provided, one inside and two outside the Containment. These valves also are actuated to the closed position upon a containment isolation or containment high radiation signal. The two intravalve spaces are pressurized by air from the Penetration and Weld Channel Pressurization System whenever they are closed.

The equipment access closure is a bolted, gasketed closure, which is sealed during reactor operation. The personnel air locks consist of two doors in series with mechanical interlocks to assure that one door is closed at all times. Each air lock door and the equipment closure were provided with double gaskets to permit pressurization between the gaskets by the Penetration and Weld Channel Pressurization System, Section 6.6.

The fuel transfer tube penetration inside the Containment was designed to present a missile protected and pressurized double barrier between the containment atmosphere and the

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atmosphere outside the Containment. The penetration closure was treated in a manner similar to the equipment access hatch. A positive pressure is maintained between the double gaskets to complete the double barrier between the containment atmosphere and the inside of the fuel transfer tube. The interior of the fuel transfer tube is not pressurized. Seal water injection is not required for this penetration.

The following lines would be subjected to pressure in excess of the Isolation Valve Seal Water System design pressure (150 psig) in the event of an accident, due to operation of the recirculation pumps:

1) Residual heat removal loop return line

2) Bypass line from residual heat exchanger outlet to safety injection pumps suction

3) Residual heat removal loop sample line

4) Recirculation pump discharge sample line

5) Residual heat removal pump miniflow line

6) Residual heat removal loop outlet line

Lines 1, 2, and 6 are isolated by double disc gate valves, while line 3, 4 and 5 are each isolated by two valves in series. These valves can be sealed by nitrogen gas from the high pressure nitrogen supply of the Isolation Valve Seal Water System.

A self contained pressure regulator operates to maintain the nitrogen injection pressure slightly higher than the maximum expected line pressure. The nitrogen gas injection is manually initiated.

Lines which communicate with the containment atmosphere at all times (normally filled with air or vapor) include:

1) Steam jet air ejector return line to containment

2) Containment radiation monitor inlet and outlet lines.

In an accident condition, the space between the two containment isolation valves in each line is sealed by pressurizing with air from the Penetration and Weld Channel Pressurization System. The air is introduced into each space above the containment calculated peak accident response pressure through a separate line from the Penetration and Weld Channel Pressurization System. Parallel (redundant) fail open valves in each injection line open on the appropriate containment isolation signal to provide a reliable supply of pressurizing air. A flow limiting orifice in each injection line prevents excessive air consumption if one of these valves spuriously fails open, or if one of the containment isolation valves fails to respond to the “trip” signal.

Class 7 (Steam and Feedwater Lines)

These lines and the shell side of the steam generator are considered basically as an extension of the containment boundary and as such must not be damaged as a consequence of Reactor

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Coolant System damage. This required that the steam generator shell, feed and steam lines within the Containment be classified and designed for the Reactor Coolant System missile-protected category. The reverse is also true in that a steam line break is not to cause damage to the Reactor Coolant System.

5.2.2.1 Isolation Valves and Instrumentation Diagrams

Plant Drawings 9321-F-27473, -27203, -27353, -27453, -27503, -27513 Sh. 1, -27363, -27193, -27233, -27473, -20253, -27263, -70453, -20173, -20193, -27293 Sh. 1 & 2, -27223, -20353, -40223, -26533, -20363, -26533, and -27243 [Formerly Figures 5.2-1 through 5.2.28] show all valves in lines leading to the atmosphere or to closed systems on both sides of the containment barrier, valve actuation and preferential failure modes, the application of "trip" (containment isolation) signals, relative location of the valves with respect to missile barriers, and the boundaries of seismic Class I designed lines. Figure 5.2-29 defines the nomenclature and symbols used. Individual containment isolation valves are listed in Section 5.2 of the FSAR and Table 5.2-3.

5.2.2.2 Normally Closed Isolation Valves

Table 5.2-3 identifies those isolation valves which are either locked closed, or normally closed, (under administrative control) in normal position and relates to Figures 5.2-1 through 5.2-29.

5.2.2.3 Valve Parameters Tabulation

A summary of the fluid systems lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 5.2-3. Each valve is described as to type, operator, position indication and open or closed status during normal operation, shutdown and accident conditions. Information is also presented on valve preferential failure mode, automatic trip by the containment isolation signal, and the fluid carried by the line.

Containment isolation valves were provided with actuation and control equipment appropriate to the valve type. For example, air operated globe and diaphragm (Saunders Patent) valves are generally equipped with air diaphragm operators, with fail-safe operation provided by the control devices in the instrument air supply to the valve. Motor operated gate valves are capable of being supplied from reliable onsite emergency power as well as their normal power source. Manual and check valves, of course, do not require actuation or control systems.

The automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals is derived in conjunction with automatic safety injection actuation, and trips the majority of the automatic isolation valves. These are valves in the so-called "non-essential"** process lines penetrating the containment. This is defined as a "Phase A" isolation and the trip valves are designated by the letter "T" in the isolation diagrams, Figures 5.2-1 through 5.2-29. This signal also initiates automatic seal water injection (See Section 6.5). The second, or "Phase B," containment isolation signal is derived upon actuation of the Containment Spray System, and trips the automatic isolation valves in the so called "essential"* process lines penetrating the containment. These trip valves are designated by the letter "P" in the isolation diagrams.

NOTE:

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* "Essential" are those lines required to mitigate an accident, or which, if unavailable, could increase the magnitude of the event. Also, those lines which, if available, would be used in the short term (24 to 36 hours) to restore the plant to normal operation following an event which has resulted in containment isolation.

** "Non-Essential" are those lines which are not required to mitigate or limit an accident, which if required at all would be required for long-term recovery only, i.e., days or weeks following an accident.

A manual containment isolation signal can be generated from the Control Room. This signal performs the same functions as the automatically derived "T" signal, i.e., "Phase A" isolation and automatic seal water injection.

Non-automatic isolation valves, i.e., remote stop valves and manual valves, are used in lines which must remain in service, at least for a time, following an accident. These are closed manually if and when the lines are taken out of service.

Standard closing times available with commercial valve models are adequate for the sizes of containment isolation valves used. Valves equipped with air-diaphragm operators generally close in approximately two seconds. The typical closing time available for large motor operated gate valves is ten to thirty seconds. These general closure times are shown on Table 5.2-1. They are not used for determining valve stroke time limits. Specific design assumptions, closure times for design basis accidents, containment response analyses and resulting off-site dose calculations are contained in specific analyses.

The large butterfly valves used to isolate the containment ventilation purge ducts are each equipped with spring-assisted air pistons capable of closing the valve in two seconds. These valves fail to the closed position on loss of control signal. They also fail closed upon loss of instrument air through use of a local air reservoir as an energy source.

5.2.2.4 Valve Operability

All containment isolation valves, actuators and controls are located so as to be protected against missiles which could be generated as a result of a Loss-of-Coolant Accident. Only valves so protected are considered to qualify as containment isolation valves.

Only isolation valves located inside containment are subject to the high pressure, high temperature, steam laden atmosphere resulting from an accident. Operability of these valves in the accident environment is ensured by proper design, construction and installation, as reflected by the following considerations:

1) All components in the valve installation, including valve bodies, trim and moving parts, actuators, instrument air and control and power wiring, were constructed of materials sufficiently temperature resistant to be unaffected by the accident environment. Special attention was given to electrical insulation, air operator diaphragms and steam packing material.

2) In addition to normal pressures, the valves were designed to withstand maximum pressure differentials in the reverse direction imposed by the accident conditions. This criterion was particularly applicable to the butterfly type isolation valves used in the containment purge lines.

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5.2.2.5 Valve Position Indication and Monitoring

In general, all remote operated valves have visual position indication in the Control Room. Table 5.2-4 lists the containment isolation valves and the location of each valve's position indicator lights. Two different types of indicating lights are used: 1) red and green lights and 2) white and red monitoring lights. The red position indicating light is on when the valve is fully open and the green position indicating light is on when the valve is fully closed. At all other positions, both the red and green position indicating lights are on. The red monitor light is on when the valve is in its safeguard position and the white monitor light is on when the motive power is available to the valve. For those valves that are normally de-energized, the white monitor light indicates that power is available to the monitor indicating circuit.

Remote operated containment isolation valves, which are under remote manual control and do not receive a signal from the ESF actuation system, were provided with visual indication of position. An audible alarm feature was provided for remote operated safeguards valves under remote manual control for safeguards functions to denote their off-normal positions.

5.2.2.6 Local Leak Rate Testing of Containment Isolation Valves

Local leak rate testing of containment isolation valves is performed in accordance with Technical Specification 5.5.15. The Containment Leak Rate Program is in accordance with the guidance contained in Regulatory Guide 1.163, except as noted in the Technical Specification.

Amendment No. 195 to the Technical Specifications relocated information concerning containment isolation valves from the Technical Specifications to the FSAR.

Subsequent to implementation of Option B of 10 CFR 50, Appendix J, a third-party review of NYPA's (Option B) implementation program was completed. That review, confirmed by NYPA Nuclear Safety Evaluation, determined the scope of the Appendix J, "Type C," Local Leak Rate Test Program was greater than required by regulation. Specifically:

1) Leakage testing of SI-MOV-888A and B and SI-MOV-1835A and B is not required. The valves are not required to be LLR tested for purposes of compliance with Appendix J. These valves do not represent potential primary containment atmospheric leak paths following a single active failure. Since IVSWS nitrogen will only be applied to these valves in the event of a passive failure, but in no case sooner than 24 hours post-LOCA, there are no requirements for performance of leak rate tests.

2) Leak rate testing of the remaining valves penetrations served by the high-pressure nitrogen sub-system IVSWS is not required for compliance with Appendix J. Continued testing is required to ensure adequate nitrogen supply to the affected CIVs for the initial twenty-four hour period following a LOCA.

3) Local leak rate (Type C) testing of SI-1814 A, B, and C is not required for compliance with Appendix J. These valves do not represent potential primary containment atmospheric leak paths following a single active failure.

    Note: It is understood the body and packing of the SI-1814 A, B, and C valves are an extension of the containment pressure boundary and are exposed to containment pressure during Type A tests. If that pressure boundary is "broken," i.e., to facilitate

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calibration of the transmitters, then appropriate testing will be performed to confirm the integrity of the pressure boundary.

4) Local leak rate (LLR) testing of AC-741 is not required for compliance with Appendix J. This valve does not represent a potential primary containment atmospheric leak path following a single active failure.

5) Local leak rate testing of SI-MOV-885 A & B is not required for compliance with Appendix J. These valves do not represent potential primary containment atmospheric leak paths following a single active failure.

6) LLR testing of the SES CIVs is not required for compliance with Appendix J. Continued testing is required to assure the potential for in-leakage of service water into the containment following a postulated breach of the SWS integrity during the long-term post-LOCA recovery phase is within analyzed limits.

5.2.2.7 Containment Isolation During Refueling Outage

The Outage Equipment Hatch (OEH) may be used during an outage, when the permanent Equipment Hatch is removed.

The OEH will maintain containment closure during core alterations and during movement of irradiated fuel assemblies within containment building. The OEH may provide penetrations for temporary services and personnel access during an outage. The penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side and are subject to the requirements of ITS 3.9.3.

Electric penetrations used will be verified for leakage and integrity of the pressure boundary connection and not its function.

The roll-up door is an alternate device that is capable of rapid closure. It is effectively an airtight, but not pressure-resistant, door that when closed prevents direct communication between the containment atmosphere and the outside atmosphere.

Subsequent to a loss of RHR cooling as defined in ITS 3.9.4 and 3.9.5, the roll-up door provides rapid containment closure until either cooling is restored, or the main equipment hatch (or OEH) may be installed within four hours.

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TABLE 5.2-1

CONTAINMENT CONTROL ISOLATION VALVES

Valve Type	Actuator	Closure*
1500# Globe	Reverse Diaphragm	6 sec
1500# Globe	Motor	10 sec
1500# D.D.V.	Motor	10/30 sec
150# Gate	Motor	10 sec
150# Saunders	Direct Diaphragm	2 sec
150# Saunders	Reverse Diaphragm	2 sec
150# Globe	Solenoid	1.5 sec
150# D.D.V.	Motor	10 sec
150# Globe	Reverse Diaphragm	6 sec
150# Butterfly	Air & Spring	2 sec
600# Plug	Air Piston	4 sec
150# Butterfly	Air Piston	3.5 sec
300# Gate (RHR V 744)	Motor	30 sec
150# Gate (Aux Coolant V 769 and V 797)	Motor	30 sec

*Note: Closure times listed are general closure times for valve types shown. They do not form the basis of the safety analysis and are not used in determining valve stroke criteria.

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TABLE 5.2-2

NORMALLY CLOSED ISOLATION VALVES

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TABLE 5.2-3
(Sheet 1 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR. DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-1	PRESSURIZER RELIEF TANK TO GAS ANALYZER Penetration "V"	RC-AOV-549 RC-AOV-548	1	GLOBE GLOBE	AIR AIR	FC FC	T T	Yes Yes	G	H	C C	O O	C C	No No	Water (A)(4) Water (A)(4)	47	W W
5.2-1	PRESSURIZER RELIEF TANK N_2 SUPPLY Penetration "Y"	RC-518 RC-AOV-550	3	CHECK DIA.	- AIR	- FC	- T	No Yes	G	C	- O	- O	- C	No No	- -	43	G G
5.2-1	PRESSURIZER RELIEF TANK MAKE-UP Penetration "Y"	RC-AOV-552 RC-AOV-519	3	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	W	C	C(9) C(9)	C C	C C	No No	Water (A)(4) Water (A)(4)	47	W W
5.2-2	RESIDUAL HEAT REMOVAL RETURN Penetration "J"	AC-741 AC-MOV-744	6	CHECK DDV	- MOTOR	- FAI	- -	No Yes	W	H	- O(8)	- O	- O	No Yes	(5) Nitro(M)(32)	N/A 43 (15)	N/A N
5.2-2	RESID. HEAT REMOVAL LOOP TO SI PUMPS Penetration "QQ"	SI-MOV-888A SI-MOV-888B CS	6	DDV DDV -	MOTOR MOTOR -	FAI FAI -	- - -	Yes Yes	W	H	C(8) C(8)	LC(28) LC(28)	O O	Yes Yes	Nitro(M)(31) Nitro(M)(31)	N/A	N/A N/A
5.2-2	RESID. HEAT REMOVAL LOOP TO SAMPLING SYS. Penetration "QQ"	SP-AOV-958 SP-AOV-959 SP-990C	6	GLOBE GLOBE GLOBE	AIR AIR MANUAL	FC FC -	T T -	Yes Yes No	W	H	C C LC(8)	C(12) C(12) C(12)	C(12) C(12) C(12)	No(12) No(12) No(12)	Nitro(M)(32) Nitro(M)(32) Nitro(M)(32)	50	N N N
5.2-2	RESID. HEAT REMOVAL LOOP TO RHR PUMP MINIFLOW Penetration "QQ"	AC-MOV-1870 AC-MOV-743	6	GLOBE GATE	MOTOR MOTOR	FAI FAI	- -	Yes Yes	W	H	LTh(8) O(8)	LTh O	O O	Yes Yes	Nitro(M)(32) Nitro(M)(32)	50	N N
5.2-2	RESID. HEAT REMOVAL LOOP OUT Penetration "K"	AC-732	6	DDV	MANUAL	-	-	No	W	H	LC(8)	O	C	No	Nitro. (M)(32)	50 (15)	N
5.2-2	CONTAINMENT SUMP RECIRC. LINE Penetration "OO"	SI-MOV-885A SI-MOV-885B	5	DDV(23) DDV(23)	MOTOR MOTOR	FAI FAI	- -	Yes Yes	W	H	C(8) C(8)	C LC(28)	C(18) C(18)	No(18) No(18)	(5) (5)	N/A	N/A N/A
5.2-3	LETDOWN LINE Penetration "X"	CH-AOV-201 CH-AOV-202 CS	1	GLOBE GLOBE -	AIR AIR -	FC FC -	T T -	Yes Yes -	W	H	O O	C(9) C(9)	C C	No No	Water (A)(4) Water (A)(4)	47	W W
5.2-3	CHARGING LINE Penetration "R"	CH-MOV-205 CH-MOV-226 CH-227 CS	3	GATE GATE GLOBE -	MOTOR MOTOR MANUAL -	FAI FAI - -	- - - -	No No No -	W	C	O(8) O(8) LC(8)	C(9) C(9) C	C C C	No No No	Water(M)(4) Water(M)(4) Water(M)(4)	47	W W W
5.2-4	REACTOR COOLANT PUMP SEAL WATER SUPPLY LINES Penetration "Z"	CH-MOV-250A CH-MOV-250B CH-MOV-250C CH-MOV-250D CH-MOV-441 CH-MOV-442 CH-MOV-443 CH-MOV-444	3	GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE	MOTOR MOTOR MOTOR MOTOR MOTOR MOTOR MOTOR MOTOR	FAI FAI FAI FAI FAI FAI FAI FAI	- - - - - - - -	No No No No No No No No	W	C	O(8) O(8) O(8) O(8) O(8) O(8) O(8) O(8)	C(9) C(9) C(9) C(9) C(9) C(9) C(9) C(9)	C(11) C(11) C(11) C(11) C(11) C(11) C(11) C(11)	No(11) No(11) No(11) No(11) No(11) No(11) No(11) No(11)	Water(M)(4) Water M)(4) Water(M)(4) Water(M)(4) Water(M)(4) Water(M)(4) Water(M)(4) Water(M)(4)	47	W W W W W W W W

TABLE 5.2-3
(Sheet 2 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-4	REACTOR COOLANT PUMP SEAL WATER RETURN Penetration "R"	CH-MOV-222	2	DDV	MOTOR	FAI	P	Yes	W	C	O(11)	O	C(11)	No(11)	Water (M)(4)	47	W
5.2-5	REACTOR COOLANT SYSTEM SAMPLE LINES Penetration "W"	SP-AOV-956E SP-AOV-956F	1	GLOBE GLOBE	AIR AIR	FC FC	T T	Yes Yes	W	H	O O	C C	C C	No No	Water (A)(4) Water (A)(4)	47	W
5.2-5	FUEL TRANSFER TUBE Penetration "HH"	-	6	BLIND FLANGE (27)	-	-	-	-	W	H	-	-	-	-	(17)	-	-
5.2-6	CONTAINMENT SPRAY HEADERS Penetrations "GG" and "P"	SI-869A SI-869B SI-867A SI-867B SI-878A SI-878B	3	DDV DDV CHECK CHECK GLOBE GLOBE	MANUAL MANUAL - - MANUAL MANUAL	- - - - - -	- - - - - -	No No No No No No	W	C	LO(8) LO(8) - - LC(8) LC(8)	C C - - C C	O O - - C C	Yes Yes Yes Yes Yes Yes	Water(M)(4) Water(M)(4) - - - -	47 47 43 43 43 43	W W G G G G
5.2-7	SAFETY INJECTION HEADERS Penetrations "Q" and "NN"	SI-MOV-1835A SI-MOV-1835B SI-MOV-851A SI-MOV-850C SI-MOV-850A	3	DDV DDV DDV GATE GATE	MOTOR MOTOR MOTOR MOTOR MOTOR	FAI FAI FAI FAI FAI	S S - - -	Yes Yes Yes Yes Yes	W	H	O(8) O(8) O(8) LO(8) LO(8)	C C C C C	O(19) O(19) O(19) O(19) O(19)	Yes(33) Yes(33) Yes(19) Yes(19) Yes(19)	Nitro.(M)(33) Nitro.(M)(33) Water (M)(4) Water (M)(4) Water (M)(4)	N/A N/A 47 47 47	N/A N/A W W W
5.2-7	SAFETY INJECTION TEST Penetration "Y"	SI-859A SI-859C	5	GLOBE GLOBE	MANUAL MANUAL	- -	- -	No No	W	C	LC(8) LC(8)	C C	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-8	ACCUMULATOR NITROGEN SUPPLY Penetration "RR"	NNE-1610 NNE-AOV-863	5	CHECK GLOBE	- AIR	- FC	- T	No Yes	G	C	- C(9)	- C	- C	No No	- -	43 43	G G
5.2-8	ACCUMULATOR SAMPLE Penetration "RR"	SP-AOV-956G SP-AOV-956H	2	GLOBE GLOBE	AIR AIR	FC FC	T T	Yes Yes	W	C	C(12) C(12)	C C	C(12) C(12)	No(12) No(12)	Water (A)(4) Water (A)(4)	47 47	W W
5.2-9	PRIMARY SYSTEM VENT AND NITROGEN SUPPLY Penetration "V"	WD-AOV-1786 WD-AOV-1787 WD-AOV-1610 WD-1616	2 3	DIA DIA DIA. CHECK	AIR AIR AIR -	FC FC FC -	T T T -	Yes Yes Yes No	G	H	O(9) O(9) O -	C C O -	C C C -	No No No -	Water (A)(4) Water (A)(4) - -	47 47 43 43	W W G G
5.2-9	REACTOR COOLANT DRAIN TK. TO GAS ANALYZER Penetration "V"	WD-AOV-1788 WD-AOV-1789	2	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	G	H	C(13) C(13)	O C(13)	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-9	RCDT PUMP DISCHARGE Penetration "Z"	WD-AOV-1702 WD-AOV-1705	2	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	W	C	C(9) C(9)	O O	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-10	REACTOR COOLANT PUMP COOLING WATER IN Penetration "N"	AC-MOV-797 AC-MOV-769	3	GATE GATE	MOTOR MOTOR	FAI FAI	P P	Yes Yes	W	C	O(11) O(11)	C(11) C(11)	C(11) C(11)	No(11) No(11)	Water (M)(4) Water (M)(4)	47 47	W W

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TABLE 5.2-3
(Sheet 3 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-10	REACTOR COOLANT PUMP COOLING WATER OUT 6" Penetration "O"	AC-MOV-784 AC-MOV-786	2	GATE GATE	MOTOR MOTOR	FAI FAI	P P	Yes Yes	W	C	O(11) O(11)	C(11) C(11)	C(11) C(11)	No(11) No(11)	Water (M)(4) Water (M)(4)	47 47	W W
5.2-10	REACTOR COOLANT PUMP COOLING WATER OUT 3" Penetration "O"	AC-FCV-625 AC-MOV-789	2	GATE GATE	MOTOR MOTOR	FAI FAI	P P	Yes Yes	W	C	O(11) O(11)	C(11) C(11)	C(11) C(11)	No(11) No(11)	Water (M)(4) Water (M)(4)	47 47	W W
5.2-11	RESIDUAL HEAT EXCHANGERS COOLING WATER IN Penetrations "KK" and "VV"	AC-751A AC-751B CS	4	CHECK CHECK -	- - -	- - -	- - -	No No -	W	C	- -	- -	- -	Yes Yes	- -	N/A N/A	N/A N/A
5.2-11	RESIDUAL HEAT EXCHANGERS COOLING WATER RETURN Penetrations "JJ" and "UU"	AC-MOV-822A AC-MOV-822B CS	4	GATE GATE -	MOTOR MOTOR -	FAI FAI -	S S -	Yes Yes -	W	C	C(8) C(8)	O O	O O	Yes Yes	- -	N/A N/A	N/A N/A
5.2-12	RECIRC. PUMP COOLING WATER SUPPLY Penetration "LL"	AC-752F AC-753F CS	4	GLOBE GLOBE -	MANUAL MANUAL -	- - -	- - -	No No -	W	C	O(8) O(8)	O O	O O	Yes Yes	- -	N/A N/A	N/A N/A
5.2-12	RECIRC. PUMP COOLING WATER RETURN Penetration "LL"	AC-752J AC-753J CS	4	GLOBE GLOBE -	MANUAL MANUAL -	- - -	- - -	No No -	W	C	O(8) O(8)	O O	O O	Yes Yes	-	N/A N/A	N/A N/A
5.2-13	EXCESS LETDOWN HEAT EXCHANGER COOLING WATER IN Penetration "U"	AC-AOV-791 AC-AOV-798	4	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	W	C	C(9) C(9)	O O	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-13	EXCESS LETDOWN HEAT EXCHANGER COOLING WATER OUT Penetration "R"	AC-AOV-796 AC-AOV-793	4	GLOBE DIA.	AIR AIR	FC FC	T T	Yes Yes	W	C	C(9) C(9)	O O	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-13	CONTAINMENT SUMP PUMP DISCHARGE Penetration "Y"	WD-AOV-1728 WD-AOV-1723	2	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	W	C	O O	O O	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-14	CONTAINMENT AIR SAMPLE IN- RAD. MONITORING SYSTEM Penetration "RR"	VS-PCV-1234 VS-PCV-1235	6	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	G	C	O O	O O	C(20) C(20)	No(20) No(20)	Air (A)(7) Air (A)(7)	43 43	G G
5.2-14	CONTAINMENT AIR SAMPLE OUT – RAD. MONITORING SYSTEM Penetration "RR"	VS-PCV-1236 VS-PCV-1237	6	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	G	C	O O	O O	C(20) C(20)	No(20) No(20)	Air (A)(7) Air (A)(7)	43 43	G G
5.2-14	AIR EJECTOR DISCHARGE TO CONTAINMENT Penetration "R"	CA-PCV-1229 CA-PCV-1230	6	GLOBE GLOBE	AIR AIR	FC FC	T T	Yes Yes	G	C	C C	C C	C C	No No	Air (A)(7) Air (A)(7)	43 43	G G

TABLE 5.2-3
(Sheet 4 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-15	MAIN STEAM HEADERS Penetrations "A,B,C and D"	CS	7	-	-	-	-	-	G	H	-	-	(22)	Yes(22)	-	-	-
	MAIN STEAM TO AUX. FW PUMP TURBINE	CS	-	-	-	-	-	-	G		-	-	-	Yes	-	-	-
5.2-15	MAIN FEEDWATER HEADERS Penetrations "E,F,G and H"	CS	7	-	-	-	-	-	W	H	-	-	-	Yes	-	-	-
	AUXILIARY FW TURBINE DRIVEN	CS	-	-	-	-	-	-	W		-	-	-	Yes	-	-	-
	AUXILIARY FW MOTOR DRIVEN	CS	-	-	-	-	-	-	W		-	-	-	Yes	-	-	-
5.2-15	STEAM GENERATOR BLOWDOWN Penetrations "AA,BB, CC, and DD"	BD-PCV-1214		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1215	2	GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1216		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1217		GLOBE	AIR	FC	T	Yes	W	H	O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1214A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1215A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1216A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1217A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
5.2-15	STEAM GENERATOR BLOWDOWN SAMPLE Four Lines @ Penetration "W"	BD-PCV-1223	2	GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1224		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1225		GLOBE	AIR	FC	T	Yes	W	H	O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1226		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1223A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1224A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1225A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1226A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
5.2-16	VENTILATION SYSTEM COOLING WATER IN Penetrations "La,Lb,Lc, Ld and Le"	SWN-41-1		BV	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-41-2		BV	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-41-3		BV	MANUAL	-	-	No	W	C	O(8)	O	O	Yes	(6)	47	W
		SWN-41-4		BV	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-41-5		BV	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-43-1		GATE	MANUAL	-	-	No			C(8)	C	C	No	(6)	47	W
		SWN-43-2		GATE	MANUAL	-	-	No			C(8)	C	C	No	(6)	47	W
		SWN-43-3	4	GATE	MANUAL	-	-	No			C(8)	C	C	No	(6)	47	W
		SWN-43-4		GATE	MANUAL	-	-	No			C(8)	C	C	No	(6)	47	W
		SWN-43-5		GATE	MANUAL	-	-	No			C(8)	C	C	No	(6)	47	W
		SWN-42-1		RV	-	-	-	No			-	-	-	-	(6)	47	W
		SWN-42-2		RV	-	-	-	No			-	-	-	-	(6)	47	W
		SWN-42-3		RV	-	-	-	No			-	-	-	-	(6)	47	W
		SWN-42-4		RV	-	-	-	No			-	-	-	-	(6)	47	W
		SWN-42-5		RV	-	-	-	No			-	-	-	-	(6)	47	W
		CS		-	-	-	-	-			-	-	-	-			

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TABLE 5.2-3
(Sheet 5 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-16	VENTILATION SYSTEM COOLING WATER OUT Penetrations "Ma, Mb, Mc, Md, Me, and SS"	SWN-44-1		BV	MANUAL	-	-	No			LTh(8)	LTh	LTh	Yes	(6)	47	W
		SWN-44-2		BV	MANUAL	-	-	No	W	C	LTh(8)	LTh	LTh	Yes	(6)	47	W
		SWN-44-3		BV	MANUAL	-	-	No			LTh(8)	LTh	LTh	Yes	(6)	47	W
		SWN-44-4		BV	MANUAL	-	-	No			LTh(8)	LTh	LTh	Yes	(6)	47	W
		SWN-44-5		BV	MANUAL	-	-	No			LTh(8)	LTh	LTh	Yes	(6)	47	W
		SWN-51-1		GATE	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-51-2	4	GATE	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-51-3		GATE	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-51-4		GATE	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-51-5		GATE	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-71-1		GLOBE	MANUAL	-	-	No			Th(8)	Th	Th	Yes	(6)	47	W
		SWN-71-2		GLOBE	MANUAL	-	-	No			Th(8)	Th	Th	Yes	(6)	47	W
		SWN-71-3		GLOBE	MANUAL	-	-	No			Th(8)	Th	Th	Yes	(6)	47	W
		SWN-71-4		GLOBE	MANUAL	-	-	No			Th(8)	Th	Th	Yes	(6)	47	W
		SWN-71-5		GLOBE	MANUAL	-	-	No			Th(8)	Th	Th	Yes	(6)	47	W
		CS		-	-	-	-	-									
5.2-17	STATION AIR Penetration "Y"	SA-24-1	3	DIA.	MANUAL	-	-	No	G	C	LC(8)	LC(8)	LC	No	Water (A)(4)	47	W
		SA-24-2		DIA.	MANUAL	-	-	No			LC(8)	LC(8)	LC	No	Water (A)(4)	47	W
5.2-17	WELD CHANNEL PENETRATION PRESSURE SYSTEM Penetration "Y"	PS-PCV-1111-1	4	BALL	MANUAL	-	-	No			LO(8)	LO	LO	Yes	(17)	N/A	N/A
		PS-PCV-1111-2		BALL	MANUAL	-	-	No	G	C	LO(8)	LO	LO	Yes	(17)	N/A	N/A
		CS (inside)		-	-	-	-	-			-	-	-	-			
		CS (outside)		-	-	-	-	-			-	-	-	-			
5.2-19	PURGE SUPPLY DUCT VENTILATION Penetration "EE"	VS-FCV-1170	6	BV	AIR	FC	T (2)	Yes	G	C	C	O	C	No	Air (A)(7)	43	G
		VS-FCV-1171		BV	AIR	FC	T (2)	Yes			C	O	C	No	Air (A)(7)	43	G
5.2-19	PURGE EXHAUST DUCT VENTILATION Penetration "FF"	VS-FCV-1172	6	BV	AIR	FC	T (2)	Yes	G	C	C	O	C	No	Air (A)(7)	43	G
		VS-FCV-1173		BV	AIR	FC	T (2)	Yes			C	O	C	No	Air (A)(7)	43	G
5.2-19	CONTAINMENT PRESSURE RELIEF VENTILATION Penetration "PP"	VS-PCV-1190		BV	AIR	FC	T (2)	Yes	G	C	C(14)	C	C	No	Air (A)(7)	43	G
		VS-PCV-1191	6	BV	AIR	FC	T (2)	Yes			C(14)	C	C	No	Air (A)(7)	43	G
		VS-PCV-1192		BV	AIR	FC	T (2)	Yes			C(14)	C	C	No	Air (A)(7)	43	G
5.2-20	RECIRCULATION PUMP DISCHARGE SAMPLE LINE Penetration "TT"	SP-MOV-990A	6	GATE	MOTOR	FAI	-	No	W	C	LC(8)	C	LC (12)	No	Nitro(M)(32)	50	N
		SP-MOV-990B		GATE	MOTOR	FAI	-	No			LC(8)	C	LC (12)	No	Nitro(M)(32)	50	N
5.2-20	PRESSURIZER STEAM SAMPLE LINE Penetration "W"	SP-AOV-956A	1	GLOBE	AIR	FC	T	Yes	W	H	C	C	C	No	Water (A)(4)	47	W
		SP-AOV-956B		GLOBE	AIR	FC	T	Yes			C	C	C	No	Water (A)(4)	47	W
5.2-20	PRESSURIZER LIQUID SAMPLE LINE Penetration "W"	SP-AOV-956C	1	GLOBE	AIR	FC	T	Yes	W	H	C	C	C	No	Water (A)(4)	47	W
		SP-AOV-956D		GLOBE	AIR	FC	T	Yes			C	C	C	No	Water (A)(4)	47	W

TABLE 5.2-3
(Sheet 6 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-21	CONTAINMENT PRESSURE INSTRUMENTATION LINE Penetration "RR"	SI-1814A CS	6	GLOBE -	MANUAL -	- -	- -	No -	G	C	LO(8)	O	O	Yes	- -	(34)	N/A
5.2-21	CONTAINMENT PRESSURE INSTRUMENTATION LINE Penetration "LL"	SI-1814B CS	6	GLOBE -	MANUAL -	- -	- -	No -	G	C	LO(8)	O	O	Yes	- -	(34)	N/A
5.2-21	CONTAINMENT PRESSURE INSTRUMENTATION LINE Penetration "O"	SI-1814C CS	6	GLOBE -	MANUAL -	- -	- -	No -	G	C	LO(8)	O	O	Yes	- -	(34)	N/A
5.2-22	POST ACCIDENT CONTAINMENT SAMPLING SUPPLY AND RETURN LINES Penetrations "R, TT, LL, Z, and O"	SP-SOV-506 SP-SOV-507 SP-SOV-508 SP-SOV-512 SP-SOV-513 SP-SOV-511 SP-SOV-516 SP-SOV-509 SP-SOV-510 SP-SOV-514 SP-SOV-515	5	GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE	SOL. SOL. SOL. SOL. SOL. SOL. SOL. SOL. SOL. SOL. SOL.	FC FC FC FC FC FC FC FC FC FC FC	T(10) T(10) T(10) T(10) T(10) T(10) T(10) T(10) T(10) T(10) T(10)	Yes Yes Yes Yes Yes Yes Yes Yes Yes Yes Yes	G	C	C(8) C(8) C(8) C(8) C(8) C(8) C(8) C(8) C(8) C(8) C(8)	C C C C C C C C C C C	C(12) C(12) C(12) C(12) C(12) C(12) C(12) C(12) C(12) C(12) C(12)	Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12)	Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7)	43 43 43 43 43 43 43 43 43 43 43	G G G G G G G G G G G
-	CONTAINMENT SUMP RECIRCULATION (SPARE) Penetration "O_1 O_1"	CS (3)	6	-	-	-	-	-	-	C	-	-	-	-	(17)	-	-
5.2-25	INSTRUMENT AIR – P. A. VENTING SYSTEM SUPPLY Penetration "Y"	IA-39 IA-PCV-1228	6	CHECK DIA.	- AIR	- FC	- T	No Yes	G	C	- O	- O	- C (24)	No No (24)	- -	43 43	G G
5.2-25	POST ACCIDENT VENTING SYSTEM EXHAUST LINE Penetration "LL"	PS-7 PS-8 PS-9 PS-10	5	DIA. DIA. DIA. DIA.	MANUAL MANUAL MANUAL MANUAL	- - - -	- - - -	No No No No	G	C	LC(8) LC(8) LC(8) LC(8)	LC LC LC LC	C (24) C (24) C (24) C (24)	No (24) No (24) No (24) No (24)	(17) (17) (17) (17)	43 43 43 43	G G G G
5.2-26	CONTAINMENT LEAK TEST INSTRUMENT SENSOR LINE Three lines @ Penetration "RR"	CS (3)	-	-	-	-	-	-	G	C	-	-	-	No	(17)	-	-
5.2-26	CONTAINMENT LEAK TEST AIR LINE Penetrations "XX and YY"	CS (3)	6	-	-	-	-	-	G	C	-	(30)	-	No	(17)	-	-
5.2-27	EQUIPMENT ACCESS	CB-7 CB-8 CB-5 CB-6	6	BALL BALL CHECK(26) CHECK(26)	MANUAL MANUAL - -	- - - -	- - - -	(29)	G	C	C(8) C(8) - -	C C - -	C C – -	No No	(17) (17) - -	43 43 43 43	G G G G
5.2-27A	PERSONNEL AIR LOCK	CB-3 CB-4 CB-1 CB-2	6	BALL BALL CHECK(26) CHECK(26)	MANUAL MANUAL - -	- - - -	- - - -	(29)	G	C	C(8) C(8) - -	C C - -	C C - -	No No	(17) (17) - -	43 43 43 43	G G G G
5.2-28	DEMIN. WTR. INTO CONTAINMENT Penetration "Y"	DW-AOV-1 DW-AOV-2	6	PLUG PLUG	AIR AIR	FC FC	T T	Yes Yes	W	C	C C	C(21) C(21)	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W

TABLE 5.2-3
(Sheet 7 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

ABBREVIATIONS:

A	Automatic
AMB	Ambient
BV	Butterfly Valve
C	Cold
CS	Closed System
COL	Check Off List
DDV	Double Disc Gate Valve
DIA	Diaphragm Valve
FAI	Fail As Is
FC	Fail Closed
FO	Fail Open
G	Gas
H	Hot
LC	Locked Closed
LO	Locked Open
LTh	Locked Throttled
M	Manual
N	Nitrogen
POP	Plant Operating Procedures
P	Containment Isolation Signal Phase B
T	Containment Isolation Signal Phase A
Th	Throttled
RV	Relief Valve
S	Safety Injection Signal (Opens valves on SI signal)
SOP	System Operating Procedures
SOL	Solenoid Operated Valves
W	Water

TABLE 5.2-3
(Sheet 8 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

DEFINITIONS:

NORMAL POSITION: Defined as RCS operation above 200^{0}F to Full Power. Valve positions as defined by POP's, SOP's, and COL's

SHUTDOWN POSITION: Defined as RCS 200^{0}F and below, not in refueling and not at reduced inventory. Valve positions as defined by POP's and SOP's.

POST ACCIDENT POSITION: Defined as SI with Phase A and B isolation. Note: valve position may differ based on the accident in progress (i.e. phase B may not be required).

POST ACCIDENT USAGE: Defined as Design Basis Accident valve usage based on position during long term recirculation, assuming no failures. Note: valve position may differ based on the accident in progress, equipment failure, and recommendations during the recovery phase.

NOTES:

1.Penetration class is described in subsection 5.2.2.	**16**. Test Fluid "G" signifying Gas indicates either air or nitrogen as test medium.
2.Also tripped closed by high radiation in containment.	**17**. Seal air via WCCPP, continuously pressurized.
3.Penetration sealed at both ends.	**18**. May be opened Post Accident if normal path from recirc. pumps not available.
4. Sealed by Isolation Valve Seal Water System.	**19**. Valves may be closed Post Accident if not in service.
5. "Sealed" by Residual Heat Removal System or recirculation sump fluid. Not a "seal system" as defined in 10 CFR 50, Appendix J.	**20**. May be opened Post Accident when the containment pressure is below 5 psig.
6. "Sealed" by Service Water System fluid. Not a "seal system" as defined in 10 CFR 50, Appendix J. LLR testing is not required for Appendix J compliance but is required to limit in-leakage to the containment given a postulated breech of SWS integrity during the long-term recovery phase.	**21**. Valves may be opened for maintenance.
7. Sealed by Weld Channel and Containment Penetration Pressurization System.	**22**. Valves outside containment in these lines will automatically isolate for steamline break or Hi-Hi containment pressure.
8.Non-Automatic Containment Isolation Valves open continuously or intermittently for plant operation (under administrative control).	**23**. DDV modified due to press. locking to function as a std.gate valve. 885A upstream disc drilled with 3/16"dia. hole, 885B bonnet connection bypasses downstream disc.
9.Valves may be operated as required to support plant operation.	**24**. Valves may be opened intermittently during Post Accident venting.
10.These series valves have non-redundant phase A automatic signals and therefore are treated as non-automatic containment isolation valves.	**25**. Penetrations identified as H (hot) indicates designed with expansion bellows or expansion coil, C (cold) indicates designed without an expansion bellows or expansion coil.
11.Isolated when Reactor Coolant Pumps are stopped.	**26**. Spring-loaded check valves (pressure relieving).
12.Valves opened intermittently to take samples.	**27**. Flange is double gasketed type, located in refueling canal.
13.Valve opened periodically by the Gas Analyzer.	**28**. Necessary to LC & de-energize if AC-730 & 731 are de-energized open.
14.Opened intermittently for pressure relief.	**29**. Control Rm. Annunciator "Personnel hatches not shut" alarm indication provided.
15. Testable only at Cold Shutdown.	

NOTES:

30. A Seismic Class I QA CAT M temporary fiber optic penetration flange (TFP) may be installed in cold shutdown / refueling conditions to satisfy containment isolation function for refueling operations.	
31. Once opened to facilitate high head or hot leg recirculation, valves would	

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TABLE 5.2-3
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CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

remain open unless closed to isolate a postulated passive failure during the long-term recovery phase. LLR testing is not required.	
32. LLR testing performed to verify adequacy of on-site nitrogen inventory. LLR is not required for Appendix J compliance.	
33. Valves remain open to facilitate high head or hot leg recirculation unless closed to isolate a postulated passive failure during the long-term recovery phase. LLR testing is not required.	
34. LLRT is not required. Valve / penetration is open during Type A ILR test.	

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TABLE 5.2-4

CONTAINMENT ISOLATION
VALVE POSITION INDICATION

Valve Number	Control Board Panel Location Red & Green Indicating Lights	Control Board Panel Location (Two-is-True) Monitor Lights
AC-MOV-743	SNF	SB1F
AC-MOV-744	SGF & SB1F	SB1F
CH-AOV-201	SFF & SNF	SNF
CH-AOV-202	SFF & SNF	SNF
CH-MOV-222	SFF & SNF	SNF
RC-AOV-519	SAF	SNF
RC-AOV-548	SNF	SNF
RC-AOV-549	SNF	SNF
RC-AOV-552	SAF	SNF
AC-FCV-625	SGF & SNF	SNF
AC-MOV-769	SGF & SNF	SNF
AC-MOV-784	SGF & SNF	SNF
AC-MOV-786	SGF & SNF	SNF
AC-MOV-789	SGF & SNF	SNF
AC-AOV-791	SGF & SNF	SNF
AC-AOV-793	SGF & SNF	SNF
AC-AOV-796	SGF	SNF
AC-MOV-797	SGF & SNF	SNF
AC-AOV-798	SGF & SNF	SNF
SP-AOV-956A	Sampling System Panel *	SNF
SP-AOV-956B	Sampling System Panel *	SNF
SP-AOV-956C	Sampling System Panel *	SNF
SP-AOV-956D	Sampling System Panel *	SNF
SP-AOV-956E	Sampling System Panel *	SNF
SP-AOV-956F	Sampling System Panel *	SNF
SP-AOV-956G	Sampling System Panel *	SNF
SP-AOV-956H	Sampling System Panel *	SNF
SP-AOV-959	Sampling System Panel *	SNF
VS-FCV-1170	SLF and Fan Room Ctr. Cab.*	SNF
VS-FCV-1171	SLF and Fan Room Ctr. Cab *	SNF
VS-FCV-1172	SLF and Fan Room Ctr. Cab *	SNF
VS-FCV-1173	SLF and Fan Room Ctr. Cab *	SNF
VS-PCV-1190	SLF and Fan Room Ctr. Cab *	SNF
VS-PCV-1191	SLF and Fan Room Ctr. Cab *	SNF
VS-PCV-1192	SLF and Fan Room Ctr. Cab *	SNF
BD-PCV-1214	SCF	SNF
DW-AOV-1	SKF	SNF
DW-AOV-2	SKF	SB1F
RC-AOV-550	SKF	SB1F

* Not located in control room

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TABLE 5.2-4
(Cont.)

CONTAINMENT ISOLATION
VALVE POSITION INDICATION

Valve Number	Control Board Panel Location Red & Green Indicating Lights	Control Board Panel Location (Two-is-True) Monitor Lights
SP-AOV-958	CR Isolation Valve Panel JK1	-
WD-AOV-1610	SKF	SNF
SI-MOV-850A	SB2F&WD Extension *	-
SI-MOV-850C	SB2F&WD Extension *	-
BD-PCV-1214A	SCF	SNF
BD-PCV-1215	SCF	SNF
BD-PCV-1215A	SCF	SNF
BD-PCV-1216	SCF	SNF
BD-PCV-1216A	SCF	SNF
BD-PCV-1217	SCF	SNF
BD-PCV-1217A	SCF	SNF
BD-PCV-1223	Sampling System Panel*	SNF
BD-PCV-1223A	Sampling System Panel *	SNF
BD-PCV-1224	Sampling System Panel *	SNF
BD-PCV-1224A	Sampling System Panel *	SNF
BD-PCV-1225	Sampling System Panel *	SNF
BD-PCV-1225A	Sampling System Panel *	SNF
BD-PCV-1226	Sampling System Panel *	SNF
BD-PCV-1226A	Sampling System Panel *	SNF
IA-PCV-1228	SNF	SNF
CA-PCV-1229	SNF	SNF
CA-PCV-1230	SNF	SNF
VS-PCV-1234	SNF	SNF
VS-PCV-1235	SNF	SNF
VS-PCV-1236	SNF	SNF
VS-PCV-1237	SNF	SNF
WD-AOV-1702	Waste Disposal System Panel*	SNF
WD-AOV-1705	Waste Disposal System Panel *	SNF
WD-AOV-1723	Waste Disposal System Panel *	SNF
WD-AOV-1728	Waste Disposal System Panel *	SNF
WD-AOV-1786	Waste Disposal System Panel *	SNF
WD-AOV-1787	Waste Disposal System Panel *	SNF
WD-AOV-1788	SNF	SNF
WD-AOV-1789	SNF	SNF
AC-MOV-822A	SGF & SB1F	SB1F
AC-MOV-822B	SGF & SB1F	SB1F
SI-MOV-851A	SB2F	SB2F
NNE-AOV-863	SMF	SNF

*Not located in control room

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TABLE 5.2-4
(Cont.)

CONTAINMENT ISOLATION
VALVE POSITION INDICATION

Valve Number	Control Board Panel Location Red & Green Indicating Lights	Control Board Panel Location (Two-is-True) Monitor Lights
SI-MOV-885A	SB1F	SB1F
SI-MOV-885B	SB1F	SB1F
SI-MOV-888A	SB1F	SB1F
SI-MOV-888B	SB1F	SB1F
SI-MOV-1835A	SB2F	SB2F
SI-MOV-1835B	SB2F	SB2F
AC-MOV-1870	SNF	SB1F
CH-MOV-205	WD Extension *	-
CH-MOV-226	WD Extension *	-
CH-MOV-250A	WD Extension *	-
CH-MOV-250B	WD Extension *	-
CH-MOV-250C	WD Extension *	-
CH-MOV-250D	WD Extension *	-
CH-MOV-441	WD Extension *	-
CH-MOV-442	WD Extension *	-
CH-MOV-443	WD Extension *	-
CH-MOV-444	WD Extension *	-
SP-MOV-990A	WD Extension *	-
SP-MOV-990B	WD Extension *	-
SP-SOV-506	CR Isolation Valve Panel JK1	-
SP-SOV-507	CR Isolation Valve Panel JK1	-
SP-SOV-508	CR Isolation Valve Panel JK1	-
SP-SOV-509	CR Isolation Valve Panel JK1	-
SP-SOV-510	CR Isolation Valve Panel JK1	-
SP-SOV-511	CR Isolation Valve Panel JK1	-
SP-SOV-512	CR Isolation Valve Panel JK1	-
SP-SOV-513	CR Isolation Valve Panel JK1	-
SP-SOV-514	CR Isolation Valve Panel JK1	-
SP-SOV-515	CR Isolation Valve Panel JK1	-
SP-SOV-516	CR Isolation Valve Panel JK1	-

*Not located in control room

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5.3 CONTAINMENT VENTILATION SYSTEM

5.3.1 Design Basis

5.3.1.1 Performance Objectives

The Containment Ventilation System was designed to accomplish the following:

a) Remove the normal heat loss from all equipment and piping in the Reactor Containment during plant operation and maintain a normal ambient temperature of 130°F or less

b) Provide sufficient air circulation and filtering throughout all containment areas to permit safe and continuous access to the reactor containment within two hours after reactor shutdown, assuming defects exist in 1% of the fuel rods.

c) Provide for positive circulation of air across the refueling water surface to assure personnel access and safety during shutdown

d) Provide a minimum containment ambient temperature of 50°F during reactor shutdown

e) Provide for purging of the containment vessel to the plant vent for dispersion to the environment. The rate of release is controlled by IP3 RECS / ODCM, such that automatic termination of release occurs prior to impacting 10 CFR 20 limits.

f) Provide for depressurization of the containment vessel following an accident. The post-accident design and operating criteria are detailed in Chapter 6

g) Provide ventilation to remove radiogas when steam generator primary man rays are removed

h) Provide means for measurement of flow in main plant ventilation exhaust duct

In order to accomplish these objectives the following systems were provided:

a) Containment Air Recirculation Cooling and Filtration System

b) Control Rod Drive Mechanism Cooling System

c) Reactor Compartment Cooling System

d) Containment Purge System

e) Containment Auxiliary Charcoal Filter System

f) Containment Post-Accident Charcoal Filter System (Described in Section 6.4)

g) Steam Heating System

h) Steam Generator Maintenance Exhaust System.

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5.3.1.2 Design Characteristics – Sizing

The design characteristics of the equipment required in the Containment for cooling, filtration and heating to handle the normal thermal and air cleaning loads during normal plant operation are presented in Table 5.3-1. In certain cases where engineered safeguards functions also are served by the equipment, component sizing was determined from the heavier duty specifications associated with the Design Basis Accident (DBA), detailed further in Chapter 6.

The fan motors match the power requirements of the fans, which require a maximum power input of 219 horsepower under accident operation. The fan cooler heat removal rate, as a function of the containment pressure, is presented in Section 14.3.6 covering the Containment Integrity Evaluation. For example, this rate at 271°F and 47 psi containment temperature and pressure is 49.0 x 10^6 Btu/hr per air handling unit. As noted in the Containment Integrity Evaluation, the ability of the Containment Air Recirculation Cooling and Filtration System to function properly in the accident environment was demonstrated by the computer code "HECO." The code determines the plate fin coil heat removal rate when operating in a saturated steam-air mixture.

5.3.2 System Design

5.3.2.1 Piping and Instrumentation Diagram

The containment ventilation, purging and recirculation cooling and filtration systems flow diagram is shown in Plant Drawing 9321-F-40223 [Formerly Figure 6.4-2]. The containment ventilation systems and main plant vent were designed as seismic Class I structures.

5.3.2.2 Containment Recirculation Ventilation

Air recirculation cooling and filtering during normal operation is accomplished using all five air handling units discharged to a common headered ductwork distribution system to assure adequate flow of filtered and cooled air throughout the Containment. The cooling coils in each air handling unit transfer up to 2.3 x 10^6 Btu/hr to the Service Water System during normal plant operation and 49.0 x10^6 Btu/hr/FCU in the event of an accident when supplied with 1400 gpm cooling water at 95°F inlet temperature.

Each air handling unit consists of the following equipment arranged so that during normal operation air flows through the unit in the following sequence: cooling coils, centrifugal fan with direct-drive motor, and distribution header.

The fans and motors of these units are equipped with vibration sensors to detect abnormal operating conditions in the early stages of the disturbance. In the event of an accident, the flow path will be diverted automatically by air operated dampers through a compartment containing moisture separators, HEPA filters and charcoal filters. It will then flow through the cooling coils and centrifugal fan and into the distribution header. The normal air flow rate per air handling unit is approximately 70,000 cfm and the post-accident flow rate will be approximately 34,000 cfm, with a 8,000 cfm through the filtration section. Section 6.4.2 provides additional information on the operation of this system.

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The recirculating ductwork located in the annulus of the Containment Building was provided with spring loaded relief dampers designed to open inward when the external pressure on the ductwork reaches 2 psig. This is discussed in Section 6.4

The Control Rod Drive Cooling System supplements the main containment recirculation system. The Control Rod Drive Cooling System consists of fans and ductwork to circulate air through the control drive mechanism shroud and discharge it to the main containment volume. Four 1/3 capacity direct driven axial flow fans are used.

5.3.2.3 Containment Purge System

The Containment Purge System includes provisions for both supply and exhaust air. The purge system is maintained isolated whenever the plant is above the cold shutdown condition. The supply system includes roughing filters, heating coils, fan, supply penetration with two butterfly valves for bubble tight shutoff, and a purge supply distribution header inside containment. The exhaust system includes exhaust penetration with two butterfly valves identical to those above, exhaust ductwork, filter bank with roughing, HEPA and charcoal filters, fans and exhaust vent. Provision was made to measure isokinetic flows at the radiation monitor using pitot tubes. The purge system flow rate is 28,000 cfm; however, the isolation valves will be shut prior to going above cold shutdown and will remain closed during normal operation. The quick closing purge isolation valves are capable of closing within two seconds of receipt of the accident signal. The weld channel and penetration pressurization system pressurizes the space between the purge valves and therefore serves as a continuous on-line monitoring system for valve leakage.

During power operation, containment integrity is maintained with no release from the containment ventilation system to the atmosphere. Prior to purging the Containment, air particulate and gas monitor indications of the closed containment activity levels are used as a guide to making routine releases from the Containment. During power operation, the containment air particulate and gas monitor indications help determine the desirability of using either one or both of two auxiliary particulate and charcoal filter units installed in the Containment primarily for pre-access cleanup.

When the containment purging for access following reactor shutdown is in progress, releases from the plant vent are continuously monitored for radiogas and particulates and sampled for iodine and tritium. A wide range plant vent gas monitor (Section 11.2.3.1) provides continuous indication of noble gas releases passing through the plant vent to the atmosphere.

5.3.2.4 Isolation Valves

The purge supply and exhaust ducts butterfly valves, both inside and outside the containment, are closed during power operation. The spaces between the closed valves are pressurized with air by the Penetration and Weld Channel Pressurization System. The valves were designed for rapid automatic closing by the containment isolation signal (derived from any automatic safety injection signal), or upon a signal of high activity level within the Containment in the event of a radioactivity release when the purge line is open.

5.3.2.5 Containment Pressure Relief Line

The normal pressure changes in the Containment during reactor power operation will be handled by the containment pressure relief line. This line is equipped with three quick-closing butterfly type isolation valves, one inside and two outside the Containment. The valves will be

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automatically actuated to the closed position by the containment isolation signal, or by a containment high radioactivity signal. The two intra-valve spaces are pressurized with air by the Penetration and Weld Channel Pressurization System when the valves are closed. The pressure relief line discharges through roughing, HEPA, and charcoal filters to the plant vent. While the valves are fully capable of closing from a 60° open position during accident conditions, mechanical stops prevent the valves from opening more than 40° (90° = full open).

5.3.2.6 Steam Generator Maintenance Exhaust System

Steam generator maintenance ventilation is accomplished by use of two 3000 cfm fans driven by 5 hp motors. These fans connect to 14" diameter exhaust ducts, which allow maintenance on the steam generators when the manways are removed. The fans exhaust into the containment purge exhaust duct.

5.3.2.7 Pressurizer Relief Tank Venting

During shutdown conditions, the potential exists for radioactive gases to be vented from the Pressurizer Relief Tank. These gases are therefore routed to the containment purge exhaust duct where their radioactive content can be monitored (see Section 11.2).

The system uses a jet eductor, using station air to vent the tank. The system is shut down during normal operation.

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TABLE 5.3-1

PRINCIPAL COMPONENT DATA SUMMARY

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Containment Recirculation			
Demister	5	8,000 cfm	0
Cooling Coils – Normal	5	2.3 x 10^6 Btu/hr	5
Cooling Coils – DBA	5	49.0 x 10^6 Btu/hr	0
HEPA Filters	5	8,000 cfm	0
Fans	5	70,000* cfm	5
Fans Pressure – Normal	-	6.3 in H20	
Fan Motors (440 V, 3 phase)	5	225 hp	5
DBA Charcoal Filters	5**	8,000 cfm	0
Temperature Switches	30**		
Control Rod Drive Mechanism Cooling			
Fans, Standard Conditions	4	15,000 cfm	3
Fan Pressure	-	5-1/2 in H20	
Fan Motors	4	25 hp	3
Reactor Compartment Cooling			
Part of CB Recirculation System	-	12,000 cfm	
Refueling Canal Air Sweep			
Part of CB Recirculation System	-	17,5000 cfm	

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TABLE 5.3-1
(Cont.)

PRINCIPAL COMPONENT DATA SUMMARY

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Containment Ventilation/Purge Supply			
Fans, Standard Conditions	1	40,000 cfm	Optional
Fan Pressure	-	Approximately 2.75 in H_20	
Fan Motors	1	40 hp	
Pre-heat Coils	1 Set		Optional
Air Filters, Roughing	1	40,000 cfm	1
Exhaust			
Fans,* Standard Conditions	2	70,000 cfm**	Optional
Fan Pressure	-	12.5 in H_20	
Fan Motors	2	150 hp	
Plenums	2	40,000 cfm	
HEPA Filters	1 Bank	40,000 cfm	Optional
Roughing Filters	1 Bank	40,000 cfm	Optional
Charcoal Filters	1 Bank	40,000 cfm	Optional
Containment Auxiliary Charcoal Filters			
Fans, Standard Conditions	2	8,000 cfm	Optional
Fan Pressure	-	4.75 in H_20	
Fan Motors	2	10 hp	
Filters; Roughing, HEPA and Charcoal Filters	2	8,000 cfm	Optional

*Note: The two exhaust fans are used interchangeably or as backup for:
1. Ventilation of Primary Auxiliary Building (70,000 cfm)
2. Containment Building Purge System (40,000 cfm)

**Note: Normal System Flow for Containment Building Purge Exhaust is 28,000 cfm.

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TABLE 5.3-1
(Cont.)

PRINCIPAL COMPONENT DATA SUMMARY

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Steam Heating			
Heaters, 25 psig steam	2	400,000 Btu/hr each	Optional
Steam Generator Maintenance Exhaust System			
Centrifugal Fan	2	3000 cfm	2
Fan Pressure	-	4.5 in H_20	-
Fan Motors	2	5 hp	2
Containment Building Pressure Relief			
Fan, Standard Conditions	1	1500 cfm	Optional
Fan Pressure	-	3.5 in H_20	
Fan Motor	-	5 hp	
Filters; Roughing, HEPA and Charcoal Filters	1	1500 cfm	Optional

Note: The operating configuration for the Containment Building Pressure Relief system involves limiting the three containment isolation valves to a minimum position of 40° open. This causes a decrease in system flow.

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5.4 POST ACCIDENT CONTAINMENT VENTING SYSTEM [Historical Information]

*NOTE: The Post Accident Containment Venting System was retired 04/02/03.

5.4.1 Function

Following a Design Basis Accident, hydrogen gas may be generated inside the Containment by reactions such as zirconium metal with water, corrosion of materials of construction, and radiolysis of aqueous solution in the sump and core. The Post-Accident Containment Venting System permits controlled venting of the containment atmosphere to maintain the hydrogen concentration at a safe level.

5.4.2 Design Basis

The Post-Accident Containment Venting System was designed to limit the hydrogen concentration in the Containment to three percent by volume.

5.4.3 System Description

The Post-Accident Containment Venting System consists of a single line penetrating the Containment, which will be used alternately to supply hydrogen free air to the Containment or exhaust hydrogen bearing gases from the Containment. These exhaust gases are directed through roughing, HEPA, and charcoal filters to the plant vent. The major components of the Post-Accident Containment Venting System are as follows:

5.4.3.1 Containment Air Supply

Hydrogen free air is admitted to the Containment through the single supply/exhaust line. The supply air is provided from the Instrument Air System, which is in use during normal plant operation. The nominal flow rate from either of the two instrument air compressors is 200 scfm. If the Instrument Air System is not available, the Station Air System with a nominal capacity of 600 scfm is available as a backup.

5.4.3.2 Containment Air Exhaust

From inside the Containment, hydrogen bearing gases are exhausted through the single supply/exhaust line. Outside containment is a normally closed, manually operated containment isolation valve followed by a branch connection with an additional manual isolation valve in each branch. Between these adjacent isolation valves, there is a connection through parallel redundant manual valves to the containment penetration pressurization system. Thus, the exhaust line at the containment penetration can be manually sealed with air following a LOCA. Following this valve in each line are:

a) Local pressure indicator for containment pressure

b) Remote manual air operated stop valve

c) Self-contained pressure control valve

d) Manual flow control valve.

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The two branches go into a common header and the single line to the plant vent passes through the following:

a) Flow indicator/integrator which provides remote readout of both instantaneous flow rate and integrated flow

b) Local temperature indicator

c) Roughing filter

d) HEPA filter

e) Carbon filter.

The latter three components are shielded by 16 inches of normal concrete.

All the components inside the Containment Building, the penetrations, and the piping and valves to the second isolation valves are seismic Class I design. The remaining components in the exhaust line (except the plant vent, which is seismic Class I) are seismic Class III.

All active components of the system, namely, valves, instruments, controls and associated electrical supplies, are redundant. All passive components, namely, piping and the three filters, are not and need not be redundant.

The system was designed to obtain a flow of 200 scfm with containment pressure at 1.9 psig. For this flow rate, the residence time in the charcoal filters is approximately 0.4 seconds.

5.4.4 Operation

The flow rate and the duration of venting required to maintain the hydrogen concentration at or below 3 percent of the containment volume are determined from the containment hydrogen concentration measurements and the hydrogen generation rate (Section 6.8). The containment pressure necessary to obtain the required vent flow is then determined. Using one of the two instrument air compressors, hydrogen free air is pumped into the Containment until the required containment pressure is reached. The air supply is then stopped and the supply/exhaust line is isolated by valves outside the Containment.

The addition of air to pressurize the Containment will also dilute the hydrogen; therefore, the Containment will remain isolated until analysis of samples indicates that the concentration is again approaching 3 percent by volume. Venting is then started by opening either the primary or bypass exhaust line, and adjusting the hand controlled throttle valve to obtain the required flow.

This process of containment pressurization followed by venting is repeated as may be necessary to maintain the hydrogen concentration at or below 3 percent by volume.

Post Loss-of-Coolant Accident purging provides a backup method to the hydrogen recombiners in the Containment (Section 6.8) for controlling the potential hydrogen accumulation in the Containment. The analysis of offsite doses using purging to control hydrogen was based on the Westinghouse model for hydrogen production and accumulation discussed in Section 14.3.7.

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The purging system requires a differential pressure between the Containment and the outside atmosphere in order to permit purging. If required, the containment is pressurized with diluent air when the hydrogen reaches 3 percent by volume after the Loss-of-Coolant Accident. The hydrogen concentration is reduced by this pressurization. Purging is thus delayed until the hydrogen concentration in the Containment has once again built up to 3 percent by volume. The 3 percent hydrogen level was selected as the point of starting the purge because of the following factors:

1) This level allows a sufficient margin of safety below the lower flammability limit of 4.1 percent

2) It provides a sufficient margin so that purging could be delayed a few days if so desired. With neither containment purging nor recombiner operation, the hydrogen generation rate is sufficiently low so that 45 days are required for the hydrogen concentration in the Containment to build up from the 3 percent level to the 4.1 percent level

3) The optimum starting time for the purge, from the standpoint of minimizing the doses, is the latest time.

The hydrogen concentration in the Containment will slowly decrease from 3 percent as purging continues. The required purge rate is based on the hydrogen production rate at the time of purge initiation.

The dose analysis is based on the activity released from the Containment after the time of the postulated Loss-of-Coolant Accident until all the activity in the Containment is either removed or released. The infinite-time thyroid, beta and gamma doses as a function of distance from the plant due to activity release from containment leakage following the postulated Loss-of-Coolant Accident are computed using the core activity release model described in Section 14.3.5. Then the analysis is repeated except that the doses are based on activity released from both containment leakage and purging. The offsite doses due to purging are then determined by subtraction of the doses due to containment leakage and purging. The parameters used to compute the activity releases from containment leakage and from purging are given in Tables 5.4-1 and 5.4-2.

The dose models discussed in Reference (1) and the atmospheric dispersion factor given in Section 14.3.5 are used in determining doses following the Loss-of-Coolant-Accident. In the evaluation of doses from activity released to the atmosphere after 720 hours (30 days), the annual average dispersion factor at the site boundary of 2.6×10^{-5} sec/m^3 was used. The thyroid beta and gamma doses at the site boundary due to containment purging to control hydrogen are given in Table 5.4-4.

The thyroid, beta and gamma doses due to containment purging to control hydrogen were also determined using the assumptions outlined in Reference (2) and the dos models given in Reference (1). The parameters used to determine the activity release resulting from leakage and purging using the model in Reference (2) are given in Tables 5.4-1 and 5.4-3, and the boundary doses are listed in Table 5.4-4.

<u>References</u>

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1) "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant-Accident for Pressurized Water Reactors," Safety Guides for Water Cooled Nuclear Power Plants, Safety Guide No. 4, Division of Reactor Standards, U.S. Atomic Energy Commission, November 1971.

2) "The Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident," Safety Guides for Water Cooled Nuclear Power Plants, Safety Guide No. 7, Division of Reactor Standards, U.S. Atomic Energy Commission.

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TABLE 5.4-1

PARAMETERS USED TO DETERMINE CONTAINMENT LEAKAGE ACTIVITY RELEASE

Plant Power	3216 MWt	
Containment free volume	2.61×10^6 ft^3	
Unsprayed containment volume	5.22×10^5 ft^3	
Mixing rate between sprayed and unsprayed containment volume	2.4×10^4 cfm (filtered) 1.26×10^5 cfm (unfiltered)	
Spray removal coefficient for elemental iodine	32 hr^{-1} until DF=100	
Containment design leak rate	0.1% per day (0-24 hours)	
	0.045% per (>24 hours)	
Containment Air Recirculation and Cooling System filter efficiencies	Westinghouse Model	AEC Model
Elemental Iodine	90%	90%
Methyl Iodine	70%	5%
Particulate Iodine	90%	90%

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TABLE 5.4-2

PARAMETERS USED TO DETERMINE
HYDROGEN PURGING ACTIVITY RELEASE – W MODEL

1. Westinghouse Basis H_2 Generation*

2. Pressurize Containment to 2.14 psig, if required, when H_2 reaches 3.0 percent by volume (day 36)

3. Purge at 15.3 scfm continuously once H_2 reaches 3.0 percent by volume again (day 50)

4. Containment Air Recirculation and Cooling System filter efficiencies

Elemental Iodine	90%
Organic Iodine	70%
Particulate Iodine	90%

*NOTE: Discussed in Section 14.3.7.

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TABLE 5.4-3

PARAMETERS USED TO DETERMINE HYDROGEN PURGING ACTIVITY RELEASE – AEC MODEL

1. AEC Basis H_2 Generation*

2. Pressurize Containment to 2.14 psig, if required, when H_2 reaches 4.0 percent by volume (day 23)

3. Purge at 21.8 scfm continuously once H_2 reaches 4.0 percent by volume again (day 33)

4. Containment Air Recirculation and Cooling System filter efficiencies

Elemental Iodine	90%
Organic Iodine	5%
Particulate Iodine	90%

*Safety Guide No.7

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TABLE 5.4-4

DOSES FROM CONTAINMENT PURGING TO CONTROL HYDROGEN

Dose at Site Boundary (Rem)

	Westinghouse Model	AEC Model
Thyroid	4.5×10^{-1}	5.1
Beta	1.4	1.6
Gamma	1.7×10^{-2}	5.0×10^{-2}

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5.5 CONTAINMENT PARAMETERS

The description of the instrumentation system included in the Indian Point 3 design for remote monitoring of post-accident conditions within the primary containment is presented in Appendix 6F. Non-nuclear process instrumentation of the containment is described in Section 7.5.

Containment Building Pressure

The containment pressure is transmitted to the main control board for post accident monitoring. Six transmitters, two in each of three safety channels, are installed outside the containment to prevent potential missile damage. The pressure is indicated on the main control board; the range is –5 psig to 75 psig.

In addition, monitoring of the containment building pressure during and following an accident is effected by two Safety Category I redundant systems. Pressure signals are obtained at the pipe penetration area and brought to transmitters outside containment. These same signals are transmitted to the two-recorders at the control room recorder cabinet. Continuous monitoring of containment pressure is possible in the –5 to 200 psig range. Power requirements for the two systems are met from vital instrument buses. The installation of cable and conduit is consistent with separation criteria, as outlined in Section 8.4.

Containment Building Water Level Monitoring

There are three sumps in the Containment Building: Reactor Pit Sump, Recirculation Sump and Containment Sump. Associated with the Recirculation Sump and Containment Sump, there are two redundant, separately channeled and powered level measurement loops. Associated with the Reactor Pit is a level sensor, alarmed on the Control Room Supervisory Panel. These provide continuous level and alarm indication in the Control Room. Additionally, a water level transmitter installed at the top of Containment Sump will provide a Containment Sump overflow alarm indication in the Control Room.

Containment Building Hydrogen Concentration

Hydrogen concentration indication is provided by a measuring system which consists of the following: redundant analyzers and continuously recording two (2) single pen recorders. The recorders are located in the control room and the analyzers are located in the pipe penetration area of the fan house. Samples are drawn from containment recirculation fans via the retired post-accident sample system and returned to the general area of the containment building.

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APPENDIX 5-A

[Historical Information]

WESTINGHOUSE NUCLEAR ENERGY SYSTEMS
UNITED ENGINEERS AND CONSTRUCTORS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

CONTAINMENT DESIGN REPORT

September 1970

B. Scott
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1.0 INTRODUCTON

1.1.0 PURPOSE & SCOPE OF REPORT

The object of this report is to illustrate the design adequacy of the containment structure for the Indian Point Nuclear Generating Unit No. 3. To this end, this documentary report describes the design of the structure, as well as the construction procedures, to demonstrate fulfillment of the design criteria.

The following sections of this report enumerate the basic criteria that were used, the analyses that were developed to satisfy these criteria, the various loading combinations under normal and postulated accident conditions (including seismic effects), and the construction and testing procedures that were employed to ultimately construct the containment structure at the site.

1.2.0 FUNCTION OF CONTAINMENT STRUCTURE

The containment structure completely encloses the entire reactor and reactor coolant system and ensures that essentially no leakage of radioactive materials to the environment would result even if gross failure of the reactor coolant system were to occur. The structure will provide biological shielding for normal and accident situations.

The containment structure is designed to safely withstand several conditions of loading and their credible combinations. The limiting extreme conditions are:

a) Occurrence of a gross failure of the reactor coolant system which creates a high pressure and temperature condition within the containment.

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b) Coincident failure of the reactor coolant system with an earthquake or wind.

The design pressure and temperature of the containment will be, as a minimum, equal to the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to and including the hypothetical severance of a reactor coolant pipe. Energy contribution from the steam system is included in the calculation of the containment pressure transient due to reverse heat transfer through the steam generator tubes. The supports for the reactor coolant system will be designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that the coincidental rupture of the steam system is not considered credible. In addition, the design pressure will not be exceeded during any subsequent long term pressure transient determined by the combined effects of heat sources such as residual heat and limited metal-water reactions, structural heat sinks and the operation of the engineered safeguards; the latter utilizing only the emergency electric power supply.

The design pressure and temperature on the containment structure will be those created by the hypothetical loss-of-coolant accident. The reactor coolant system will contain approximately 512,000 lbs. of coolant at a weighted average enthalpy of 595 Btu/lb. for a total energy of 304,000,000 Btu. In a hypothetical accident, this water is released through a double-ended break in the largest reactor coolant pipe, causing a rapid pressure rise in the containment. The reactor coolant pipe used in the accident will be the 29-in. ID section because rupture of the 31-in. ID section requires that the blowdown go through both the 29-in. and the 27-1/2-in. ID pipes and would, therefore, result in a less severe transient.

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Additional energy release was considered from the following sources:

a) Stored heat in the reactor core.

b) Stored heat in the reactor vessel piping and other reactor coolant system components.

c) Residual heat production.

d) Limited metal-water reaction energy and resulting hydrogen-oxygen reaction energy.

The following loadings will be considered in the design of the containment in addition to the pressure and temperature conditions described above:

a) Structure dead load.

b) Live loads.

c) Equipment loads.

d) Internal test pressure.

e) Earthquake.

f) Wind. (Tornado)

The containment structure is inherently safe with regard to common hazards such as fire, flood and electrical storm. The thick concrete walls are invulnerable to fire and only an insignificant amount of combustible material, such as lubricating oil in pump and motor bearings, is present in the containment.

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Internal structures consist of equipment supports, shielding, reactor cavity and canal for fuel transfer, and miscellaneous concrete and steel for floors and stairs. All internal structures are supported on the containment mat.

A 3-ft. thick concrete ring wall serving as a partial radiation shield surrounds the reactor coolant system components and supports the polar-type reactor containment crane. A 2-ft. thick reinforced concrete floor covers the reactor coolant system with removable gratings in the floor provided for crane access to the reactor coolant pumps. The four steam generators, pressurizer and various piping penetrate the floor. Spiral stairs provide access to the areas below the floor.

The refueling canal connects the reactor cavity with the fuel transport tube to the spent fuel pool. The floor and walls of the canal are concrete. The floor is 5 ft. thick. The concrete walls and floor are lined with ¼ inch thick stainless steel plate. The linings provide a leak- proof membrane that is resistant to abrasion and damage during fuel handling operation.

1.3.0 CONTAINMENT DESCRIPTION

The reactor containment structure is a reinforced concrete vertical right cylinder with a flat base and a hemispherical dome. A welded steel liner with a minimum thickness of ¼ inch is attached to the inside face of the concrete shell to insure a high degree of leak-tightness. The design objective of the containment structure is to contain all radioactive material which might be released from the core following a loss-of-coolant accident. The structure serves as both a biological shield and a pressure container.

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The structure consists of side walls measuring 148-feet from the liner on the base to the springline of the dome, and has an inside diameter of 135-feet. The side walls of the cylinder and the dome is 4-ft. 6-in. and 3-ft. 6-in. thick respectively. The inside radius of the dome is equal to the inside radius of the cylinder so that the discontinuity at the springline due to the change in thickness is on the outer surface. The flat concrete base mat is 9-ft. thick with the bottom liner plate located on top of this mat. The bottom liner plate is covered with 3-ft. structural slab of concrete which serves to carry internal equipment loads and forms the floor of the containment. The internal pressure within the containment is self-contained in that the vector sum of the pressure forces is zero; therefore, there is no need for mechanical anchorage between the bottom mat and underlying rock. The base is supported directly on rock.

The basic structural elements considered in the design of the containment structure is the base slab, side walls and dome acting as one structure under all possible loading conditions. The liner is anchored to the concrete shell by means of stud anchors so that it forms an integral part of the entire composite structure under all membrane loadings. The reinforcing in the structure has an elastic response to all primary loads with limited maximum strains to insure the integrity of the steel liner. The lower 20 feet of the cylindrical liner is insulated to avoid excess deformation of the liner due to restricted radial growth when subjected to a rise in temperature.

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2.0 CONTAINMENT STRUCTURAL DESIGN BASIS

2.1.0 DESIGN LOAD CRITERIA

The following loads were considered to act upon the containment structure creating stresses within the component parts.

2.1.1 DEAD LOADS

Dead load consists of the weight of the concrete wall, dome, liner, insulation, base slab and the internal concrete. Weights used for dead load calculations were as follows:

a) Reinforced Concrete : 150 lb/ft^3

b) Steel Lining : 490 lb/ft^3 using nominal cross-sectional area

d) Insulation : 6 lb/ft^3 including stainless steel jacket.

2.1.2 OPERATING LIVE LOADS

Operating live loads consist of the weight of major components of equipment in the containment. Equipment loads were those specified on the drawings supplied by the manufacturers of the various pieces of equipment.

All major pieces of equipment are supported on the 3'-0" base slab or on the interior concrete, which in turn bears directly on the 9'-0" mat.

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Item		Flooded Operating Weight, lb.
Pressurizer	-1	346,000
Steam Generators	-4	3,746,000
Reactor	-1	
a) Vessel		868,000
b) Internals		420,000
RCS Piping		1,000,000
Reactor Pumps	-4	824,000
Accumulator Tanks	-4	529,000
175 Ton Polar Crane	-1	650,000
Ventilation Fans	-4	656,000
Reactor Coolant Drain Tank	-1	20,000
Pressure Relief Tank	-1	100,000
Other Misc. Equipment		100,000
		9,259,000

Other Uniform Live Loads

@ El. 68' – 10 ft. strip adjacent to crane wall = 600 psf

Remaining strip = 100 psf

@ El. 95' - 0" – Concrete Slab = 500 psf

Grating areas = 100 psf

2.1.3 SNOW LOADS

Snow and ice loads have been applied uniformly to the top surface of the dome at an estimated value of 20 pounds per square foot of horizontal projection of the dome. This loading represents approximately 2-ft. of snow, which was considered to be a conservative amount since the slope of the dome tends to cause much of the snow to slide off.

2.1.4 CONSTRUCTION LOADS

A construction live load of 50 pounds per square foot has been used on the dome, but was not considered to act concurrently with the snow load.

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A load equivalent to the weight of wet concrete, placed in sections during construction of the concrete dome, was used for the design of the stiffened dome liner plate. During the pressure test of containment, the concrete will crack and thereby relieve the effects of shrinkage and creep.

2.1.5 WIND LOADS

The American Standards Association "American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures" (A58.1-1955) designates the site as being in a 25 psf zone. In this code, for height zones between 100 and 499 feet, the recommended wind pressure on a flat surface is 40 psf. Correcting for the shape of the containment by using a shape factor of 0.60, the recommended pressure becomes 26 psf. The State Building and Construction Code for the State of New York stipulates a wind pressure up to 30 psf on a flat surface for heights up to 300 feet. For design, a uniform 30 psf basic wind load has been used from ground level up.

The tornado loads considered in design are as follows:

a. cyclonic wind velocity = 300 mph
b. translational wind velocity = 60 mph
c. differential pressure drop = 3 psi in 3 seconds
d. missile - 4" x 12" x 12' plank at 300 mph horizontal, or at 90 mph, vertical.
e. Missile – 4000 lb passenger car, not exceeding 25 feet above the ground, at 50 mph horizontal or at 17 mph vertical (25 ft^2 contact area).

2.1.6 OPERATING TEMPERATURE LOADS

The operating temperature assumed in the design of the containment structure in 120°F, with a -5°F outside winter temperature. Thermal loads induced in the containment as a result of operating temperature effects are composed of a) the steady state temperature gradient through the wall as shown in Figure 2.1 for Winter conditions for both the insulated and

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uninsulated portions of the liner and b) the effective load induced in the concrete shell as the concrete acts to restrain the steel liner when the mean temperature of the concrete differs from that of the liner.

2.1.7 CREEP AND SHRINKAGE LOADS

The containment structure has been investigated for end of life creep and shrinkage factor as follows:

(a) $k_{(creep)} = 0.22 x 10^{-6} in / in / psi$

(b) $k_{(shrinkage)} = 70 x 10^{-6} in / in$

The maximum stress induced in the steel reinforcement by this maximum condition is less than 4000 psi. Since the limiting case for design is accident pressure load which effectively cracks the concrete and places the reinforcement into membrane tension creep and shrinkage induced stress are not a limiting factor in design.

2.1.8 SEISMIC LOADS

The ground acceleration for the Operational Basis Earthquake, "OBE" was determined to be 0.1g applied horizontally and 0.05 applied vertically. These values were resolved as conservative numbers based upon recommendation from Dr. Lynch, Director of Seismic Observatory, Fordham University. A dynamic analysis has been used to arrive at equivalent design loads. Additionally, a Design Basis Earthquake, "DBE" acceleration of 0.15 horizontally and 0.10 vertically has been used to analyze for the no-loss of function.

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A damping factor of 2 percent was assumed for the reinforced concrete containment structure for the OBE and 5 percent for the DBE. The response spectra used were based on the Spectrum curves presented in Figures A.1-1 and A.1-2 of Appendix A1 normalized to 0.15g zero period ground acceleration as required.

2.1.9 ACCIDENT PRESSURE LOADS

The design basis accident pressure load is shown in Figure 5.1-8 of the FSAR as a function of time. This design value is at least 5 percent in excess of maximum calculated containment pressure.

2.1.10 ACCIDENT TEMPERATURE LOADS

The design basis accident containment temperature assumed in the design of the containment is also shown in Figure 5.1-8 of the FSAR as a function of time. This containment temperature induces loads in the concrete shell as the concrete acts to restrain liner thermal expansion. This thermal load effect on the liner is combined with pressure load effects to develop design basis accident design load requirements as a function of time. Accident temperature induced thermal gradients through the wall are not a factor in concrete shell design since the accident temperature effect penetrates approximately 10 percent of the containment wall thickness during the significant overpressure phase of the accident and the cracking of the concrete shell due to containment pressurization acts to relieve secondary stresses induced by thermal gradient effect.

2.1.11 LOADS AT PENETRATIONS

The effect of growth of the liner due to accident conditions has been considered in the design of penetrations and sleeves together with the effects of lateral loads due to thermal expansion of pipes, seismic motion, pipe break loads and pressure loads. In addition, stress concentration effects on large penetrations have been considered.

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2.1.12 MISSILE LOADS

Potential external missiles (tornado; turbine failure) have been considered in design as described in Appendix 14A to the Report.

2.1.13 TEST PRESSURE LOADS

Internal pressure will be applied to test the structural integrity of the vessel up to 115 per cent of the design pressure of 47 psi. For this structure the test pressure will be 54 psig.

2.1.14 COMBINED FACTORED LOAD EQUATIONS

The design was based upon limiting load factors which were used as the ratio by which loads were multiplied for design purposes to assure that the loading formation behavior of the structure was one of elastic, tolerable strain behavior. The load factor approach was used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. In the case of the containment structure, therefore, this approach places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads. The loads utilized to determine the required limiting capacity of any structural element on the containment structure are computed as follows:

a) $C = 1.OD \pm 0.05D + 1.5P + 1.0 (T + TL)$ (2.1.1)

b) $C = 1.OD \pm 0.05D + 1.25P + 1.0 (T' + TL') + 1.25E$ (2.1.2)

c) $C = 1.OD \pm 0.05D + 1.OP + 1.0 (T'' + TL'') +1.OE'$ (2.1.3)

d) $C = 1.OD \pm 0.05D + 1.OW$ (2.1.4)

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Symbols used in these formulas are defined as follows:

C: = Required load capacity section.

D: = Dead load of structure and equipment loads.

P: = Accident pressure load as shown on pressure-temperature transient curves.

T: = Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with 1.5 times accident pressure.

TL: = Load exerted by the liner based upon temperatures associated with 1.5 times accident pressure.

T': = Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with 1.25 times accident pressure.

TL': = Load exerted by the liner based upon temperatures associated with 1.25 times accident pressure.

E: = Load resulting from operational basis earthquake.

T'': = Load due to maximum temperature gradient through the concrete shell, and mat based upon temperature associated with the accident pressure.

TL'': = Load exerted by the liner based upon temperatures associated with the accident pressure.

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E': = Load resulting from design basis earthquake.

W': = Tornado wind load and the pressure drop effect.

Load condition a) indicates that the containment has the capacity to withstand loadings at least 50 per cent greater than those calculated for the postulated loss-of-coolant accident alone.

Load condition b) indicates that the containment has the capacity to withstand loadings at least 25 per cent greater than those calculated for the design basis accident with a coincident operational basis earthquake.

Load condition c) indicates the containment will withstand loads at least equal to those calculated for the design basis accident coincident with a design basis earthquake. The Indian Point Unit No. 3 containment has the capacity to withstand loadings associated with the design basis accident and a coincident earthquake within ACI specified Ultimate Strength Design stress level allowable limits.

Load condition d) indicates the containment will withstand loads at least equal to those calculated for the design basis tornado within ACI specified Ultimate Strength Design stress level allowable limits.

All structural components have been designed to have a capacity required by the most severe loading combination. The loads resulting from the use of these equations will hereafter be termed "factored loads." Specific resultant loading diagrams are presented in Figures 2.2 through 2.9.

The load factors utilized in these equations are based upon the load factor concept employed in Part IV-B, "Structural Analysis and Proportioning of Members Ultimate Strength Design" of ACI 318-63. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors in the design primarily provide for a safety margin on the load assumptions.

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2.2.0 STRESS, STRAIN OR DEFORMATION CRITERIA

The containment is designed such that under all factored load conditions the behavior of the structure will be in the small deformation elastic range. This behavior range is defined by the stress limits contained in the ACI-318-63 code to include additional margin as provided by the capacity reduction factor, ∅.

2.2.1 CAPACITY REDUCTION FACTOR ∅

The theoretical member capacity is lowered by the reduction factor ∅ to recognize variation in quality of materials and permissible tolerances in bar and plate areas and section dimensions, as well as approximations inherent in theoretical analysis. In theory the capacity reduction factor should be divided into the calculated load effect to determine actual design load requirements. Since ∅ is less than one this always results in a design load requirement in excess of calculated requirements.

As a practical matter in the design of this containment the capacity reduction factor has been applied as a multiplier to the theoretical stress criteria. This has the result of reducing the allowable stress as a function of the type of load being carried.

The following ∅ factors for both concrete and steel are used in design:

∅ = .95 (tension)
∅ = .90 (flexure)
∅ = .85 (diagonal tension, bond and anchorage)

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2.2.2 CONCRETE STRESS CRITERIA

The stress criteria governing behavior are as specified in Part IV-B of the ACI-318-63 Code. Specifically the code limitations on concrete compression, tension, shear strength with and without web reinforcement, bond and anchorage are followed. These values are further reduced by applicable capacity reduction factors.

2.2.3 CONCRETE REINFORCING STEEL

The calculated structural capacity of reinforced concrete sections is based on the specified minimum yield strength of the reinforcement using the design methods specified in Part IV-B of the ACI-318-63 Code. This limiting stress value is further reduced by the applicable capacity reduction factor.

2.2.4 STEEL LINER PLATE

The maximum steel stress is limited to 0.95 yield under all primary loading conditions. In regions of local stress concentrations or stresses due to localized secondary load effects the maximum liner strain is limited to 0.5 per cent. (Detailed finite element computer analysis has identified regions of high localized liner stresses which would not have been detected using conventional analytical techniques.)

2.2.5 PENETRATIONS

The steel penetration elements not backed up by concrete are designed to carry design basis accident loads plus operational basis earthquake loads (unfactored) within the stress limitations of the ASME Section VIII Unfired Pressure Vessel Code stress limitations. It should be noted the ASME Code is a "working stress" design code and as such has safety margin contained in the reduced stress levels rather than in the factored load concept.

2.2.6 SUMMARY OF MATERIAL STRESS STRAIN PROPERTIES

The materials used in containment conform to stress-strain limitations as follows:

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	Item	Specification	Min. Yield Strength (PSI)	Min Ultimate Strength (PSI)	Elongation
1.	Concrete	ACI-318	-	3,000	-
2.	Reinforcing Steel	ASTM A432 ACI-318	60,000	90,000	7% in 8"
3.	Liner Plate	ASTM A442, Gr.60	32,000	60,000	22% in 8"
4.	Mech. Penetration Sleeve-12" Dia. & under	ASTM A333 Gr. I	30,000	55,000	35% in 2"
5.	Mech. – Over 12" Dia.	ASTM A201 GR. B to A300	32,000	60,000	22% in 8"
6.	Rolled Shapes	ASTM A36	36,000	58,000	20% in 8"
		ASTM A131 GR. C	32,000	58,000	21% in 8"
7.	End Plates a)	ASTM A300 C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
	b)	ASTM A240 Tp. 304L	25,000	70,000	40% in 2"
8.	Fuel Transfer Tube	ASTM A240 Tp. 304L	25,000	70,000	40% in 2"
9.	Bellows a)	ASTM A312 Tp. 304L	25,000	70,000	35% in 2"
	b)	ASME SB168 Inconel 600	35,000	80,000	30% in 2"
10.	Elec. Penetrations	ASTM A333 Gr.1	30,000	55,000	35% in 2"
11.	Equip. Hatch Insert	ASTM A300 C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
12.	Equip, Hatch Flanges	ASTM A300 C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
13.	Equip. Hatch Head	ASTM A300 C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
14.	Personnel Hatch	ASTM A300 C1.1-Firebox	32,000 A201, Gr. B	60,000	22% in 8" 5A-16

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3.0 CONTAINMENT ANALYSIS METHODS AND COMPARISON WITH CRITERIA

3.1.0 GENERAL CONTAINMENT LOADS

3.1.1 DEAD LOAD

The weight of the concrete structure above the point under consideration based on a density of 150#/ft^3 which includes only the weight of the reinforced concrete structure. Since the maximum rebar stress occurs in tension it is conservative not to consider snow loads or any other non-permanent load which will add to the dead load.

The formula for dead load in k/ft at any point is

$$^{T}DL_i = .150V_i/2\pi R \qquad (3.1.1)$$

where:

V_i = the volume of concrete in feet cubed above point i

R = mean radius in feet

$^{T}DL_i$ = the dead load at any point in the structure (k/ft) of wall

i = the point under consideration

The horizontal thrust from the dead weight of the dome is computed by considering

$$H = -T + wr\cos\varnothing_O \qquad (3.1.2)$$

and

$$T = W/2\pi r\sin^2\varnothing_O \qquad (3.1.3)$$

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where:

W = $2\pi r^2 w(1 - \cos \varnothing_o)$; the total weight of the dome above the point defined by $\varnothing_O$ in kips (3.1.4)

H = the horizontal or hoop thrust in the dome in k/ft of shell

r = mean radius of dome in feet

$\varnothing_O$ = the central angle measured from the top of the dome to the point under consideration

w = the dead load per unit surface area of shell in k/ft.^{2}

T = the vertical or meridinal thrust in the dome in k/ft of shell

3.1.2 DESIGN BASIS ACCIDENT PRESSURE LOAD

Membrane pressure loads in the vertical direction in the cylinder and either direction in the dome are determined by

$$P = \frac{pR}{2} \tag{3.1.5}$$

For the horizontal or hoop direction in the cylinder

$$P = pR \tag{3.1.6}$$

where:

P = pressure load in #/in of wall

p = internal design pressure in #/in^2

R = mean radius in inches

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3.1.3 DISCONTINUITY MOMENT AND SHEAR LOAD

The bending moments, shears and deflections induced in the cylindrical shell by the restraint provided by the base are found by considering a cylindrical shell with a uniform internal pressure. [1] Using the general equations for deflection and slope for a cylinder with end moment and shear, and substituting boundary conditions of $w = \delta$ and $\theta = 0$ at $x = 0$ (the built in end) where δ = the unrestrained growth of a cylinder under uniform internal pressure, one obtains formula for the moment and shear at the built-in end to cause zero deflection and rotation

$$Mo = P/2\beta^2 \quad \text{and} \quad Q_O = -P/\beta \qquad (3.1.7)$$

where:

P	=	the internal pressure in #/in^2	
β^4	=	$E_s h_s / 4a_c^2 D$	(3.1.8)
D	=	$E_c h_c^3 / 12(1-\mu)$ (flexural rigidity of the shell)	(3.1.9)
h_s	=	area of horizontal steel and liner in the cylinder which acts as a spring constant (in^2 / in)	
a_c	=	mean radius of the containment cylinder in inches	
h_c	=	effective depth or thickness of the wall	
μ	=	Poisson's ratio = 0 for cracked concrete	
E_s	=	modulus of elasticity of steel = 29 x 10^6 psi	
E_c	=	modulus of elasticity of concrete = 3.2 x 10^6 psi	
M_o	=	moment at built in Section to cause 0 rotation	
Q_o	=	shear at built in Section to cause 0 deflection	

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Substituting these values in the following expressions, values for bending moment, shear and deflection at any distance from the end can be found:

$$\Delta_x = w = \frac{-1}{2\beta^3 D} \left[\beta M_0\, \gamma(\beta x) + Q_o\, \theta(\beta x)\right] \qquad (3.1.10)$$

$$\theta_x = \frac{dw}{dx} = \frac{1}{2\beta^2 D} \left[2\beta M_0\, \theta\ \ (\beta x) + Q_o\, \phi(\beta x)\right] \qquad (3.1.11)$$

$$M_x = D\frac{d^2 w}{dx^2} = \frac{-1}{2\beta D} \left[2\beta M_0\, \phi(\beta x) + 2Q_o\, \delta(\beta x)\right] D \qquad (3.1.12)$$

$$V_x = D\frac{d^3 w}{dx^3} = \frac{1}{D} \left[2\beta M_0\, \delta(\beta x) - Q_o\, \gamma(\beta x)\right] D \qquad (3.1.13)$$

where:

$$\phi(\beta x) = e^{-\beta x}(cos\,\beta x + Sin\,\beta x)$$
$$\gamma(\beta x) = e^{-\beta x}(cos\,\beta x + Sin\,\beta x)$$
$$\theta(\beta x) = e^{-\beta x} cos\,\beta x$$
$$\delta(\beta x) = e^{-\beta x} Sin\,\beta x$$

Δ_x	=	the deflection of the shell at x
θ_x	=	the slope of the shell at x
M_x	=	the moment of the shell at x
V_x	=	the shear in the shell at x

From these values Figures 3.1 and 3.2 are plotted showing moment and shear v. height of wall in inches. Since no backfill is present shifts in moment and shear, due to backfill restraint, will not occur.

The problem of determining the discontinuity moment and shear at the springline is similar to that at the base. Discontinuity forces at the dome-cylinder junction are only a function of the relative deformation at this point, since

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the rotations of the cylinder and the dome due to the internal pressure are zero and therefore present no discontinuity. The extension of the radius of the cylindrical shell due to the internal pressure is given by

$$\delta_c = \left(Pa_c^2 / E_s h_s^c\right)(1 - \mu/2) \qquad (3.1.14)$$

and the unrestrained extension of the dome (δ_D) is given by

$$\delta_d = \left(Pa_d^2 / 2E_s h_s^d\right)(1 - \mu) \qquad (3.1.15)$$

where

a_d = mean radius of the containment dome in inches

h_s^d = area of horizontal steel and liner of the dome which acts as a spring constant (in^2 / in)

Since the area of the hoop steel per foot in the dome is approximately one half that of the cylinder, the values of δ_c and δ_d are nearly equal and therefore the relative deformation is insignificant.

In calculating the discontinuity effects, the bending is of a local character so that an approximate solution can be obtained by assuming that the bending is of importance only in the zone of the dome close to the springline and that this zone can be treated as a portion of a long cylindrical shell. Equations of continuity for deflection and rotation are written such that the values of M_o and Q_o at the springline may be found. The distribution of the moment and shear into the dome and the cylinder are then found by substituting M_o and Q_o into equations 3.1.12 and 3.1.13. The resulting moments and shears are insignificant.

3.1.4 BASE MAT LOADS

The beam shears and moments in the base mat can be calculated by considering the loads shown in Figure 3.3 acting on a 1' –0 wide beam. The 1' –0 strip of mat to be considered is located at the point where the uplift from the overturning moment in the containment due to earth quake is maximum.

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This gives the maximum moments and shears in the strip.

The loads considered as shown in Figure 3.3. are:

$$U_T = P + T_{EQ} + T_v - T_{DL} \text{ in k/ft} \quad (3.1.16)$$

where:

P = design basis loss-of-coolant accident pressure effect load in the wall in k/ft of wall

T_{EQ} = the tensile load k/ft of wall developed by the earthquake overturning moment

T_v = the effective tensile load or reduction of dead load in k/ft of wall caused by response of the containment structure to vertical earthquake motion

T_{DL} = the dead load in the wall in k/ft

M = base discontinuity moment defined in Section 3.1.3

V'_u = base discontinuity shear defined in Section 3.1.3

D = the dead weight of the base mat on the outside of containment cylindrical wall centerline in k/ft

C = the reaction of the internal structural support columns which are based on the 3'–0 reinforced concrete fill mat; in all cases equal to 50K spaced every 23'–0

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w = $12p + \frac{\rho z}{12}$; the effective uniform load acting on a 1' wide segment of the base slab per inch of segment length where

p = the containment internal pressure in kips/in^2

ρ = the density of reinforced concrete in #/ft^3 = 150#/ft^3

z = the total depth of section including the 3' –0 fill slab.

The crane wall reaction in k/ft is determined by

$$R = D_c + D_o + P_c \quad (3.1.17)$$

where:

D_c = $\rho t_1 \ H$; or the dead weight of the crane wall in k/ft

t_1 = the thickness of the crane wall = 3.0 ft.

H = the height of the crane wall = 50.0 ft.

D_o = $\pi R_1^2 t_2 \rho / 2\pi R_2$ or the approximate dead weight of the operating floor in k/ft

R_1 = the outside radius of the operating floor = 53' –0

R_2 = the mean radius of the crane wall

= 51' –6

P_c = 12 pt_1 or the pressure load acting on the top of the crane wall with t_k given in inches

t_2 = the thickness of the operating floor = 2' –0

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Moments and shears are calculated by writing equations for moment and shear in terms of x as the origin, with x increasing toward the center of the containment building and x measured in inches.

The formulas are as follows:

For $0 \le x \le 201$

$$V_x = U_T - D - 2C^* - wx \qquad (3.1.18)$$

* $-2C$ when $x \ge 201$

$-c$ when $x < 201$

with V_x assumed constant and equal to the value of V_x at 201 inches for the region under the crane wall $201 \le x \le 237$.

For $x \ge 237$

$$V_x = U_T - D - C - w(201) - R - Cw(x - 237) \qquad (3.1.19)$$

or

$$V_x = U_T - 2C - D - R - wx + 36w \qquad (3.1.20)$$

where:

V_x = uplift shear at any point x (inches) in k/ft

Equation 3.1.20 is applicable until $V_x \le 0$.

The design moment in the base slab is determined for $0 \le x \le 201$

$$M_x = M + V'_u e + D(x + 19.5) + \frac{wx^2}{2} + C(x - 27) - U_T x \qquad (3.1.21)$$

With M_x assumed constant and equal to the value of M_x at 201 inches for the region under the crane wall $201 \le x \le 237$.

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For $x \geq 237$

$$M_x = M + V'_u e + D(x + 19.5) + w(201)(x - 105.5) + C(x-27)$$

$$+ C(x-201) + R(x-219) + \frac{w(x-237)^2}{2} - U_T x \qquad (3.1.22)$$

or

$$M_x = M_u + V'_u e + D(x + 19.5) + x(1/2\, x^2 - 36x + 6850) + 2Cx - 228C$$

$$+ R(x-219) - U_T x \qquad (3.1.23)$$

where:

e = the effective depth of the 9' –0 base mat divided by 2 and

M_x = the base moment at any point x (inches) in in-k/ft

At the point where $V_x \leq 0$ flexural beam action is no longer considered since uplift is 0 and the mat acts as a flat circular plate supported on a rigid non-yielding foundation.

Again it should be noted that these maximum values for shear and moment occur at only one point on the base slab circumference where the uplift from the horizontal earthquake is maximum and decreases to zero 90° from this point; therefore, it is considered that the calculations shown are conservative.

A gradient with an operating temperature of 120°F inside the containment and a 50°F temperature at the mat-rock interface was considered and the stresses determined are negligible. Accident temperatures have no appreciable effect on the base slab.

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3.1.5 SEISMIC LOAD

Horizontal Earthquake

The loads on the containment structure caused by the earthquake are determined by Dynamic Analysis of the structure. The Dynamic Analysis is made on an idealized structure of lumped masses and weightless elastic columns acting as spring restraints. The model representation is essentially that of a cantilever beam. Since the containment is founded on rock, no translation or rotation of the structure as a rigid body is considered.

The analysis is performed in two stages: The determination of the natural frequencies of the structure and its mode shapes, and the modal response of these modes to the earthquake by the spectrum response-method.

The natural frequencies and mode shapes are computed from the equations of motion of the lumped masses. These equations are solved by iteration techniques by a fully tested digital computer program. The form of the equation is:

$$(k)\,\Delta = \omega^2 (M)\,\Delta \qquad (3.1.24)$$

(k)	=	Matrix of stiffness coefficients including the combined effects of shear and flexure.
(M)	=	Matrix of concentrated masses. Each mass may have up to six degrees of freedom.
Δ	=	Matrix of mode shapes
ω	=	angular frequency of vibration

The results of this computation are the several values of ω_n and mode shapes $(\Delta)_n$ for n = 1, 2, 3...N, where N is the number of degrees of freedom assumed in the idealized structure.

The response of each mode of vibration to the earthquake ground motion is computed by the response spectrum technique as follows:

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The participation of each mode, P_n is computed from:

$$P_n = \sum_{m=1}^{N} \Delta_{mn}^{x} M_m \Big/ \sum_{m=1}^{N} \Delta_{mn}^{2} M_m \qquad (3.1.25)$$

Where Δ_{mn} is the deflection of mass point m in mode n.

Δ_{mn}^{x} is the component of Δ_{mn} in the direction of the earthquake.

The relative deflection of each mass is determined from

$$Y_{mn} = \Delta_{mn} x P_n S_{an} \Big/ \omega_n^2 \qquad (3.1.26)$$

Where S_{an} = the spectral acceleration of a single degree of freedom system with a frequency ω_n and damping coefficient, a.

The shear at any section of the cylinder is determined by

$$V_s = RMS\left(V_{im} = \Sigma F_{rim}\right)$$

where:

V_{im} = the seismic shear at point i for mode m

F_{rim} = the horizontal inertial force at r nodes above elevation i for mode m

V_s = the root mean square of V_1 for each mode

Figures 3.4 and 3.5 show the base shear and moment distribution up the wall.

The shear flow is determined by consideration of a hollow ring with a total thickness of 2t.

$$S_f = V_s Q/I \qquad (3.1.28)$$

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where:

S_f = shear flow in the wall

V_s = shear at the elevation under investigation as determined by (Eqt. 3.1.27)

$$Q = 2\int_o^{\pi/2} y dA$$

where:

y = distance from element under consideration to the neutral axis of the circular tube cross section of R sin θ

R = radius of containment in inches

θ = angle from neutral axis to the element under consideration in radians

dA = the area of the element under consideration of Rtdθ

t = the thickness of the containment shell in inches

I = moment of inertia of section about neutral axis in in.^{4}

Vertical Earthquake:

The frequency of the containment structure in the vertical direction is determined as described in the dynamic analysis for horizontal earthquake loads.

Using this frequency of , 12.0 cps and the given value response spectral acceleration curve with 5% critical damping a coefficient of spectral acceleration of 0.11g (1.OE') is obtained for the design basis earthquake For the operational basis earthquake with 2% critical damping a coefficient of spectral acceleration of 0.065 is obtained. Multiplying this coefficient by the total mass of the structure yields the vertical earthquake reaction in k/ft of wall. The model in the vertical direction assumes a single degree of freedom response.

$$T_{V_i} = kM_i / 2\pi R \qquad (3.1.29)$$

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where:

k = coefficient of seismic acceleration in the vertical direction:
(0.111 for 1.OE'), (0.0813g for 1.25E)
M_i = mass of containment shell above point i in kips/g
R = radius of containment in feet

Uplift from the Horizontal Earthquake:

The horizontal inertial forces on the containment structure produce overturning movements which in turn produce tension on one side of the containment and compression on the other side in the direction of the earthquake. These forces per foot of wall section are computed by dividing the overturning moment on the section, considering the containment a cantilever beam, by the moment of inertia of the containment as a hollow cylinder. Since the concrete shell is assumed cracked and in tension under the loss-of-coolant accident pressure condition, only the area of the containment vertical rebar and liner are considered in determining the moment of inertia.

The seismic overturning moment above a point i about point i is determined:

M_i = RMS ($M_{im} = F_{irm}\, h_{ir}$) (3.1.30)

where:
M_{im} = the seismic overturning moment above a point i about point i for mode m
h_{ir} = the distance from the location of forces F_{irm} to the point i
F_{irm} = the horizontal inertial forces on the r segments above point i for mode m
M_i = the root mean square of M_{im} for each node

The moment of inertia is computed by
$I = \pi t_1 r^3$ (A hollow circular ring) (3.1.31)

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where:

t_1 = equivalent thickness of vertical reinforcing steel, including liner in sq. in. per inch of wall

r = mean radius of containment in inches

and $$T_{EQ} = M_i C t_1 / I \qquad (3.1.32)$$

where:

T_{EQ} = vertical force in k/in induced in the containment wall by the seismic overturning moment

C = distance from neutral axis to outermost fiber of containment cross section.

Torsional effects from an earthquake are negligible due to the symmetry of the containment structure and therefore are not considered.

3.1.6 TEMPERATURE EFFECT LOAD

An increase in internal temperature caused by a loss-of-coolant accident has been considered. The maximum temperatures, which do not occur at the same time as the maximum pressures, related to the design (P), 1.25P and 1.5P cases are 247°F, 285°F and 306°F respectively. This increase in temperature causes compressive forces in the restrained liner which in turn induces tensile stresses into the rebar. The equivalent force induced in the containment wall is determined:

$$F_c = A_L \varepsilon_{TL} E_S \qquad (3.1.33)$$

where:

F_c = the equivalent tensile load induced in concrete containment shell by the attempted expansion of the liner

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ε_{TL} = final compressive strain in the liner after pressure and temperature conditions and elastic relaxation of the concrete shell have been considered

E_S = modulus of elasticity for the liner steel

A_L = area of liner steel in in^2/ft

In addition to the liner temperature effect on the containment shell the effect of operating thermal gradients through the wall have been considered in analysis of the containment as shown in Section 3.2.5.

The effect of accident thermal gradients has been investigated and found to penetrate less than 10 percent of the containment wall thickness during the maximum temperature-pressure transient following a loss-of-coolant accident. For this reason, the accident temperature transient thermal gradient effect has not been considered in design analysis.

3.1.7 WIND LOAD

The wind load will be determined by considering a conservative wind pressure of 30 psi for ground level up as stipulated in the state building and construction code for the State of New York.

The forces due to the wind loading are given by

$$V_i = P_1 A_i \qquad (3.1.34)$$

where:

V_i = the wind shear at point i

P_1 = the wind pressure of 30 psf

A_i = the projection, perpendicular to the direction of the wind, of the area of containment above the point i

and

$$M_i = P_1 A_i L \qquad (3.1.35)$$

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where:

M_1 = overturning moment about point i determined from the wind load
L = the moment arm from the centroid of the projected area above point i to point i

In all cases the magnitude of the design wind loads are less than the seismic loads as shown in Table 4.1; therefore no stresses are calculated.

3.1.8 TORNADO WIND AND MISSILE LOADS

Tornado loads consist of extreme wind including associated pressure difference and missiles. They are assumed to occur independent of any other extreme load condition.

The wind load is considered for three tornado conditions. One includes a tangential velocity of 300 mph and a translational velocity of 60 mph. This load superposition (Case I, Fig. 3.9) depicts a tornado condition where the funnel coincides with the center of the containment. Load pressure distribution patterns that will result due to various locations of the funnel are considered. The structure will be designed for a triangular (Case II, Fig. 3.9) and a rectangular (Case III, Fig. 3.9) wind distribution of 360 mph.

<u>Case I</u>

For Case I, a torsional effect is induced into the containment structure. This torsional effect results from the tangential wind striking the containment building at an angled α from the normal (See Fig. 3.10). The torsional force is due to the component of the wind tangential to the surface of the containment building and is equal to

$$F_t(\text{lbs.}) = AC_D q \quad Sin\alpha \qquad (3.1.36)$$

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where:

A = surface area of the containment (FT^2)

C_D = $0.45^{(2)}$ (coefficient of drag)

q = $0.002558\ V^{2\ (2)}$ in pounds per square foot

where V = the wind velocity in miles per hour = 300 mph

α = 45°

This assumption is conservative in that the actual tangential force would be the result of skin friction and the effects would be negligible. The shear force F_t which is a maximum at the juncture of the walls and base slab and varies to zero at the top of the dome is distributed over the containment circumference to obtain a maximum shear force per foot.

The average shear force from the translational velocity of 60 mph is equal to

$$F^{(lbs)} = C_D q\ A^{(2)} \qquad (3.1.37)$$

where:

C_D = $.45^{(5)}$ (coefficient of drag)

q = $0.002558\ V^{2\ (2)}$ in pounds per sq. ft.

where V = wind velocity in miles per hr. = 60 mph

A = projected area of the containment normal to the wind direction (FT^2)

The maximum shear equals twice the average shear from above since the shape factor for a hollow circular ring is equal to two. This maximum shear force which occurs at the juncture of the walls and base slab and varies to zero at the top of the dome is distributed over the containment circumference to obtain a maximum shear force per foot.

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The total shear per ft. for Case I is equal to 18^k/FT + $.583^k$/FT = 18.583^k/FT. This is less than the smallest earthquake shear of 46.5^k/FT. The seismic steel is designed to resist an earthquake causing 61.5^k/FT which is 3.3 times greater than the tornado shear.

Case II

For Case II, a torsional effect is induced into the containment structure by the tangential wind which is assumed to strike the containment in such a way that one-half of the containment surface is affected by a frictional force F_t.

F_t is calculated by equation 3.1.36

where:

A = one-half the surface area of the containment and V = 360 mph.

The average shear force from the translational velocity of 360 mph is calculated by Eqt. (3.1.37). The average shear force is equivalent to one-half the force from Eqt. 3.1.37 since a triangular load distribution is assumed rather than the rectangular distribution assumed for the 60 mph wind in Case I. The maximum shear force equals twice the average shear from above.

The torsional and translational wind forces which are greatest at the juncture of the walls and base slab and vary to zero at the top of the dome are distributed over the containment circumference to obtain a maximum shear force per foot.

The total shear per ft. for Case II is equal to 12.85^k/FT + 10.4^k/FT = 23.25^k/FT which has a factor of safety of 2.65 with the maximum earthquake shear force for which the seismic steel is designed.

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Case III

Case III, considers a 300 mph tangential wind traveling with a forward velocity of 60 mph for a total load of 360 mph with a rectangular distribution. The average shear force from the translational velocity of 360 mph is calculated by Eqt. (3.1.37). The maximum shear force equals twice the average shear from above.

The maximum translational wind force which is greatest at the juncture of the walls and base slab and varies to zero at the top of the dome is distributed over the containment circumference to obtain a maximum shear force per foot.

The total shear per foot for Case III is equal to 21.2^k/FT which has a factor of safety of 2.9 with the maximum earthquake shear force for which the seismic steel is designed.

Since the maximum base shear for Case II is smaller than the base shear from both earthquakes, the seismic steel, which is sized for the earthquake producing the largest base shear, provides an adequate mechanism for resisting all tornado shear loads. Since the tornado acts independently of other severe loads, it is not necessary to do a stress analysis for these smaller loads.

Overturning Moment from Wind Load

The maximum overturning moment is produced by the 360 mph wind with a rectangular distribution in Case III. The overturning moment is calculated from Eqt. 3.1.35.

where:

$$P_i = q\, C_D{}^{(5)} \qquad (3.1.38)$$

q and C_D are as defined in Eqt. (3.1.37) and equal .45 and 330 psf respectively.

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The maximum M_i occurs at the wall base mat juncture and varies to zero at the top of the dome. The maximum overturning moment equals 6.15 x 10^9 #-in which is less than the overturning moment for the smallest earthquake (17.2 x 10^9 #-in).

Missile Loads

The containment structure is designed to resist the following missiles:

1. 4" x 12" wood plank @300 mph
2. 4000# auto at 50 mph less than 25'-0 above the ground

Only one missile is considered acting at any time simultaneously with the 360 mph wind load and 3 psi negative pressure if it is conservative to consider the 3 psi negative pressure. The capability of the containment shell to withstand missile impact was calculated by the procedures presented in Reference 5.

The results of this analysis indicate a percentage depth of penetration equal to $\left(\frac{\text{Pent.}}{3.5'}\right)$ and $\left(\frac{\text{Pent.}}{4.5'}\right)$ for the plank and automobile respectively.

3 psi Negative Pressure

The 3 psi negative pressure is not a design consideration when acting independently or in combination with the wind and/or missile. The containment is designed for a maximum no loss of function factored load pressure of 70.5 psi. In combination with the missile or external wind load the negative uniform pressure is assumed to act radially hence does not contribute to the rigid body failure modes of containment.

Vertical Missile Loads

Vertical missile loads are not a factor in the containment design. Since the height of the containment above grade is more than 25'-0 the auto is not a factor.

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A wood plank falling at 90 mph, in the unlikely event that it reaches heights greater than the containment, would produce very small loads in comparison to the horizontal auto missile and therefore is not a design consideration.

3.1.9 LOAD COMBINATIONS

The loads discussed above were combined to design the containment structure as given in Section 2.1.12.

3.2.0 GENERAL STRESS/STRAIN FORMULA

3.2.1 DEAD LOAD STRESS

$$\sigma_{T_i} = T_{DLi} / A_{s_i} \text{ when overall effect is tension} \qquad (3.2.1)$$

$$\sigma_{c_i} = T_{DLi} / A_{c_i} \text{ when overall effect is compression} \qquad (3.2.2)$$

where:

A_{s_i} = area of vertical steel including liner, per foot of wall
A_{c_i} = area of concrete per foot of wall
T_{DLi} = dead load as defined in Section 3.1.1

3.2.2 DESIGN BASIS ACCIDENT PRESSURE LOAD STRESS

$$\sigma = P / A_s \qquad (3.2.3)$$

where:

A_s = area of vertical steel or hoops, including liner, per foot of wall
P = pressure induced membrane force per foot of wall.

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3.2.3 DISCONTINUITY MOMENT AND SHEAR LOAD STRESS

The stress induced in the containment shell wall from the discontinuity moment is calculated by considering formula (16-1) of the ACI 318-63 Code "Ultimate Strength Design."

$$M = A_{S_1} \ f_s(d - a/2) \qquad (3.2.4)$$

$$f_s = M / A_{S_1} \ (d - a/2) \qquad (3.2.5)$$

where:

$$a = A_{S_1} f_y / .85 f'_c b \qquad (3.2.6)$$

and

A_{S_1} = area of steel on the tension side of the containment wall in in²/ft

f_s = stress in the steel in k/in²

f_y = yield strength of the steel in k/in²

f'_c = 3000 psi 28 day design compressive stress of concrete in k/in²

b = width of cross section: in all cases assumed equal to 12"

d = effective depth of cross section in inches = 45"

M = resisting moment in inch-kips per foot. The basis for this number is Figure 3.1. This is less that the ultimate moment since $f_s < f_y$.

The stress in the stirrups is computed from Equation (17-6) of the ACI 318-63 Code – Ultimate Strength Design.

$$A_v = V's / F_s d(Sin\alpha + Cos\alpha) \qquad (3.2.7)$$

$$f_s = V's / A_v d(Sin\alpha + Cos\alpha) \qquad (3.2.8)$$

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where:

A_v = total area of web reinforcement in tension within a distance, s measured in a direction parallel to the longitudinal reinforcement in in^2.
V' = total shear to be carried by web reinforcement in kips
s = spacing of stirrups or bent bars in a direction parallel to the longitudinal reinforcement in inches
f_s = stress in the stirrups in k/in^2
d = effective depth of cross section in inches = 45"
α = angle between inclined web bars and longitudinal axis of member = 45°

3.2.4 BASE MAT STRESS

Stress from the moment is calculated by considering formula (16-1) of the ACI-318-63 code ultimate strength design as shown in Eqs. 3.2.4, 3.2.5 and 3.2.6.

where:

A_s = area of steel on the tension face of the containment base slab in in^2/ft
f_s = stress in the steel in k/in^2
f_c = 3000 psi 28 day design compressive stress of concrete in k/in^2
b = width of cross section – in all cases assumed equal to 12"
d = effective depth of cross section in inches = 100"
M = resisting moment in inch – kips per foot

Stress from the uplift shear is computed from Eqt. (17-6) of the ACI-318-63 code as shown in Eqts. 3.2.7 and 3.2.8.

where:
$V' = V - V_c$ (3.2.9)

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where:

v = total shear

$v_c = v_c^{bd}$ (3.2.10)

and

v_c = the allowable concrete shear stress or $2\varnothing\sqrt{f'_c} = 93k/in^2$

ϕ = capacity reduction factor = .85

f_s = stress in the stirrups in k/in^2

α = angle between inclined web bars and longitudinal axis member = 45°

b = width of the section = 12 inches

d = effective depth of the cross section = 100"

Additional web reinforcement was also provided on the basis of a minimum spacing of s equal to 0.75d.

Bond stresses in the stirrups are computed by considering the formula

$$\mu = A_v f_s / \varepsilon_o L \quad (3.2.11)$$

where:

μ = the bond stress in k/in^2

ε_o = sum of perimeters of all effective bars crossing the section on the tension side

L = the anchorage length above or below the mid height of the mat. No credit is taken for additional anchorage provided by the bend in the bar.

The allowable bond stress for tension bars with deformations conforming to ASTMA-408 and other than top bars is

$$\mu_A = (.8) \quad \sqrt[6]{f'_c} \quad (3.2.12)$$

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where:

μ_A = the allowable bond stress in k/in^2

.8 is the factor allowed by the ACI-318-63 ultimate strength design code for anchorage bond

A finite element analysis was performed on the Unit No. 2 base mat utilizing loads for the three basic loading conditions specified in the Containment Design Report. Since earthquake loads are smaller for Unit No. 3 than for Unit No. 2, due to differences in percent critical damping for the design basis earthquake and the fact that a modal analysis is performed on Unit No. 3, the results of the Unit No. 2 analysis can conservatively be used for Unit No. 3. Maximum hoop moment caused by lack of symmetry of the seismic loading was found to be 454 in.-k/in. This compares with a capacity of 690 in.-k/in. for the in place hoop reinforcing. In all cases tornado loads are smaller than earthquake loads, therefore, no tornado analysis is required.

3.2.5 SEISMIC LOAD STRESS

Horizontal or Vertical Earthquake Effects

$$\sigma = \frac{Load}{A_s} = \quad \text{(For structure in membrane tension)} \qquad (3.2.13)$$

$$\sigma = \frac{Load}{A_c} = \quad \text{(For structure in membrane compression)} \qquad (3.2.14)$$

where:

A_s = area of vertical reinforcing steel, per foot of shell

A_c = area of concrete per foot of shell

Load = force per foot of shell resulting from dead load response to vertical earthquake acceleration or overturning moment induced by horizontal earthquake acceleration.

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The basic assumptions considered in the seismic analysis are:

1) Maximum stress in the seismic reinforcing occurs under the action of seismic shear at 90° points from the direction of seismic motion.

2) The liner does not participate in resisting seismic shear.

3) The stress limitations on intersection bars under the combination of pressure plus earthquake shear in one bar may reach 95% of yield and the opposing bar may relieve stress to 0 ksi. Under this consideration only half of the seismic diagonal steel is considered active in resisting earthquake shear at any given instant.

4) The concrete in the containment does not participate in resisting membrane seismic shear.

Thus, the stress can be calculated by considering the shear flow in the wall being resisted by diagonal bars in a hollow ring.

$$A_{s_s} = 1.414 S_f / 2 f_s \tag{3.2.15}$$

$$f_s = 1.414 S_f / 2 A_{s_s} \tag{3.2.16}$$

where:

A_{s_s}	=	area of diagonal steel per foot, in one direction, measured along a horizontal plane
f_s	=	stress in the steel in k/in^2
S_f	=	the shear flow as determined from Eq. 3.1.27

The 1.414 take the 45° angle of inclination of the diagonal bars into account.

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3.2.6 TEMPERATURE EFFECT STRESSES

As discussed in Section 3.1.6 temperature considerations must involve both temperature gradient and the interaction effects of the liner on the containment shell. The following development for interaction takes both of these phenomena into account.

Temperature effects as shown in Figure 3.6 are combined with dead load, pressure, and earthquake uplifts in the following manner.

Due to the redistribution of stresses in the rebar, the reinforcing steel is considered to carry an equal amount of tension which must balance the compression in the liner to satisfy $\Sigma F_x = 0$

To satisfy equilibrium conditions:

$$F_{Liner} = F_{Wall} \qquad (3.2.17)$$

$$A_L\,{}^{\varepsilon}TLE = -A_s\,{}^{\varepsilon}TL'^{E}$$

$$A_L\left[\frac{{}^{\varepsilon}TL_x + {}^{\mu\varepsilon}TL_y}{1-\mu^2}\right]E = -A_s\,{}^{\varepsilon}TL'^{E} \qquad (3.2.18)$$

$${}^{\varepsilon}TL'_x = \frac{A_L}{A_S}\left[\frac{{}^{\varepsilon}TL_x + {}^{\mu\varepsilon}TL_y}{1-\mu^2}\right]$$

The 2nd condition which must be satisfied is the deformation compatibility

$${}^{\varepsilon}TLx + {}^{\varepsilon}\Delta T = {}^{\varepsilon}T + {}^{\varepsilon}TL'$$

$${}^{\varepsilon}TLx + {}^{\varepsilon}\Delta T = {}^{\varepsilon}T - \frac{A_L}{A_S}\left[\frac{{}^{\varepsilon}TLx + {}^{\mu\varepsilon}TLy}{1-\mu^2}\right] \qquad (3.2.19)$$

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Let $\varepsilon_x = \varepsilon_T - {}^{\varepsilon}\Delta_T$

$${}^{\varepsilon}TLx\left[1+\frac{A_L}{A_s\left(1-\mu^2\right)}\right] = {}^{\varepsilon}x - \frac{A_L}{A_s}\left[\frac{{}^{\mu\varepsilon}TLy}{1-\mu^2}\right]$$

$${}^{\varepsilon}TLx = \frac{{}^{\varepsilon}x}{1+\frac{A_L}{A_S\left(1-\mu^2\right)}} - \frac{{}^{\mu\varepsilon}TLy}{\frac{A_S}{A_L}\left(1-\mu^2\right)+1} \qquad (3.2.20)$$

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Let $\mu = .25$

$$^{\varepsilon}TLx = \frac{\varepsilon_x}{1+1.067\frac{A_L}{A_S}} - \frac{.25\,{}^{\varepsilon}TLy}{.9375\frac{A_S}{A_L}+1} \quad (3.2.21)$$

$$^{\varepsilon}TLy = \frac{\varepsilon_y}{1+1.067\frac{A_L}{A_S}} - \frac{.25\,{}^{\varepsilon}TLy}{.9375\frac{A_S}{A_L}+1} \quad (3.2.22)$$

to solve Eq. 3.2.18 for the strain in the rebar induced by liner compression solve Eq. 3.2.21 and 3.2.22 simultaneously and insert values for $^{\varepsilon}TLx$ and $^{\varepsilon}TLy$ into Eq. 3.2.18.

The definitions of the terms used in the above derivations are:

$^{\varepsilon}T$ = strain in the rebar induced by the dead load, pressure and uplift from horizontal and vertical earthquakes.

$^{\varepsilon}TL$ = Final strain in liner causing stress or the restrained portion of the potential strain of the liner due to the temperature increase (X or Y direction)

$^{\varepsilon}TL'$ = strain in rebar from stress induced by liner compression. (X or Y direction)

μ = Poissons Ratio = .25

A_L = Area of liner in in^2/ft

A_s = Area of rebar in in^2/ft

E = the modulus of elasticity of steel when the section is in tension $\left({}^{\varepsilon}T + {}^{\varepsilon}TL' \geq 0\right)$ and modulus of elasticity of concrete when the section is in compression $\left({}^{\varepsilon}T + {}^{\varepsilon}TL' \leq 0\right)$.

All preceding developments are for the section in tension since this will yield the maximum rebar stress.

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$^{\varepsilon}\Delta T$ = the strain in the liner if unrestrained growth were allowed or $\alpha\Delta T$

where:

α = coefficient of thermal expansion in inch/inch/degree F = 6.5×10^{-6}

ΔT = the difference in temperature between the accident temperature felt by the liner and the temperature of the neutral surface (or the point through the wall where no thermal stress exists because of a thermal gradient through the wall).

The gradient is assumed linear with the inside temperature equal to the operating temperature of 120°F and the outside surface temperature of 0°F.

ΔT can be considered in two steps

ΔT gradient = 120° - ^{T}neutral surface

ΔT interaction – ^{T}Max - 120°

This shows the contribution of both the gradient and interaction effects.

The effect of accident temperatures on thermal gradients has not been considered since analysis has shown only 10 percent of the wall located on the inner face of the containment sees any change of thermal gradient during the pressure phase of the accident. In actuality the stresses induced by thermal gradients in the concrete shell are secondary in nature and are largely relieved by the shell cracking under design accident pressure load conditions. For conservatism, however, the operating temperature gradient was included in the stress analysis.

The location and temperature at the neutral surface as shown in Figure 3.7 is found by equating tension on the outside of the neutral surface to compression on the inside

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assuming the concrete carries no tension. This development of thermal stresses in the rebar is based on the method presented in ACI chimney code [4].

The total compressive force is equal to

$$\tfrac{1}{2}\alpha k^2 t T_x E_c + \alpha k T_x \frac{E_s A_L}{b} + \alpha (k - Z_q) \frac{T_x E_s A_s}{b} + \alpha (k - Z_8) \frac{T_x E_s A_s}{b} \tag{3.2.23}$$

and the total tensile force is equal to

$$\alpha \frac{T_x E_s A_s}{b} (Z_7 + Z_6 - 2k) \tag{3.2.24}$$

When equating total tension to total compression the result is the following

$$k^2 + \frac{2k\, n t_L}{t} + \sum_{l}^{i} \frac{2n A_s}{bt} \quad (k - Z_i) = 0 \tag{3.2.25}$$

where:

i = number of layers of reinf. type

k = distance from the liner to the neutral surface divided by the total thickness of the wall

b = rebar spacing in inches

n = $\frac{E_s}{E_c}$

t = total wall thickness

t_L = liner thickness in inches

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Z = distance from the liner to the rebar under consideration divided by the total thickness of the wall.

α = coefficient of thermal expansion in inch/inch/degree F = 6.5×10^{-6}

The temperature at the neutral surface = $(1 - k)\,\Delta T_1$ (3.2.26)

where:

$\Delta T_1 = 120^\circ - 0^\circ = 120^\circ$

to get the final stress in the rebar due to temperature, pressure, earthquake and dead load.

$$\sigma = \left({}^{\varepsilon}t + {}^{\varepsilon}TL' \right) E_s \qquad (3.2.27)$$

3.2.7 TORNADO WIND AND MISSILE STRESSES

Tornado- caused base shears, overturning moments, and internal pressures are all less than design loads used for the containment and, therefore, stresses are not computed for these loads.

The local stresses caused by tornado generated missiles (4000# auto at 50 mph) can be quite large depending on the area of the containment assumed engaged by the missile and mechanisms considered for absorbing the kinetic energy of the missile. Gross shear and overturning effects have been considered in Section 3.1.18. Local structural integrity of the shell is assured by application of emperically derived penetration formulas[5] to determine structural adequacy.

3.3.0 DETAILED ANALYSIS OF CONTAINMENT AT REPRESENTATIVE LOCATIONS

In order to perform a specific comparison between actual stress-strain levels and limiting behavior criteria several representative points on the containment shell to include the base, cylinder and dome are selected for analysis.

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The selected points are shown in Figure 3.8 and described in Section 3.3. and through 3.3.8. Detailed tabulation of design loads for the eight points listed are found in Section 3.3.9 with the resultant stresses and allowable stress criteria presented in Section 3.3.10. The detailed determination of the loads and stresses shown in Sections 3.3.9 and 3.3.10 are based on the equations given in Sections 3.1 and 3.2. The actual calculations are in the files of United Engineers and Constructors, Inc., Philadelphia, Pennsylvania.

3.3.1 POINT 1

Point 1 is located in the base mat at a point adjacent to the outside face of the crane wall in a region of negligible uplift, where the mat begins to act as a flat circular plate supported on a rigid non-yielding foundation, and high positive moment, point 1 is located at coordinates
H = 53 ft., V = 43 ft.

3.3.2 POINT 2

Point 2 is located in the base mat near the containment wall in a region of high uplift and negative moment adjacent to the knuckle of the liner. Point 2 is located at coordinates
H = 67 ft. and V = 43 ft.

3.3.3 POINT 3

Point 3 is located in the cylindrical portion of the containment shell in a region of very high negative discontinuity moment at a point adjacent to the knuckle at the cylinder-base mat junction which is insulated against any thermal effects. It is located at coordinates
H = 67.5 ft. and V = 45.7 ft.

3.3.4 POINT 4

Point 4 is located in the cylindrical portion of the containment shell in a region of relatively high positive discontinuity moment adjacent to the cut off point for liner insulation at coordinates H = 67.5 ft. and V = 64 ft.

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3.3.5 POINT 5

Point 5 is located in the cylindrical portion of the containment shell, about half way between the base mat and the springline, in a region of membrane stresses only at coordinates H = 67.5 ft. and V = 117 ft.

3.3.6 POINT 6

Point 6 is located in the cylindrical portion of the containment shell at a point just below the springline. It is an area of membrane stress only since the discontinuity effects at the springline are insignificant because the deflection of the dome and cylinder are essentially equal due to the changing steel areas. It is located at coordinates H = 67.5 ft. and V = 191.0 ft.

3.3.7 POINT 7

Point 7 is located in the dome portion of the containment shell at a point just above the springline. It is an area of membrane stress only since the discontinuity effects at the springline are insignificant because the deflection of the dome and cylinder are essentially equal due to the changing steel areas. Point 7 is located at coordinates H = 67.5 and V = 191.0 + ft.

3.3.8 POINT 8

Point 8 is located in the dome portion of the containment shell at a point approximately defined by a 30° arc from the springline in a region of membrane stresses only. The seismic bars are terminated at this point and seismic shear is resisted by hoop and meridional rebar. Point 8 is located at coordinates H = 57.8 ft. and V = 225.8 ft.

3.3.9 SUMMARY OF CONTAINMENT DESIGN LOADINGS

In this Section are presented two tables relative to the design Points 1 through 8 shown in Figure 3.8. In Table 3.1 is shown the material and section properties relative to the eight

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design points selected while Table 3.2 shows the resultant loads for the points selected which were developed from the equations given in Section 3.1 for the load factors and combinations presented in Section 2.1.12.

3.3.10 SUMMARY OF CONTAINMENT DESIGN STRESSES COMPARED TO CRITICAL STRESS LEVELS

In Table 3.3 is presented the stress resultants for the loads given for selected points in Table 3.2 Section 3.3.9. The Table also presents a comparison between resultant stress and allowable stress levels.

3.4.0 EQUIPMENT HATCH & PERSONNEL LOCK— BOSS DESIGN

3.4.1 INTRODUCTION

There are two large openings in the Indian Point – Unit No. 3 Containment Structure. The Personnel Lock is located in the South East quadrant with a center line elevation of 83' –6 and an opening size of 8' –6 diameter. The Equipment Hatch is located in the North East quadrant of the Containment with a center line elevation of 101' –6 and opening size of 16' –0 diameter. Both of these openings along with their thickened reinforced concrete bosses are located a sufficient distance above the fixed base mat at El. 43' –0 that all moments and shears created at this discontinuity have substantially dissipated in the hatch area.

Both hatch and lock are constructed of ASTM 516 GR 60 (formerly A201 GRB) steel normalized to meet the requirements of ASTM A300. The material has been impact tested to meet the requirements of Section N331 of Section III of the ASME Boiler and Unfired Pressure Vessel Code.

All reinforcing steel in the cylindrical wall and the heavily reinforced hatch areas is high – strength deformed billet steel bars conforming to ASTM Designation A432-65 "Specification For Deformed Billet Steel Bars For Concrete Reinforcement With 60,000 psi Minimum Yield Strength." This steel has a minimum tensile strength of 90,000 psi and a minimum elongation of 7% in an 8-in. specimen.

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Bars No. 14S and 18S are spliced by the Cadweld process only. The splices used to join these bars are designed to develop at least 125% of the minimum yield point stress of the bar.

The plate steel liner inside the cylindrical wall including the hatch areas is carbon steel conforming to ASTM Designation A442-65 Grade 60 "Standard Specification for Carbon Steel Plates With Improved Transition Properties." This steel has a minimum yield strength of 32,000 psi and a minimum tensile strength of 60,000 psi with an elongation of 22% in an 8-in. gauge length at failure. The liner material is tested to assure an NDT temperature more than 30°F lower than the minimum operating temperature of the liner material. Impact testing was done in accordance with Section N331 of Section III of the ASME Boiler and Pressure Vessel Code.

Internal forces and stresses in the concrete containment shell were determined for the factored load combinations listed in Section 3.4.3.1 of the Containment Design Report for Unit #2 (Docket 50-247). In verifying the adequacy of resistance to these factored loads, capacity reduction factors recommended in ACI 318-63 Building Code Requirements for Reinforced Concrete were applied where applicable.

Under loadings which include incident pressure and temperature, some local yielding of the liner may occur; however, this has no adverse strength implications for the containment wall. Moreover, the ductility of the liner fastening studs is sufficient to tolerate local inelastic buckling without stud failure.

Under load combinations a, b, and c on Page 5A-11 dropping the thermal effects, and with the liner contribution to strength disregarded, calculated rebar stresses do not exceed $\varnothing f_y$ (where $\varnothing$ is the capacity reduction factor). Under load combination a involving a factor not greater than 1.0 on reactor incident, and with the liner stress (and temperature) accounted for, calculated rebar stresses do not exceed $\varnothing f_y$. Under factored load combinations b and c involving a factor greater than

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1.0 on reactor incident, and with the liner (and temperature loads) accounted for, a limited amount of local rebar yielding is permitted. These criteria guarantee not only assured resistance to the active loads but also minimize any local inelastic strains which may be associated with stress redistribution due to local rebar yielding.

The hatch and lock are anchored into reinforced concrete bosses by means of stud anchors. Along the Equipment Hatch there are 16 rows of 5/8" ∅ x approximately 15" long studs to extend beyond the first row or hoop rebar with 100 per row around the hatch for a total of 1600 studs. Along the Personnel Lock there are 9 rows of 5/8" ∅ x approximately 15" long studs to extend beyond the first row of hoop rebar with 44 per row around the lock for a total of 396 studs. In the areas adjacent to the penetrations, the liner is thickened to 3/4" and is anchored into the concrete by hooked L – anchors of 1/2" ∅ x 9" long (minimum including 2" hook).

The reinforced concrete bosses are thickened to 7'-6 at the Equipment Hatch and 5'-6 at the Personnel Lock. The bosses have flat outside faces and a smooth transition to the dimensions of the wall beyond the effects of the discontinuities (see Figure 3.11).

The hatch and lock have been designed to withstand the internal Containment pressure plus operating and earthquake loads associated with the design accident in accordance with Section III Subsection B of the ASME Boiler & Pressure Vessel Code – Nuclear Vessels. The anchors have been designed to transmit these loads back into the reinforced concrete boss.

Both the Equipment Hatch and Personnel Lock penetrate the concrete shell. In the case of the 16' ∅ Equipment Hatch, a personnel lock is mounted in the head of the hatch and transmits all pressure loads thru the barrel to the concrete when the inside door is closed. Should the personnel door be left open on this lock, the temperature and pressure loads are transmitted to the lock but not into the concrete due to the space between the lock and hatch. Where the 8'-6" Personnel Lock is mounted in the concrete, the temperature and pressure loads inside the lock are transmitted to the concrete if the inside door is left open. (See Figure 3.12.)

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3.4.2 DESCRIPTION OF OPENING REINFORCEMENT

The thickened boss has been heavily reinforced in addition to the dense reinforcing which already exists in the 4'-6 thick Containment cylinder wall. The hoop, vertical and seismic wall reinforcing are bent around the openings to provide continuity of reinforcing and assure flow of membrane forces around the openings. All splices will be by the Cadweld process only. The splices are designed to assure that they will develop at least 125% of the minimum yield point stress of the rebar. Several secondary bars have been terminated by means of mechanical anchorage. At the continuous bar bends, hooked bars are provided to prohibit any local crushing of the concrete. In addition the radius of the bar bends is such that crushing of the concrete will not occur. Due to bending the main bars around the large openings, a void in reinforcing is created on the horizontal and vertical center lines. To prevent any cracking and spalling of concrete and to resist membrane tensions, these voids are filled with added rebar which are terminated by hooks at each end.

To accommodate stress concentrations and discontinuity effects of the opening hoop reinforcing is provided around the opening.

In addition to the membrane forces a moment on the ring is produced by the shear load from the pressure on the door of the hatch tending to cause the ring to rotate inside out. Since the ring is restrained from warping, bending moments occur in the cross section of the ring which are resisted by the additional hoops in the reinforced boss. The hoops are designed to resist the tensile loads in addition to bending mentioned above. Since the ring tends to rotate inside out and detach itself from the Containment shell about its outer boundary, a tensile load is induced on the inside surface of the ring and containment. This is resisted by the main vertical and horizontal reinforcing in the Containment cylinder continuous wall.

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Since there is an eccentricity between the center of the wall and the center of the thickened ring, moments causing tension on the inside face of the ring develop. These moments are resisted in tension by the main vertical and horizontal bars which are continuous around the opening. In addition these bars assist in resisting membrane tensile loads.

In addition to the main vertical and horizontal reinforcing in the Containment cylinder wall, the two-way seismic reinforcing in the wall is continuous around the opening, thus increasing the steel area available to carry discontinuity forces and moments.

Transverse shears radial to the center of the containment and in plane shears are resisted by #8 stirrups placed radially to the opening at 6" centers around the opening. Popout shears along the circumference of the opening caused by edge reactions from the pressure against the barrel head are resisted by 2-#9 bars @12" around the opening placed through the cross section perpendicular to the reference plane. These bars are spaced at d/3 to insure that at least one bar will cross a potential diagonal crack through the cross section. One end of the bar will be hooked in order to develop adequate anchorage from the point of crack formation to the end of the bars. In addition to the above mentioned stirrups; concrete, extra stirrups at the voids created by the main horizontal and vertical rebar bending around the opening and inclined horizontal and vertical rebar are also available to resist shear loads. See Figures 3.11 and 3.13.

3.4.3 DESIGN OF OPENINGS

The design of the Unit #3 Equipment Hatch and Personnel Lock is identical to that used in Unit #2 Hatches.

The resultant stresses in the containment shell are modified slightly due to the movement of the containment shell seismic reinforcement toward the outer face to facilitate placement and the 6 percent reduction in total containment reinforcement resulting from the reduced seismic load based on 5 percent rather than 2 percent damping.

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The methods used in the design of the equipment hatch and personnel lock were verified by a finite element analysis, the details of which are presented in Section 3.4 of the Containment Design Report of the FSAR for Unit #2.

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References

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(2) Nuclear Reactors and Earthquakes, TID 7024, Division of Technical Information, USAEC August, 1963.

(3) Blume, J., Newmark, N., et al., Design of Multistory Reinforced Concrete Buildings for Earthquake Motion, Portland Cement Association, 1961.

(4) American Concrete Institute, "Specification for the Design and Construction of Reinforced Concrete Chimneys (ACI 505-54)," ACI Manual of Concrete Practice, Part 2, 1967.

(5) Trexel, C.A., Tests and Design of Bombproof Structures of Reinforced Concrete, Navy Department, U.S. Government Printing Office, Washington, 1961.

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4.0 CONTAINMENT COMPONENT DESIGN

4.1.0 CONTAINMENT SUMPS

There are three containment sumps that cause projections of the bottom of the containment base mat. The largest is the containment reactor sump which is a key shaped reinforced concrete pit located in the center of the base slab (Figure 4.1). This sump, which is 52.5 feet long and 25'-6" deep, encloses the bottom section of the reactor vessel and the in-core instrumentation leads. The side walls and floor of the sump are 4.5 feet thick supporting the ¼" steel liner. An additional 2 feet of concrete is poured over the liner.

Since the reactor sump walls and floor are poured directly against the rock foundation, rigid support conditions have been considered in the design at the sump structural elements to withstand load. Also, since this sump is located in the central portion of the base slab which is poured directly on the rigid rock foundation, negligible bending shears and moments exist in the base slab at the sump location under all load conditions. The reinforcing steel in the sump includes an extension of the reinforcement with the standard detailing procedures specified in ACI-315 being followed. Temperature steel is included in the sump to meet the requirements of ACI-318.

The next largest sump encloses the intakes for the recirculating pumps and consists of a rectangularly shaped reinforced concrete pit 18 feet by 12 feet in plan and 12 feet deep. The side walls and floor of this sump are 9 feet thick supporting the sump liner with an additional 3 feet covering the pit liner floor and 1 foot covering the liner enclosing the sides of the sump (Figs. 4.2, 4.3, 4.4, and 4.5). As in the case of the reactor sump, the walls and floor of this sump are supported by the rigid rock foundation and the sump is located in a region of negligible bending stresses in the base mat. The walls and floor of the sump are considered structurally as part of the base mat.

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The smallest sump encloses the containment sump intake and measures 7.5 feet by 7.5 feet by 5.75 feet deep. It has side walls and floor 7.25 feet thick with a 1 foot covering on the liner. As in the case of the recirculating water sump, the walls of the sump are considered as part of the base mat and are located in a region of negligible bending moment and shear.

The three sumps and in particular the concrete cover over the sump liners, also serve as excellent shear keys in transferring seismic or thermal shear loading from the containment internal structure to the base mat. While it is anticipated most of the shear load would be transmitted by friction between the containment base liner and the containment mat the concrete cover area of the sumps acting alone is capable of transmitting full seismic shear load for a 0.15 earthquake at an average shearing stress of 120 psi.

4.2.0 CONTAINMENT BASE MAT

The containment base mat is a reinforced concrete slab 146 ft. in diameter and 9 ft thick (Figure 4.6). The base slab is designed as a flat circular plate supported on a rigid non-yielding foundation. For loads applied uniformly around the slab, the analysis considers a one foot wide beam fixed at a point where the vertical shear is equal to zero. This is the point where the downward pressure on the mat and the dead weight overcome the uplift at the containment wall base mat juncture from pressure and earthquake loadings.

4.2.1 SHEAR REINFORCEMENT DESIGN OF SLAB

The limiting loading condition for shear is defined by the 1.25P factored load equation which results in the base mat loads as shown in Figure 4.7. The external shear load per foot of 1 foot wide section of the mat is determined from Eqs. 3.1.18 and 3.1.20.

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The maximum shear stress permitted on an unreinforced web subjected to combined shear and bending is given by (ACI-318; Eq. 17-2)

$$v_c = \varnothing\left(1.9\sqrt{f'_c} + 2500\frac{P_w V d}{M}\right) \qquad (4.2.1)$$

where:

v_c = shear stress carried by concrete

$\varnothing$ = capacity reduction factor for shear (0.85)

p_w = reinforcement ration (As/bd)

V = total shear at section

M = bending moment at section

d = depth of section from compression fact to centroid of tensile steel (100 in)

Solving Eqs. 4.2.1 and 4.2.3 for the loads defined in Figure 4.7 the distance x determined as the cut off point for shear reinforcement is 16.0 ft. or just inside of the crane wall.

The shear load V used for the design of shear reinforcement is determined at a distance d from the edge of the slab. This value for the loading given in Figure 4.7 is 183 k/ft. The shear load which is assumed carried by other than shear reinforcement is determined as shown in Eq. 3.2.10 equal to 108 k/ft.

V_c = vbd = 108 k/ft

where:

v = $2\varnothing\sqrt{fc'}$ (ACI-318, section 1701); $\varnothing = 0.85$

d = effective depth of base mat slab (100 in.)

b = width of wedge shaped section at x = d = 100 in.

The required area of shear reinforcement per foot is determined by Eqs. 3.2.9 and 3.2.7 as shown in Figure 4.8.

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4.2.2 MOMENT REINFORCEMENT DESIGN OF SLAB

As in the case for shear, moment was calculated by writing equations for moment in terms of x using the center of the containment wall-base slab juncture as the origin with x increasing toward the center of the containment building. For the 1.5P limiting case the discontinuity moment is 1210 k. ft/ft, the discontinuity shear is 157 k/ft as shown in Figure 4.9. The expressions for the moment as a function of x are shown in Equations 3.1.21 and 3.2.23.

The loading diagram in the mat is shown in Figure 4.9. The equation for moment as a function of x is set equal to zero and the distance x at which the condition of tension in the top of the mat would discontinue is found to be 6.5 feet. The expression for shear is also set equal to zero and the distance x at which the maximum positive moment (1208 k. ft) occurs is found to be 20.6 feet.

The moment steel provided for the maximum negative moment of 1210 k ft/ft which occurs along the perimeter of the slab is also assumed to carry one half of the discontinuity shear of 157 k/ft as an axial load which results in a direct stress of 18.4 psi. The section is designed according to Part IV-B Structural Analysis and Proportioning of Members – Ultimate Strength Design of the ACI-318-63 Code as shown in Section 3.2.3 of this report.

The value of fy used is reduced to 41.6 KSI since 18.4 KSI is taken by the discontinuity shear and the ultimate moment is found to be 1,250 K ft/ft which is greater than the maximum applied negative moment value of 1210 K ft/ft.

For all combinations of pressure, dead load and earthquake loadings which tend to cause uplift in the base slab, the dead weight of the crane wall greatly reduces uplift. This forms a rigid central region in the base slab which is supported on an essentially rigid non-yielding foundation. The model used to analyze this condition is a circular and solid flat plate with a central rigid portion subjected to an external moment (Figure 4.10). The maximum radial stress at the inner edge is given by [1].

$$\sigma_R = \beta \frac{M_{EXT}}{a\,t^2} \qquad (4.2.2)$$

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where:

M_{EXT} = external overturning moment

β = parameter which depends on ratio of a to the radius of the central rigid portion of the slab.

a = radius of the circular slab (875 in)

t = thickness of the slab

The radial stress due to an internal moment is:

$$\sigma R = \frac{{}^{M}INT^{C}}{I} = \frac{6^{M}INT}{t^{2}} \qquad (4.2.3)$$

where:

M_{INT} = internal moment in base slab

C = distance from neutral axis to outer fiber of section

I = moment of inertia of section

By equating Eqs. 2.11 and 2.12, the expression for internal moment as a function of the external overturning moment is:

$$^{M}INT = \frac{\beta^{M}EXT}{6a} \qquad (4.2.4)$$

The external overturning moment ^{M}EXT is that due to the seismic shear forces. The maximum positive moment acting on the slab base occurs for the 1.25 P factored load case at the crane wall. The uplift pressure is added to the internal moment due to the seismic overturning moment.

Temperature steel was also added in the base mat to meet the requirements of article 807 of the ACI 318 Code. In the circumferential direction reinforcement is placed in the top and bottom of the base slab. In the central region of the base slab for a radius of 28 feet the temperature steel is placed in an orthogonal grid pattern.

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4.3.0 CONTAINMENT CYLINDER WALLS

The analysis of the cylinder was accomplished by the superposition of membrane forces resulting from gravity, internal design basic accident, temperature and pressure and overturning due to earthquake using the factored load equation presented in Section 2.1.12. The cylindrical walls are reinforced circumferentially with steel hoops and vertically with straight bars.

For the vertical axial load in the cylinder the 1.25 P loading condition governs the design. The axial force in the cylinder due to the pressure loading on the dome is given by Eq. 3.1.5.

The uplift force in the cylinder due to the horizontal earthquake is given by Eqs. 3.1.29 and 3.1.32. The dead weight force in the cylinder is obtained by taking the total weight of the dome as the force acting at the top of the cylinder and the total weight of the dome and cylinder as the force acting at the base. The uplift force in the cylinder due to the pressure loading on the dome and the uplift due to the horizontal earthquake are combined with the dead weight load in the cylinder. The resultant load diagram varies from an uplift force of 330 k/ft at the base of the cylinder to 276 k/ft at the springline.

For the hoop direction, the 1.5P case controls since the dead weight and earthquake effects are zero. The force in the hoop direction is given by Eq. 3.1.6.

The seismic loads were determined as described in Section 3.1.5. To provide for the seismic steel, diagonal bars are placed in the cylinder walls in both directions at an angle of 45°.*
Seismic steel reinforcement is as shown in Figures 4.11 and 4.12.

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* The design of the diagonal steel is such that its horizontal component is equal to the maximum value of the shear flow which is equal to twice the average shear on the cross-section. Since the diagonals are assumed to act in diagonal tension only, half of the total area of the 45° diagonal seismic bars as assumed active to resist seismic shear effects at any given instant.

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4.4.0 CONTAINMENT DOME

The thickness of the dome is small in comparison with the radius of curvature (1/15) and there are no discontinuities such as sharp bends in the meridional curves, therefore the stresses due to dead weight, pressure, or earthquake, were calculated by considering a uniform distribution across the wall thickness. All membrane tensile stresses are assumed taken by the steel reinforcement and none by the concrete unless they are compressive stresses since the concrete is assumed to have no tensile strength.

The membrane analysis of the hemispherical dome has been performed by the superposition of forces resulting from gravity and accident pressure. The dead weight forces in the dome are computed by using the procedure outlined in the Portland Cement Association Bulletin ST55, "Design of Circular Domes." The total vertical dead load acting downward for a given central angle from the apex is given by Eq. 3.1.4.

The meridional thrust (T) is given by Eq. 3.1.3 and the circumferential thrust (H) is given by Eq. 3.1.2.

The membrane force due to the internal design pressure is equal throughout the dome and is given by Eq. 3.1.5.

Analysis has shown that the earthquake effects are small in the dome, therefore the critical design condition is the 1.5P factored load case. The membrane forces due to the 1.5 factored internal design pressure of 70.5 psi are added vectorily to the membrane forces due to 95 percent of dead weight and the total force per foot is divided by the allowable yield stress of the rebar (57 KSI) to determine the area of steel required. All of the combined direct stresses are developed in the reinforcing steel encased in the concrete.

The vertical steel in the cylindrical concrete wall is extended into the dome such that a continuity between the dome and cylinder is achieved. At an angle of 60° from the springline, the 18S bars come together to a 6 inch spacing. The bars are connected to splice plates by means of Cadwell mechanical splices such that for every two bars coming together there is one 18S bar extending beyond this point.

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At an angle of 75° from the springline the bars again come together to a 6 inch spacing and are cadwelded to a splice plate to increase the spacing to 12 inches. Similarly, the 18S bars are connected to splice plates at an angle of 83° and 86.5° from the springline. At the apex of the dome 18S bars at a constant spacing of 12 inches connect the splice plates which are 3.5° from the center of the dome as shown on Figure 4.13.

To provide the required earthquake resistance the seismic steel in the cylinder is extended into the dome to a point which is 30° above the springline as shown in Figure 4.14.

Above 30° from the springline the membrane steel in the dome is sufficient to carry the seismic shear . The maximum stress in the rebar due to an earthquake is determined by resolution of the principal tensile stress into components parallel the rebar. In addition, the dome liner has sufficient capacity to carry seismic loads under operation or accident conditions.

4.5.0 CONTAINMENT LINER

4.5.1 PURPOSE OF LINER

The purpose of the steel liner, which is attached to the inside face of the concrete shell, is to ensure a high degree of leak tightness in the event of an accident resulting in the loss of reactor coolant and potential release of radioactive material. The liner is attached to the concrete by means of stud anchors so that it forms an integral part of the entire composite structure under all loadings.

4.5.2 DESIGN LOAD CRITERIA

The loads considered in the design of the containment structure, which can create stresses within the component parts such as the liner, are enumerated in Section 2.1.0. The resultant limiting loads in the liner from these specified load combinations for the typical points specified in Section 3.3.0 are shown in Table 4.1

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4.5.3 STRESS CRITERIA

The design stress criteria for the liner is based on the philosophy that no gross deformation beyond the elastic limit occurs for all primary membrane loading conditions defined previously.

In this reinforced concrete structure, the design limits for tension member (i.e., the capacity required for the design loads) are based upon ASTM specified minimum allowable stresses for reinforcing steel.

This reinforcement has also been designed so that it is not subject to average stresses beyond the yield point across any section due to the factored loads.

4.5.3.1 Additional Safety Provisions Regarding Stresses

As an additional safety factor, the allowable stress under any given load is reduced from the values referred to above by a capacity reduction factor, denoted as "∅." This reduction provides for the possibility that small adverse variations in material strength, workmanship, dimensions, control and degree of supervision, while individually within required tolerances and the limits of good practice, occasionally may combine to result in undercapacity.

The values of "∅" used for the liner is 0.95.

Thus, for principal compression and tension, the liner stresses are maintained below 0.95 specified minimum yield at normal operating temperature, i.e., 0.95 x 32,000 = 30,400 psi. For shear, the liner stresses are maintained below 0.6 specified minimum yield at normal temperature. (The actual shear stresses are well below this limit.) The actual proportioning of seismic shear between the liner and the concrete shell is dependent upon the relative stiffness of the two elements. Conservatively assuming only the relative stiffness of the steel reinforcement in the concrete shell versus the stiffness of the liner approximately 30 percent of the seismic shear could be transmitted into the liner.

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The limiting case governing the contribution of shear stress to direct stress in the liner to determine the maximum principal compressive stress shows the liner capable of carrying 40 percent of the seismic shear before principal yield compression would be reached. The liner plate material is ASTM A-442, Grade 60.

For the Structural Proof Test, primary membrane stresses are maintained within elastic limits.

4.5.4 MISSILE PROTECTION

High pressure reactor coolant system equipment which could be the source of missiles is suitably shielded from impacting on the liner either by the concrete shield wall enclosing the reactor coolant loops and pressurizer or by the concrete operating floor to block any passage of missiles to the containment walls. A structure is provided for the control rod drive mechanism to block any missiles generated from fracture of the mechanisms.

4.5.5 DESIGN AND STRESS ANALYSIS

The reactor containment is a reinforced concrete shell in the form of a vertical right cylinder with a hemispherical dome and a generally flat base, supported on rock. The inside surface of the structural concrete is lined with steel plate anchored in the concrete shell.

Anchorage of the liner to the concrete shell is effected as shown on Figure 4.15 and described below.

Attachment of the dome liner to the concrete is made by a combination of structural steel tee sections welded to the exterior face of the dome plate in two directions at approximately five foot intervals and Nelson Studs which are provided between the tees. The liner for the cylindrical portion of the concrete shell is anchored by means of Nelson Studs welded to the plate at 14 inches vertical spacing and 24 inches horizontal spacing on the 3/8 inch thick plate and 28 inches by 24 inches on the ½ inch thick plate.

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The first course of studs is approximately 18 inches above the base slab. Results of the analysis performed for the base slab preclude the need for anchorage of the bottom horizontal liner plate to the concrete base.

The basic design concept for the liner utilizing stud anchorage ductility assures that the studs fail due to shear, tension or bending stress without the stud connection causing failure or tearing of the liner plate.

The design has also taken into consideration the possibility of daily stress reversals due to ambient temperature changes for the life of the plant. Fatigue limit of the studs, verified by extensive testing of the fatigue life of plates with stud shear connectors will exceed the design requirements. Moreover, to accommodate possible fatigue failure in the plate-to-stud weldment, the depth of weld to the liner plate is controlled to avoid impairment of liner integrity.

In general, the stresses in the liner have been determined assuming deformation compatibility with the containment concrete. The exception to this assumption is the base of the cylindrical wall at the juncture with the base slab. The shear capacity of the studs in the vicinity of the juncture points is less than 10 per cent of the shear capacity required to transfer total discontinuity bending stresses into the liner. For this reason stresses induced in the liner by bending of the concrete shell have been neglected.

The design of the liner takes into consideration buckling of the plate under loading. In order to determine the critical buckling stress, the plate is assumed to be hinged along EFGH as shown in Figure 4.16. This assumption corresponds to buckling mode type III as identified in reference 2. The critical buckling stress for the case of equal bi-axial compression of the assumed hinged plate EFGH is 38.1 ksi. The maximum calculated stresses as shown in Table 4.2 and F are -30.4 ksi vertically and –25.0 ksi horizontally and from a Mohr's circle consideration, the normal stress on the assumed hinged plate is –29.0 ksi and the shear stress 2.34 ksi. The shear stresses on the assumed hinged plate is of such low magnitude that no reduction of normal critical buckling stress results.

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Since the maximum applied stress of 29.0 ksi is less than the critical buckling stress of 38.1 ksi, the plate will not buckle.

It will be assumed that during the 115% pressure test of the containment at 54 psig, the liner will contribute to the net overall cross-sectional strength of the structure to resist membrane forces. Since the liner will be anchored to the shell by Nelson Studs at appropriate intervals, elastic stability will be assured and the liner will not be loaded beyond a 95% yield. Results of the calculations for the overpressure test indicate maximum stresses of 30.3 ksi in the liner which are within the allowance of 95% of yield.

The liner will make only a small contribution to the structural capability of the total containment under an accident loading condition. It will tend to expand faster than the concrete at increased temperature and therefore will be stressed first in tension due to pressure build up, and then in compression as a result of temperature rise. Insulation material will be applied to the lower 20 on the inside of the liner cylinder to maintain stresses within the design criteria and to ensure elastic stability.

The maximum liner stresses, computed for this condition, is 30.4 ksi, which is within the design criteria.

The stress values at different points in the liner due to the three loading conditions (a), (b) and (c) on the containment structure, described in Section 2.1.12, are summarized in Table 4.2. The results indicate that the calculated maximum liner stresses are in conformance with the criteria. In determining the final stress state both Poisson ratio effects and elastic deformation of the concrete are considered. In all cases, the seismic loads exceed tornado induced loads hence stresses for load case (d) has not been included.

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Columns 3 through 9 of Table 4.2 show the stresses resulting from individual load components of the factored load equations. In column 10 is found total resultant liner stress considering the containment wall rigid, that is neglecting the deformation of the liner due to elastic straining of the concrete shell, and no Poisson ratio effects. The stresses corrected for the interaction between liner and concrete shell are presented in column 11 and final liner stress intensities including Poisson ratio effects are shown in column 12.

4.6.0 PENETRATIONS

In general, a penetration consists of a sleeve embedded in the concrete wall and welded to the containment liner. Piping penetrations pass through on embedded sleeve and the ends of the resulting annulus are closed off, either by welded end plates, bolted flanges or a combination of these.* Provision is made for differential expansion and misalignment between pipe or cartridge, and sleeve. The cartridges, however, have no expansion provisions as they are only connected at one end.

Penetrations are designed with double seals so as to permit continuous pressurization during plant operation to prevent outleakage in the event of a loss-of-coolant accident. In addition, small steel channels are welded over all joints in the containment vessel liner to form chambers which also permit continuous pressurization to demonstrate the integrity of the penetration-to-liner weld joint. Pressurizing connections are provided to continuously demonstrate the integrity of the penetration assemblies. Pressure in the penetrations and liner joint channels is maintained at a minimum pressure greater than the calculated peak accident pressure. This is accomplished by the Containment Penetration Pressurization System. This system also allows introduction of Freon or a similar tracer gas for leak detection as may be required should consumption of pressurizing air be excessive. These provisions, in addition to the Isolation Valve Seal Water System, effectively block all containment leakage paths.

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* Electrical penetrations and the equipment hatch pass through an embedded sleeve but the ends are not closed off outside the containment building.

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4.6.1 TYPES

4.6.1.1 Electrical Penetrations

"Cartridge" type penetrations are used for all electrical conductors passing through the containment. The penetrations are provided with a pressure connection to allow continuous pressurization. Ceramic type seals are used to provide a pressure barrier for the conductors. Typical electrical penetrations are shown in Figure 4.17.

4.6.1.2 Piping Penetrations

Double barrier piping penetrations are provided for all piping passing through the containment. The pipe is centered in the embedded sleeve which is welded to the liner. End plates are welded to the pipe at both ends of the sleeve. Several pipes may pass through the same embedded sleeve to minimize the number of penetrations required. In this case, each pipe is welded to both end plates. A connection to the penetration sleeve is provided to allow continuous pressurization of the compartment formed between the piping and the embedded sleeve. In the case of piping carrying hot fluid, the pipe is insulated and cooling is provided to maintain the concrete temperature adjoining the embedded sleeve at or below 150°F. Typical piping penetrations are shown in Figure 4.18.

4.6.1.3 Equipment and Personnel Access Hatches

An equipment hatch is provided which is fabricated from welded steel and furnished with a double-gasketed flange and bolted dished door. The hatch barrel is embedded in the containment wall and welded to the liner. Provision is made to continuously pressurize the space between the double gaskets of the door flanges and the weld seam channels at the liner joint, hatch flanges and dished door. Pressure is relieved from the double gasket spaces prior to opening the joints. The personnel hatch is a double door, mechanically latched, welded steel assembly. A quick-acting type, equalizing valve connects the personnel hatch with the interior of the containment vessel for the purposes of equalizing pressure in the two systems when entering or to prevent both being open simultaneously and to ensure that one door is completely closed before the opposite door can be opened.

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Remote indicating light and annunciator situated in the control room indicate the door position status. An emergency lighting and communication system operating from an external emergency supply is provided in the lock interior. Emergency access to either the inner door, from the containment interior; or to the outer door, from outside is possible by the use of special door unlatching tools.

4.6.1.4 Fuel Transfer Penetration

A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment and the spent fuel pit. The penetration consists of a 20 inch stainless steel pipe installed inside a 24 inch pipe. The inner pipe acts as the transfer tube and is fitted with a pressurized double-gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pit. The arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment liner and provision is made, by use of special seal ring, for pressurizing all welds essential to the integrity of the penetration during plant operation. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures. The fuel transfer penetration is shown in Figure 4.19.

4.6.1.5 Containment Supply and Exhaust Purge Ducts

The ventilation system purge ducts are each equipped with two quick-acting tight-sealing valves (one inside and one outside of the containment) to be used for isolation purposes. The valves are manually remotely opened for containment purging but are automatically closed upon a signal of high containment pressure or high containment radiation level. The space between the valves is pressurized above design pressure while the valves are normally closed during plant operation.

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4.6.1.6 Sump Penetrations

The piping penetration in the containment sump area is welded directly to the base liner. The weld to the liner is shrouded by a test channel which is used to demonstrate the integrity of the liner.

4.6.1.7 Dome Penetration

An opening is located n the dome at the top of the vessel. This opening is for construction ventilation and will be permanently closed at the conclusion of the construction work.

4.6.1.8 Temporary Construction Openings

There are no temporary construction openings.

All personnel locks and any portion of the equipment access door extending beyond the concrete shall conform in all respects to the requirements of ASME Section VIII Nuclear Vessels Code. The weldments of liner to penetration sleeve are of sufficient strength to accommodate stress concentrations and adhere strictly to ASME Code Section VIII requirements for both type and strength. Liner reinforcements are designed to support penetrations in the appropriate portion of the liner plate during shop testing, shipping and field erection.

The adequacy of penetrations in retaining strength and ductility while preventing leakage is ensured by the following measures:

1. The materials for all components are selected primarily because of their high ductility.

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2. By design, all penetrations can withstand all stresses imposed on the as a result of normal plant operation and the hypothetical loss-of-coolant accident. Specifically, the joint between the penetration sleeve and the building liner plate is reinforced with a thickened plate. The sleeve is anchored to the concrete by means of stud anchors welded to a steel ring which is, in turn, welded to the sleeve. The penetration end plates through which the pipes or electric cable pass are designed to withstand the penetration's internal air pressure during normal operation and also containment internal pressure during the hypothetical loss-of-coolant accident.

3. Load transfer around penetrations is based on maintaining continuity of main reinforcing bars which is accomplished by bending of reinforcing to ensure the transfer of tensions, bending moments and shares. At the equipment access opening, a reinforced concrete boss is provided to carry stresses around the opening and to resist bending and torsional moments created by the load transfer. Again, main reinforcement is bent to maintain continuity of stress to ensure load transfer.

4. The line is basically not a load-carrying member and because of its integral relationship with the reinforced concrete is subjected to the strains which the reinforced concrete imposes upon it. Therefore, the criterion at penetrations is one of consistent deformations rather than transfer of load. Nevertheless, the liner is reinforced at each penetration according to the rules set forth in the ASME Unfired Pressure Vessel Code, Section VIII UG-36. An additional conservatism is that the reinforcing requirements set forth in the ASME Code are based on unequal bi-axial stresses, whereas the liner principal stresses, being dependent on reinforcing bar strains, are essentially equal. For the penetrations the maximum stress at the opening is essentially the same as the average nominal stress of the liner.

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5. The weldments of liner to penetration sleeve are of sufficient strength to accommodate the stress raisers around the openings. These welds shall adhere strictly to ASME Section VIII requirements for both type and strength. In addition, each weld has a channel placed over it (for pressurization and ultimate leak testing) which adds strength and stiffness to the welded area and assists in reducing stress in the weld and liner plate.

4.6.2.1 Penetration Loading

The penetration sleeves and end plates are designed to accommodate all loads imposed on them. These loads include the following:

1. Internal pressure

2. Concentrated loads imposed by the sleeve anchors to the concrete as the anchors strain in conjunction with wall movement under both operating and accident conditions.

3. Thermal effects due to both gradient and thermal reactions of the particular item passing through the sleeve.

4. Shear, bending and compression due to accident end pressures.

5. Shear and bending due to seismic movements of the particular item passing through the penetration.

The sleeve and expansion joint are designed to remain within the stress limitations imposed by ASME Code Section VIII.

In addition, pipes which penetrate the containment building wall and which are subject to machinery originated vibratory loadings, such as the reactor coolant pumps, will have their supports spaced in such a manner that the natural frequency of the piping system immediately adjacent to the penetrations will be greater than the dominant frequencies of the pump. Pipe line

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vibration will be checked during preliminary plant operation; and where necessary vibration dampers will be fitted. This checking and fitting will effectively eliminate vibrating loads as a design consideration.

4.6.2.2 Design Computations

Stresses in the penetration sleeves and the liner to which they are attached is determined by compatibility of deformation between the liner and the sleeve. The radial deformation in a plate subject to biaxial stresses is determined by performing an integration of the tangential strains around the periphery of the hole.

$$\sigma_\theta = S - 2S\cos 2_\theta + \left[S - 2S\cos(2\theta - T)\right] \qquad (4.6.1)$$

Where:

σ_θ = tangential stress at the boundary of the hole defined at the angle θ from the horizontal axis

S = horizontal stress in the liner

S' = vertical stress in the liner

The displacements are determined

$$\delta D = \frac{1}{E}\int_0^\pi \left(S - 2S\cos 2\theta + \left[S' - 2S'\cos(2\theta - \pi)\right]\right) r\sin\theta d\theta \qquad (4.6.2)$$

$$\delta D = \frac{r}{E}\int_0^\pi S\sin\theta d\theta - 2S\int_0^\pi \cos 2\theta\sin\theta d\theta + S'\int_0^\pi \sin\theta d\theta - 2S'\int_0^\pi \cos(2\delta - \pi)\sin\theta d\theta \qquad (4.6.3)$$

$$\begin{aligned}\int \cos(2\theta - \pi)\sin\theta d\theta &= -\int \cos 2\theta \sin\theta d\theta \\ &= -\int\left(1 - 2\mathrm{Sin}^2\theta\right)(\mathrm{Sin}\,\theta)\, d\theta \\ &= -\int\left(\mathrm{Sin}\,\theta - 2\mathrm{Sin}^3\theta\right)d\theta \\ &= -\left[(-\cos\theta) - 2\left(\frac{\mathrm{Sin}^2\theta\cos\theta}{3} + \frac{2}{3}\int \sin\theta d\theta\right)\right] \qquad (4.6.4) \\ &= -\left(-\cos\theta + \tfrac{2}{3}\sin^2\theta\cos\theta + \frac{4}{3}\cos\theta\right)\Big]\end{aligned}$$

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$$\int cos(2\theta-\pi)sin\theta d\theta = \frac{-cos\theta}{3} - \frac{2}{3}sin^2\theta cos\theta$$

$$\delta = \frac{r}{E}\left[-S\,cos\theta - 2S\left(\frac{cos\theta +}{3}\frac{2}{3}sin^2\theta cos\theta\right) - S'cos\theta - 2S'\left(\frac{-cos\theta}{3} - \frac{2}{3}sin^2\theta cos\theta\right)\right]_o^{\pi} \quad (4.6.5)$$

$$\delta = \frac{r}{E}\left[\left(S + \frac{2}{3}S + S' - \frac{2}{3}S'\right) - \left(-S - \tfrac{2}{3}S - S' + 2/3S'\right)\right]$$

$$\delta = \frac{r}{E}\left[2S + \frac{4}{3}S + 2S' - \frac{4}{3}S'\right]$$

$$\delta = \frac{r}{E}\left[\frac{10}{3}S + \frac{2}{3}S'\right]$$

$$\delta = \frac{2}{3}\frac{r}{E}[5S + S'] \quad \text{(For Stresses in the Same Direction)} \quad (4.6.6)$$

$$\delta = \frac{2}{3}\frac{r}{E}(5S - S') \quad \text{(For Stresses in Opposite Directions)} \quad (4.6.7)$$

The composite deformation determined in the plate and the sleeve and the resultant stress in the liner and sleeve is determined:

$$^{\Delta}UN = {}^{\Delta}P \;\text{(restrained)}\; + {}^{\Delta}S \quad (4.6.8)$$

$$^{\Delta}UN = \frac{S_1}{E}(1-\nu)R + \frac{S_1(t)R^2\lambda}{2Et_s}$$

$$^{\Delta}UN = \frac{S_1}{E}\left[R(1-\nu) + \frac{{}^{t}pR^2\lambda}{2\quad t_s}\right]$$

$$S_1 = \frac{{}^{\Delta}UN^{E}}{R\left[(1-v)+\left(\frac{{}^{t}p\,R\lambda}{2t_s}\right)\right]} \quad (4.6.9)$$

$$S_s = \frac{S_1 t_p\,R\lambda}{2t_s} \quad (4.6.10)$$

$$S_S = \frac{{}^{\Delta}UN^{E}\,{}^{t}p\,R\,\lambda}{R\left[(1-\upsilon)+\left(\frac{{}^{t}p\,R\lambda}{2t_s}\right)\right]} \quad 2t_s$$

where:

$$\lambda = 4\sqrt{\frac{3(1-v^2)}{R^2 t_s^2}}$$

${}^{\Delta}UN$ = unrestrained deflection

${}^{\Delta}p$ = deflection of liner plate

${}^{\Delta}s$ = deflection of sleeve

S_1 = stress in liner

R = radius of penetration

υ = poissons ration

t_p = thickness of liner plate

t_s = thickness of sleeve

E = modulus of elasticity

A summary of liner and penetration stresses is shown in Table 4.3. The assumptions assumed in design are as follows:

1. The liner alone was designed for stress concentration effects while the cracked concrete was ignored.

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2. The unrestrained growth is based on maximum growth from a stress concentration consideration.

3. The main stream and mechanical penetrations have been considered in a non-insulated zone when they are just inside the insulated zone. The compression in the hoop direction will be greatly reduced or perhaps go into tension, thus reducing the stresses.

4. The allowable stress in the sleeve = 56,700 psi except for the stainless steel fuel transfer penetration = 49,500 psi. These values come from Table N-421 and Figure N-414 of the ASME Nuclear Vessel Code Section III.

In addition, thermal loads have been investigated for their effect on the shell adjacent to the penetration sleeve and found to be insignificant (38 psi bearing stress on the concrete is the maximum stress on the concrete shell).

4.7.0 CONTAINMENT CYLINDER, BASE AND DOME AT POINTS OF DISCONTINUITY

Discontinuity stresses occur at changes in section or direction of the containment shell. The juncture of the cylinder to the dome is a point of discontinuity since, under the internal pressure and temperature design conditions, the cylinder will tend to increase in diameter somewhat differently than the dome. To compute the unrestrained dimensional changes, the dome and cylinder have been considered as steel membranes equivalent to the area of reinforcing steel in the hoop direction. As shown in Section 3.1.3, the unrestrained radial deformation of the dome and cylinder are nearly equal therefore the discontinuity moments and shears are insignificant and there is no steel required at the dome to cylinder juncture due to the discontinuity effects.

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The juncture of the cylindrical wall and the base mat is also a point of discontinuity. In determining the discontinuity moments and shears, the base mat was considered as offering complete fixity, therefore the only discontinuity is that due to the unrestrained radial expansion of the cylinder. As for the dome to cylinder juncture, the unrestrained radial expansion of the cylinder has been computed by considering the cylinder to be a steel membrane equivalent to the area of reinforcing steel in the hoop direction. The method of analysis for the discontinuity moment and shear and its distribution into the cylindrical walls is given in Section 3.1.3.

The maximum discontinuity moment at the base occurring under the 1.5P factored load condition is 1210 K.FT/FT and the maximum discontinuity shear is 157/K/FT. The limiting discontinuity moments and shears are distributed as shown in Figures 2.2., 2.3 and 2.4 of this report. The placement of steel to carry discontinuity shears and moments is shown in Figures 4.20 and 4.21.

The required area of shear reinforcing as determined from Eq. 17-6 of the ACI Code is given in Eq. 3.2.7. The allowable value of f_y used as the basis for f_y in Eq. 3.2.7 is reduced from 60 KSI to 47 KSI since part of the stress is assumed taken by the axial force due to uplift. The point where the minimum web reinforcement required is less than the .15 percent of the area b_s the provisions of ACI-318 Code Article 1706b apply.

The allowable shear which may be taken by the concrete alone is found from Article 1701 e) of the ACI Code and is given by:

$$v_c = 3.5 ø \quad f_c \; (1 + 0.002N/A_g)$$

where:

v_c = allowable shear stress carried by the concrete
f_c = concrete design compressive strength
N = load normal to the cross section where N is negative for tensile loads
A_g = gross area of cross section

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References

(1) Roark, R. J., Formulas for Stress and Strain, 4th Ed. McGraw Hill Book Co., New York, 1965.

(2) Diable Canyon Unit No. 1, Pacific Gas and Electric Company, Docket No. 50-275, Supplement No. 4, Section III.

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5. CONTAINMENT MATERIAL PROPERTIES, FABRICATION AND ERECTION PROCEDURES

5.1 CONCRETE

Concrete used in the containment structure was designed to have minimum compressive strengths in 28 days of 3000 psi and 4000 psi. The concrete mixes were designed to produce strengths of fifteen percent above the minimum design strengths as determined by the average strengths of three laboratory tests of the specified design mixes including satisfactory plasticity qualities.

The minimum cement factor specified was 5 sks/cu. yd. for 3000# Class and 6 ¼ sks/cu. yd for 4000# Class. The maximum slump permitted was limited to 5 inches, except in localized regions of extreme congestion where 7 inch slump was permitted. Concrete was prepared in ready mix equipment conforming to ASTM Specifications C94.

5.1.1 CEMENT

The cement used was Portland Cement Type II conforming to ASTM designation C-150. Cement used in the ready mix batch process was stored in weather-proof bins so as to prevent deterioration or contamination.

5.1.2 WATER

Concrete mix water was supplied from the drinking water supply of the city of Verplank, New York, and as such is clean, clear and free of significant impurity.

5.1.3 AGGREGATES

Fine aggregate consisted of sand conforming to the requirements of ASTM Specification C-33.

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Typical properties of the sand are as follows:

SIEVE ANALYSIS

Sieve Sizes	% Passing by wt.	ASTM C-33 Specifications
3/8"	100	100
#4	97.8	95-100
#8	84.9	80-100
#16	61.8	50-85
#30	42.7	25-60
#50	18.1	10-30
#100	3.3	2-10
Fineness Modulus	2.91	
Specific Gravity (SSD)	2.67	
Absorption %	0.7	
Clay Lumps %	Negative	1.0 Max.
Coal & Lignite %	Negative	0.5 Max.
Material Finer than		
No. 200 Sieve %	0.6	3.0 Max.
Organic Impurities	Standard	Standard
Soundness 5 Cycles, % Loss	10.9	
Unit wt. (dry-rodded) lbs/ft^3	104.3	

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Coarse aggregate consisted of crushed gravel conforming to the requirements of ASTM Specification C-33. Typical properties of the crushed gravel are as follows;

COARSE AGGREGATE

30% — 40% Crushed Gravel

SIEVE ANALYSIS

Sieve Sizes	% Passing by wt.	ASTM C-33 Specifications
1 ½"	100.0	100
1"	97.2	95-100
3/4"	71.5	-
1/2"	30.9	25-60
3/8"	12.4	-
#4	0	0-10
Fineness Modulus	7.16	
Specific Gravity	2.67	
Absorption %	0.7	
Clay Lumps %	Negative	0.25 Max.
Soft Particles %	Negative	5.0 Max.
Unit wt. (dry-rodded) lbs/ft^3	102.2	
Magnesium Sulfate Soundness		
5 Cycles, % loss	14.8	18 Max.
Los Angeles Abrasion, % loss	41.7	50 Max.

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5.1.4 ADMIXTURES

The admixtures used in the concrete mix design was a plasticizer "Placewell" manufactured by the Union Carbide Corporation. The plasticizer is provided to increase ease of concrete placement in highly congested areas. Air entraining admixture used was Aircon when specified in concrete design mix.

5.1.5 PLACEMENT AND CURING

Placing and Curing of concrete conform to the provisions of Chapter 6 of the ACI 318-63.

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5.2. REINFORCING STEEL

Reinforcing steel used for the dome, cylindrical walls and base mat is high-strength deformed billet steel bars conforming to ASTM Designation A-615 "Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength." This steel has a minimum yield strength of 60,000 psi, a minimum tensile strength of 90,000 psi, and a minimum elongation of 7 percent in an 8-in. specimen. The design limit for a tension member (i.e., the capacity required for the design load) was based upon the yield stress of the reinforcing steel. No steel reinforcement experiences average strains beyond the yield point at the factored load except in local areas when subjected to temperature at accident conditions. The load capacity so determined has been reduced by a capacity reduction factor "ø" which provides for the possibility that small adverse variations in material strengths, workmanship, dimensions, and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in under capacity. For tension numbers, the factor "ø" was 0.95, 0.90 for flexure and 0.85 for diagonal tension, bond and anchorage.

5.2.1 CADWELD SPLICES

All reinforcing bar design to carry membrane tension or in the size range 14S and 18S where jointed by means of mechanical butt splices known as a Cadweld splice which is a standard commercial product manufactured by Erico Products Inc., Cleveland, Ohio. All splices used are designed to develop the specified minimum ultimate strength of the ASTM A-615 reinforcing bar or greater even though the specified requirement on splice strength was set at 125 percent of specified minimum yield (83.3 percent of minimum ultimate).

The mean value of the ultimate strength of splices made during any time period shall be equal (as a minimum) to 75,000 psi, plus the standard deviation in strength from the mean ultimate strength. In addition, the mean value of the ultimate strength and the standard

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deviation shall show, by statistical analysis, that at least 99.0% of all of the splices will have an ultimate strength of 60,000 psi or greater.

Splices shall be monitored by the following procedure. Any splice which, in the judgment of the inspector, does not pass visual inspection shall be cut out and replaced.

a. Bar ends shall be approximately square. They may be torch-cut, sawed or sheared. The cut faces of both re-bar, when inserted into the sleeve, shall be entirely within the specified limits for the size of the bar.

b. Bar ends shall be cleaned of dirt, oil, moisture, concrete, or heavy rust, to a degree of cleanliness as represented by heating the end of the bar uniformly to a surface temperature of 200°F to 300°F, power wire brushing to bare metal, reheating to the same temperature range, and hand wire brushing to remove any resulting dust and/or loose material.

c. The re-bars shall be assembled with their sleeve immediately after cleaning and properly aligned.

d. Preheating is not generally required; however, if the air temperature is below 40°F and/or the humidity is above 80%, the bar ends and sleeve shall be preheated to 100°F in order to remove moisture.

e. If it is necessary to remove a portion of the longitudinal rib on the re-bar in order to fit it into the splicing sleeve, the metal shall be removed by grinding only. In no case shall the entire rib be removed nor shall there be any under-cutting of the rib into the stock material of the re-bar.

Containment wall splices are staggered as specified on the UE & C drawings.

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5.3 FORMWORK

Concrete form work was erected to conform to the shape, lines and dimensions of the concrete elements as called for on the drawing and sufficiently tight to prevent leakage of mortar.

For all permanently exposed surfaces of concrete the form facing was constructed of new unscarred plywood, re-used plywood in good condition or metal pans. Forms were removed in such a manner and at such a time as to insure the complete safety of the structure. No areas of the containment concrete structure are in contact with backfill.

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5.4 CONTAINMENT LINER

5.4.1 MATERIAL

The steel liner plate is carbon steel conforming to ASMT Designation A-442 "Standard Specification for Carbon Steel Plates with Improved Transition Properties," Grade 60. This steel has a minimum yield strength of 32,000 psi and a minimum tensile strength of 60,000 psi with an elongation of 22 percent in an 8-in gauge length at failure. The liner is ¼-in. thick at the bottom, ½-in. thick in the first three courses except ¾-in. thick at penetrations and 3/8-in. thick for remaining portion of the cylindrical walls and ½-in. thick in the dome. The liner material was impact tested at a temperature 30°F lower than the minimum operating temperature of the liner material. For the liner steel the factor "ø" was 0.95 for tension and compression.

5.4.2 FABRICATION

The steel liner plate was fabricated from hot rolled plate in the Greeneville, Pennsylvania and New Castle, Delaware shop of the Chicago Bridge and Iron Co. The plate was shop fabricated into approximately 9' by 30' section and rolled to desired curvature. The Nelson stud anchors were welded to the containment liner shell after the plate was erected.

5.4.3 ERECTION

The difference between the minimum and maximum inside diameters at any cross section does not exceed 0.25 percent of the nominal diameter at the cross section under consideration. Maximum diameter 135'-2", minimum diameter 134'-10" below elevation +95. Above +95 tolerance does not exceed .50 percent of the nominal diameter of cross section under consideration. The liner was erected true and plumb not to exceed 1/500 of height at cross section under consideration with allowance for 2" buckling in the plates.

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Particular care was taken in matching edges of cylindrical and hemispherical sections to insure that all joints were properly aligned. Maximum permissible offset of completed joints was 25 percent of nominal plate thickness.

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5.5 LINER INSULATION

To protect the lower portion of the containment liner from severe temperature changes under accident conditions, the first 18 feet (approximately) of the liner is covered with insulation. The basic insulation selected is 7/8" thick urethane foam covered with a 1/2" thick gold bond fire shield gypsum board and a .019" thick stainless steel jacket and backed with asbestos paper cover on the unexposed side.

The insulation was designed to meet the following operational requirements:

1. Normal operating temperature - 120°F.

2. Under accident conditions rise in liner temperature not to exceed 80°F above ambient.

3. Insulation panels rated non-burning in accordance with ASTM procedure D-1692.

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5.6 PENETRATIONS

In general, containment penetrations for pipe, electrical conduit, duct or access hatches consist of sleeves imbedded in the concrete section and welded to the containment liner. The weld to the liner is shrouded by a continuously pressurized channel which is used to assure the leak tightness of the penetration to liner weld joint. Differential expansion between sleeve and pipes passing through is accommodated by bellows type expansion joints between the outer end of the sleeve and the outer plate.

5.6.1 MATERIALS

The materials for penetrations, including the personnel and equipment access hatches together with mechanical and electrical penetrations, will be carbon steel, conform with the requirements of the ASME Nuclear Vessels Code and exhibit ductility and welding characteristics compatible with the main liner material. As required by the Nuclear Vessels Code, the penetration materials were Charpy V-notch impact tested to a minimum of 15 ft-lbs at 50°F.

The stainless steel bellows of the hot penetration expansion joints will be protected from damage in transit and during construction by sheet metal covers fastened in place at the fabricator's shop. These can be left in place permanently if there is no interference with nearby piping or equipment.

The specific materials used in penetrations may be found in Section 2.2.6.

5.6.2 DESIGN

Those portions of penetrations not backed up by concrete are designed to meet the requirements of ASME Code Section VIII. Those portions of penetrations backed up by concrete are designed considering strains and stresses compatible with the deformation

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of the concrete wall sections and as such have the same governing design criteria as does the containment liner. As such, no primary load strains greater than the guaranteed yield point under factored loads are permitted. However, strains due to stress concentrations and other localized secondary load effects are limited to 0.5 percent strain.

5.6.3 FABRICATION

The qualifications of welding procedures and welders have been in accordance with Section IX, "Welding Qualifications" of the ASME Boiler and Pressure Vessel Code. The repair of defective welds has been in accordance with Para. UW-38 Section VIII "Unfired Pressure Vessels."

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6. QUALITY CONTROL METHOD AND PREOPERATIONAL TEST PROCEDURES

6.1 QUALITY CONTROL ORGANIZATION AND CHAIN OF COMMAND

The responsibility for implementation of the on-site quality control program for WEDCO rests with the Manager – Site Quality Control who reports directly to the Reliability Manager who in turn reports directly to the WEDCO executive Vice President.

Reporting directly to the Manager-Site Quality Control at the project site are Quality Control Engineers assigned primarily to a specific discipline (e.g., Concrete, Structural, Mechanical, Electrical, and Piping/Welding), Quality Control Inspectors, Clerks, and subcontracted testing service personnel.

No one in this quality control chain of command is directly responsible for production or construction schedules.

WEDCO Site Quality Control and/or the subcontracted testing service personnel conduct the first level inspection and test of all construction of structural elements of the vapor containment building except for the field fabrication and erection of the containment liner and penetrations where the construction subcontractor to WEDCO has first level responsibility subject to audit and surveillance by the WEDCO Site Quality Control.

In all cases, all quality control activity is audited by the Prime Contractor, Westinghouse Electric Corp., the Owner, Consolidated Edison Co., and the owner's Surveillance Group, United States Testing Laboratories.

All necessary records and documentation are compiled and maintained by the WEDCO Site Quality Control.

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6.2. SUMMARY OF MATERIAL TEST RESULTS

6.2.1 CONCRETE

Minimum design strengths of 3000 psi and 4000 psi are specified. To date no test cylinders strengths under 3000 psi or 4000 class 28 day strength have been determined. One set of six test cylinders to include three 7 day and three 28 day test cylinders have been tested per each 100 cubic yards placed. Approximately 10,000 cubic yards of 3000# Class and 2000 cubic yards of 4000# Class of concrete have been placed in the containment structure to date.

6.2.2 REINFORCING STEEL

Material mill test reports are required for each heat of steel received. Results of all tests show conformance with ASTM specification requirements. In addition to the mill test reports, random heats of no's 11, 14 and 18 bars are user tested. All tests have met minimum specified strength requirements.

6.2.3 STRUCTURAL STEEL

Various types of structural steel were furnished and erected. Structural steel was furnished to ASTM Specification in job lots substantiated by mill certification covering each job lot.

6.2.4 INSULATION

Letters of certification covering material requirements substantiated by test results are furnished by the manufacturer.

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6.2.5 CONTAINMENT LINER

All heats of steel used in the fabrication of the liner plate are covered by mill test certificates showing chemical analysis, mechanical test results, and Charpy impact test results.

Each liner plate is marked or coded to a specific heat of steel. These heat numbers are recorded on the as-built drawings. Material control (heat number) continuity is maintained by subcontractor and checked by WEDCO.

The same method of heat identification, certification, and recordation is maintained for the penetration material as for the liner plate.

Weld rod control (only E 7018 rod used on liner plate) is maintained by subcontractor and audited by WEDCO Site Quality Control.

Dimensions of erected material are checked by the WEDCO engineers and recorded on marked-up drawings. Any dimension found out of tolerance is reported to the subcontractor, corrected and rechecked by the survey group. The correction procedure is approved by Engineering.

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6.3 QUALITY CONTROL TESTS ON FABRICATED ELEMENTS

6.3.1 LINER, PENETRATIONS, LOCKS, AND EQUIPMENT HATCH

Nondestructive testing of these items consists of the following:

Coupon Testing – In locations on the liner where radiography is not possible, such as floor plates, and lower course of the shell where back-up plates are used, the subcontractor welds a 2" long overrun coupon which is broken off, marked for location and given to WEDCO for destructive examination or radiography.

a. Vacuum Box Test

Bottom liner plate welds and all liner plate seam welds in the cylindrical walls utilizing back-up plates are vacuum box tested with at least 5 psi pressure differential by the subcontractor. No leaks are permitted. (If any portion of a weld seam is inaccessible for vacuum box testings other forms of NDT will be applied.)

b. Strength Tests

After successful vacuum box testing or spot radiography all liner plate weld channels (bottom, cylinder, and dome) are welded on the seam weld and the channel welds tested by pressurizing the channel with air at 54 psig for 15 minutes. No leaks are permitted. Strength testing shall be by predetermined zones, and includes channels and gaskets of the personnel locks.

c. Leak Test

After strength tests of liner seam welds and channels, these welds and penetration sleeve weld channels, and personnel lock weld channels are leak tested by

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pressurization to 47 psig with a 20% by weight Freon-Air mixture. The entire run of plate weld and the channel to plate welds are then traversed with a halogen leak detector.

The sensitivity of the leak detector is 1×10^{-9} standard cc per second. Any halogen indication indicates a leak requiring repair and retest. In addition, the zone of channels tested is held at test pressure for at least 2 hours, with no indication of drop in pressure.

The strength and leaks tests are also performed on the gaskets and seals on the lock penetrations by pressurizing the space between the gaskets and seals as above.

6.3.2 CADWELDS

All Cadwelds are visually inspected by WEDCO Site Quality Control or its Q.C. Sub-contractor site. Details of Cadwelding operations, operator qualification criteria, testing frequencies and criteria, inspection procedures and acceptance standards are included in the UE &C procedure - "Recommended Procedure for the Testing of Mechanical Type Splices for Concrete" reinforcing bars. Each splice shall be visually inspected in accordance with the following procedure:

a. Properly made splices will have filler metal visible at both ends of the sleeve and at the tap hole in the center of the sleeve.

b. Filler metal will not flow to the very edge of the sleeve due to the gasket action of the asbestos wicking used to seal in the molten filler metal. A recess less than 1/2" will not be cause for rejection.

c. As a result of the Cadweld process, a shrinkage bubble may be visible at the tap hole where the molten metal is introduced and shrinkage fissures and pinholes may be

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visible at the top of splices. These casting flaws do not adversely affect the physical performance of the splice and, therefore, do not constitute cause for rejection.

Bars or splices which do not meet the requirements above shall be rejected and removed from the structure.

Before any Cadweld crew can be assigned to production work, they shall demonstrate their ability to produce splices meeting the specification requirements.

Each new crew shall be qualified using approved materials and procedures by making five splices of each type and tested to destruction.

Each Cadweld crew shall be qualified to do specific work only to the extent or having performed satisfactory qualification splices, for each category of crew can make only this type of splice. Any crew having prior qualification for the four types of splices (horizontal-straight, horizontal-reducing, vertical-straight, vertical-reducing) shall be deemed capable of making any type of splice required by the project.

Each crew shall be assigned an identification number and this number shall not be re-issued during the life of the project.

For purposes of this work a crew is defined as an operator who has been qualified in accordance with the above procedure and who shall be assigned a competent helper.

Cadweld splices shall be capable of developing tension at least 125 per cent of the specified yield strength of the reinforcing bar, in accordance with the requirements of ACI 318-63, Section 805-d.

Individual splices which do not meet 125% of yield shall be rejected.

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6.3.3 STUD ANCHORS ON THE LINER

A procedure is set up whereby after qualification, the first stud welded each day by each welder is tested by cold bending the stud to an angle of 45°. This is repeated after the lunch break.

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6.4 PREOPERATIONAL PERFORMANCE TESTING

6.4.1 STRUCTURAL INTEGRITY TEST

After completion of the vapor containment structure the building will be pressurized with air to 54 psig (115% of the design pressure of 47 psig). At pressure levels of 12, 21, 41, and 54 psig, gross deformations are determined and visual inspections are performed. Crack pattern and spacing measurements will be made to determine deformation behavior of the containment. These results will be correlated with the results obtained from the structural test behavior of Unit No. 2 as the test proceeds to ensure that structural behavior of Unit No. 3 is comparable to the successful testing of Unit No. 2.

Instrumentation will consist of invar wire extensometers inside the containment capable of measurement of movement of ± 0.01 in. and mechanical feeler gages used to measure crack width with ± 0.002 in. accuracy.

The range of strains and deformations expected vary from 0 to the following expected maximums:

	In.
Vertical elongation (top of mat to top of dome)	1.5
Increase in Diameter	2.0
Crack Width	1/16
Uniform Strain	.002 in/in.

In the quadrant of "boss" around the equipment hatch and personnel lock and in the 10' wide by 5' high areas of the base wall intersection, at the mid-height of the wall, and at the wall dome intersection, concrete surface will be sandblasted or acid etched and detailed measurements of crack width and spacing shall be recorded prior to pressurization at 54 psig, and immediately following depressurization.

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6.4.2 CONTAINMENT INTEGRATED LEAK RATE TEST

Following the completion of the Structural Integrity Test (at 54 psig), the containment pressure will be lowered to 50% (min) of the calculated peak accident pressure for purposes of performing the reduced pressure Integrated Leak Rate Test. The 24-hour test will be conducted with the Weld Channel and Penetration Pressurization System depressurized and open to the inside of the containment building. The containment pressure will then be raised to 100% (min) of the calculated peak accident pressure and leakage again determined under the same conditions established during the 50% integrated leak test to correlate leakage rates for purposes of in-service testing.

In addition, a Sensitive Leak Rate Test will be performed to assess the capability of the Weld Channel and Penetration Pressurization System to limit building outleakage. This leak test will be conducted with the containment building at ambient conditions, and the Weld Channel and Penetration Pressurization System pressurized at a pressure greater than the calculated peak accident pressure. Leakage will be limited to less than equivalent 0.2% of the building free volume per day during accident conditions.

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CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1 GENERAL DESIGN CRITERIA

Criteria applying in common to all engineered safety features are given in Section 6.1.1. Criteria which are related to engineered safety features, but which are applicable to specific features or systems, are listed and cross referenced in Section 6.1.2.

The engineered safety features are discussed in detail in this Chapter. In each section a separate safety feature is described and evaluated. In the evaluation section for each engineered safety feature, a single failure evaluation is provided which delineates the components of that safety feature system and the interconnected auxiliary systems that must function for the proper operation of that engineered safety feature. An examination of these tables shows that some components of the Residual Heat Removal System, Component Cooling Water System, and the Service Water Systems are necessary for proper operation of the Engineered Safety Features. These systems and their components are discussed in Sections 9.3 and 9.6; the instrumentation associated with these systems is also discussed in the referenced sections. Since the auxiliary system components, both inside and outside the containment, and their instrumentation and power systems are not required for actuation of the engineered safety features, neither IEEE-279 nor the General Design Criteria apply.

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, and in effect at the time of study, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

6.1.1 Engineered Safety Features Criteria

Engineered Safety Features Basis for Design

Criterion: Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends. (CDC 37 of 7/11/67)

The design, fabrication, testing and inspection of the core, reactor coolant pressure boundary and their protection systems give assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions. However, engineered safety features are provided in the facility to back up the safety provided by these components. These engineered safety

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features were designed to cope with any size reactor coolant pipe break, up to and including the circumferential rupture of any pipe, assuming unobstructed discharge form both ends, and to cope with any steam or feedwater line break, up to and including the main steam or feedwater headers.

Limiting the release of fission products from the reactor fuel is accomplished by the Safety Injection System which, by cooling the core, keeps the fuel in place and substantially intact, and limits the metal water reaction to an insignificant amount.

The Safety Injection System consists of high and low head centrifugal pumps driven by electric motors, and passive accumulator tanks which are self actuated and which act independently of any actuation signal or power source.

The release of fission products from the containment is limited in three ways:

Blocking the potential leakage paths from the containment. This is accomplished by:

A steel-lined, reinforced concrete Reactor Containment with testable, doubly sealed penetrations and most liner weld channels, the spaces of which are continuously pressurized above accident pressure, and which form a virtually leak-tight barrier to the escape of fission products should a loss-of-coolant accident occur.

Isolation of process lines by the Containment Isolation System which imposes double barriers in each line which penetrates the containment except for lines utilized during the accident. An Isolation Valve Seal Water System provides a water or nitrogen seal at the isolation valves thus sealing some of the pipes penetrating the containment.

Reducing the fission product concentration in the containment atmosphere. This is accomplished by:

a) Containment Air recirculation filters which provide for rapid removal of particles and iodine vapor from the containment atmosphere.

b) Chemically treated spray which removes elemental iodine vapor from the containment atmosphere by washing action.

Reducing the containment pressure and thereby limiting the driving potential for fission product leakage. This is accomplished by cooling the containment atmosphere by the following independent systems:

a) Containment Spray System

b) Containment Air Recirculation and Cooling System

Reliability and Testability of Engineered Safety Features

Criterion: All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public. (GDC 38 of 7/11/67)

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A comprehensive program of plant testing was formulated for all equipment, systems and system controls vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, integrated tests of the system as a whole, and periodic tests of the actuation circuitry and mechanical components to assure reliable performance, upon demand, throughout the plant lifetime.

The initial tests of individual components and the integrated test of the system as a whole complemented each other to assure performance of the system as designed and to demonstrate the proper operation of the actuation circuitry.

Routine periodic testing of the engineered safety features components is performed as specified in the Technical Specifications.

Missile Protection

Criterion: Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40 of 7/11/67)

A Loss-of-Coolant Accident or other plant equipment failures might result in dynamic effects or missiles. For engineered safety features which are required to ensure safety in the event of such an accident or equipment failure, protection is provided primarily by the provisions which are taken in the design to prevent the generation of missiles. In addition, protection is also provided by the layout of plant equipment or by missile barriers in certain cases. See Chapter 5 for a discussion of missile protection. The dynamic effects associated with postulated pipe breaks in the Primary Coolant System (hot legs, cold legs, crossover legs) need not be a design basis (NCR SER dated March 10, 1986).

Injection paths leading to unbroken reactor coolant loops are protected against damage as a result of the maximum reactor coolant pipe rupture by layout and structural design considerations. Injection lines penetrate the main missile barrier, which is the crane wall, and the injection headers are located in the missile-protected area between the crane wall and the containment wall. Individual injection lines, connected to the injection header, pass through the barrier and then connect to the loops. Separation of the individual injection lines is provided to the maximum extent practicable. Movement of the injection lines, associated with rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.

The containment structure is capable of withstanding the effects of missiles originating outside the containment, and which might be directed toward it, so that no Loss-of-Coolant Accident can result from these missiles.

All hangers, stops and anchors were designed in accordance with ANSI B31.1 Code for Pressure Piping and ACI 318 Building Code Requirements for Reinforced Concrete which provide minimum requirements on material, design and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Additional information on the design and re-analyses of hangers, stops and anchors is presented in Section 16.3.

Where necessary to prevent pipe whip, restraints were installed with the proper arrangement and spacing to prevent a plastic hinge mechanism from forming as a result of the forces associated with a pipe rupture. Restraint spacing was determined by calculation of the unsupported pipe length resulting in a plastic hinge formation for two basic support arrangement

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and break location cases. Both slot and guillotine breaks were considered. Slot breaks are defined as instantaneous openings in the pipe parallel to the axis of the pipe with an opening length twice the length of the nominal pipe diameter and with an opening area equal to the area of the pipe interior cross-section.

Slot breaks were assumed to occur anywhere in the piping system, including fittings. Guillotine breaks are defined as instantaneous severance of the pipe cross-section and are assumed to occur any point of discontinuity in the piping system (such as valves, fittings and elbows). The pipe break loads were determined from

$$P' = P_O A$$

where P_O = system pressure

A = inside cross sectional area,

except for the main steam lines downstream of the flow limiting device, where force resultants are limited by the restriction of the flow limiting device. Such loads for the steam lines were taken as P' = 340 kips; and for the feedwater lines, P' = 200 kips. For both slot and guillotine breaks, restraints were spaced such that plastic hinge mechanisms cannot form in the piping system which would permit unrestrained rotation of the piping.

The restraints were designed such that the maximum applied load or stress be less than the lesser of the yield strength of the material or 0.67 times the rated ultimate load capacity of the support. High strength cable restraints were designed such that the maximum applied load be less than 0.4 times the rated ultimate load capacity of the cable. In those instances where the integrity of the restraint is also dependent on reinforced concrete anchorage, the concrete behavior limits are in accordance with ACI-318-63, Part IV-B, requirements and bearing stress is limited to 0.8 f'_c.

Vital equipment is protected from pipe whip by locating restraints on nearby high pressure lines such that the two free ends of a broken pipe cannot reach the equipment.

The plant arrangement provides the basic protection against pipe whip. The four loops of the primary coolant system are spaced to the maximum extent possible; the crane wall protects the reactor compartment from pipe whip in the annulus; pipe lines are run radially outward from the reactor compartment. Wherever possible, redundant engineered safeguards piping is physically separated so that a failure of one pipe and subsequent whipping cannot cause the failure of the second pipe. Where physical separation is impossible, for instance the Accumulator Tanks' discharge piping, both pipes are restrained in such a way that a plastic hinge cannot form in case of a double ended rupture.

Containment fan cooler units are separated from high pressure pipe lines by the floor at Elev. 68' –0".

Small lines are treated no differently from large lines in so far as containment isolation, separation, pipe whip protection, etc. Separation is provided where whipping of larger lines would otherwise result in damage to many small lines.

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Small lines having significant internal pressure are supported and restrained in a manner that would preclude any failure of the containment vessel from the failure of the small line. In addition, see Section 5.2 for the containment isolation provisions for these lines.

Engineered Safety Features Performance Capability

Criterion: Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public. (GDC 41 of 7/11/67)

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in manner to avoid undue risk to the health and safety of the public.

The extreme upper limit of public exposure is taken as the levels and time periods presently outlined in 10 CFR 100, i.e., 300 rem to the thyroid in two hours at the exclusion radius and 300 rem to the thyroid over the duration of the accident at the low population zone distance. The accident condition considered is the hypothetical case of a release of fission products per TID 14844. Also, the total loss of all outside power is assumed concurrently with this accident. With all engineered safety features systems functioning at full capacity, the offsite exposure would be within 10 CFR 20 limits.

Under the above accident conditions, the Containment Air Recirculation Cooling and Filtration System and the Containment Spray System are designed and sized so that either system operating with partial effectiveness is able to supply the necessary post-accident cooling capacity to assure the maintenance of containment integrity, that is, keeping the pressure below design pressure at all times, assuming that the core residual heat is released to the containment as steam. Partial effectiveness is defined as operation of a system with at least one active component failure. Both systems together, each operating with partial effectiveness, are capable of providing the necessary post-accident iodine removal such that the resulting off-site exposures are within the guidelines of 10 CFR 100.

Engineered Safety Features Components Capability

Criterion: Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a Loss-of-Coolant Accident to the extent of causing undue risk to the health and safety of the public. (GDC 42 of 7/11/67)

Instrumentation, pumps, fans, filters, cooling units, valves, motors, cables and penetrations located inside the containment were selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or were designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

The Safety Injection System pipes serving each loop are anchored at the crane wall, which constitutes the missile barrier in each loop area, to restrict potential accident damage to the portion of piping beyond this point. The anchorage was designed to withstand, without failure, the thrust force of any branch line, severed from the reactor coolant pipe and discharging fluid to the atmosphere; and to withstand a bending moment equivalent to that which produces failure

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of the piping under the action of free discharge to atmosphere or motion of the broken reactor coolant pipe to which the injection pipes are connected. This prevents possible failure at any point upstream from the support point including the branch line connection into the piping header.

Accident Aggravation Prevention

Criterion: Protection against any action of the engineered safety features which would accentuate significantly the adverse after-effects of a loss of normal cooling shall be provided. (GDC 43 of 7/11/67)

The reactor is to be maintained subcritical following a pipe rupture accident. Introduction of borated cooling water into the core results in a net negative reactivity addition. The control rods are inserted and remain inserted.

The supply of water by the Safety Injection System to cool the core cladding reduces the potential for significant metal-water reaction (less than 1.0%).

The delivery of cold safety injection water to the reactor vessel following accidental expulsion of reactor coolant does not cause further loss of integrity of the Reactor Coolant System boundary.

Sharing of Systems

Criterion: Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public (GDC 4 of 7/11/67)

The residual heat removal pumps and heat exchangers serve dual functions. Although the normal duty of the residual heat exchangers and residual heat removal pumps is performed during periods of reactor shutdown, during all plant operating periods these residual heat removal pumps are aligned to perform the low head safety injection function. In addition, during the recirculation phase of a Loss-of-Coolant Accident, the residual heat exchangers of this system perform the core cooling function and the containment cooling function as part of the Containment Spray System, and the residual heat removal pumps, which are part of the external recirculation loop, provide back-up capability to the recirculation pumps which comprise part of the internal recirculation loop.

Demonstration checking of the system, performed as dictated by the Technical Specifications, provides assurance of correct system alignment for the safety injection function of the components.

During the injection phase, the safety injection pumps do not depend on any portion of other systems. During the recirculation phase, if Reactor Coolant System pressure stays high due to a small break accident, suction to the safety injection pumps is provided by the internal recirculation pumps, and can also be provided by the Residual Heat Removal pumps.

The Containment Air Recirculation and Filtration System also serves the dual function of containment cooling during normal operation and containment cooling after an accident. Since the method of operation for both cooling functions is the same, the dual aspect of the system does not affect its function as an engineered safety feature.

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The steam supply and city water systems at the Indian Point site were shared by all three reactor facilities. However, independent steam supply and city water systems have been installed at Indian Point 3 (See Chapter 9); the city water system for Indian Point 2 is presently used by Indian Point 3 as a backup supply. The steam supply and city water systems are used for the following purposes:

a) Steam for unit heaters for standby heating.

b) Steam to valved hose connections for maintenance purposes.

c) Water to emergency showers.

d) Water to hose connections for maintenance purposes.

e) (Deleted)

f) Water supply to fire protection tanks.

g) Water supply for make-up demineralizers in Condensate Polishing Facility (CPF).

h) Redundant source of makeup water to the spent fuel pit.

i) Backup water supply to Charging Pumps' Fluid Drive Coolers.

6.1.2 Related Criteria

The following are criteria which, although related to all engineered safety features, are more specific to other plant features or systems, and therefore are discussed in other sections, as listed:

Title of Criterion (7/11/67 issue)	Reference
Quality Standards (GDC 1)	Chapter 4
Performance Standards (GDC 2)	Chapter 4
Records Requirements (GDC 5)	Chapter 4
Instrumentation and Control Systems (GDC 12)	Chapter 7
Engineered Safety Features Protection Systems (GDC 15)	Chapter 7
Emergency Power (GDC 39 and GDC 24)	Chapter 8

6.2 SAFETY INJECTION SYSTEM

6.2.1 Design Basis

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. On November 22, 1965, the Atomic Energy Commission (AEC) published and requested comments on Proposed

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General Design Criteria which were developed to assist in the evaluation of applications for nuclear power plant construction permits. On July 11, 1967, a revised set of General Design Criteria were published for comment. The revision reflected extensive public comments, suggestions from meetings with the Atomic Industrial Forum (AIF) and review within the AEC. In the July to October 1967 time frame, AIF Incorporated assembled nuclear industry comments and transmitted to the AEC revised wording of the 1967 Draft General Design Criteria along with a description of the changes. It was the AIF version of the 1967 General Design Criteria which formed the bases of the Indian Point 3 design and are discussed in this section. The AEC subsequently revised the 1967 version of the General Design Criteria and incorporated them into 10 CFR 50, Appendix A in 1971.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Emergency Core Cooling System Capability

Criterion 44: An Emergency Core Cooling System with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

Adequate emergency core cooling is provided by the Safety Injection System (which constitutes the Emergency Core Cooling System) whose components operate in three modes. These modes are delineated as passive accumulator injection, active safety injection and residual heat removal recirculation.

The system assures that the core will remain intact and in place with its essential heat transfer geometry preserved following a rupture in the Reactor Coolant System. It also assures that the extent of metal-water reaction is limited such that the amount of hydrogen generated from this source in combination with that from other sources, is tolerable in the Containment.

This capability is provided during the simultaneous occurrence of a Design Basis Earthquake. This protection is afforded for:

1) All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends,

2) or loss of coolant associated with the rod ejection accident,

3) or steam generator tube rupture.

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The primary function of the emergency Core Cooling System (ECCS) for the ruptures described is to remove the stored and fission product decay heat from the core such that fuel damage to the extent that would impair effective cooling of the core is prevented. This implies that the core remain intact and in place with its essential heat transfer geometry preserved. To assure effective cooling of the core, limits on peak clad temperature and local metal-water reaction will not be exceeded. It has been demonstrated in the Westinghouse Rod Burst Program that for conditions within the area of safe operation, fuel rod integrity is maintained.

To limit the production of hydrogen in the Containment, the overall metal water reaction is limited to 1%.

In evaluating ECCS performance, consideration was given to core geometry distortion caused by swelling or fuel rod bursting.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the Safety Injection System (SIS) adds shutdown reactivity so that with a stuck rod, no offsite power and minimum engineered safety features, there is no consequential damage to the Reactor Coolant System and the core remains in place and intact.

Redundancy and segregation of instrumentation and components are incorporated to assure that postulated malfunctions will not impair the ability of the system to meet the design objectives. The system is effective in the event of loss of normal station auxiliary power coincident with the loss of coolant, and is tolerant of failures of any single component or instrument channel to respond actively in the system. During the recirculation phase, the system is tolerant of a loss of any part of the flow path since backup alternative flow path capability is provided.

The ability of the Safety Injection System to meet its capability objectives is presented in Section 6.2.3. The analysis of the accidents is presented in Chapter 14.

Inspection of Emergency Core Cooling System

Criterion 45: Design provisions shall, where practical, be made to facilitate inspection of all physical parts of the Emergency Core Cooling System, including reactor vessel internals and water injection nozzles.

Design provisions are made to the extent practical in order to facilitate access to the critical parts of the reactor vessel internals, pipes, valves and pumps for visual or boroscopic inspection for erosion, corrosion and vibration wear evidence and for non-destructive test inspection where such techniques are desirable and appropriate as detailed in Section 6.2.5.

Testing of Emergency Core Cooling System Components

Criterion 46: Design provisions shall be made so that components of the Emergency Core Cooling System can be tested periodically for operability and functional performance.

The design provides for periodic testing or active components of the Safety Injection System for operability and functional performance as detailed in Section 6.2.5.

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Power sources are arranged to permit individual actuation of each active component of the Safety Injection System.

The safety injection pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The residual heat removal pumps are used every time the residual heat removal loop is put into operation and can be tested periodically. All remote operated valves can be exercised and actuation circuits can be tested during routine plant maintenance.

Testing of Emergency Core Cooling System

Criterion 47: Capability shall be provided to test periodically the operability of the Emergency Core Cooling System up to a location as close to the core as is practical.

An integrated system test is performed when the plant is cooled down and the residual heat removal loop is in operation. This test would not introduce flow into the Reactor Coolant System but would demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

Level and pressure instrumentation are provided for each accumulator tank, and accumulator tank pressure and level are continuously monitored during plant operation. Flow from the tanks can be checked at any time using test lines.

The accumulators and the safety injection piping up to the final isolation valve are maintained full of borated water at boron concentrations consistent with the accident analysis while the plant is in operation. The accumulators and injection lines are refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection headers. A small bypass line and a return line are provided for this purpose.

Flow in each of the high head injection branch lines and in the main flow line for the residual heat removal pumps is monitored by a flow indicator.

Pressure instrumentation is also provided for the main flow paths of the high head and residual heat removal pumps.

Testing of Operational Sequence of Emergency Core Cooling System

Criterion 48: Capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the Emergency Core Cooling System into action, including the transfer to alternate power sources.

The design provides for capability to test, to the extent practical, the full operational sequence up to the design conditions for the Safety Injection System to demonstrate the state of readiness and capability of the system. Details of the operational sequence testing are presented in Section 6.2.5, Tests and Inspections.

Engineered Safety Features

The Engineered Safety Features are discussed in detail herein. In each section of this Chapter 6 a separate safety feature is described and evaluated. In the evaluation section for each Engineered Safety Feature, a single failure table is provided which lists the components of that safety feature system and the interconnected auxiliary systems that must function for the proper

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operation of that Engineered Safety Feature. An examination of these tables shows that some components of the Residual Heat Removal System, Component Cooling System, and Service Water System are necessary for proper operation of the Engineered Safety Features. These systems and their components are discussed in Section 9.3 and 9.6. The instrumentation associated with these systems is also discussed in those sections. As the auxiliary system components outside the Containment, as well as those inside the Containment, their instrumentation and power systems are not required for actuation of the Engineered Safety features; neither IEEE-279 nor the General Design Criteria apply.

Codes and Classifications

Table 6.2.1 tabulates the codes and standards to which the Safety Injection System components were designed.

Service Life

All portions of the system located within the Containment were designed to operate without benefit of maintenance and without loss of functional performance for the duration of time the component is required.

6.2.2 System Design and Operation

System Description

Adequate emergency core cooling following a Loss-of-Coolant Accident is provided by the Safety Injection System as shown in Plant Drawings 9321-F-27353 and -27503 [Formerly Figures 6.2-1A & 6.2-1B]. The system components operate in the following possible modes:

1) Injection of borated water by the passive accumulators.

2) Injection of borated water from the Refueling Water Storage Tank with the safety injection pumps. (NOTE: Technical Specification Amendment 139 eliminates the requirement to maintain a boron injection tank.)

3) Injection by the residual heat removal pumps also drawing borated water from the Refueling Water Storage Tank.

4) Recirculation of spilled reactor coolant, injected water and Containment Spray System drainage back to the reactor from the recirculation sump by the recirculation pumps. (The residual heat removal pumps provide backup recirculation capability.)

The initiation signal for core cooling by the safety injection pumps and the residual heat removal pumps is the safety injection signal which is actuated by any of the following:

Low pressurizer pressure (2/3)

High containment pressure (2/3, High Pressure)

High differential pressure between any other two steam generators (2/3)

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After time delay (maximum of 6 seconds): high steam flow in any two of the four steam lines (1/2 per line) coincident with low T_{avg} (2/4) or low steam pressure (2/4)

Manual Actuation

High-High containment pressure (two sets of 2/3, High-High pressure) [energize to actuate]

In the Technical Specifications, limits are set on minimum number of operable channels and required plant status for all reactor protection and ESF instrumentation.

Injection Phase

The principal components of the Safety Injection system which provide emergency core cooling immediately following a loss of coolant are the accumulators (one for each loop), the three safety injection (high head) pumps and the two residual heat removal (low head) pumps. The safety injection and residual heat removal pumps are located in the Primary Auxiliary Building.

The accumulators, which are passive components, discharge into the cold legs of the reactor coolant piping when pressure decreases below the N_2 cover gas operating pressure (approximately 650 psig), thus rapidly assuring core cooling for large breaks. They are located inside the Containment, but outside the crane wall; therefore, each is protected against possible missiles.

The safety injection signal starts the safety injection and residual heat removal pumps and opens the Safety Injection System isolation valves (certain valves have their motor leads disconnected and are locked open). The valves on Plant Drawings 9321-F-27353 and -27503 [Formerly Figures 6.2-1A & -B] marked with a "S" receive the safety injection signal.

Separate and independent key-lock switches one for each SI train are provided in series to each of the auto SI actuation relays to allow manual blocking of the automatic Engineered Safeguards System actuation when the unit is in cold shutdown.

The operation of the key-lock switches into the "defeat" position will activate the existing separate annunciation for each train (Safeguard Train "A" in test and Safeguard Train "B" in test) and separate status lights (one for each train) in the Control Room. While the operator can deactivate the alarm, the individual status lights and the alarm windows will stay lit as long as the key-lock switches are in the defeat position.

The considerations involved insure that:

1) The operation of the key-lock switch to defeat the auto SI is normally carried out during the plant conditions which do not require the actuation of auto SI.

   The key-lock switch will be used only during normal plant operation, with the plant in the cold shutdown condition. The Technical Specifications do not require the operability of the SI system or any of its components during the cold shutdown conditions.

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2) The operation of the key-lock switch to defeat auto SI is also permitted following an SI activation if the normal method or resetting SI is unavailable. This action is required to restore control of plant equipment to the operators.

3) Annunciation devices are provided to augment the administrative procedures.

   The operation of the key-lock switch will activate the individual annunciations and individual status lights. During the time auto SI is in the "defeat" position, the "alarm windows" and the status lights will stay lit.

The safety injection pumps (high head) deliver borated water to two separate discharge headers. The flow from each header can be injected into each of the three available cold legs (one of four cold leg lines per header has been permanently isolated by locking closed valves SI-856A on 2" Line #56 and SI-856F on 1-½" Line #754, as evaluated in Reference 2) and one hot leg of the Reactor Coolant System. Isolation valves in each of the three available cold leg injection lines are open and valves in the hot leg injection lines are closed during normal plant operation. The hot leg injection lines are provided for later use during hot leg recirculation following a reactor coolant pressure boundary break.

One high head injection header contains the retired-in-place Boron Injection Tank (BIT), which formerly contained concentrated boric acid for rapid insertion of negative reactivity in the safety injection mode. A modification replaced the contents of the BIT with water from the Refueling Water Storage Tank (RWST), (Reference 3).

> NOTE: Technical Specification Amendment 139 eliminates the requirement to maintain a BIT.

However, the BIT is an in-line, passive component of the Safety Injection System, therefore it is only "functionally eliminated," but not physically. Furthermore, in the event of a safety injection scenario, the BIT will continue to "function" in a passive mode, to convey refueling water to the reactor core.

No credit is taken for any boron concentration in the BIT, or in any Safety Injection piping downstream of the RWST.

The BIT inlet and outlet isolation valves, (two pairs of motor operated valves, each pair arranged in parallel), are maintained in the open position, as their function to isolate the BIT is not required since implementation of the BIT elimination modification (References 2 through 5). Maintaining the BIT isolation valves open provides the benefits of eliminating an active safety function and potentially minimizing the time delay in delivery of safety injection flow. The BIT isolation valves may be individually closed for testing (one of each pair at a time) during normal power operation. Only one inlet isolation valve (SI-1852A or B) and one outlet isolation valve (SI-1835A or B) must be open to achieve the emergency core cooling safety function.

However, closing a single BIT isolation MOV presents the potential for loss of the function of the BIT header should a coincident spurious or inadvertent closure of the parallel BIT isolation MOV occur. Spurious or inadvertent mis-positioning of MOVs are considered to be credible single failures. When configured with both parallel BIT inlet or outlet MOVs closed simultaneously, the motor actuators' calculated capabilities lack the opening margin required by the GL 89-10 program. Therefore, in accordance with the IP3 GL 89-10 program, if any of the BIT isolation valves are to be closed in support of maintenance or testing, then the potential for loss of BIT

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header function via closure of the parallel valve (as caused by any reason including single failure) must be eliminated by administrative means.

A Safety Injection Signal still generates a signal to open the BIT isolation valves. However, based on the limited margin available in the capabilities of the motor actuators of these valves, opening in response to an SI signal would require modification in order to meet the margins required under the GL 89-10 program. Such modifications are unnecessary provided the normal position of the BIT isolation valves is open.

While refueling water has relatively low boron concentration (nominally 2,500 ppm), analyses performed to support implementation of the modification, assumed zero boron concentration in the BIT and associated piping, for conservatism. The Westinghouse "Revised Feasibility Report for BIT Elimination for Indian Point Unit 3," (July 1988) determined that the concentration of boron in the BIT may be reduced to that in the RWST while continuing to meet applicable safety criteria.

The high-head safety injection system is configured as follows:

1. The three available cold leg (one of the four cold leg lines has been permanently isolated by locking closed valve SI-856F on 1-½" Line #754) injection lines on the discharge header containing the retired-in-place BIT are physically connected to the reactor coolant pressure boundary.

2. The three available cold leg (one of the four cold leg lines has been permanently isolated by locking closed valve SI-856A on 2" Line #56) injection lines on the Non-BIT discharge header are physically connected to the accumulator discharge lines upstream of the reactor coolant pressure boundary.

3. The two hot leg injection lines on both discharge headers are physically connected to the reactor coolant pressure boundary.

This configuration was implemented in a modification installed during the 3R13 Refueling Outage to accommodate Stretch Power Uprate to provide for Hot Leg Switchover (HLSO) prior to 6.5 hours following a LBLOCA, and to resolve sump particle and ECCS valve erosion concerns identified in NRC Information Notice 96-27 and 97-76.

Since a small break in the reactor coolant pressure can include a cold leg injection line, safety injection flow capability can be limited by the resulting flow from only five available intact cold leg (Note: one of the original four cold leg lines per header has been isolated by a locked closed valve) injection lines. Depending on the assumed single failure, either two or three safety injection pumps can be operating. To maximize the fraction of safety injection flow delivered to the reactor coolant system with a broken cold leg injection line, the six available cold leg (Note: one of the original four cold leg lines per header has been isolated by a locked closed valve) injection lines are flow balanced to within an allowable range. The resulting system flow capability is sufficient for the makeup of coolant following a small break that does not immediately depressurize the reactor coolant system to the accumulator discharge pressure. Credit is not taken for operator action to isolate a broken cold leg injection line.

For large breaks, the Reactor Coolant System would be depressurized and voided of coolant rapidly (about 26 seconds for the largest design break) and a high flow rate is required to quickly recover the exposed fuel rods and limit possible core damage. To achieve this

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objective, one residual heat removal pump and two safety injection pumps are required to deliver borated water to the cold legs of the reactor coolant loops. Two pumps are available in order to provide for an active component failure. Delivery from these pumps supplements the accumulator discharge. Since the Reactor Coolant System back pressure is relatively low (rapid depressurization for large breaks), a broken injection line would not appreciably change the flows in the other injection lines delivering to the core.

The residual heat removal pumps take suction from the refueling water storage tank.

Because the injection phase of the accident is terminated before the refueling water storage tank is completely emptied, all pipes are kept filled with water before recirculation is initiated. Water level indication and alarms on the refueling water storage tank give the operator ample warning to terminate the injection phase. Additional level sensors are provided in the containment sump which also give backup indication when injection can be terminated and recirculation initiated.

Recirculation Phases

After the injection operation, coolant spilled from the break and water collected from the containment spray is cooled and returned to the Reactor Coolant System by the recirculation system.

Following a Loss-of-Coolant Accident (LOCA), sampling is accomplished as necessary from outside of the Containment via the sampling connection from the recirculating pump discharge.

When the break is large, depressurization occurs due to the large rate of mass and energy loss through the break to Containment. In the event of a large break, the recirculation flow path is within the Containment. The system is arranged so that the recirculation pumps take suction from the recirculation sump in the containment floor and deliver spilled reactor coolant and borated refueling water back to the core through the residual heat exchangers. The system is also arranged to allow either of the residual heat removal pumps to take over the recirculation function. The residual heat removal pumps would only be used if backup capacity to the internal recirculation loop is required. Water is delivered from the Containment to the residual heat removal pumps from a separate sump inside the Containment.

Although the residual heat removal pump is an acceptable alternative for providing core cooling and containment spray flow in lieu of the recirculation pump, there is no single failure that would require its use. The residual heat removal pump(s) would be used only in scenarios beyond the design basis involving multiple active failures. Use of a residual heat removal pump during the long-term recovery phase could be required in the event of ECCS leakage outside Containment.

The motor operated valves in the recirculation suction lines from the containment sump are maintained in the normally closed position at all times, however, they could be opened to allow for residual heat removal pump recirculation operation if that mode was required.

The valves are exercised in accordance with Technical Specification requirements. The valves are operated one at a time and each valve is returned to its normal position before exercising the next one.

No automatic opening features are provided; hence, the probability of a spurious signal to open the valves is nil. The only time these valves are opened is for periodic testing and the

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procedure ensures that both valves are closed immediately after the test. In addition, the two valves are provided in series to protect against the inadvertent opening of one valve.

The procedure used for periodic testing of these valves ensures that the only water which would be drained from these lines is the small amount trapped between the two valves. This water will discharge to the containment sump. The sump contains two sump pumps which operate on level control and will periodically pump the sump contents to the waste holdup tank during normal plant operation.

For small breaks the depressurization of the Reactor Coolant System is augmented by steam dump and auxiliary feed water addition to the Steam System. For the small breaks in the Reactor Coolant System where recirculated water must be injected against higher pressures for long term core cooling, the system is arranged to deliver the water from the residual heat exchangers to the high-head safety injection pump suction and, by this external recirculation route, to the reactor coolant loops. Thus, if depressurization of the Reactor Coolant System proceeds slowly, the safety injection pumps may be used to augment the flow-pressure capacity of the recirculation pumps in returning the spilled coolant to the reactor.

The recirculation pumps, the residual heat exchangers, piping and valves vital to the function of the recirculation loop are located in a missile-shielded space inside the polar crane support wall on the west side of the reactor primary shield.

There are two recirculation related sumps within the Containment, the recirculation sump and the containment sump. Both sumps collect liquids discharged into the Containment during the injection phase of the design basis accident.

The recirculation sump contains two screens through which the recirculated water must flow before entering the pumps. The first screen consists of a floor grating (1" x 4") which covers the sump on the basement floor. The purpose of the grating is to prevent large particles form entering the sump. The second screen is located in the sump and has the capability to exclude particles greater than 1/8 inch in diameter from the recirculation pump suction. This floor grating has a total surface area of 48.3 ft^2. Since all recirculated water passes through both screens before entering the pumps, particles in excess of 1/8 inch diameter are precluded from entering these lines. The water velocity through the sump is less than one foot per second.

The containment sump contains two screens for the purpose of preventing particles greater than 1/8 inch diameter from entering the residual heat removal pump suction. The first screen consists of 1" x 4" floor grating with an area of 41.3 ft^2; the second screen is located in the sump. The water velocity through the sump is less than one foot / second.

The low head external recirculation loop via the containment sump line and the residual heat removal pumps provides backup recirculation capability to the low head internal recirculation loop. The containment sump line is contained within a concentric guard pipe which is connected to the containment liner and terminates within a leak tight compartment. This sump line has two remote motor operated normally closed valves for containment isolation purposes, one of which is within this leak tight compartment.

The high head external recirculation flow path via the high head safety injection pumps is only required for the range of small break sizes for which the Reactor Coolant System pressure remains in excess of the shutoff head of the recirculation pumps (or residual heat removal pumps) at the end of the injection phase or to provide hot leg flow during hot leg recirculation.

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The external recirculation flow paths within the Primary Auxiliary Building are designed so that external recirculation can be initiated immediately after the accident. Those portions of the Safety Injection System located outside of the Containment which are designed to circulate under post-accident conditions radioactively contaminated water collected in the Containment meet the following requirements:

Shielding to maintain radiation levels within the guidelines set forth in 10 CFR 100

Collection of discharges from pressure relieving devices into closed systems

Means to detect and control radioactivity leakage into the environs to the limits consistent with guidelines set forth in 10 CFR 100.

This criterion is met by minimizing leakage from the system. External recirculation loop leakage is discussed in Section 6.2.3.

One pump (either recirculation or residual heat removal) and one residual heat exchanger of the recirculation system provides sufficient cooled recirculated water to keep the core flooded with water by injection through the cold leg connections while simultaneously providing, if required, sufficient containment spray flow to prevent the containment pressure from rising above design limits because of the boiloff from the core. Only one pump and one heat exchanger are required to operate for this capability at the earliest time recirculation is initiated. The design ensures that heat removal from the core and Containment is effective in the event of a pipe or valve body rupture.

Cooling Water

The Service Water System (Section 9.6.1) provides cooling water to the component cooling loop, which in turn, cools the residual heat exchangers, all of which are part of the Auxiliary Cooling Systems (Section 9.3). Three conventional service water pumps are available to take suction from the river and discharge to the two component cooling heat exchangers. Three component cooling pumps are available to discharge through their heat exchangers and deliver to the two residual heat exchangers. With the component cooling water system in long term recirculation mode, the following components are required in order to meet core cooling requirements, one residual heat removal pump and heat exchanger, one component cooling water pump, one component cooling water heat exchanger, one service water pump on the nonessential header, and two essential service water pumps on the essential header. All of this equipment with the exception of the residual heat exchangers is located outside Containment.

Containment Building Water Level Monitoring

Continuous indication of containment water level during and after an accident is provided by three systems with redundant measuring loops distributed as follows:

Containment Sump (El. 38' 3"), narrow range, 0' to 10' of water.

Recirculation Sump (El, 34' 0"), narrow range, 0' to 14' of water.

Containment Building (El, 46' 0"), wide range, 0' to 8' of water.

Each loop consists of a sensor and a transmitter located inside the containment building, a

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recorder and power supply at the control room. Refer to Plant Drawing 9321-F-27353 [Formerly Figure No. 6.2-1A].

Change-Over from Injection Phase to Recirculation Phase

Assuming that the three high head safety injection pumps, the two residual heat removal pumps, and the two containment spray pumps (Section 6.3) are running at their maximum capacity, the time sequence for the changeover from injection to recirculation in the case of a large rupture beginning form the time of the safety injection signal:

In approximately ten minutes, sufficient water has been delivered to provide the required NPSH to start the recirculation pumps.

In approximately 15 to 20 minutes, (1) one of two low level alarms on the RWST sounds, and the redundant containment recirculation sump level indicators show the sump water level. The alarm serves to alert the operator to start the switchover to the recirculation mode. The redundant containment recirculation sump level indicators provide verification the RWST water has been delivered during the injection phase, in addition to providing consideration to the case of a spurious (i.e., early) RWST low level alarm. The operator would see on the control board that the redundant recirculation sump level indications are at the appropriate points; switch-over to the recirculation phase of safety injection is performed at this time.

With the initiation of the switch sequence (e.g., Switch No. 1), only one spray pump will continue in operation. This spray pump will continue to draw from the RWST for approximately 25 minutes to assure that the contents of the spray additive tank have been completely mixed with the spray liquid.

Recirculation pump motors are 2'-2" above the highest water level after addition of the injected water to the spilled coolant.

The entire switchover from injection to recirculation phase is carried out by manually initiating equipment starts/stops and closing a series of switches (each of which carries out several operations) located in the Control Room. At only two points in the switchover routine is any reliance on instrumentation necessary:

1) When the low level alarm occurs from the refueling water storage tank low level (LT-920 and/or LIC-921 outside containment), the operator is alerted to begin closing the series of recirculation switches. The low level alarm for each instrument is set to actuate when there is between 10.5 feet and 12.5 feet of water in the tank.

2) After closing switch 4, the operator is required to make a decision whether to close switch 5 or switch 6. The basis for this decision is the flow reading on the flow meters FT-946A, 946B, 946C and 946D. If three or more of these flow meters each indicate greater than zero and the lowest of these readings is at least 360 gpm, $\pm$ 10 gpm, the operator will close switch 6; otherwise, the operator will close switch 5.

Analysis indicates that approximately 662 gpm to the core is required to match boil-off at 1398 seconds (the earliest time at which recirculation could be initiated). This includes a 20% penalty to allow for the effects of hot metal quenching. The flow rates that follow ensure an actual flow of $\geq$662 gpm. Accordingly, a requirement of 360 gpm, $\pm$ 10 gpm minimum flow rate

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on the lowest indicating loop has been specified to account for uncertainties in flow measurement and to provide margin.

The decision making process with regard to the flow to the Reactor Coolant System via the low head injection lines is based on readings of the four injection line flowmeters. The rationale for this basis is the following:

For four flow meters each reading greater than zero (i.e., none indicating zero flow):

1) Assume one flow meter fails to an inaccurate high reading, (as a result of a single failure); if the flow rate reads greater than 360 gpm, ± 10 gpm then this meter is not used as a basis for the 360 gpm, ± 10 gpm setpoint, but rather, the lowest indicating meter (greater than 0 gpm) is used.

2) Of the four injecting lines, the highest flow line is assumed to be connected to the spilling line; therefore, flow from this line is ineffective.

3) For the three remaining intact lines, their total flow of 751 gpm is delivered to core; if at least 360 gpm, ±10 gpm is indicated by the lowest line, total flow requirement of 662 gpm is satisfied using low head recirculation.

For one (or two) flow meters each reading greater than zero (i.e., two indicating zero flow):

4) Assume failure of one or two flow meters due to loss of common power supply (i.e., single failure). If the flow rate reads greater than 360 gpm, ± 10 gpm then this meter is not used as a basis for the 360 gpm, ± 10 gpm setpoint, but rather, the lowest indicating meter (greater than 0 gpm) is used.

5) Of the four injecting lines, the highest flow line is assumed to be connected to the spilling line; therefore, flow from this line is ineffective.

6) For the three remaining intact lines, their total flow of 751 gpm is delivered to the core; if at least 360 gpm, ± 10 gpm is indicated by the lowest line; total flow requirement of 662 gpm is satisfied using low head recirculation.

Thus, the entire recirculation switchover sequence requires only the aforementioned instrumentation. Manual switchover, that is, without use of the "eight-switch sequence," does not require any additional instrumentation; it requires only that the operator close a switch for each and every operation in the switchover sequence.

The control circuitry for the associated valves and pumps is located external to the Containment. The motor operators, associated power cables, and instrumentation inside the Containment have been designed to withstand the LOCA environment as stated in Appendix 6F.

Several motor operated valves operated during the transfer to cold leg or hot leg recirculation are maintained de-energized in their safeguards position during normal power operation in accordance with the Technical Specifications. The subject valves are 856B, 856G, 1810, 882, 744, 842, 843, 883, 1870, 743, 894A, 894B, 894C, and 894D. Each valve (except for the submerged 894C) may be energized during or following the transfer to recirculation at a point in

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time when spurious or inadvertent mispositioning would not defeat a safety function relied upon to mitigate the consequences of the event.

The manual switchover by the operator which accomplishes the changeover from injection to recirculation is listed below. This switchover takes place when the level indicator or level alarms on the refueling water storage tank indicates that the fluid has been injected. The level indicators in the containment sump will verify that the level is sufficient within the Containment. The sequence is followed regardless of which power supply is available. The time required to complete the switchover to recirculation is the time for the switch gear to function. All the recirculation switches are grouped together on the safeguard control panel. The service water pumps and component cooling water pumps are located on the auxiliary coolant panel. The component position lights verify when the function of a given switch has been completed. Should an individual component fail to respond, the operator can take corrective action to secure appropriate response from the back-up component.

The following sequence maintains the loads on the 480V buses within analyzed limits, maintains sufficient core cooling flow during and following the transfer to recirculation and ensures that components are operated within their analyzed limits. While this sequence does not attempt to mirror each procedural step, the major steps listed below must be performed in the sequence described:

1) Terminate safety injection actuation signal and containment spray actuation signal in order that the control logic permits manipulation of the system (at any time following completion of the auto start sequence)

2) Close switch one (remove and isolate unnecessary loads from the diesels).

   Trips high head safety injection pump No. 32 if all three are operating (no action if two are operating). Isolates pump No. 32 from the Refueling Water Storage Tank.

   Trips spray pump No. 32 if both are operating (no action if only one is operating).

   Closes isolation valve at the inoperative spray pump discharge.

3) Close switch three (remove and isolate unnecessary loads from the diesels).

   Trips both residual heat removal pumps.

   Closes isolation valves at pump suction and discharge headers (the Technical Specifications require the motor operators for these valves to be de-energized).

4) Secure electric auxiliary feedwater pumps prior to closing switch two.

   a) If continued Auxiliary Feedwater flow is required, the turbine driven pump is used.

   If continued, Auxiliary Feedwater flow is required and the turbine driven pump is unavailable, only one motor driven Auxiliary Feedwater pump may be run.

5) Close switch two (establish cooling flow for Residual Heat Exchangers)

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a) Starts on one non-essential service water pump (the second or third pump is given a start signal if the first or second pump fails to start).

b) Starts one component cooling water pump (the second or third pump is given a start signal if the first or second pump fails to start).

6) Manually initiate internal recirculation flow.

a) Manually start recirculation Pump A (if Pump A fails to start, use manual start for Pump B; Pump B control switch is adjacent to switch four).

b) Close switch four to open valves on discharge of recirculation pumps. Starting a Recirculation Pump prior to closing switch four minimizes the potential pressure differential across these motor operated valves.

c) Valves SI-HCV-638 and / or SI-HCV-640 are throttled to maintain recirculation flow. For one pump operation, throttling is required to maintain recirculation pump flow within maximum pump flow limits.

7) Check Flow to Reactor Coolant System via the low head injection lines.

a) For the preferred operating mode of omitting switch five and closing switch six (i.e., provides recirculation at low system pressure), the following flow conditions must be verified:

1) With flow in three or more lines greater than zero, the lowest of these flows is at least 360 gpm, $\pm$ 10 gpm.

b) If the above flow conditions are met, the following actions are taken:

1) Direct operators in the field to throttle service ware valves SWN-35-1 and 35-2 to maintain CCW temperature within prescribed limits.

2) Close switch six, which trips operating safety injection pumps.

c) If the above flow conditions are not verified, close switch five and omit switch six (provides recirculation at elevated system pressure).

1) Aligns flow from residual heat exchanger to high head safety injection pumps. (The motor-operated valves on the outlet of the residual heat exchangers to the suction of the high-head safety injection pumps are opened. The motor-operated valves on the outlets of the residual heat exchangers to the low-head injection lines are closed together with the safety injection pump mini-flow and residual heat removal pump mini-flow).

2) Direct operators in the field to throttle service water valves SWN-35-1 and 35-2 to maintain CCW temperature within prescribed limits.

8) Close switch seven

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Starts a second non-essential service water pump.

Starts a second component cooling water pump only if the second service water pump successfully started.

Starts a second recirculation pump only if the second component cooling water pump successfully started.

9) Close switch eight (complete the isolation of the safety injection system and containment spray system test lines to the refueling water storage tank).

a) Close the valve on the spray test line.

b) Close the valve in the safety injection pumps suction line from the Refueling Water Storage Tank (control power for this valve is de-energized as required by the Technical Specifications).

If an RHR pump is used for low head recirculation, then valves SI-HVC-638 and/or SI-HCV-640 are throttled to maintain RHR pump flow to the cold legs (and recirculation spray) less than the maximum pump flow limit. Section 9.6.1 describes additional requirements to be met relating to alignment and operation of the service water system at the beginning of the post- LOCA recirculation phase.

Although the listed recirculation switches are manual, each automatically causes the operations listed. An indicating lamp is provided to show the operator when the operations of a given switch have been performed and when he should proceed with the next switching operation. In addition, lamps indicating completion of the individual functions for a given switch are provided. These lamps are adjacent to the switches. The time required to complete the switchover is just the time for the switch gear to operate. Should an individual component fail to respond, the operator can take corrective action to secure appropriate response from controls within the Control Room. Remote operated valves for the injection phase of the Safety Injection System (Table 6.2-11) which are under manual control (that is, valves which normally are in their ready position and do not receive a safety injection signal) have their positions indicated on a common portion of the control board. At any time during operation when one of these valves is not in the ready position for injection, it is shown visually on the board. Reference is made to Table 6.2-11 which is a listing of the instrumentation readouts on the control board which the operator can monitor during recirculation. In addition, an audible annunciation alerts the operator to the condition.

Hot leg recirculation is initiated after 4 hours but prior to 6.5 hours.

<u>Location of the Major Components Required for Recirculation</u>

The residual heat removal pumps re located in the residual heat removal pump of the Primary Auxiliary Building (El. 15'). The residual heat exchangers are located on a platform above the basement floor of the Containment Building (El. 66').

The recirculation pumps are located directly above the recirculation sump in the Containment Building (El. 46').

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The component cooling pumps and heat exchangers are located in the Primary Auxiliary Building (El. 41' and 73', respectively).

The service water pumps are located in the intake structure and the redundant piping to the component cooling heat exchangers is run underground.

Steam Break Protection

A large break of a steam system pipe rapidly cools the reactor coolant causing insertion of reactivity into the core and depressurization of the system. Compensation is provided by injection of borated water from the refueling water storage tank (RWST). Redundant isolation valves open upon a safety injection signal, providing a supply of borated water with a boron concentration of 2500 ppm nominally. Even assuming all of the safety injection lines downstream of the RWST, including the BIT, contain unborated water, this is sufficient to terminate the reactor power transient before any clad damage results. The analysis of the steam line rupture accident is presented in Section 14.2.5.

Components

All associated components, piping, structures, and power supplies, of the Safety Injection System were designed in accordance with the seismic criteria provided in Section 16.1.1 and were predominately designated as seismic Class 1. Refer to Plant Drawing 9321-F-27353 and -27503 [Formerly Figures 6.2-1A and 1B] for indication of the seismic class piping boundaries.

All components inside the Containment are capable of withstanding or are protected from differential pressure which may occur during the rapid pressure rise to 47 psig in 10 seconds.

Emergency core cooling components are austenitic stainless steel, and hence, are quite compatible with the spray solution over the full range of exposure in the post-accident regime. While this material is subject to crevice corrosion by hot, concentrated caustic, the NaOH additive cannot enter the containment or Emergency Core Cooling Systems without first being diluted and partially neutralized with boric acid to a mild solution. Corrosion tests performed with simulated spray showed negligible attack, both generally and locally, in stressed and unstressed stainless steel at containment and ECCS conditions. These tests are discussed in WCAP-7153[1].

The quality standards of all Safety Injection System components are tabulated in summary form in Table 6.2-12.

Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation each accumulator is isolated from the Reactor Coolant System by two check valves in series. Should the Reactor Coolant System pressure fall below the accumulator operating pressure, the check valves open and borated water is forced into the Reactor Coolant System. Mechanical operation of the swing-disc check valves is the only action required to open the injection paths from the accumulators to the core via each cold leg.

Indian Point 3 does not utilize hot leg injection. No timer is involved or provided. The two hot leg connections are provided to allow hot leg recirculation. However, these connections are closed at all times during plant operation.

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The level of borated water in each accumulator tank is adjusted remotely as required during normal plant operation. During normal plant operation, the fluid level can be reduced by draining through the Sampling System to the Sample Sink in the PAB. The water level can also be reduced by draining to the reactor coolant drain tank or to the VC sump; however, these drain paths degrade the accumulator function by exposing the affected accumulator(s) to non-seismic piping from which it can not be isolated in accordance with design criteria. Draining accumulators to the Reactor Coolant Drain tank or the VC sump may only be performed under the conditions delineated by the plant Technical Specifications. To increase and/or maintain the accumulator water level, refueling water is added using a safety injection pump. Samples of the solution in the tanks are taken at the sampling station for periodic checks of boron concentration.

The accumulators are passive Engineered Safety Features because the gas forces injection; no external source of power or signal transmission is needed to obtain fast-acting, high flow capability when the need arises. On accumulator is attached to each of the cold legs of the Reactor Coolant System.

The design capacity of the accumulators is based on the assumption that flow from one of the accumulators spills onto the containment floor through the ruptured loop. The flow from the three remaining accumulators provides water after the end of blowdown, to reflood the core. (Section 14.3)

The accumulators are carbon steel, internally clad with stainless steel and designed to ASME Section III, Class C. Connections for remotely draining or filling the fluid space during normal plant operation are provided.

Redundant level and pressure indictors are provided with readouts on the control board. Each indicator is equipped with high and low level alarms.

For the Accumulator Discharge Valves (894 A, B, C, D), the following indications are provided to supervise the administrative procedures and to highlight the existence of an incorrect configuration:

1) Red (open) and Green (closed) position indicating lights at the control switch for each valve. These lights are powered by valve control power and actuated by valve motor operator limit switches.

2) An additional indicating system of lights is used, whereby each valve has a two light sugar cube. The right side of the cube is a WHITE light (which glows PINK if the adjacent RED light is lit) that indicates power applied to the indicating system; the left side of the cube is a RED light to indicate when the respective valve is in its proper position enabling safeguards operation. This grouping highlights a valve not properly lined up. These lights are energized from a separate monitor light supply and the RED light is actuated by a valve motor operator limit switch.

3) In the event a valve is closed for accumulator or valve testing at the time injection is required, a safety injection signal is applied to open the valve (if power is available), overriding the test closure.

Prior to commercial operation, the AEC required that the electric power to these valves be locked out to prevent a spurious closure. The lockout will be implemented whenever RCS

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temperature is above 350°F. These valves are closed during plant shutdown conditions to isolate the pressurized accumulators from the depressurized reactor coolant system.

The accumulator design parameters are given in Table 6.2-2.

Boron Injection Tank

The Boron Injection Tank (BIT) is "functionally" retired-in-place. That is, it is no longer relied upon to provide concentrated boric acid for injection into the reactor core during emergency core cooling. It does, however, remain a passive component of the Safety Injection System, and as such, it is relied upon for its properties as a pressure vessel. Because the BIT no longer contains concentrated boric acid, the specialized handling requirements associated with that substance, such as heating and recirculation, no longer need to be met. The heaters which reside at the bottom of the BIT have been permanently de-energized. Furthermore, the recirculation flowpath between the BIT and the Boric Acid Storage Tanks has been valved off.

The BIT inlet and outlet isolation valves (two pairs of motor operated valves, each pair arranged in parallel) are maintained in the open position, as their function to isolate the BIT is not required since implementation of the Reference 3 modification. A Safety Injection Signal still generates a signal to open the BIT isolation valves. However, based on the limited margin available in the capabilities of the motor actuators of these valves, opening in response to a SI signal would require modification in order to meet the margins required under the GL 89-10 program. Such modifications are unnecessary provided the normal position of the BIT isolation valves is open.

The BIT remains in the safety injection flowpath, and continues to be relied upon to convey the water contained in it, as well as water from the Refueling Water Storage Tank, in the same manner as a section of piping would. Although the BIT contents are identical to the contents of the RWST, that is, borated water with a nominal boron concentration of 2,500 ppm, no credit is taken by the core response analyses for any boron in the BIT, or the safety injection piping downstream of the RWST, for conservatism.

The design parameters of the BIT are presented in Table 6.2-3.

Refueling Water Storage Tank

In addition to its normal duty to supply borated water to the refueling canal for refueling operations, this tank provides borated water to the safety injection pumps, the residual heat removal pumps and the containment spray pumps for the Loss-of-Coolant Accident. During plant operation, it is aligned to these pumps.

The capacity of the refueling water storage tank is based on the requirement for filling the refueling canal. When filled to Technical Specification requirements, approximately 342,200 gallons is available for delivery. One low level alarm is set to actuate at between 10.5 feet and 12.5 feet of water in the tank. This tank capacity and these alarm settings provide an amount of borated water to assure:

1) A sufficient volume of water on the floor to permit the initiation of recirculation (195,800 gal).

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2) A volume sufficient to allow switchover to recirculation pumps, containment pressure relief, and sump pH control via containment spray system following a reactor coolant pressure boundary break (66,700 gal).

3) Adequate volume to allow for instrument uncertainties (total 52,100 gallons between the Technical Specifications minimum RWST level of 35.4' and the nominal containment spray shutoff point of 1.5'. Of this volume, 26,100 gallons are eventually added to the Containment, but the remaining 26,000 gallons are considered by the analysis to be unusable.

4) The total RWST volume, when added with accumulator discharge to the reactor coolant system, will assure no return to criticality with the reactor at cold shutdown and no control rods inserted into the core.

The water in the tank is borated to a concentration which assures reactor shutdown by at least 5% $\in$k/k when all RCC assemblies are inserted and when the reactor is cooled down for refueling. The maximum boric acid concentration is approximately 1.5 weight percent boric acid. At 32°F the solubility limit of boric acid is 2.2%. Therefore, the concentration of boric acid in the refueling water storage tank is well below the solubility limit at 32°F.

The contents of the Refueling Water Storage Tank are kept above 32°F by a steam heated, austenitic stainless steel pipe coil in the bottom of the tank. Steam is supplied to this coil through a single header from the auxiliary boilers which are used to supply all required auxiliary steam to Indian Point 3.

The passive heating coil and passive single supply header are supplied with steam from any one of five sources. In the remote case of loss of steam to this tank, there would be a time period of at least 24 hours available for repair or connection to another steam source before freezing problems would arise, even under the most severe weather conditions. If the electrical heat tracing on the tank discharge line remains operable it is very probable that a freezing problem would not arise.

The steam to the heating coil is automatically flow controlled to maintain a minimum tank water temperature of 35°F. In response to low RWST temperature, steam is admitted by temperature control valve TCV-1116, and the pressure is controlled automatically by pressure control valve PCV-1250 to maintain a nominal 7 psig steam pressure in the coil (see Plant Drawing 9321-F-27273 [Formerly Figure 9.6-16]). During normal and shutdown operations when the tank is filled with borated water, the water pressure outside the heating coil will be approximately 15 psig, thereby preventing leakage of steam out of the coil and subsequent dilution of the borated water.

All outdoor piping connected to the Refueling Water Storage Tank is electrically heat traced. The failure of any section of heat tracing is annunciated in the Control Room. The power source for the heat tracing can be manually switched between two MCC's, each powered automatically by different emergency diesel generator buses.

The design parameters are presented in Table 6.2-4

<u>Pumps</u>

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Class I (seismic) pumps in the Emergency Safeguards Systems, their required Net Positive Suction Head (NPSH) at extreme operating conditions, the fluid operating temperature, the NPSH available, the atmospheric pressure assumption, and the elevation of each pump are given in Table 6.2-13.

The Internal Recirculation Pump NPSH data in Table 6.2-13 are given for a single pump operation, which represents the most limiting case for maximizing flow and NPSHR. As shown on the Figure 6.2-4 pump curve, an NPSHR of 12.7 ft is required for a flow of 3530 gpm. These pumps were evaluated to operate under cavitating conditions. The pump vendor has confirmed that reduced levels of NPSH are acceptable (Reference 7), such that the Recirculation Pumps can operate indefinitely with an NPSH value @90% of that required on pump curve (Figure 6.2-4). The effective limit for NPSHR thus becomes 11.4 ft. at an actual flow of 3530 gpm at the pump nozzles. See additional discussion of Recirculation Pump NPSH in Section 6.2.3.

An analysis predicts that, for the large break LOCA, there will be 11.4 ft. of NPSH available, which credits the remainder of the RWST water delivered to the containment prior to the start of recirculation containment spray (References 8, 9, 10). In the case of a small-break LOCA, when elevated RCS pressure would preclude direct low head recirculation, high head recirculation would then be established using the Recirculation Pump(s) to deliver a suction supply to the SI Pumps. During high head recirculation, the Recirculation Pumps operate at lower flow rates and the NPSH requirements are correspondingly lower.

NPSH calculations assume saturated water in the sumps so that no credit is taken for containment pressure exceeding the vapor pressure of the sump water. While this conservative assumption is appropriate at accident initiation, it does not allow any credit for the increase of NPSHA which would result from the gradual cooling of the sump fluid to below saturated conditions.

A review performed pursuant to NRC Generic Letter 85-22 had also established that the actual containment water level would be above the minimum switchover level indicated in Table 6.2-13. This actual water level would provide sufficient additional NPSH available to overcome the head loss effects of debris which has been postulated to result from the destruction of steam generator thermal insulation by LOCA jet forces.

The three (high head) safety injection pumps for supplying borated water to the Reactor Coolant System are horizontal centrifugal pumps driven by electric motors. Parts of the pump in contact with borated water are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event the pumps are started with the normal flow paths blocked. The bypass line joins a common miniflow line shared by the other pumps. Each safety injection pump is sized at 50% of the capacity required to meet the design criteria outlined in Section 6.2.1. The design parameters are presented in Table 6.2-5, and Figure 6.2-2 gives the performance characteristics of these pumps.

The two residual heat removal (low head) pumps of the Auxiliary Coolant System are used to inject borated water at low pressure to the Reactor Coolant System. The two recirculation pumps are used to recirculate fluid from the recirculation sump and send it back to the reactor, the spray headers or to suction of the safety injection pumps. All four of these pumps are of the vertical centrifugal type, driven by electric motors. Parts of the pumps which contact the borated water and sodium hydroxide solution during recirculation are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on the discharge of the

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residual heat exchangers to recirculate cooled fluid to the suction of the residual heat removal pumps should these pumps be started with their normal flow paths blocked. Additionally, each residual heat removal pump is provided with a dedicated recirculation line. These recirculation lines prevent either pump from operating at shutoff conditions and, also, during dual-pump operation preclude the stronger residual heat removal pump from dead heading the weaker pump as described in IE Bulletin 88-04. A minimum flow bypass discharging back into the recirculation sump, is provided to protect the recirculation pumps should their normal flow paths be blocked. Figure 6.2-3 and 6.2-4 give the performance characteristics of these pumps. The design parameters are presented in Table 6.2-5.

The safety injection pump bearings are cooled by booster pumps using component cooling water. The booster pumps are directly connected to the injection pump motor shaft. The pump seals were designed to operate at accident conditions without cooling water. Pump data is provided in chapter 9.3.

The recirculation pump motors are enclosed fan cooled. The air is cooled by coils utilizing component cooling water and four auxiliary component cooling pumps located outside the Containment. During recirculation the sump water cools the pump bearings. The four (i.e., two pairs) auxiliary component cooling pumps are started during the injection phase; either pump of a pair is capable of protecting its recirculation pump motor from the containment atmosphere. The fans are directly connected to the recirculation pump motor shafts. The auxiliary component cooling pumps are a part of the Component Cooling Water System and pump data is provided in Chapter 9. The component cooling water volume constitutes a large heat sink so that the main component cooling pumps are not needed during the injection phase (with loss of offsite power).

Details of the component cooling pumps and service water pumps, which serve the Safety Injection System, are presented in Chapter 9.

The pressure containing parts of the high head safety injection pumps are castings, conforming to ASTM A-296, Grade CA-15. The pressure containing parts of the Residual Heat Removal Pumps and the Recirculation Pumps are castings conforming to ASTM-296, Grade CF-8a (chromium content 21.0 to 22.5) and ASTM-296, Grade CF-8, respectively. Stainless steel forgings were procured per ASTM A-182, Grade F304 or F316, or ASTM A-336, Class F8 or F8M, and stainless plate was constructed to ASTM A-240, type 304 or 316. All bolting material conforms to ASTM A-193. Materials such as weld-deposited Stellite or Colmonoy were used at points of close running clearances in the pumps to prevent galling and to assure continued performance ability in high velocity areas subject to erosion.

All pressure containing parts of the pumps were chemically and physically analyzed and the results were checked to ensure conformance with the applicable ASTM specification. In addition, all pressure containing parts of the pump were liquid penetrant inspected in accordance with Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code.

The acceptance standard for the liquid penetrant test is ANSI B31.1, Code for Pressure Piping, Case N-10.

The pump design was reviewed with special attention to the reliability and maintenance aspects of the working components. Specific areas include evaluation of the shaft seal and bearing design to determine that adequate allowances had been made for shaft deflection and clearances between stationary parts.

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Where welding of pressure containing parts was necessary, a welding procedure including joint detail was submitted for review and approval by Westinghouse. The procedure included evidence of qualification necessary for compliance with Section IX of the ASME Boiler and Pressure Vessel Code Welding Qualifications. This requirement also applied to any repair welding performed on pressure containing parts.

The pressure-containing parts of the pump were assembled and hydrostatically tested to 1.5 times the design pressure for 30 minutes.

Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shut-off head and three additional points to verify performance characteristics. Where NPSH is critical, this value was established at design flow by means of adjusting suction pressure.

Pump Cooling Water Supply	
Pump	Source of Cooling Water
1. Internal Recirculation Pumps	Auxiliary Component Cooling Water pumps are used to deliver – 40 gpm to each motor cooler.
2. High Head Safety Injection Pumps	Booster pumps connected to the shafts of the SI pumps are designed to circulate – 40 gpm of CCW per pump.
3. Residual Heat Removal Pumps	Cooling water for seals is not required when the temperature of the pumped fluid is less than 150° F. This is the case during the injection phase after a LOCA. During the recirculation phase, when the pumped fluid temperature may be more than 150° F, a component cooling pump will be running to supply cooling water to the RHR pump.
4. Containment Spray Pumps	This pump pumps fluid with a temperature never in excess of 100° F. Therefore no cooling water is required.

The only period of concern when these pumps experience a lack of cooling water is during the injection phase following a LOCA, since during the recirculation phase and at all other times, the component cooling pumps will be available.

During the injection phase, the only heat removal requirement is for the high head safety injection pumps and for the internal recirculation pumps.

<u>Safety Injection Pumps</u>:

Component cooling water is required for cooling the bearings of these pumps. The heat load is estimated to be 75,000 Btu/hr per pump or a total of 225,000 Btu/hr for 3 pumps.

<u>Internal Recirculation Pumps</u>:

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Cooling water is required to protect the pump motors from the containment environment during a LOCA. The heat load is estimated to be approximately 150,000 Btu/hr per pump or a total of 300,000 Btu/hr for two pumps.

Since the component cooling pumps do not run during the injection phase, (with loss of offsite power), the water volume of the component cooling system is used as a heat sink. This heat load causes a temperature rise of approximately 7°F/hour in the component cooling water (no credit is taken for the water volume in the surge tank). With 110° F cooling water at the start of the accident, 6 hours are available before the cooling water temperature reaches 150° F; 10 hours are available before reaching 180° F.

Heat Exchangers

The two residual heat removal heat exchangers of the Auxiliary Coolant System cool the recirculated sump water. These heat exchangers were sized for the cool-down of the Reactor System. Table 6.2-6 gives the design parameters of the heat exchangers.

The ASME Boiler and Pressure Vessel Code has strict rules regarding the wall thickness of all pressure containing parts, material quality assurance provisions, weld joint design, radiographic and liquid penetrant examination of materials and joints, and hydrostatic testing of the unit as well as requiring final inspection and stamping of the vessel by an ASME Code inspector.

The designs of the heat exchangers also conform to the requirements of TEMA (Tubular Exchanger Manufacturers Association) for Class R heat exchangers. Class R is the most rugged class of TEMA heat exchangers and is intended for units where safety and durability are required under severe service conditions. Items such as: tube spacing, flange design, nozzle location, baffle thickness and spacing, and impingement plate requirements are set forth by TEMA Standards.

In addition to the above, additional design and inspection requirements were imposed to ensure rugged, high quality heat exchangers. The design and inspection requirements included: confined-type gaskets, main flange studs with two nuts on each end to ensure permanent leak tightness, general construction and mounting brackets suitable for the plant seismic design requirements, tubes and tube sheet capable of withstanding full shell side pressure and temperature with atmospheric pressure on the tube side, ultrasonic inspection in accordance with Paragraph N-324.3 of Section III of the ASME Code of all tubes before bending, penetrant inspection in accordance with Paragraph N-627 of Section III of the ASME Code of all welds and hot or cold formed parts, a hydrostatic test duration of not less than thirty minutes, the witnessing of hydro and penetrant tests by a qualified inspector, a thorough final inspection of the unit for good workmanship and the absence of any gouge marks or other scars that could act as stress concentration points, and a review of the radiographs and of the certified chemical and physical test reports for all materials used in the unit.

The residual heat exchangers are conventional vertical shell and U-tube type units. The tubes are seal welded to the tube sheet. The shell connections are flanged to facilitate shell removal for inspection and cleaning of the tube bundle. Each unit has a SA-285 Grade C carbon steel shell, a SA-234 carbon steel shell end cap, SA-213 TP-304 stainless steel tubes, a SA-240 type-304 stainless steel channel, a SA-240 type 304 stainless steel channel cover, and a SA-240 type 304 stainless tube sheet.

Valves

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All parts of valves used in the Safety Injection System in contact with borated water are austenitic stainless steel or equivalent corrosion resistant material. The motor operators on the injection line isolation valves are capable of rapid operation. All valves required for initiation of safety injection or isolation of the system have remote position indication in the Control Room.

Valving is specified for exceptional tightness, and where possible, instrument valves and packless diaphragm valves are used. All valves, except those which perform a control function, are provided with backseats which are capable of limiting leakage to less than 1.0 cc per hour per inch of stem diameter, assuming no credit for valve packing. Backseats can also be employed to facilitate repacking the valve stem. As a general rule, the plant relies on packing to minimize valve stem leakage. Normally closed globe valves are installed with recirculation flow under the seat to prevent leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed. Control and motor-operated valves which are 2-1/2 in and larger and which are exposed to recirculation flow are provided with double-packed stuffing boxes and stem leakoff connections which are piped to the Waste Disposal System.

The check valves which isolate the Safety Injection System from the Reactor Coolant System are installed immediately adjacent to the reactor coolant piping to reduce the probability of a safety injection line rupture causing a Loss-of-Coolant Accident.

A relief valve is installed in the safety injection pump discharge header discharging to the pressurizer relief tank in order to prevent overpressure in the lines which have a lower design pressure than the Reactor Coolant System. The relief valve is set at the design pressure of the safety injection piping.

The gas relief valves on the accumulators protect them from pressures in excess of the design value.

Motor Operated Gate Valves

The pressure containing parts (body, bonnet and discs) of the valves employed in the Safety Injection System were designed per criteria established by the ANSI B16.5 (1955) or MSS SP66 specifications. The materials of construction for these parts were procured per ASTM A182, F316 or A351, GR-CF8M or CF8. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or equivalent corrosion resisting material. The pressure containing cast components were radiographically inspected as outlined in ASTM E-446 Class 1 or Class 2. The body, bonnet and discs were liquid penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code Section VIII, Appendix VIII. The liquid penetrant acceptable standard was as outlined in ANSI B31.1, Case N-10.

When a gasket is employed, the body-to-bonnet joint was designed per ASME Boiler and Pressure Vessel Code Section VIII or ANSI B16.5 with a fully trapped, controlled compression, spiral wound asbestos or suitable material, gasket with provisions for seal welding, or of the pressure seal design with provisions for seal welding. The body-to-bonnet bolting and nut materials were procured per ASTM A193 and A194, respectively.

The entire assembled unit was hydrotested as outlined in MSS SP-61 with the exception that the test pressure was maintained for a minimum period of 30 minutes. The seating design of the Darling parallel disc design, the Crane flexible wedge design, or the equivalent. These designs have the feature of releasing the mechanical holding force during the first increment of

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travel. Thus, the motor operator has to work only against the frictional component of the hydraulic unbalance on the disc and against the packing box friction. The discs are guided throughout the full disc travel to prevent shattering and provide ease of gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear.

The stem material is ASTM A276 type 316 condition B or precipitation hardened 17-4 pH stainless, procured and heat treated to Westinghouse specifications. These materials were selected because of their corrosion resistance, high tensile properties, and their resistance to surface scoring by the packing. The valve stuffing box was designed with a lantern ring leakoff connection with a minimum of a full set of packing below the lantern ring and a maximum of one-half of a set of packing above the lantern ring; a full set of packing is defined as a depth of packing equal to 1-1/2 times the stem diameter. The experience with this stuffing box design and the selection of packing and stem materials has been very favorable in both conventional and nuclear power plants.

Valves 744, 882, 1810, are required to be open during the injection phase of the LOCA and then must be closed for long-term recirculation. There is no time that these valves would be closed during plant power operation. The motors for these valves are normally de-energized with their breakers locked open. In addition, these valves are provided with red/green position indicating lights and monitor lights to highlight valves configuration as described in Section 6.2.2 "Accumulators," items 1 and 2.

Valves 856B and 856G are required to be closed during the injection and cold leg recirculation phases. The motors for these valves are normally de-energized with their breakers locked open and normal indicating lights de-energized. In addition, these valves are interlocked with corresponding cold leg injection line valves on each header to prevent simultaneous opening of all high-head safety lines on each header. The valves are equipped with a position monitor light, via limit switch and separate DC circuit, and an alarm via limit switch.

The motor operator is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a "hammer blow" feature that allows the motor to move the valve off its main seat or backseat while allowing the motor to attain its operational speed.

The valve was assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned, and packaged per specifications. All manufacturing procedures employed by the valve supplier, such as hard facing, welding, repair welding and testing, were submitted to Westinghouse for approval.

For those valves which function on the safety injection signal, 10 seconds operation or other justified times are typically required. The BIT isolation valves are maintained open, but still receive a safety injection signal to open based on their former application as normally closed valves. An opening stroke time of 11 seconds had been justified for these valves when they were maintained normally closed. However, changing their position from normally closed to normally open has superseded the requirement for a 10 or 11 second opening stroke time. For all other valves in the system, the valve operator completes its cycle from one position to the other typically in 120 seconds.

Valves which must function against system pressure were typically designed such that they function with a pressure drop equal to full system pressure across the valve disc.

<u>Manual Valves</u>

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The stainless steel manual globe, gate and check valves were designed and built in accordance with the requirements outlined in the motor operated valve description above.

The carbon steel valves were built to conform with ANSI B16.5. The materials of construction of the body, bonnet and disc conform to the requirements of ASTM A105 Grade II, A181 Grad II, or A216 Grade WCB or WCC. The carbon steel valves pass only non-radioactive fluids and were subjected to hydrostatic test as outlined in MSS SP-61 except that the test pressure was maintained for at least 30 minutes. Since the fluid controlled by the carbon steel valves is not radioactive, the double packing and seal weld provisions are not provided.

Accumulator Check Valves

The pressure containing parts of this valve assembly were designed in accordance with MSS SP-66. All parts in contact with the operating fluid are of austenitic stainless steel or of equivalent corrosion resistant materials procured to applicable ASTM or WNES specifications. The cast pressure-containing parts were radiographed in accordance with ASTM E-94 and with the acceptance standard as outlined in ASTM E-446 Class 1 or Class 2. The cast pressure-containing parts, machined surfaces, finished hard facings, and gasket bearing surfaces were liquid penetrant inspected per ASME B&PV Code, Section VIII and the acceptance standard as outlined in ANSI B31.1 Code Case N-10. The final valve was hydrotested per MSS SP-66 except that the test pressure was maintained for at least 30 minutes. The seat leakage was conducted in accordance with the manner prescribed in MSS SP-61.

The valve was designed with a low pressure drop configuration with all operating parts contained within the body, which eliminates those problems associated with packing glands exposed to boric acid. The clapper arm shaft was manufactured from 17-4 pH stainless steel, heat treated to Westinghouse Specifications. The clapper arm shaft bushings were manufactured from Stellite No. 6 material. The various working parts were selected for their corrosion resistant, tensile, and bearing properties.

The disc and seat rings were manufactured from a forging. The mating surfaces are hard faced with Stellite No. 6 to improve the valve seating life. The disc is permitted to rotate, providing a new seating surface after each valve opening.

The valves are operated in the closed position with a normal differential pressure across the disc of approximately 1700 psi. The valves remain in this position except for testing and safety injection. Since the valve will not be required to normally operate in the open condition, which would subject the valve to impact loads caused by sudden flow reversal, this equipment does not have difficulties performing its required functions.

When the valve is required to function a differential pressure of less than 25 psig will shear any particles that may attempt to prevent the valve from functioning. Although the working parts are exposed to the boric acid solution contained within the reactor coolant loop, a boric acid “freeze up” is not expected with this low a concentration.

The experience derived from the check valves employed in the Emergency Injection System of the Carolina-Virginia Tube Reactor in a similar system indicated that the system is reliable and workable. The CVTR Emergency Injection System, maintained at atmospheric conditions, was separated from the main coolant piping by one six inch check valve. Check valve leakage was

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not a problem. This was further substantiated by the satisfactory experience obtained from operation.

Relief Valves

The accumulator relief valves were sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves can also pass water in excess of the expected leak rate, but this is not necessary because the time required to fill the gas space gives the operator ample opportunity to correct the situation. For an inleakage rate 15 times the manufacturing test rate, there will be about 1000 days before water will reach the relief valves. Prior to this, level and pressure alarms would have been actuated.

The safety injection test line relief valves are provided to relieve any pressure above design that might build up in the high head safety injection piping. The valve can pass a nominal 15 gpm (2.25×10^5 cc/hr), which is far in excess of the manufacturing design leak rate of 24 cc/hr.

Leakage Limitations of Valves

Valving was specified for exceptional tightness and, where possible, instrument valves, packless diaphragm valves were used.

Normally open valves have backseats which are capable of limiting leakage to less than one cubic centimeter per hour per inch of stem diameter assuming no credit for packing in the valve. Backseats can also be employed to facilitate repacking the valve stem. As a general rule, the plant relies on packing to minimize valve stem leakage. Normally closed globe valves were installed with recirculation flow under the seat to prevent stem leakage from the more radioactive fluid side of the seat.

Motor operated valves which are exposed to recirculation flow were provided with double-packed stuffing boxes and stem leakoff connections which are piped to the Waste Disposal System.

The specified leakage across the valve disc required to meet the equipment specification and hydrotest requirements is as follows:

Conventional globe – 3 cc/hr/in of nominal pipe size

Gate valves – 3 cc/hr/in of nominal pipe size; 10/cc/hr/in for 300 and 150 pound USA Standard

Motor operated gate valves – 3 cc/hr/in of nominal pipe size: 10/cc/hr/in for 300 and 150 pound USA Standard

Check valves – 3 cc/hr/in of nominal pipe size: 10/cc/hr/in for 300 and 150 pound USA Standard

Accumulator check valves – 10 cc/hr/in of nominal pipe size; relief valves are totally enclosed.

Piping

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All Safety Injection System piping in contact with borated water is austenitic stainless steel. Piping joints are welded except for the flanged connections at the safety injection pumps and recirculation pumps.

The piping beyond the accumulator stop valves was designed for Reactor Coolant System conditions (2485 psig, 650°F). All other piping connected to the accumulator tanks was designed for 700 psig and 400°F.

The safety injection pump and residual heat removal pumps suction piping (210 psig at 300°F) from the refueling water storage meets NPSH requirements of the pumps.

The safety injection high pressure branch lines (1500 psig at 300°F) were designed for high pressure losses to limit the flow rate out of the branch line which may have ruptured at the connection to the reactor coolant loop.

The system design incorporated the ability to isolate the safety injection pumps on separate headers such that full flow from at least one pump is ensured should a branch line break. Two SI pump discharge headers are provided in a configuration which allows 2 of 3 SI Pumps to deliver into either header. The suction flow paths are configured to allow isolation of the 32 SI Pump suction piping from the common suction flow path, with an alternate suction piping alignment dedicated to the 32 SI Pump. The common and alternate suction flow paths are cross-tied via a 0.75" pressure equalization pipe, with two normally closed valves and a normally closed vent valve.

The piping was designed to meet the minimum requirements set forth in (1) the ANSI B31.1 Code (1955) for the Pressure Piping, (2) Nuclear Code Case N-7, (3) ANSI Standards B36.10 and B36.19 and (4) ASTM Standards with supplementary standards plus additional quality control measures.

Minimum wall thickness were determined by the ANSI Code (1955) formula found in the power piping Section 1 of the ANSI Code (1955) for Pressure Piping. This minimum thickness was increased to account for the manufacturer's permissible tolerance of minus 12-1/2 percent on the nominal wall. Purchased pipe and fittings had a specified nominal wall thickness that is no less than the sum of that required for pressure containment, mechanical strength and manufacturing tolerance.

Thermal and seismic piping stress analyses were performed in accordance with ANSI B31.1 code (1967). Special attention was directed to the piping configuration at the pumps with the objective of minimizing pipe imposed loads at the suction and discharge nozzles. Piping is supported to accommodate expansion due to temperature changes during the accident.

Pipe and fittings materials were procured in conformance with all requirements of the ASTM and ANSI specifications. All materials were verified for conformance to specification and documented by certification of compliance to ASTM material requirements. Specifications imposed additional quality control upon the suppliers of pipes and fittings as listed below:

1) Purchased pipe and fittings required the submittal of actual heat chemical and physical test results. Each item or part of a fabrication required identification to an individual test report. Welding materials required the submittal of heat or manufacturers' lot reports showing heat chemical and physical test results.

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2) Pipe branch lines 2-1/2 inch and larger between the reactor coolant pipes and the isolation stop valves conform to ASTM A376 and meet the supplementary requirement S6 ultrasonic testing. Fittings conform to the requirements of ASTM A403. Fittings 2-1/2 inch and larger had requirements for UT inspection similar to S6 of A376.

Shop fabrication of piping subassemblies was performed by reputable suppliers in accordance with specifications which defined and governed material procurement, detailed design, shop fabrication, cleaning, inspection, identification, packaging and shipment.

Welds for pipes sized 2-1/2 inch and larger were butt welded. Reducing tees were used where the branch size exceeds ½ of the header size. Branch connections of sizes that are equal to or less than ½ of the header size were of a design that conforms to the ANSI rules for reinforcement set forth in the ANSI B31.1 Code for Pressure Piping. Bosses for branch connections are attached to the header by means of full penetration welds.

All welding was performed by welders and welding procedures qualified in accordance with the ASME Boiler and Pressure Vessel Code Section IX, Welding Qualifications. The Shop Fabricator was required to submit all welding procedures and evidence of qualifications for review and approval prior to release for fabrication. All welding materials used by the Shop Fabricator required prior approval.

All high pressure piping butt welds containing radioactive fluid at greater than 600°F temperature and 600 psig pressure of equivalent were radiographed. The remaining piping butt welds were randomly radiographed. The technique and acceptance standards were those outlined in UW-51 of the ASME B&PV Code Section VIII. In addition, butt welds were liquid penetrant examined in accordance with the procedure of ASME B&PV Code, Section VIII, Appendix VIII and the acceptance standard as defined in the ANSI Nuclear Code Case N-10. Finished branch welds were liquid penetrant examined on the outside and where size permitted, on the inside root surfaces.

A post-bending solution anneal heat treatment was performed on hot-formed stainless steel pipe bends. Completed bends were then completely cleaned of oxidation from all affected surfaces. The Shop Fabricator was required to submit the bending, heat treatment and cleanup procedures for review and approval prior to release for fabrication.

General cleaning of completed piping subassemblies (inside and outside surfaces) was governed by basic ground rules set forth in the specifications. For example, these specifications prohibited the use of hydrochloric acid and limited the chloride content of service water and demineralized water.

Packaging of the piping subassemblies for shipment was done so as to preclude damage during transit and storage. Openings were closed and sealed with tight-fitting covers to prevent entry of moisture and foreign material. Flange facings and weld end preparations were protected from damage by means of wooden cover plates and securely fastened in position. The packing arrangement proposed by the Shop Fabricator was subject to approval.

Field Run Piping

Field running of small diameter piping for essential system including all Engineered Safety Features was not permitted. All seismic Class I and II piping ¾ inch diameter and larger was

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designed by the architect-engineer. All supports and restraints for that piping were located by the architect-engineer and designed by either the architect-engineer or the subcontractor supplying the pipe support hardware. All seismic Class I stainless steel piping sub-assemblies were prefabricated offsite at one or more subcontractors' pipe fabrication shops. All seismic Class I carbon steel piping subassemblies 2-1/2 inches in diameter and smaller were fabricated in the field.

Certain seismic Class I and II systems comprised of small diameter tubing were field run, for example, the N.S.S.S. Sampling System, which is Class II and utilizes 3/8 diameter tubing.

In instrumentation design, virtually all tubing was field run including tubing for engineered safety related devices. However, the following detailed information was supplied by the architect-engineer where critical design requirements were to be met:

1) Physical location of tubing where separation is required for redundant measurements

2) Detailed design of tubing where thermal expansion of vessels to which tubing is attached requires special expansion loops, etc.

3) Detailed design of typical instrument tubing supports and anchors

4) Detailed design of missile protection of small diameter tubing

5) Detailed design of tubing where proper operation of the instrument is dependent upon adequate slope of lines, etc.

It was found practical to eliminate field running of all seismic Class I and II piping ¾ inch in diameter and larger. However, it was not found practical to limit the use of field running to a greater extent, namely all small diameter seismic Class I and II tubing. Most tubing was erected near the end of the construction phase. At that time, the tubing erection forces had access to more potential support points than were known to the Architect-Engineer. Also, at that time, the construction forces necessarily erected the tubing around objects which would otherwise have been unknown interference during the design phase.

Fabrication of all Class I and II piping, 3/4" diameter and larger, was done by piping subcontractors following orthographic piping drawings prepared by the Architect-Engineer. These same piping drawings were also used by the A-E to prepare isometric piping drawings which were then used for both the stress analysis by the designer and for installation by the field groups. These isometrics also showed locations of pipe supports and restraints. Any deviation from the drawing required approval by the Architect-Engineer. A copy of the final as-built information was directed to the Architect-Engineer for final design review.

Site Quality Control Procedures were followed embracing: purchasing and receiving inspection to assure that all weld filler materials, field run pipe and fittings met Class I and II quality requirement; joint by joint inspection to assure cleanliness; proper weld fit-up; proper welding and welder certification; and performance of required NDT. All these procedures were in accordance with A-E specifications for installation, ASA B31.1, and Section IX of the ASME Code. Detailed Quality Control records were maintained on a piece by piece basis and were also recorded on approved spool and line isometric drawings to assure that complete Quality Control coverage was obtained.

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Hydrostatic tests plus hot and/or cold functional tests were performed on completed systems as required and at the appropriate time. No other special quality assurance measures were necessary.

Pump and Valve Motors

Motors Outside the Containment

Motor electrical insulation systems were supplied in accordance with ANSI, IEEE and NEMA standards and tested as required by such standards.

Temperature rise design selection was such that normal long life is achieved even under accident loading conditions.

Criteria for motors of the Safety Injection System required that under any anticipated mode of operation, the motor name plate 1.15 service factor rating is not to be exceeded. Design and test criteria ensure that motor loading does not exceed the application criteria.

Motors Inside the Containment

The SI Recirculation pumps are three stage, vertical pumps driven by 3 phase, 60 cycle, 350 HP motors, and are powered from 480V bus 5A (31) and 480V bus 6A (32). The recirculation pump motors were designed to operate in an ambient condition of saturated steam at 271°F and 47 psig pressure for one day, followed by operation for at least one year at 155°F and 5 psig in a steam atmosphere. The motors are mounted directly to their respective pumps, approximately 2 ft above the highest anticipated water level.

The SI Recirculation Pump motors are provided with thermalastic epoxy insulation and with a heat exchanger. The motors have Class F insulation, temperature rating of 155°C. However, the motor insulation was derated to Class B (130°C) level to provide a safety margin. The operating temperature of the motor insulation is dependent on cooling water temperature rather than the ambient temperature.

The recirculation pump motors are cooled by radiator type coolers using CCW as the cooling medium. Fans are directly connected to the recirculation pump motor shafts. Rotation of the motor rotor and its end fans forces air through the heat exchanger, and air is contained and returned to the ends of the rotor via ducts. A pressure equalizing device permits incident pressure to enter the motor air system so that the bearings are not subject to differential pressures.

The motors are equipped with high temperature grease lubricated ball bearings which would not break down if the bearings were subjected to incident ambient temperatures.

The motors for the valves inside Containment were designed to withstand containment environment conditions following the Loss-of-Coolant Accident so that the valves can perform the required function during the recovery period.

Periodic operation of the motors and tests of the insulation ensure that the motors remain in a reliable operating condition.

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Although the motors which are provided only to drive Engineered Safety Features equipment are normally run only for test, the design loading and temperature rise limits are based on accident conditions. Normal design margins were specified for these motors to make sure the expected lifetime included allowance for the occurrence of accident conditions.

Valve Motor Operators

A production line valve motor has been irradiated to a level of 2 x 10^8 rads using a cobalt-60 irradiation source. The irradiated motor and an identical unirradiated motor have undergone series of reversing tests at room temperature, followed by a series of reversing tests at 275F. The room temperature test was repeated while vibrating the motors at a frequency of 30 cycles per second. Both motors operated satisfactorily during all of the tests. No significant difference was evident in the comparison of the data for the two units throughout the test period.

Two independent valve operator manufacturers conducted loss of coolant environmental tests on units similar to those used in this plant. Reports of results indicated that all units operated satisfactorily at test conditions more severe than those expected in the loss-of-coolant or steam-break environment for this plant.

In addition, Westinghouse performed environmental tests on a unit similar to that being used in this plant. The results of the Westinghouse tests indicated that the equipment would perform its required function in the post-LOCA environment.

Electrical Supply

Details of the normal and emergency power sources for the Safety Injection System are presented in Chapter 8.

Protection Against Dynamic Effects

The injection lines penetrate the Containment adjacent to the Primary Auxiliary Building.

For most of the routing, these lines are outside the crane wall, hence, are protected from missiles originating within these areas. Each line penetrates the crane wall near the injection point to the reactor coolant pipe. In this manner, maximum separation, hence, protection is provided in the coolant loop area.

In the event of a Loss-of-Coolant Accident, all piping systems required to function are designed to remain within acceptable stress limits. The stresses due to dead weight, pressure, operational or design basis earthquake, and maximum motions of the Reactor Coolant Loop imposed on the attached Safety Injection Piping were evaluated in accordance with the stress limits in Section 16.1. The inclusion of the stresses in the injection lines required to function due to movements of the Reactor Coolant Loop assures that these lines maintain their integrity during a Loss-of-Coolant Accident.

All piping supports were designed for the loads imposed by the supported system. The rated loads of allowable stress limits for standard manufactured support components are in accordance with requirements of MSS-SP-58-1967. For non-standard supports designed by analysis, the requirements of AISC-1969 were followed. Where support integrity is dependent on reinforced concrete anchorage, the design was in accordance with the Requirements of ACI-318-63.

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These standards provide minimum requirements on materials, design and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Specifically, these standards required that:

1) All materials used be in accordance with ASTM specifications which establish quality levels for the manufacturing process, minimum strength properties, and for test requirements which ensure compliance with the specifications

2) There be proper qualification of welding processes and welders for each class of material welded and for types and positions of welds

3) Maximum allowable stress values be established which provide an ample safety margin on both yield strength and ultimate strength.

6.2.3 Design Evaluation

Range of Core Protection

The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of this performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met. (See Section 6.2.1)

With minimum onsite emergency power available (two-of-three diesel generators), the emergency core cooling equipment consists of two out of three safety injection pumps, one or two out of two residual heat pumps, and three out of four accumulators for a cold leg break and four accumulators for a hot leg break. With these systems, the calculated maximum fuel cladding temperature is limited to a temperature less than that which meets the emergency core cooling design objectives for all break sizes up to and including the double-ended severance of the reactor coolant pipe. (See Section 14.3)

For large area ruptures analyzed (see Section 14.3) the clad temperatures are turned around by the accumulator injection. The active pumping components serve only to complete the refill started by the accumulators. Either two safety injection pumps or one residual heat removal pump provides sufficient addition of water to continue the reduction of clad temperature initially caused by the accumulator.

System Response

To provide protection for large area ruptures in the Reactor Coolant System, the Safety Injection System must respond to rapidly reflood the core following the depressurization and core voiding that is characteristic of large area ruptures. The accumulators act to perform the rapid reflooding function with no dependence on the normal or emergency power sources, and also with no dependence on the receipt of an actuation signal.

Operation of this system with three of the four available accumulators delivering their contents to the reactor vessel (one accumulator spilling through the break) prevents fuel clad melting and limits metal-water reaction to an insignificant amount (less than 1%).

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The function of the safety injection or residual heat removal pumps is to complete the refill of the vessel and ultimately return the core to a sub-cooled state. The flow from either two safety injection pumps or one residual heat removal pump is sufficient to complete the refill with no loss of level in the core.

The design features applied to the Residual Heat Removal System (RHRS) Valves 730 and 731, that isolate it from the Reactor Coolant System provide a diverse combination of control interlock and mechanical limitations preventing improper opening of these valves and also pressure relief capacity capable of limiting pressure if the valves are not closed upon startup of the plant. These features are:

1) That the valves that are separately interlocked with independent pressure control signals to prevent their being opened whenever the Reactor Coolant System pressure is greater than a designated setpoint (which is below the RHRS design pressure).

   The pressure interlock was not specifically designed to meet the requirements of IEEE Standard 279-1971. However, each valve, its associated pressure channel and related circuitry are powered from separate instrument buses, and wiring separation is provided to preclude any single failure from rendering both of the valves' control circuits inoperable. Each of the pressure channels is provided with separate Control Room indication to show channel operability.

   A separate pressure interlock is provided for each of the two Valves Nos. 730 and 731. Each pressure interlock prevents its valve from being opened when the Reactor Coolant System pressure is greater than a designated open permissive setpoint and also automatically closes the valve whenever the Reactor Coolant System pressure is above a designated auto-close setpoint. These setpoints are below the design pressure of the RHRS.

   While the automatic closure interlock for MOV-730 and –731 will prevent over-pressurizing the RHR system piping during an RCS pressure increase transient, this interlock will isolate the suction source of the operating RHR pump(s), potentially causing pump failure. In order to prevent inadvertent isolation of the RHR pump suction, this auto-closure interlock may be defeated by de-energizing the motor operators to MOV-730 and –731. Prior to de-energizing these MOV's Reactor Coolant System T_{ave} must be below 200°F, depressurized and vented through a minimum equivalent opening of two (2) square inches.

2) That the Reactor Coolant System pressure interlocks meet single failure criteria.

3) That the motors are qualified in accordance with IEEE 323-1974, IEEE 344-1975, IEEE 382-1972 for increased reliability and operability in the normal and accident containment environment.

   The Residual Heat Removal System was designed for a pressure of 600 psig and 400°F and was hydrostatically tested at a pressure of 900 psig prior to initial operation. Insofar as the piping itself is concerned, the piping code (USAS B31.1) allows a rating of 700 psig at 400°F for schedule 40 stainless steel pipe. Thus the piping system, as presently designed, incorporates a considerable margin in that it is rated at a pressure-temperature condition which is less than that allowed by Code. It

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