SECTION 6

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SECTION 6

6.0 ENGINEERED SAFETY FEATURES

6.1 GENERAL

Engineered safety features are provided to fulfill three functions in the unlikely event of a serious loss-of-coolant accident:

a. Protect the fuel cladding.

b. Ensure Containment Vessel integrity.

c. Reduce the driving force for containment leakage.

Emergency injection of borated water into the Reactor Coolant System satisfies the first function above, while Containment Vessel atmosphere cooling satisfies the latter two functions. Each of these operations is performed by two or more systems which, in addition, employ multiple components to ensure operability. All equipment requiring electric power for operation is supplied by the essential electric power buses described in Chapter 8.

The engineered safety features include core flooding tanks, high pressure injection, low pressure injection, the containment cooling system, and the containment spray system. The Emergency Core Cooling System (ECCS) includes the Core Flooding Tanks, High Pressure Injection, and Low Pressure Injection systems. A comprehensive listing of engineered safety features is included in Table 1.2-1.

6.1.1 Protective Coating Systems (Paints) - Organic Materials

Protective coating material has been applied to carbon steel and concrete surfaces within the containment. The function of the materials is to provide surfaces which resist exposures due to both normal operating and Design Basis Accident conditions. Exposures include ionizing radiation, high temperature, and impingement from sprays.

Davis-Besse commits to the regulatory position of Regulatory Guide 1.54 (Revision 0, June 1973) with the following clarifications.

1. This Regulatory Guide and its associated ANSI Standard implies that a significant amount of coating work is required at the plant site. Although this is correct for construction sites, the coating work at an operating site generally consists of repair and touchup work following maintenance and repair activities or the initial coating of components such as hangers, supports, and piping during facility modifications. Therefore, in lieu of the full requirements of the Regulatory Guide and ANSI N101.4, Davis-Besse shall impose the following requirements:

    a. The quality assurance requirements of Section 3 of ANSI N101.4 applicable to the coating manufacturer shall be imposed on the coating manufacturer through the procurement process.

    b. Coating application procedures shall be developed based on the manufacturer's recommendations for application of the selected coating systems.

c. Coating applicators shall be qualified to demonstrate their ability to satisfactorily apply the coatings in accordance with the manufacturer's recommendations.

d. Quality control personnel shall perform inspections to verify conformance of the coating application procedure. Section 6 of ANSI N101.4 shall be used as guidelines in the establishment of the inspection program.

e. Quality control personnel shall be qualified to the requirement of Regulatory Guide 1.58 (Revision 1).

f. Documentation demonstrating conformance to the above requirements shall be maintained.

2. The requirements of Item 1 above apply to surfaces within containment with the following exceptions:

   a. Surfaces to be insulated

   b. Surfaces contained within a cabinet or enclosure.

   c. Repair/touchup areas less than 30 square inches or surface areas such as: cut ends; bolt heads; nuts and miscellaneous fasteners; and damage resulting from spot, tack or arc welding.

   d. Small items such as small motors, handwheels, electrical cabinets, control panels, loud speakers, motor operators, etc. where special painting requirements would be impracticable.

   e. Stainless steel or galvanized surfaces.

   f. Banding used for insulated pipe.

Davis-Besse commits to the requirements of ANSI N101.4-1972 for activities comparable in nature and extent to construction phase activities as modified by the commitment to Regulatory Guide 1.54.

Two DBA qualified coating systems are currently used on ferrous metal and concrete surfaces within the containment.

The first system consists of an inorganic zinc prime coat with an epoxy top coat. This system is applied to the containment vessel, structural steel, and equipment.

The second system consists of an epoxy primer and epoxy topcoat. This system is applied to ferrous metal surfaces and concrete surfaces.

Non-DBA qualified coating materials have also been applied to structures and components within the containment. These materials are standard manufacturer's paints or unqualified coating systems and epoxy materials applied to structures or components with inadequate surface preparation. These materials have been quantified and are tracked by a non-DBA qualified protective coating inventory. Coating material exclusions to this inventory include

surfaces which are insulated or are contained within a cabinet or enclosure. The documented quantity of non-DBA qualified coating material must remain below the limit of coating material debris identified by the ECCS emergency sump debris analysis.

Coating condition assessment inspections are performed each refueling outage to identify and correct degraded coating materials.

6.2 CONTAINMENT SYSTEMS

Isolation valves are employed to maintain and/or re-establish the containment system integrity by the automatic isolation of all Containment Vessel fluid penetrations, thereby eliminating potential leakage paths.

6.2.1 Containment Vessel Functional Design

6.2.1.1 Design Bases

6.2.1.1.1 Containment Design Basis Accident

The containment system is designed to withstand the effects of a Maximum Possible (larger) Earthquake including a loss-of-coolant accident (LOCA) concurrent with any single failure in a safety-related system. The containment system is designed to contain the pressure generated as a result of the most serious LOCA for the Containment Vessel, a 14.14 ft^2 hot leg double-ended guillotine break at the inlet of the steam generator. Subsection 6.2.1.3 gives a comparison of the severity of various size pipe ruptures resulting in a LOCA.

License Amendment No. 278 increased core rated thermal power by 1.63% from 2772 MWt to 2817 MWt, based on the use of more accurate instrumentation for heat balance measurement. The heat balance measurement uncertainty, based on use of the Caldon CheckPlus™ instrumentation, is 0.37%.

6.2.1.1.2 Containment Energy Release Assumptions

The mass and energy release rates to the containment following a LOCA are calculated using RELAP5/ MOD2-B&W computer program. Subsection 6.2.1.3 provides the assumptions and the methodology. The energy released to the Containment Vessel during and subsequent to a LOCA is tabulated in Table 6.2-8 and discussed in subsections 6.2.1.3 and 6.2.2. Table 6.2-9 gives a chronology of events during a LOCA.

Table 6.2-1 provides summary data for the analyzed spectrum of breaks including the break location, break size, peak containment pressure, the time of peak containment pressure, and the energy released to the Containment Vessel at the time of the peak pressure. Figure 6.2-7 contains the mass release rate and the enthalpy as a function of time throughout the blowdown and core reflood phases of the 14.14 ft^2 hot leg double-ended guillotine break at the inlet of the steam generator. Tables 6.2-2 and 6.2-3 provide tables of mass release rates and enthalpies as a function of time for breaks in the reactor coolant pump suction and discharge lines.

For large break sizes, the peak Containment Vessel pressure occurs at about the same time as the end of the initial blowdown phase. Thus, the values provided in Table 6.2-1 for the energy released to the containment at the time of peak pressure, also corresponds to the energy released at the end of the initial blowdown.

6.2.1.1.3 Limiting Value of Energy Released

The maximum value of energy released to the Containment Vessel in Subsection 6.2.1.1.2 is determined assuming that both core flooding tanks functions, one of the two redundant high pressure injection trains functions, and one of the two redundant low pressure injection trains functions.

6.2.1.1.4 Subcompartment Differential Pressure Considerations

Subcompartment differential pressure considerations and capability including the theoretical mass and energy input that might result from a design basis accident is discussed in Subsection 6.2.1.3. The structural design of the vital compartments with respect to the effect of pressure differential and jet force is described in Subsections 3.8.2.3.4 and 3.8.2.3.5.

6.2.1.1.5 Parameters Affecting the Assumed Capability for Post Accident Pressure Reduction

Post-accident Containment Vessel pressure reduction is effected through the use of the Containment Air Coolers and the Containment Spray System. It is assumed that one cooler and one spray header are operating.

Factors Affecting the Containment Air Coolers:

The ability of the Containment Air Coolers to remove heat and thus reduce containment pressure is affected by several variables:

a. Air flow rate

b. Containment Vessel ambient temperature

c. Service water temperature

d. Service water flow rate

e. Air cooler fouling

f. Containment Vessel relative humidity

g. Tube pluggage

Factors Affecting the Containment Spray System:

The ability of the Containment Spray System to remove heat and thus reduce containment pressure is also affected by several variables:

a. Spray water temperature

b. Spray flow rate

c. Spray header reliability

d. Spray droplet size

e. Containment Vessel air temperature

f. Containment Vessel relative humidity

Discussion:

The operation and reliability of the containment heat removal systems are discussed in Subsection 6.2.2. Table 6.2-21 presents a summary of the single failure analyses performed.

6.2.1.2 System Design

6.2.1.2.1 Design Parameters

The Containment Vessel design internal pressure, temperature, and free volume are:

36 psig (design) /40 psig (max)
264°F
2,834,000 ft^3

The Shield Building annulus volume is 678,700 ft^3. The Emergency Ventilation System limits the temperature induced pressure transients in the annular space to 6 inches H_2O during a LOCA. Following the initial pressure transient, the annulus pressure is maintained between (-)1/4 and (-)1-1/2 inches WG.

Although the maximum design pressure is 40 psig, the pressure criterion for the Containment Vessel's pressure analysis is less than or equal to 38 psig. This value corresponds to the minimum of the pressure range utilized for the Integrated Leak Rate test (i.e., 38 to 40 psig). By limiting the Containment Vessel's predicted peak pressure during a LOCA to-38 psig, the minimum Containment Vessel pressure utilized for the leak rate analysis will be protected.

Free Containment Volume Calculational Method and Accuracy:

Mensuration is the method that was used in determining the Containment Vessel free volume. This type of mathematics is the act of measuring or computing the lengths, areas, and volumes from given dimensions or angles. An accurate analysis of the free volume was obtained by using the geometry of the Containment Vessel, Containment Vessel internal structures, Nuclear Steam Supply System, cranes, miscellaneous equipment, etc. The accuracy of the results obtained should be in the range of approximately 1 to 2 percent.

Increases or decreases in the free volume are accompanied by reductions or increases in the containment peak pressure, respectively. In calculating the pressure variations with free volume variations, the I4.14 ft^2 double-ended guillotine break at the steam generator inlet, i.e, the Design Basis Accident (DBA), was used. The containment free volume sensitivity is tabulated below:

Containment Free Volume Sensitivity
(Initial Average Containment Temperature = 90°F)

Containment free volume, ft^3	$2.891x10^6$	$2.862x10^6$	$2.834x10^6$	$2.806x10^6$	$2.777x10^6$
Pressure, psig	37.3	37.6	37.9	38.3	38.6
∆ Volume, %	+2	+1	0	-1	-2
∆ Pressure, %	-1.77	-0.90	0	+.087	+1.68

6.2.1.2.2 Design Leakage Rates

The design leakage rate associated with an internal pressure of 38 psig would not exceed 0.5 percent of the containment contained weight of air and vapor in 24 hours.

6.2.1.2.3 Containment Integrity

The design method used to ensure the integrity of containment structures and subcompartments during pressure pulses is described in detail in Subsections 3.8.2.3.4 and 3.8.2.3.5.

6.2.1.3 Design Evaluation

6.2.1.3.1 Systems Ensuring Containment Leaktightness

There are no normally operating systems required to ensure leaktightness of the Containment Vessel. However, the Emergency Ventilation System is intended for use in an accident situation to establish a slight vacuum in the Shield Building annulus, the penetration rooms, and the Emergency Core Cooling System equipment rooms. Any leakage would be into these spaces, and exhaust would be through the HEPA and charcoal filters. The system description and a single- failure analysis of the system components are presented in Subsection 6.2.3. The containment leakage testing is described in Subsection 6.2.1.4.

6.2.1.3.2 Containment Pressure Transient Analysis Break Spectrum

The containment pressure response to mass and energy releases from reactor coolant system breaks has been analyzed for a spectrum of break sizes to establish the performance capability of the containment system. Break sizes from a 2 ft^2 pump suction break to a 14.14 ft^2 hot-leg pipe break have been studied.

The system parameters, initial conditions, and engineered safety features used in the containment pressure response analysis are summarized as follows:

<u>Initial Conditions</u>

Containment Vessel Free Volume	2.834×10^6 ft^3
Containment Vessel Air Temperature	90°F (DBA analysis, i.e., peak pressure analysis)
Containment Vessel Air Temperature	120°F (Equipment Qualification analysis)
Service Water Inlet Temperature	90°F (without crediting the non safety-related connection to Lake Erie, see Figure 9.2-6)
Borated Water Storage Tank Water Temperature	90°F
Partial blockage of nonsafety-related Service Water system discharge line	partial blockage is analyzed from 0 hours to 2.8 hours of the transient

When using RELAP5 mass and energy release data, as the initial average Containment Vessel air temperature is reduced, the peak pressure increases slightly. Consequently, an initial average Containment Vessel air temperature, corresponding to the lowest historical value during the winter months of 90°F, was utilized for the DBA analysis. Due to the small effect of the initial average Containment Vessel temperature on peak pressure, routine monitoring of the minimum temperature is not necessary. For determination of temperature profiles used by Equipment Qualification (EQ) evaluations, an initial Containment Vessel average air temperature of 120°F is utilized. This value corresponds to the average steady-state temperature in the vicinity of the EQ equipment during the summer months.

The simultaneous occurrence of a LOCA and partial blockage of a nonsafety-related Service Water discharge line must be postulated. The blockage is postulated for the initial 2.8 hours of the transient. The blockage will be isolated by the operators within 2.8 hours. The blockage will cause a reduction in delivered flow to the components served by the Service Water system. The components that are affected with respect to the Containment Vessel response analysis during a LOCA are: (1) Containmer Air Cooler and, (2) Component Cooling Water Heat Exchanger. Since both of these components are modeled to start after 300 seconds, the reduced flowrates only affect the long-term cooling analysis of the Containment Vessel.

Engineered Safety Features Operation

	ESF Systems	Design** Heat Removal Capacity, Btu/hr	Flow Capacity, gpm/cfm	Initial Operating Time, sec
Maximum	Air Coolers (2)	150×10^6	2300 (fluid)	300
ESF	Containment		90000 (air)	
Operation	Sprays (2)	150×10^6	2600	160
	HP Injection		1000*	30
	LP Injection		6000*	30
Minimum	Air Cooler (1)	75×10^6	1150 (fluid)	300
			805 (fluid)***	
ESF	Containment		45000 (air)	
Operation	Spray (1)	75×10^6	1300	160
	HP Injection		500*	30
	LP Injection		3000*	30

* Flow rate depends on the discharge pressure.
** Data are original design values. Predicted heat removal capacities during the 14.14 ft^2 DBA are provided by Figure 6.2-13.
***With partial blockage of nonsafety related Service Water discharge line.

The heat sinks used for the containment pressure-temperature analysis are summarized in Table 6.2-5a.

The containment base slab absorbs heat from the Containment Vessel emergency sump. All other heat sinks transfer heat to or from the containment atmosphere.

The Containment Vessel pressure response to the spectrum of postulated pipe breaks is presented in Table 6.2-1. The Containment Vessel pressure response for all of the break sizes is shown on Figure 6.2-3a and Figure 6.2-3b.

These breaks were analyzed with the minimum engineered safety features operation described previously. From the results of containment pressure analysis, the 14.14 ft^2 double-ended guillotine break at the inlet of the steam generator is established as the Design Basis Accident (DBA).

METHODS AND ASSUMPTIONS:

Mass and Energy Release to Containment During Blowdown:

The mass and energy releases to the containment during the initial 300 seconds of the loss-of-coolant accidents are evaluated using the multinode RELAP5/MOD2-B&W computer code. The noding scheme used by RELAP5/MOD2-B&W is described in BAW-10192. The piping in the vicinity of the rupture is modeled by at least four nodes. The noding scheme is selected to produce reliable thermodynamic properties at the break. The pressurizer is modeled in the intact loop. As demonstrated in BAW-10192, this modeling choice produces similar results regardless of which loop contains the pressurizer.

Loss-of-coolant-accident energy released to the reactor building is largely comprised of the following:

a. Reactor coolant system coolant.

b. Reactor coolant system metal heat.

c. Energy generated by the core and core stored heat.

d. Emergency core cooling system coolant.

e. Energy transferred to the RCS from the secondary steam system.

For the break analyses, initial conditions and calculation assumptions were chosen in order to ensure a conservative (i.e., maximum) determination of the mass and energy releases. These included:

a. 0% tube plugging

b. maximum initial pressurizer level

c. minimum Emergency Core Cooling System flowrates

d. Loss-of-offsite-power and Reactor Coolant Pumps powered with a 2-min trip delay

e. consideration of nitrogen entering the RCS via emptying of the Core Flooding Tanks.

The mass and energy release data computed by RELAP5/MOD2-B&W is input to the Containment Vessel pressure analysis. By maximizing the release data, a conservative peak Containment Vessel pressure will be computed.

The reactor was assumed to be operating at a core power level of 102 percent of 2966 MWt with a conservative zero moderator coefficient. The initial fuel temperatures were determined with NRC-approved fuel performance codes that account for the effects of fuel densification. Additional details of the fuel performance codes are provided in Appendix 4B.

RELAP5/MOD2-B&W computes the pre-critical heat flux (CHF) heat transfer by single-phase liquid convection, nucleate boiling and forced convection vaporization. Post-CHF heat transfer regimes in RELAP5/MOD2-B&W's core model include boiling, film-boiling and single-phase steam heat transfer.

RELAP5/MOD2-B&W computes decay heat based on the combination of: (1) the fission decay heat curve described in the 1971 ANS standard, "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors," for infinite operation, times a factor of 1.2 and, (2) the 1979 ANS actinides.

Heat transfer from piping, vessel walls, and non-fuel internal structures is taken into account. Appropriate metal slabs are simulated in each control volume of RELAP5/MOD2-B&W.

Sensible energy associated with the ECCS coolant was accounted for based on the injection of two Core Flooding Tanks, two Low Pressure Injection pumps, and one High Pressure Injection pump. Flow from the second High Pressure Injection pump, which injects water to the cold leg piping in the broken loop, was neglected since it would serve to condense steam flowing to the break and hence, result in less conservative mass and energy releases.

The simulation of the secondary side of the steam generators took into account the energy addition of both the Main and Auxiliary Feedwater, and modeled heat flow from the secondary system to the primary system (RCS). Main feedwater flow was held constant for 12.5 seconds following a reactor trip and was then decreased linearly for 12.5 seconds to simulate the closing of the feedwater control valve. Paths connecting nodes, which simulated the main feedwater piping up to the control valve, permitted the liquid trapped in the feedwater piping to drain into the steam generators. Auxiliary Feedwater was actuated on loss of main feedwater, and injection was assumed available after a 120 second delay.

Heat Transfer from SG Secondary to Primary Side During Blowdown and Energy Release to Containment:

Heat transfer between the primary system and the secondary-sides of the steam generators is calculated during all phases of a LOCA. During blowdown, the steam generator tubes are modeled as a boundary across which heat is transferred using the models and heat transfer correlations described in Section 2.2 of BAW-10164. During the reflood phase of the transient, the heat is transferred using the model described in Section 2.10 of BAW-10171. A conservatively high steam generator heat transfer coefficient is utilized so that all incoming primary-side fluid is vaporized and superheated to the secondary-side saturation temperature.

During the early period of blowdown the secondary side of the steam generators act as a heat sink. As the primary coolant temperature decreases, the energy stored within the secondary fluid and hot steam generator metal is transferred to the primary coolant. However, the nature

and size of the LOCA's considered are such that complete voiding occurs in the primary side of the steam generators shortly after heat flow to the primary side commences.

<u>Description of Core Reflood Model</u>:

Peak cladding temperatures are not limiting for the break spectrum evaluated for the Containment Vessel analysis (Table 6.2-1) due to three factors: (1) high positive core flows, (2) a sufficient amount of Emergency Core Cooling fluid reaching the lower plenum and, (3) no adiabatic heatup. Under these conditions, a core steam venting path is available to the core exit, therefore, the lower plenum need not be voided to provide additional venting. The liquid remaining in the lower plenum allows reflood to begin at the end of blowdown. Absence of a refill phase leaves only the blowdown and reflood phases of the transient. RELAP5/MOD2-B&W can calculate both of these phases without the REFLOD3B system calculation.

During the reflooding portion of the LOCA, emergency injection is supplied by two Core Flooding Tanks (CFTs), two Low Pressure Injection/Decay Heat (LPI/DH) removal pumps and one High Pressure Injection pump in order to maximize the mass and energy release. The injected water, after entering the Reactor Coolant system, mixes with the fluid within the cold leg piping and downcomer and is continually heated due to the primary metal heat. The liquid within the reactor vessel thus remains at saturated conditions.

The core inlet and outlet flows during the blowdown and reflood transient are based on the dynamic calculations as predicated by the RELAP5/MOD2-B&W code. By integrating the inlet and outlet flows, the carryout rate fraction (CRF) versus time may be calculated. In all cases, RELAP5/MOD2-B&W predicts a time averaged CRF greater than 0.8.

Noding of the core is described in BAW-10192. As previously discussed, Containment Vessel analysis is not limiting with respect to peak clad temperatures. Peak clad temperature analyses are described in Section 6.3.

<u>Containment Vessel Response</u>:

The response of the Containment Vessel to the mass and energy release from postulated Reactor Coolant System pipe ruptures is analyzed with the COPATTA computer code (Reference 1). The COPATTA code was developed by Bechtel based on the original CONTEMPT code (Reference 2) written by the Phillips Petroleum Company as part of the LOFT program. COPATTA is written in FORTRAN IV for the GE 635 computer. Subsequent revisions to COPATTA allow it to run on a personal computer.

COPATTA calculates a pressure-time transient with stepwise iteration between thermodynamic state points. The iterations are based on the laws of conservation of mass, momentum and energy together with their derived functions. Superposition of heat input functions is assumed so that any combination of blowdown, metal water reaction, decay heat generations, and sensible heat energy can be used with appropriate engineered safety features to determine the pressure-time history associated with any LOCA. The effect of Containment Vessel leaks and time and position dependent thermal gradients can be evaluated depending upon the input used and printout requested.

The program assumes a three-region Containment Vessel. The containment atmosphere is the vapor region, the sump is the liquid region, and the reactor, the third region, can function independently as a vapor or a liquid region. Energy is transferred between the liquid and vapor

regions by boiling, but evaporation is neglected. A convective heat transfer coefficient can be assumed between these two regions. However, since any heat transfer in this mode is small, a conservative coefficient of zero is assumed. Each region is assumed homogeneous, but a temperature difference can exist between regions. Any moisture condensed in the vapor region during the time step is assumed to fall immediately into the sump.

The Containment Vessel model includes representation of three engineered safety features: a spray system, an ECCS, and an air cooling system. Water supplied to the spray and ECCS can come from an external source at a prescribed temperature, or it can be recirculated from the liquid region in the Containment Vessel. During recirculation, water for decay heat removal is taken from the liquid region and is pumped through a heat exchanger before being returned to reactor region. The cold side of the heat exchanger can itself be part of another heat exchange system.

The air cooling system is described with start and stop times and a table of heat removal rates as a function of vapor temperature. Moisture condensed by the air coolers is assumed to fall immediately into the liquid region.

The Containment Vessel and internal structures can be separated in up to 20 heat conduction sections whose thermal behavior can be described by a one-dimensional, mtiltiregion heat-conduction equation. These heat conducting sections can act as heat sources or sinks. Any boundary conditions from insulated to zero resistance can be applied to each section as appropriate. These conditions can be constant, time-dependent, temperature-dependent, or dependent upon the steam-to-air ratio existing in the Containment Vessel atmosphere. Bulk temperatures may be the vapor region temperature, the liquid region temperature, the reactor vessel liquid temperature, a cyclical outside air temperature, or a constant.

Heat transfer to the heat sinks from the Containment Vessel atmosphere is determined by a "Modified Tagami" heat transfer coefficient. This coefficient correlates the test results of Uchida and Kolflat with a turbulence factor that depends upon the time from accident initiation to peak pressure (Reference 3, Reference 4). As time progresses and turbulence decreases, this coefficient is reduced to Uchida's steady-state heat transfer correlation by a ratio of the instantaneous mass blowdown rate to the mass blowdown rate at the time of peak pressure. No heat transfer from the Containment Vessel outer surface to the environment was assumed for the analysis. The heat rejection from the steel shell to the Shield Building annular space is not rapid enough to affect the peak pressure of the various break sizes.

The Containment Vessel pressure-time history of a LOCA calculated by the COPATTA program is conservative. The assumptions used in preparing the program input and in the program calculations are consistent with the two-phase, two-component thermodynamic model used. These assumptions are summarized below:

> The Containment Vessel atmosphere pressure is also the sump pressure and, following blowdown, the reactor coolant system pressure. Each region is thoroughly mixed, with homogeneous thermodynamic properties.
>
> All liquid condensed in the atmosphere or on the walls during any calculational time interval is removed from the atmosphere and added to the sump at the end of the interval.
>
> The sump region contains no water at the beginning of the transient.

Water entering the Containment Vessel from the Reactor Coolant System during the blowdown phase of the transient flashes, and its final temperature is the saturation temperature at total vessel pressure.

If the total pressure inside the Containment Vessel drops below the saturation pressure of sump water, boiling of sump water occurs until a new equilibrium pressure and water temperature are established. The atmosphere region is constrained to saturation conditions when the containment spray system is in operation.

The heat transfer coefficient during blowdown, for heat transfer between the Containment Vessel atmosphere and heat conducting regions in contact with it, is calculated from an empirical relationship between the maximum heat transfer coefficient, the rate of steam input into the vessel, and the duration of blowdown. During the postblowdown phase, the heat transfer coefficient is calculated as a function of the air/steam mass ratio in the atmosphere.

Heat transfer rates from a superheated atmosphere to heat sinks are calculated, using a temperature gradient corresponding to the steam saturation temperature less the heat sink surface temperature.

No leakage from the Containment Vessel is assumed.

There is no convective heat transfer between the atmosphere and the sump, and evaporation is neglected.

The Containment Vessel bottom head is assumed insulated.

The initial Containment Vessel pressure is increased by an amount equivalent to the pressure differential allowance of Technical Specifications.

Conservative overall heat transfer coefficients are utilized for the Decay Heat Cooler and the Component Cooling Water heat exchanger in order to minimize the heat transfer between the Containment Vessel's sump and the Ultimate Heat Sink.

Conservative values for tube fouling, air flowrate and Service Water flowrate are utilized to model the Containment Air Cooler. Potential tube pluggage is also considered.

Short-term Containment Analysis (Reference 42):

Subsequent to the 14.14 ft^2 DBA, the containment pressurizes in response to the mass and energy release from the reactor coolant system. The containment atmosphere reaches a maximum pressure of 37.9 psig at 14 seconds after the pipe ruptures. The maximum containment atmosphere temperature is 259.2°F (with initial temperature of 120°F) and occurs at about the time of the peak pressure.

The mass and energy releases during the blowdown period are shown in Figure 6.2-7. Containment conditions just prior to the accident and at 14 seconds, the time of peak pressure, are summarized in Table 6.2-7. The mass and energy release rates were calculated by the RELAP5/MOD2-B&W computer code. The initial pressurization of the containment during the blowdown period is shown in Figure 6.2-8.

The energy distributions prior to the accident and at the time of peak pressure are given in Table 6.2-8. Heat transfer from the containment atmosphere to the heat sink structures is governed by the "Modified Tagami" heat transfer coefficient which maximizes at 256 Btu/hr-ft^2-°F at 14 seconds. The "Modified Tagami" heat transfer coefficient as a function of time is shown in Figure 6.2-9.

Long-term Containment Analysis (Reference 42):

The sources of heat and energy that are considered during the long-term cooldown analysis of the containment include: (1) reactor vessel and RCS piping metal, (2) steam generator metal, (3) core component metal, (4) RCS fluid and, (5) decay heat (based on Standard Review Plan 9.2.5). One Containment Spray train, one Low Pressure Injection train and one Containment Air Cooler are utilized to remove heat from the containment. The Containment Vessel response presented in this section is based on the DBA, i.e., 14.14 ft^2 hot leg double-ended guillotine break at the inlet of the steam generator.

After the containment pressure reaches its peak value of 37.9 psig at 14 seconds, it decreases to 21.0 psig at 4,500 seconds. At this time, the Borated Water Storage Tank (BWST) supply is depleted. After recirculation of containment sump water commences, spray water is supplied directly from the sump, and Low Pressure Injection water is taken from the sump and cooled by a Decay Heat Removal Cooler. Due to the change in safety features operation, repressurization of the Containment Vessel occurs, reaching a maximum of 21.6 psig at 6,500 seconds after the initial break. During recirculation, the majority of heat removal from the containment is performed by the Containment Air Cooler and the Decay Heat Removal Cooler. At 100,000 seconds the pressure has been reduced to about 10 psig. A graph of containment pressure is presented in Figure 6.2-10 to display the long term pressure response of the containment. A chronology of events is presented in Table 6.2-9 for the 14.14 ft^2 hot leg break (i.e., DBA).

The containment atmosphere and containment sump temperatures also follow the change to recirculation. The containment atmosphere temperature decreases from a maximum of 259°F at 16 seconds to 227°F at the initiation of the recirculation mode. During the recirculation mode, a second peak in atmosphere temperature, 230°F, occurs coincident with the second peak pressure.

The maximum containment sump temperature of 255°F occurs at 300 seconds after the initiation of the accident. Following the peak in sump fluid temperature, the addition of 90°F Borated Water Storage Tank water continues, reducing the sump water temperature to 251°F at 4,500 seconds. After 4,500 seconds, the recirculation of sump water through the low pressure injection system and the containment spray results in a gradual increase in sump temperature to another peak, 253°F, 10,500 seconds after the accident. Following this recirculation peak, the heat removal by the Containment Air Coolers and Decay Heat Removal Coolers reduces the sump temperature to 201°F at 100,000 seconds as shown in Figure 6.2-11.

Energy distributions following the 14.14 ft^2 hot leg break are shown in Figure 6.2-12. The system total energy reaches a maximum of 9.4 x 10^8 Btu at 10,000 seconds after the pipe rupture. By 100,000 seconds, total energy in the containment is reduced to 6.7 x 10^8 BTU due to heat removal by the Containment Air Coolers and the Decay Heat Removal Coolers.

The rates of energy (heat) generation by decay of fission products and the release of heat from hot metal structures are shown in Figure 6.2-13. Also shown as dashed lines are the heat removal rates of the Containment Air Coolers and the Decay Heat Removal Coolers. These coolers provide the only heat removal mechanism for the Containment Vessel and the internals.

No heat removal capability is assumed for the heat sinks; in the analysis, they serve as energy absorbers during more severe containment atmosphere conditions and sources when the conditions are less severe.

The Decay Heat Removal Coolers performance is determined by the temperature of the Component Cooling water which is, in turn, determined by the performance of the Component Cooling Heat Exchanger. The heat exchanger effectiveness for each exchanger is determined by the method developed by Kays and London (Reference 5).

Containment Pressure Response for a Spectrum of High Energy Breaks:

The containment pressure responses for a spectrum of high energy breaks is contained in Table 6.2-10. Only one feedwater break is analyzed since the break of a smaller feedwater line will result in pressurization less severe than that of the main feedwater line. In all cases, the breaks were assumed to be at the steam generator regardless of their location.

The breaks were analyzed in the same manner as Subsection 15.4.4. with the exception that the 5.4 ft^2 break, which utilized mass and energy release data based on RELAP5/MOD2-B&W. Other break sizes were not reanalyzed with RELAP5/MOD2-B&W since the 5.4 ft^2 break is limiting. The COPATTA analytical model used to compute peak Containment Vessel pressure was the same as that previously described in this subsection.

Main Steam Line Break Analysis (Reference 42):

Pressure switches are provided, which, in the event of a main steam line rupture automatically trip the main feedwater control and stop valves. Four pressure switches are provided on each main steam line. In order to trip the isolation valves (main steam and feedwater), a 2 out of 2 condition per actuation channel must exist.

Mass and Energy Release:

Mass and energy release data for the Main Steam Line Break (MSLB) that are utilized in the Containment Vessel response analysis were generated with the RELAP5/MOD2-B&W computer code and are listed in Table 6.2-11. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod insertion, fuel temperature changes and moderator temperature changes. The secondary model includes a detailed description of the Main Steam System including steam relief to the atmosphere through the Main Steam Safety Valves and simulation of Turbine Stop Valve operation. The secondary model also includes the delivery of both main and auxiliary feedwater to the steam generators.

In order to maximize the mass and energy release to the Containment Vessel, end-of-life core conditions were assumed. The increase in secondary heat removal will cause the Reactor Coolant System (RCS) pressure, average core moderator temperature and fuel temperature to decrease. The combined effect will cause core power to decrease initially. When the core inlet temperature decreases due to the overcooling of the RCS, the negative moderator density feedback will cause a positive reactivity insertion and will lead to a power increase.

Following a double-ended Main Steam line rupture between a steam generator and the Main Steam Isolation Valve (MSIV), both steam generators will depressurize causing an RCS cooldown and depressurization. The depressurization of the steam lines initiates the Steam and Feedwater Rupture Control System's (SFRCS) low steam line pressure trip. The SFRCS low steam line pressure trip generates signals to trip the turbine, close the MSIVs, close the Main

Feedwater Stop Valves, close the Startup Control Valves, close the Main Feedwater Control Valves and start Auxiliary Feedwater flow.

A reactor trip signal via SFRCS on low pressure and ARTS (on turbine trip) is normally generated. However, these trip functions were not credited. The depressurization of the steam generators results in a Reactor Protection System trip on low RCS pressure. The depressurization of both steam generators continues until the SFRCS low steam line pressure trip is reached in the steam line of the affected steam generator. This trip generates a signal to close the MSIVs, isolate the steam-side of the unaffected steam generator and initiate AFW to both steam generators. Normally, AFW would be isolated to the affected steam generator due to the sensing of low steam line pressure following MSIV closure, however, due to the single failure of an AFW isolation valve, AFW is supplied to the affected steam generator. The remaining inventory in the affected steam generator and the AFW continue to be released to the Containment Vessel until operator action is taken at 10 minutes to isolate the AFW being supplied to the affected steam generator.

The MSLB analysis with RELAP5/MOD2-B&W resulted in some liquid carryout which results in a lower than expected enthalpy for the break effluent. In order to ensure a conservative calculation of Containment Vessel pressure and temperature, the enthalpy of the break effluent was adjusted to that corresponding to the saturated vapor enthalpy of the break donor volume or the calculated effluent, whichever is higher.

A single failure analysis was performed to determine the failure that would maximize the mass and energy release from a MSLB to the containment. The worst-case single failure was determined to be a failure to isolate the AFW flow to the affected steam generator due to an SFRCS signal. This failure allows a portion of the AFW to flow to the affected steam generator. This increases the primary-to-secondary heat transfer which increases the mass and energy release to the containment. Three other possible single failures were considered: (1) failure of one train of the High Pressure Injection system, (2) failure of one Main Steam Safety Valve to close and, (3) failure of Main Feedwater Stop Valve to close. These three failures resulted in less mass and energy release compared to the failure to isolate AFW flow to the affected steam generator.

COPATTA Analysis:

The COPATTA computer code was utilized to compute the pressure and temperature response of the Containment Vessel following a Main Steam Line break. Details of the COPATTA modeling techniques are provided in Section 6.2.1.3.2.

Based on an 8 percent revaporization assumption of NUREG-0588, Rev. 1, (Reference 44) as permitted for a Category 11 analysis, the peak steam vapor temperature following a MSLB is 364.9°F at about 33 seconds. The peak containment pressure under these assumptions is 27.2 psig at 36 seconds, while the resulting peak electrical equipment surface temperature is 292.8°F at 613 seconds. Based on a 0 percent revaporization assumption of NUREG-0588, Rev. 1, as required for a Category I analysis, the peak steam vapor temperature following a MSLB is 378.0°F at about 37 seconds. The peak containment pressure under these assumptions is 27.6 psig at 37 seconds, while the resulting peak electrical equipment surface temperature is 329.1°F at 713 seconds. Application of two temperature profiles is necessary to demonstrate environmental qualification for both existing and replacement equipment.

Inadvertent Spray Actuation Analysis:

An analysis of the Containment Vessel negative pressure transient due to inadvertent operation of one train of the containment spray system has been performed for various spray water temperatures. A conservative spray flow rate of 2100 gpm has been assumed to account for pump run-out with the Containment Vessel at ambient pressure. Initial conditions assumed for both the Containment Vessel and the annular space between the shield building and the Containment Vessel are:

	Containment Vessel	Annular Space
Pressure	14.4 psia	14.4 psia
Temperature	120°F	60°F
Relative Humidity	10%	0%

From USAR Section 3.8.2.1.4 the steel Containment Vessel is designed to withstand an external pressure differential of 0.5 psi. The design allowable external pressure differential is reevaluated to be 0.67 psi for service levels A and B of ASME code, Section III, Subsection NE, and 0.8 psi for service level C.

The transient pressure response of the Containment Vessel has been analyzed for the following cases:

Case 1 - 35°F spray water with eight vacuum breakers operational

Case 2 - 60°F spray water with six vacuum breakers operational

The analysis demonstrated that the number of vacuum breakers required to prevent the Containment Vessel from exceeding its external pressure loading design value is sensitive to spray (BWST) water temperature. For BWST water temperatures below 60°F a minimum of eight operational vacuum breakers out of the ten installed would protect the Containment Vessel from external pressure loadings that exceed the design value. When BWST water temperature exceeds 60°F only six operational vacuum breakers would be needed.

Evaluation of Heat Sinks Within the Containment:

Table 6.2-5a takes into account all of the major heat sinks within the Containment Vessel.

All of the heat sinks are exposed on both sides except the Containment Vessel, the Containment Vessel dome, the refueling canal lining, the corrugated steel bottom of concrete floors, and the pressurizer quench tank.

For heat sinks which are exposed on both sides, the surface area is twice the area of one side and the thickness is one half the thickness of the entire slab.

6.2.1.3.2.1 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 46) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation identified and addressed the replacement Steam Generator design characteristics that could impact the containment's response to the mass and energy releases from reactor coolant system breaks analyzed in this section. It documented that the primary

and secondary side fluid inventories and structural metal masses are the dominant parameters that can affect the response to this event.

In evaluating the impact of potential changes in these parameters, it concluded that the primary side inventory volume and mass of the replacement Steam Generators are bounded by the volume and mass used in the original Steam Generator analysis. Additionally, the replacement Steam Generator structural metal (upper and lower head and tubesheets) in contact with the primary coolant is less, and is, therefore, bounded by the analysis that included the original Steam Generators.

The potential for the secondary side of the replacement Steam Generators to be an increased heat source in containment was also evaluated. Since the replacement Steam Generator secondary inventory mass is essentially the same as the original Steam Generator's, and since the replacement Steam Generator secondary side metal mass is smaller than the original Steam Generator's metal mass, the original analysis remains bounding.

All other differences between the original and replacement Steam Generators were documented as having no impact on the existing analysis results. Therefore, the conclusion of the evaluation is that the existing analyses remains bounding with the replacement Steam Generators installed.

6.2.1.3.3 Subcompartment Analyses

a. Introduction:

The dynamic effects of Primary Loop pipe breaks (i.e., jet impingement, differential pressure and pipe whip) that were used in the following original design analyses are no longer considered. See Section 3.6.2.2.1 for current use of Leak-Before-Break methodology.

The containment subcompartments in which a major loss-of-coolant accident could occur are the reactor cavity and the steam generator compartments. The walls of these chambers are designed to bear the combined loads of differential pressure and jet impingement. The differential pressure analyses were performed with the Bechtel developed code COPDA to verify the adequacy of the venting and the design of the structures. The COPDA code is described in Subsection 3.6.2.7.1.1.

Vents between control volumes are taken at minimum flow area sections. If there are significant constrictions within a compartment, then that compartment is divided into two or more control volumes at these constrictions.

b. Reactor Cavity:

The cavity was nodalized for the worst case (double-ended cold leg break) as shown in Figures 6.2-14, 6.2-15, and 6.2-16. The corresponding flow diagram is shown in Figure 6.2-17 and is summarized in Table 6.2-12. Vents between control volumes were taken at the minimum flow area cross sections. The principal obstructions within the cavity are the nozzles and the vessel supports. In addition, the cable supports in the incore instrumentation tunnel and the trisodium phosphate baskets in the access tunnel were also included as obstructions.

The insulation in the reactor cavity is the metallic reflective type. For the analyses, the insulation is conservatively assumed to remain intact and uncrushed. The width of the cavity annulus precludes jamming of the insulation. Sand plugs or other removable obstructions are

not included in the design. Restraint rings around the hot and cold leg pipes of the reactor coolant system, within the primary shield pipe penetrations, were also considered as obstructions in evaluating the vent flow path areas.

Flow coefficients were calculated using standard head loss factors such as outlined in Crane Technical Paper No. 410, "Flow of Fluids Through Valves, Fittings, and Pipe." The individual factors due to contractions, expansions, bends, and friction were summed and the flow coefficients calculated by

$$C = \frac{1}{(\Sigma K)^{0.5}}$$

In most cases, friction was negligible. Table 6.2-14 shows the calculation of each coefficient. The orifice flow coefficient is calculated by the COPDA program as described in Subsection 3.6.2.7.1.1. The miscellaneous flow equation is the same as the nozzle equation but the flow coefficient is calculated by the user, using the methods described above. The mass flux is determined from the isentropic nozzle equation and then multiplied by the product of the area and the flow coefficient.

Motion of the hot leg pipe following a break is limited by restraint rings in the primary shield penetration (see Figures 6.2-18, -19 and -20). The restraint rings shown around the reactor coolant system pipes, within the primary shield pipe penetrations (see Figure 5.1-3) were based on a series of assumed pipe breaks and reactor support failures. The rings were provided to limit the movement of the reactor during such a failure to a minimum. A gap of 1/8” at the top and sides of the pipes is provided, but due to a possibility of the reactor settling, a 1/4” gap has been provided at the bottom. The removable shims are designed in such a way that these can be removed and milled to the required thickness for final fit-up.

In case of a LOCA and failure of the beam supports, the design considerations are such that the impact of the 36” and 28” diameter pipes on the rings will restrict the movement of the reactor in the horizontal and vertical directions. In addition, the pipe is prevented from backing out by restraints in the steam generator compartment (see Figure 6.2-18A). These restraints were designed utilizing jet loads from a double- ended (2A) hot leg break. Such a large break size should not exist, however. To determine the break flow area inside the reactor cavity, simplified yet conservative assumptions were used. For a postulated double ended rupture, the calculated total longitudinal movement of the pipe will not develop a circumferential break flow area more than the full cross-sectional (1A) flow area of the pipe. The assumptions used in determining the break flow area are as follows:

1. The reactor supports were considered fixed.

2. The additional shims around the hot leg restraint would restrict the movement of the pipe.

3. Insulation around the pipe will be crushed to 1/4 inch.

Therefore, the maximum hot leg break analyzed was a single-ended (1A) break. Mass and energy release rates for this case are shown in Table 6.2-15. The resulting maximum pressure in the reactor cavity is 111 psid at 0.158 seconds after the break.

For the cold leg, a double-ended break (8.55 ft^2) was assumed. Cold leg break data are listed in Table 6.2-16. The peak differential pressure for the reactor cavity is 190 psid at 0.141 seconds following the break. The design pressure for the reactor cavity is 225 psid. The penetrations are designed to accommodate a break within the penetration at full system pressure.

The Reactor Coolant System has been evaluated using the criteria of the Standard Review Plan 3.6.3, Leak-Before-Break (LBB) evaluation procedures. This criteria, in conjunction with General Design Criterion 4 (GDC-4) of 10CFR50 Appendix A, allows the exclusion of the dynamic effects of a postulated pipe rupture. LBB excludes cold leg and hot leg breaks from the Reactor Vessel (RV) cavity pressurization analysis but does not exclude the Core Flood Line (CFL) break. The CFL break will be the limiting LOCA event in the RV cavity. By using computer code CRAFT2, the maximum pressure in the RV cavity from a CFL break would be 92 psi with the Permanent Canal Seat Plate installed.

c. Steam Generator Compartment:

Only the compartment containing the pressurizer was analyzed as this one has the smallest vent areas. The nodalization is shown in Figures 6.2-21, -22 and -23. The flow diagram is in Figure 6.2-24 and is summarized in Table 6.2-13. The steam generator, pressurizer and quench tank, and the main coolant pumps were accounted for in the flow area calculations. In addition, the resulting areas were reduced by 10 percent to allow for minor obstructions such as cable trays, small pipes, and insulation. Flow coefficients were calculated as for the reactor cavity. The compressible flow equation for nozzles was used with these flow coefficients.

The maximum break analyzed in the steam generator compartment was the double-ended (14.14 ft^2) hot leg break. The corresponding mass and energy release rates are listed in Table 6.2-17. Table 6.2-18 lists the calculated and design pressures for the steam generator compartments. The design pressure for the reactor cavity is 225 psid.

d. Shield Building Annulus:

The following high energy lines pass through the Shield Building annulus:

System	Guard Pipe
Main Steam	yes
Main Feedwater	yes
Auxiliary Feedwater	no
Reactor Coolant Letdown	no
Reactor Coolant Makeup	no
Low Pressure Injection	no
Reactor Coolant Pump Seal Water	no
Containment Spray	no

The pressure response within the annulus in all cases of lines that do not have guard pipes will be less severe than the pressure response in the annulus for a main feedwater line break in the Auxiliary Building penetration rooms. The method of analysis for the main feedwater line break is described in Subsection 3.6.2.7.1.6 and the annulus pressure response curve is Figure 6.2-25. The analysis conservatively does not take into account the effect of heat sinks.

For the following systems which do not have guard pipes, a critical crack is postulated per the failure criteria of Tables 3.6-1 and 3.6-5.

Auxiliary Feedwater:

See Section 3.6.2.7.1.8

Reactor Coolant Letdown System:

During normal station operation, the letdown line penetrating the annulus is at 2200 psia and 120°F. In the event of a critical crack in the line, an increase in flow through the letdown coolers will cause the outlet temperature to increase which will be sensed by temperature switches and initiate closure of the inlet valves to the letdown coolers (see Subsection 3.6.2.7.1.7). The energy released to the annulus will cause a negligible pressure increase.

Reactor Coolant Makeup System:

During normal station operation, the makeup system pressure in the annulus area is 2250 psig and the temperature is 120°F. If a critical crack occurred within the annulus the operator will take action as indicated in Subsection 3.6.2.7.1.10. The water discharged into the annulus will not affect annulus pressure.

Low Pressure Injection System:

Exempt from critical crack postulation as discussed in Subsection 3.6.2.7.1.11.

Reactor Coolant Pump Seal Water Supply:

During normal station operation the seal injection supply is at 2800 psig and 120°F. In the event of a critical crack in the line, the operator will take action as indicated in Subsection 3.6.2.7.1.12. The energy release due to the 120°F water discharging into the annulus will not affect the annulus pressure. The station drainage system is adequate to accommodate the accumulation of water from the seal water supply system.

Containment Spray System:

Exempt from critical crack postulation as discussed in Subsection 3.6.2.7.1.14.

6.2.1.4 Testing and Inspection

On completion of the Containment Vessel fabrication and after the penetration internals were installed and the construction opening was closed, pneumatic tests were performed in accordance with the applicable requirements of the ASME Boiler and Pressure Vessel Code to demonstrate the integrity and leak tightness of the completed vessel. The bottom head of the Containment Vessel was Halide Leak Tested in accordance with Section III, Article 14,

Paragraph N-1411 of the ASME Boiler and Pressure Vessel Code prior to placing interior and exterior concrete fill.

A soap bubble inspection test was conducted with the vessel pressurized to 5 psig. Soap suds were applied to all weld seams and gaskets, including both doors of the personnel air locks. A second soap bubble inspection test was performed at 36 psig upon completion of the over-pressure test in accordance with the requirements of the ASME Boiler and Pressure Vessel Code. After successful completion of the initial soap bubble test, a pneumatic pressure test was made on the Containment Vessel and each of the personnel air locks at a pressure of 45 psig. Both the inner and outer doors of the personnel air locks were tested at this pressure. The test pressure in the Containment Vessel was maintained for at least one hour. The test pressure was maintained on each individual airlock door for at least one-half hour. Following a successful completion of the over-pressure test, a leakage test at 38 psig pressure was performed on the Containment Vessel with the personnel air lock inner doors closed. Pressure was maintained to demonstrate full compliance with the airtightness requirements. Continuous hourly readings were taken until it was shown that the total leakage during any 24-hour period did not exceed 0.5 percent of the total contained weight of air.

The tests of the airlocks included operational testing and an over-pressure test.

Penetration closure devices for electrical and hot piping penetrations were purchased by written specification from suppliers with tested closure devices for similar service. Performance data from prototype closures of similar or identical design was required as part of vendor qualifications.

Pipe penetrations which must accommodate thermal movement are provided with expansion bellows. The bellows expansion joints are designed to withstand Containment Vessel maximum internal pressure and can be checked for leak tightness when the Containment Vessel is pressurized. In addition, these joints are provided with a second seal and test tap so that the space between the seals can be pressurized to the maximum internal pressure to permit testing the individual penetrations for leakage at any time.

Penetrations which are welded directly to the Containment Vessel can be leak tested by pressurizing the entire Containment Vessel.

Electrical penetrations are provided with double seals and are separately tested. The test taps and seals are so located that the leakage tests of the electrical penetrations can be conducted without entering or pressurizing the Containment Vessel.

All Containment Vessel closures which are fitted with resilient seals or gaskets are separately tested to verify leak tightness. The covers on flanged closures are provided with double seals and with a test tap which allows pressurizing the space between the seals without pressurizing the entire containment system. In addition, provision is made so that the space between the airlock doors can be pressurized to full Containment Vessel maximum internal pressure.

6.2.1.4.1 Tests and Their Purposes

Basically, three types of tests are performed to verify containment integrity. These tests (nomenclature from Reference 7) are:

Type A Tests - Performed to verify the overall integrity of the containment under the pressure conditions that might occur following a design-basis accident. In performing Type A tests, the

containment is pressurized to the peak pressure, and the loss per unit time is measured by converting pressure loss to leakage. Measurements during the test include pressure, temperature, and vapor content of the air. Verification tests, which consist of imposing a known leak rate on the containment and comparing the measured leak rate with the known value, are run after each test.

Type B Tests - Performed on Containment Vessel components such as penetrations, air locks, and access doors. These components pass through the Containment Vessel, and pressure leak tests are performed on these components to verify their sealing capability under accident conditions.

Type C Tests - Performed on containment isolation valves and vacuum relief valves. Test procedures described in Subsection 6.2.4.4 and the Technical Specifications verify the operational capability of the valves. Type C tests are designed to measure the integrity of the valves in their isolation position.

Further descriptions of the test methods and procedures for these tests can be found in the Containment Leakage Rate Testing Program, which has been established in accordance with Technical Specifications.

a. Containment Vessel:

	Test	Purpose
1.	Overpressure Test (45 psig)	To prove the structural integrity of the Containment Vessel.
2.	Preoperational Integrated Leak Leak Rate Test (ILRT) (38 psig) Type A Test	To ensure that measured leakage is within the design bases limits.
3.	Post-operational LRT (38 psig) Type A Test	To ensure that the Containment Vessel continues to function within the leakage rates specified in the design bases.

b. Personnel Lock and Emergency Lock:

	Test	Purpose
1.	Overpressure Test (45 psig)	To prove the structural capability of the locks.
2.	Preoperational Leak Rate Test LRT (38 psig) Type B Test	To detect local leaks and ensure that measured leakage is within the design basis limits.
3.	Post operational LRT (38 psig) Type B Test	To detect local leaks and ensure that the locks and their components continue to function within the design basis leakage limits.

4.	Operational Testing of Mechanisms Type B Test	To ensure proper functioning of lock mechanisms and to detect local leaks.

c. Equipment Hatch:

	Test	Purpose
1.	Preoperational LRT (38 psig) Type B Test	To detect local leaks and to ensure that seals function within the specified leakage limits.
2.	Post operational LRT (38 psig) Type B Test	To ensure that the seals continue to function within the specified leakage limits.

d. Penetrations with Seals, Gaskets or Bellows (38 psig) Type B Tests:

Included within this group are main steam and feedwater penetrations, electrical penetrations, and the fuel transfer tube. The purpose of these tests is to find local leaks and to ensure that leakage rates are within specified limits.

e. Isolation and Vacuum Relief Valves (38 psig) Type C Tests:

All isolation valves requiring local leak testing are testable either by pressurizing between valves in series or by individual testing. Testing connections and leaktight valves are included to form test volumes where necessary to achieve this objective.

Vacuum Relief Valves are tested to ensure their proper functioning at the pressure differentials specified. These valves shall also be tested for local leaks to ensure that leakage rates are within specified limits. The Containment Leakage Rate Testing Program which has been established in accordance with Technical Specifications defines the allowable leakage limit and the test frequency.

6.2.1.4.2 Periodic Testing Frequency

Periodic Leakage Testing is performed to ensure proper maintenance and leak repair during the service life of the containment. A Containment Leakage Rate Testing Program has been established in accordance with Technical Specifications. The frequency of the Containment Vessel ILRT (Type A), and Local Leakage Rate Tests (Type B, and Type C) is in accordance with the Containment Leakage Rate Testing Program.

Operability testing of containment isolation valves and containment vessel vacuum relief valves is performed in accordance with Technical Specification requirements and the Inservice Testing Program.

In accordance with Appendix J, Option B the submittal for Technical Specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide. Davis-Besse has

included the following exception as approved by the NRC. The exception, described in Tech Spec 5.5.15, is:

- The fuel transfer tube blind flanges (containment penetrations 23 and 24) will not be eligible for extended test frequencies. Their type B test frequency will remain at 30 months. However, As-found testing will not be required.

In Technical Specification (TS) Amendment 240 (TAC No. MA6093) NRC Safety Evaluation, the NRC described the reasons for finding this testing exception justified and acceptable. The NRC staff considered the ALARA and industrial safety concerns described in DBNPS letter Serial 2572, and the excellent testing history of these flange assemblies (based on the successful as-found test history associated with the double 0-ring seal configuration). Based on this and the continuation of the 30-month testing frequency, the exception for the fuel transfer tube blind flanges was approved in (TS) Amendment 240.

6.2.1.4.3 Test Methods

Overpressure Test:

After successful completion of the initial soap bubble test, a pneumatic pressure test was made on the Containment Vessel in conformance with Article 7 of Subsection B, Section III of the ASME Code. Both the inner and the outer doors of the personnel lock were tested at the over-pressure. The overpressure test on the vessel was maintained for not less than one hour.

The containment was then subjected to a pressure test equivalent to 125 percent of the postulated maximum accident pressure (45psig) in accordance with Safety Guide No, 18. This test provided a direct verification of the structural integrity of the containment as a whole.

A second soap bubble test was conducted at 36 psig and at ambient temperature upon completion of the over-pressure test.

Preoperational Integrated Leak Rate Test (ILRT):

The test procedure was in accordance with proposed Appendix J of 10CFR50, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," and "American National Standards Leakage Rate Testing of Containment Structures for Nuclear Reactors," ANSI N45.4-1972.

Operational ILRT:

The test procedure is in accordance with the Containment Leakage Rate Testing Program, which has been established in accordance with Technical Specifications.

6.2.1.5 Instrumentation Application

SFAS instrumentation is provided to measure the Containment Vessel pressure and the reactor coolant pressure. The Safety Features Actuation System continuously monitors this instrumentation and takes appropriate action to isolate the containment upon detecting Containment Vessel pressure, or reactor coolant pressure levels indicative of a loss-of-coolant type accident. The status of all valves necessary to achieve Containment Vessel isolation is continuously displayed in the main control room. Failure of any isolation valve to close when required will result in an open status indication and an alarm. See Section 7.3 for SFAS details.

6.2.2 Containment Vessel Heat Removal Systems

6.2.2.1 Design Bases

The Containment Vessel heat removal systems are composed of the Containment Air Cooling System and the Containment Spray System. The Containment Air Cooling System is designed to remove heat from the containment atmosphere during normal operation. In the event of a LOCA, the systems provide cooling of the containment atmosphere to reduce the pressure build-up in the Containment Vessel and thus reduce the leakage of airborne and gaseous radioactivity from the Containment Vessel.

Post LOCA containment heat removal is effected through the use of the Containment Air Cooling System and the Containment Spray System. There are two containment sprays and three containment air coolers. The capacity of the containment heat removal systems is based on a containment heat load of 150 x 10^6 Btu/hr. The LOCA considered to determine the heat load was a split of the hot leg reactor coolant pipe. Each containment spray and each containment air cooler is designed for 50 percent of the heat load (75 x 10^6 Btu/hr). Two fully redundant heat removal methods composed of one containment spray train and one containment air cooler train are provided for post-LOCA heat removal.

The design parameters for the portions of the heat removal systems located outside the containment are described in Subsections 6.2.2.2 and 9.2.1.

6.2.2.2 System Design

6.2.2.2.1 Containment Air Cooling System

System Description:

The Containment Air Cooling system is composed of three air cooler units located within the Containment Vessel. Two of the three units are used for both normal and emergency cooling. Each unit consists of finned tube cooling coils and a direct driven fan. The fans are designed to operate under normal conditions at full speed, and at half speed during LOCA conditions.

The Containment Air Cooling System is designed to control the Containment Vessel ambient air temperature to a maximum of 120°F with two of the three units operating. The ductwork distribution system is designed to distribute air over and around all equipment which produces or releases heat.

The rated capacity of each cooling units is 75x10^6 Btu/hr at design post-accident conditions. The design data for the containment air coolers is shown in Table 6.2-19.

Refer to Subsection 6.2.1.3.2 for current analysis of the Containment Air Coolers for LOCA operation.

Cooling water for the air cooler units is supplied by the Service Water System. The service water supply line for each operating cooler has a normally open isolation valve. A valve is provided in the discharge line from each cooler. The valve is operated fully open during normal station operation. In the event of a LOCA, a safety features actuation signal will start the Containment Air Cooling fan in slow speed. The control valve in the discharge service water line is interlocked to the fan such that it will actuate to its full open position to allow full water flow through each air cooler unit operating in half speed.

In the event of a loss of offsite power (LOOP), a possibility exists for a water hammer in the Containment Air Coolers Service Water piping due to the stopping and subsequent restarting of

the Service Water Pumps. To mitigate the pressure transient experienced due to the water hammer event and prevent damage to the Service Water piping and Containment Air Cooler coils, this piping and the Containment Air Cooler coils are isolated following a LOOP. To accomplish this, the Containment Air Cooler Service Water inlet valves (SW1366, SW1367 and SW1368) are signaled to close upon restoration of electrical power following the LOOP event. Service Water flow to the Containment Air Coolers is then manually restored if the CAC fans were operating in fast speed prior to the LOOP. If the CAC fans were operating in slow speed prior to the LOOP, Service Water flow will be automatically restored (in the same sequence that occurs when a LOCA signal is present, as described in the following discussion). If a LOCA signal is present in conjunction with the LOOP, the inlet valves automatically open, following a time delay, to a preset throttled position, if the associated CAC fan is running, to refill the Service Water piping. Once the piping is refilled, the Containment Air Cooler Service Water inlet valves fully open to establish design flow rates through the coils. Reference Table 6.2-21 for a single failure analysis of the Containment Vessel Heat Removal systems.

During winter operation, the Service Water modulating control valve for one of the inservice Containment Air Cooling units may be kept fully open, with the associated unit fan stopped. This will help to maintain sufficient service water pump flow without inducing low containment temperatures during power operation. Two Containment Air Cooling units will continue to respond during LOCA scenarios as described above. The Service Water System is shown on Figure 9.2-1. The air cooler units are shown on Figure 9.4-12.

The design specifications for the Service Water System including design head of pumps, flow rates heat removal capabilities, and a discussion to demonstrate that the system can perform its intended function is included in Subsection 9.2.1 and Table 9.2-1.

The components of the Containment Air Cooling System are designed for operation in a post LOCA environment inside the containment. The system components are designed to withstand a maximum radiation dose of 10^8 R occurring during a loss-of-coolant accident.

The Containment Air Cooling System is designed in accordance with the recommended practices of the American Society of Heating, Refrigerating, Ventilating and Air Conditioning Engineers, Inc., the Air Moving and Conditioning Association and the National Fire Protection Association. The ductwork distribution system is designed and fabricated in accordance with the Sheet Metal and Air Conditioning Contractors National Association (SMACNA) Standards.

The redundant equipment is operated periodically to ensure operability.

Containment Air Cooler Units:

The original containment air coolers were manufactured by the American Air Filter (AAF) Company, Inc.

Replacement cooling coils were provided by Aerofin Corporation. The replacement cooling coils have very similar construction and performance characteristics to the original AAF coils. The predicted performance of the replacement cooling coils following an accident is based on validated flow tests performed by Aerofin in a simulated LOCA environment.

The original curve of air cooler performance showing energy removal rate as a function of containment atmosphere temperature is given in Figure 6.2-26.

Coil calculations to determine the size of the coils used 0.00045 fouling of the secondary side (within tube) of the cooling coils. This is equivalent to a 75 percent cleanliness factor.

The original FSAR analysis of the heat removal capability of an air cooler used 85°F cooling water inlet temperature to the coils. The heat removal capability of an air cooler was reanalyzed using service water temperatures ranging from 90 °F to 110°F to support increasing the ultimate heat sink temperature from 85°F to 90°F. The revised heat removal capacity used an overall fouling of 0.003. The heat removal capability of a Containment Air Cooler has been reanalyzed using a CAC slow speed fan air flowrate of 45,000 cfm.

Typical maximum, minimum, and monthly average temperatures of the lake water are shown below:

Lake Erie Water Temperature Data

	MEAN		MAXIMUM		MINIMUM	
	Port Clinton	Toledo	Port Clinton	Toledo	Port Clinton	Toledo
January	35.8°F	33.9°F	49°F	41°F	34°F	32°F
February	36.4	34.1	44	39	33	32
March	39.1	36.1	50	46	34	32
April	48.6	45.3	58	57	36	33
May	59.3	56.3	68	67	50	46
June	67.8	66.9	78	77	59	53
July	74.5	73.5	82	81	56	64
August	73.3	74.0	81	82	67	63
September	69.0	68.8	78	81	54	57
October	57.5	58.0	69	70	45	44
November	46.4	46.3	72	71	36	34
December	37.6	35.9	44	50	33	32

The containment Air Cooling System ductwork required to remain intact following a loss-of-coolant accident consists of the portions of the discharge air ductwork that extend between the containment air cooler fans and the backdraft dampers, upstream of the supply plenum. This ducting and the related fusible dropout registers therein are designated as Seismic Class I.

Consideration has been given in the detailed design of the air cooler unit housings and the Seismic Class I ducting to withstand the design basis pressures resulting from a loss-of-coolant accident.

Four fusible dropout registers (fusible linked patches) are located in the air discharge ducting of each containment air cooler fan. Physical details of the fusible dropout registers are shown on Figure 6.2-27.

The dropout registers are held in place by fusible links and are designed to disengage when the fusible links melt. The fusible links are rated for 165°F and are UL approved.

The post-LOCA air distribution system is designed to discharge the air in four horizontal directions from each unit through the openings left by the fusible dropout registers.

6.2.2.2.2 Containment Spray System

System Description:

Containment Spray System and Iodine Removal:

The Containment Spray System is an engineered safety feature which has the dual function of removing heat and fission product iodine from the post- accident containment atmosphere. The absorption of iodine by the containment sprays is accomplished mostly due to the large amount of surface area continuously available for mass transfer between the spray solution and the containment atmosphere. The removal of airborne iodine from the containment atmosphere serves to limit the potential dose to receptors located at the site boundary and outer boundary of the low population zone to within 10CFR100 guideline values.

A functional drawing of the system is shown on Figure 6.3-1. The system serves no function during normal operation.

Removal of heat is accomplished by directing borated water spray into the containment. The system consists of two half-capacity pumps, two half-capacity spray headers, isolation valves, and the necessary piping, instrumentation, and controls. The pumps and valves can be remote manually operated from the control room.

High Containment Vessel pressure or low reactor coolant pressure will actuate Level 2 trip to open the spray isolation valves. High-high containment pressure will actuate Level 4 trip to start two containment spray pumps. The pumps take suction initially from the Borated Water Storage Tank. The Containment Spray System shares the Borated Water Storage Tank (BWST) and the suction lines from the tank with the high and low pressure injection systems.

After the water in the Borated Water Storage Tank reaches a low level, the spray pump suction is transferred to the Containment Vessel Emergency Sump. A BWST low level alarm and permissive alarm is set at approximately nine feet (setpoint of 108.5 ± 1.5 in. H_2O) above the top of the outlet pipe (4” above the bottom of the tank) to alert the operator in sufficient time to perform the switchover without interrupting spray operation. Baskets of Na_3PO_4 are available in containment so that when sump flooding occurs, neutralization will result. The spray pH upon recirculation is then 7.0 or greater. The Containment Vessel Emergency Sump water is cooled by the Decay Heat Removal System as shown in Figure 6.3-2A.

Pump motor power is supplied from normal sources with backup supplies from the emergency diesel generators. Design data for the spray system components are given in Table 6.2-20 and Figures 6.2-28 and 6.2-29. Design data for components of the Decay Heat Removal Systems used in the recirculation phase are given in Chapter 9 and supplemented by Tables 9.3-10 and 9.3-11.

The containment spray nozzles are installed on two containment ring headers. Each header has 90 nozzles, for a total of 180 nozzles. The construction of the headers and the nozzles is shown in Figure 3.6-25.

The containment spray nozzles are constructed of AISI 316 stainless steel, and are designed, fabricated, examined, and tested, where applicable, in accordance with the requirements of ASME Section III for Class 2 components. The nozzles are designed for service with borated demineralized water containing up to 13,000 ppm boric acid solution at temperature ranges between 50 and 300°F. (Since the Containment Spray Nozzles are fabricated from type 316

stainless steel throughout, temperature excursions significantly beyond this nominal borated water temperature range would have no impact on their design or operation.) Each nozzle will have the capacity to release 15 gpm at 15.0 psig differential pressure. The spray pattern is the full cone type with completely uniform distribution throughout the spray pattern.

The histogram and particle size versus volume percentage for the spray nozzle are shown in Figures 6.2-31 and 6.2-32. The measurement was conducted by Spray Engineering Company, Burlington, Massachusetts in their Hydrodynamic Laboratory. The test was conducted at 15 psig with nozzle spray vertically downward at an elevation of 10 feet below the nozzle. The spray distribution was measured for comparison purpose by taking the distribution in two perpendicular planes to the spray axis over a time interval. The particle size was measured by high speed photography. The droplet images were measured and recorded from negatives and presented on a histogram representing the entire count. Photographs of the drops were taken in four quadrants of the spray, each quadrant was divided into zones. The number of photographs taken in each zone was determined by the frequency of drops and the percent of volume of water falling into that zone. A total observation count of 5,825 droplets was taken to assure the best possible test result.

Design Basis:

The Containment Spray System is capable of reducing elemental and particulate iodine fission product concentrations such that the offsite radiation exposures resulting from a design basis loss-of-coolant accident (LOCA) are within the guideline values of 10CFR100.

The Containment Spray System is designed so that a single active failure during injection phase, or a single active or passive failure during the recirculation phase, cannot impair the system's ability to comply with its safety design basis.

The Containment Spray System is designed to remain functional after a safe shutdown earthquake and is protected from flooding, pipe whip, and jet impingement forces.

The Containment Spray System is placed in operation automatically following a loss-of-coolant accident. The actuation system is designed in accordance with IEEE-279. The spray pattern of either of the two independent and redundant spray headers gives adequate volumetric coverage for containment fission product removal.

The Containment Spray System is designed to draw water from the BWST during the initial phase of operation. Water in the BWST is maintained at a pH of approximately 5.0.

Upon depletion of the water in the BWST, a recirculation phase is provided to maintain spray. Trisodium phosphate baskets in the containment maintain the spray solution pH at a minimum of 7.0 or greater during the recirculation phase.

System Design:

The spray removal of elemental and particulate iodine is discussed in Subsection 15.4.6.4. The Containment Spray System does not have a provision for additive injection for iodine removal.

The BWST contains 500,100 gallons of borated water of which 360,000 gallons are available to serve one low pressure injection/decay heat pump (3,000 gpm), one high pressure injection pump (500 gpm), and one containment spray pump (1,300 gpm). The BWST will be available for Emergency Core Cooling System operation for approximately 75 minutes.

The emergency function of reactor coolant recirculation is performed when the tank level decreases to approximately 9 feet above the bottom of the tank. The spray operation will continue without interruption during switchover. The containment spray pump will be operated throughout the operation. To assure that there is adequate NPSH available for the pump, the downstream motor-operated globe valve (isolation valve) will be automatically throttled to a preselected opening. A flow indicator and high and low flow alarms are provided to monitor the proper function of the system.

Codes and Standards:

The system components and equipment are designed in accordance with the applicable codes and standards listed in Table 9.0-1 for the Decay Heat Removal System.

Material Compatibility:

The components in contact with the borated water are constructed of stainless steel. None of the active components of the Containment Spray System is located within the Containment Vessel; so none is required to operate in the steam-air environment produced by a LOCA.

Component Design:

a. Pumps

The containment spray pumps are liquid penetrant tested and are hydrotested and qualified to be able to withstand pressures greater than 1.5 times the design pressure. The systems are designed so that periodic testing of the pumps may be performed to ensure operability at all times. The operating characteristics of the pumps are verified by shop testing before installation.

b. Valves

A remotely operated valve is located on the discharge side of each pump to provide containment isolation and to prevent cavitation when suction is shifted from the Borated Water Storage Tank.

The Containment Spray System valves are designed and inspected to the same code requirements as the valves in the Emergency Core Cooling Systems; refer to Section 6.3.

c. Spray Headers and Nozzles

Ninety full cone nozzles are arranged on each of the two spray headers. The spray nozzles are spaced on the headers to provide uniform spray coverage of the containment volume above the operating floor. The minimum spray drop height is approximately 138 feet, the droplet trajectories will, however, yield an effective drop height greater than this.

The spray nozzles on each header are so arranged that there will be adequate coverage at 100 feet below the headers.

The spray nozzles are liquid penetrant tested in accordance with ASME Code Section III, Class 2.

d. Piping

The entire system is of welded construction except for the sections of piping requiring flanged connections for maintenance. The design conditions for the containment spray system are shown in Table 6.2-20.

The piping for this system is designed, fabricated and inspected in accordance with the ASME Code Section III, Class 2.

e. Coolant Storage:

The Containment Spray System shares BWST capacity with the HPI system and the LPI system.

The total volume of borated water which will be retained below the elevation at which water would begin to overflow into the Containment Emergency Sump, following a LOCA, is approximately 200,000 gallons.

The volume of borated water above the emergency sump, based on a contained volume of 500,100 gallons in the BWST (360,000 gallons available from the BWST) will be approximately 160,000 gallons, which is over 2.0 feet above the sump.

f. Motors:

The containment spray pump motors are designed to meet the requirements listed in Section 3.10 and Table 6.2-20.

Pump Characteristics:

At the design flow of 1300 gpm the required NPSH for the Containment Spray pumps is 9 feet. Pump curves showing total dynamic head and NPSH versus flow are shown in Figures 6.2-28 and 6.2-29.

Available NPSH to the Containment Spray Pumps:

The available NPSH to the containment spray pumps calculated in accordance with NRC Regulatory Guide 1.1 is described in Subsection 6.3.2.14.

6.2.2.3 Design Evaluation

In establishing the design of the containment heat removal systems, the following factors have been considered:

6.2.2.3.1 Missile Protection

Protection against missile damage is provided by direct shielding or by physical separation of duplicate equipment. The air cooler units are located outside the secondary shield at an elevation above the water level in the bottom of the Containment Vessel at post-accident conditions. In this location the units are protected against credible missiles and from being flooded. The spray headers are located outside and above the primary and secondary concrete shield.

6.2.2.3.2 Environment

All equipment, piping, valves and instrumentation associated with the Containment Air Cooling System are designed to withstand the temperature and pressure transient conditions resulting from a LOCA and the seismic forces resulting from the Maximum Possible (larger) Earthquake. None of the active components of the containment spray system are located within the Containment Vessel, so none are required to operate in the steam-air environment produced by the accident.

6.2.2.3.3 Loss of Power

The failure of the normal and reserve electrical power supply automatically places all containment air cooler units and containment spray pumps on the emergency power source from the emergency diesel generators.

6.2.2.3.4 Fan Operating Speed

The containment air cooler unit fans and motors are designed to operate under normal conditions at full speed, and at half speed during LOCA conditions. The motors are directly connected to the fan wheel.

6.2.2.3.5 Excessive Water Flow

Excessive service water flow due to leakage from the containment air cooler unit coil can be determined by containment sump level changes and isolation of each individual unit.

6.2.2.3.6 Availability Monitoring

The components of the Containment Air Cooling System are used during normal operation and hence are continuously monitored for availability for post-accident operation.

6.2.2.3.7 Previous Experience

Containment air cooler unit cooling coils of similar design have been analyzed under simulated accident conditions utilizing a software program validated by flow tests representing a post LOCA environment.

6.2.2.3.8 Capacity

The Containment Spray System heat removal capacity is based on the spray water reaching thermal equilibrium with the steam-air mixture within the Containment Vessel.

6.2.2.3.9 Response Time

The anticipated delay time for the containment spray system to deliver borated water into the containment atmosphere consists of the following time intervals:

	Time Interval (sec)
Total instrumentation lag	5
Signal to start pump	30
Isolation valve opening time	30 (non additive)
Water delivered to highest spray header (includes 4 seconds for pump acceleration at 70% voltage)	44

Therefore, the time required for the first delivery of borated water into the containment atmosphere would be approximately 79 seconds.

The Containment Spray System is designed to deliver full flow to the spray nozzles within 80 seconds after the LOCA. To account for potential increases in the time required for full Containment Spray flow to be delivered to containment, 160 seconds is assumed in the containment response analysis after a LOCA. This reanalysis was performed as part of increasing the ultimate heat sink temperature from 85°F to 90°F.

6.2.2.3.10 Physical Location

The spray pumps are located in separate rooms in the lowest level of the Auxiliary Building.

6.2.2.3.11 Independence

The containment spray system can be operated independently or with the Containment Air Cooling System to accomplish the heat removal capability as described in Subsection 6.2.2.1.

6.2.2.3.12 Safety Evaluation

Safety evaluation of the service water system is discussed in Subsection 9.2.1.3. A single failure analysis has been made on components of the systems to show that failure of any single component as shown in Tables 6.2-21 and 9.2-3 will not prevent fulfilling of the design functions. A discussion of the consequences of accidents under which the containment function becomes essential is included in Chapter 15.

6.2.2.4 Testing and Inspections

The equipment, piping, valves and instrumentation are arranged so that all items can be visually inspected. The air cooler units and associated piping are located outside the secondary concrete shield around the Reactor Coolant System loops. The service water piping and valves outside the Containment Vessel are inspectable at all times. Operational tests and inspections were performed prior to initial startup.

Periodic testing of the containment air cooler units is given as specified in the Technical Specifications.

The Containment Spray System is not normally operating. The Containment Spray System is tested in accordance with Technical Specifications.

Components of the Containment Spray System are tested as follows:

Containment Spray Pumps	These pumps are tested singly by closing the spray header valves and opening the valves in the test line. Each pump in turn is started by operator action and checked for flow to each of the spray headers. This test also verifies flow through each of the Borated Water Storage Tank outlet valves.
Borated Water Storage Tank Outlet Valves	These normally open valves are cycled by remote operator action to ensure isolation closure.
Containment Spray Isolation Valves	With the pumps shut down and the normally locked-open block valves upstream closed, these valves are each opened and closed by operator action.
Spray Nozzles	When the unit is shut down, air or smoke is blown through the test connections with visual observation of the nozzles.
Containment Vessel Emergency Sump Line Valves	These valves are each cycled opened and closed by operator action.

Testing and inspections of the service water system are discussed in Subsection 9.2.1.4.

6.2.2.5 Instrumentation Application

Instrumentation is provided to measure the containment pressure and the reactor coolant pressure. The Safety Features Actuation System continuously monitors this instrumentation and takes appropriate action to initiate the containment heat-removal systems upon detecting containment pressure or reactor coolant pressure indicative of a loss-of-coolant accident. The status of the containment air coolers and the containment spray pumps and spray isolation valves are continuously displayed in the main control room. Failure of any of these components to go to its proper safety features position when required will result in an incorrect status indication and an alarm. The containment pressure and containment air coolers inlet and outlet temperatures are monitored and displayed in the main control room during operation of the containment heat-removal systems. See Section 7.5.

6.2.2.6 Containment Vessel Emergency Sump

6.2.2.6.1 Design Bases

The Containment Vessel Emergency Sump, located inside the Containment Vessel, is an open-top concrete structure. One exit nozzle in the horizontal position in the sump is provided for each recirculation line.

Following a LOCA, after the BWST has been exhausted, the Containment Vessel Emergency Sump will serve continuous injection of the reactor coolant, through the low pressure injection/decay heat pump, into the Reactor Coolant System. This will maintain long-term core

cooling by recirculating the spilled reactor coolant back to the reactor vessel and/or through the containment spray pump, into the Containment Vessel atmosphere to remove the heat and decrease the pressure and temperature in the Containment Vessel.

Each of the two recirculation loops is designed to provide a separate and independent flow path. Each loop is designed for the maximum heat load.

The Containment Vessel Emergency Sump does not serve any normal function.

6.2.2.6.2 Description and Design Evaluation

The Containment Vessel Emergency Sump (upper strainer) consists of one sump, two horizontal exit openings, vertical strainer assemblies and antivortexing grates.

The Containment Vessel Emergency Sump provides the suction for the post-LOCA recirculating of the reactor coolant for long-term emergency core cooling. Each of the two exit lines is sized for carrying the maximum flow rate of one low-pressure injection and one containment spray pump.

The Containment Vessel Emergency Sump is located at El. 565 feet, which provides an adequate net positive suction head (NPSH) based on the maximum flow rate of the pumps.

The sump is equipped with cylindrical shaped strainers to prevent large particles from getting into the recirculating line and plugging up the spray nozzles and/or damaging the pump. The perforated plate strainers have 3/16-inch diameter openings and are fabricated of perforated stainless steel sheets rolled into cylinders. To supplement the "upper" strainers installed over the sump at El. 565 feet additional "lower" strainer are installed in the incore instrumentation tunnel. An opening is provided in the sump wall to provide a flowpath between the tunnel and sump. Adequate free-flow area is provided to ensure minimal flow resistance under conservative debris loading conditions following a LOCA. The strainer is designed to accommodate postulated accident conditions inside Containment under which transient material (fibrous and metallic insulation) and unqualified coatings are created and is specifically designed to preclude these items from jeopardizing sump performance and long term recirculation operation. A system of trash racks is also provided in key passageways inside Containment to intercept large debris before reaching the sump area. The refueling canal drain to the reactor cavity and sump floor drain features similar protection to maintain unobstructed flow paths supporting overall sump operation.

The strainer, trash racks, and the recirculation lines are constructed of stainless steel.

The recirculation lines are fabricated and tested in accordance with ASME Section III, Class 2.

The sump, strainers, anitvortexing grating, trash racks, and recirculation lines are Seismic Class I. The strainers and supporting structure are designed to withstand the combined effects of deadweight, thermal and hydraulic loading conditions, including seismic effects.

The Containment Vessel Emergency Sump, Incore tunnel and strainer arrangement are shown in Figures 6.2-33 and 6.2-33A.

The Incore Tunnel lower strainer and Containment Vessel Emergency Sump upper strainer are supported by steel frames securely anchored to the concrete. The location of the emergency

sump is such, that, the strainers are protected from missiles by the secondary shield walls, refueling canal walls, floor at El. 578 feet 0 inches, and Containment Vessel, making it impossible for missiles to penetrate this area. A jet shield deflector is provided to protect the sump strainers from the effects of a high energy line break due to the proximity of Decay Heat system piping on El. 656 feet.

The entire Containment Vessel Emergency Sump consisting of the El. 565 feet and the incore tunnel area portions are required to mitigate a LOCA; however, for breaks located inside the reactor cavity the incore portion of the strainer is assumed to fail. For breaks located inside the reactor cavity with the incore (lower) portion failed, the El. 565 feet (upper) strainer remains intact and is designed to handle the postulated debris loading.

The structural frame supporting the strainers assures that large debris carried in the water following a LOCA will not readily damage the strainers.

6.2.2.6.3 Inspection and Test

The Containment Vessel Emergency Sump is not flow tested for recirculating mode, since the flooding of the water inside the Containment Vessel is not possible during station operation. The Containment Vessel Emergency Sump is inspected periodically to ensure the sump is free of debris. The screens are inspected for signs of distress. This inspection will include verification that no foreign material capable of strainer blockage remains in the Emergency Sump prior to plant startup.

The position of the isolation valves are monitored by the valve position lights in the control room.

6.2.3 Containment Vessel Air Purification and Cleanup Systems

6.2.3.1 Design Bases

The Containment Air Purification and Cleanup System is composed of the Emergency Ventilation System and the Purge System. The function of the Emergency Ventilation System is to collect and process potential leakage from the Containment Vessel to minimize environmental activity levels resulting from all sources of containment leakage following a loss-of-coolant-accident. The Purge System was designed to purge the Containment Vessel with clean fresh air whenever access is desired. In accordance with Amendment 221 to the Technical Specifications, the containment purge isolation valves are administratively maintained closed and control power removed in Modes 1 through 4.

The Emergency Ventilation System (EVS) is designed to provide a negative pressure within the annular space between the Shield Building and the Containment Vessel and in the penetration rooms following a loss-of-coolant-accident and to reduce airborne fission product leakage to the environment by filtration prior to release of air through the station vent.

The system has two redundant, independent subsystems, each fully capable of the functional requirement. A single failure of an active component in either subsystem does not affect the functional capability of the other subsystems.

Each of the two redundant subsystems is capable of maintaining the annulus at a measurable minimum negative pressure of 1/4 inch water gauge. The Containment Vessel leakage rate is conservatively estimated to be 0.5 percent per day of the contained air weight for determination

of EVS flow requirements. The system exhausts air as required from the annulus and penetration rooms to provide a negative pressure in the annulus. Makeup air is induced into these areas by infiltration through the mechanical penetration and ECCS room, Shield Building leakage and by potential Containment Vessel leakage. The annulus is maintained at a pressure below the outside barometric pressure and well below the Containment Vessel post-LOCA design pressure. In all cases, the exhaust capability from this region exceeds the total maximum Containment Vessel leakage rate along with the normal in-leakage.

Each of the two redundant subsystems is provided with prefilters, high- efficiency particulate air (HEPA) filters and charcoal adsorbers to remove airborne particles and methyl iodide as well as elemental iodine contaminants resulting from a LOCA. These units have a total efficiency not less than 95 percent.

The Emergency Ventilation System can also be operated in conjunction with the Containment Purge System for containment cleanup during normal operation, if required. Per Technical Specifications, the containment purge isolation valves are administratively maintained closed and control power removed in Modes 1 through 4.

The Emergency Ventilation System is designed as Seismic Class I

In addition to the Emergency Ventilation System, the Containment Spray System provides Containment Vessel air purification and cleanup. A description of the Containment Spray System is given in Subsection 6.2.2.2.2. The iodine removal function is incidental to the primary heat removal function of the sprays. When the borated water is sprayed into the Containment Vessel atmosphere there is a mass transfer of the iodine, from the air to the water droplets. The iodine remains in water solution rather than desorb to the Containment Vessel atmosphere. The boric acid contents in the BWST are maintained at a minimum of 2600 ppm and have an initial pH of approximately 5.

The Purge System is designed to provide approximately one air change per hour in the Containment Vessel. This system has the capability of tempering the outside air to maintain a temperature of 50°F to 60°F in the containment with(-)10°F outside air.

Usually, the Purge System is aligned and operated to the mechanical penetration rooms. When access to containment is desired in Modes 5 and 6, the system is realigned to containment by opening the containment isolation valves and closing the mechanical penetration room valves.

The Containment Vessel Purge System fans and ductwork are designed as Seismic Class II. The containment isolation valves are designed as Seismic Class I.

Engineered Safety Feature of Air Filtration System:

The Emergency Ventilation System filter units, and the Control Room Emergency Ventilation System filter units, are in accordance with the positions in the NRC Regulatory Guide 1.52, Revision 2, except as follows:

a. The filter systems are not equipped with demisters, heaters, nor after-HEPA filters, as described in Section C, Paragraph 2.a of the Regulatory Guide.

b. There are no pressure relief valves in the event of pressure surges, as described in Section C, Paragraph 2.d of the Guide.

c. Paragraph 2.1 of Section C of the Guide states: "ESF atmosphere cleanup system housings and ductwork should be designed to exhibit on test a maximum total leakage rate …" (i.e. 0.1% of system flow).

   The filter system housings are of leaktight construction. After completion of all welding, each housing Section must pass an internal air pressure test of 1.0 psig for a 5-minute period.

d. Deleted

e. Deleted

f. Paragraph 4.d of Section C of the Guide states: "Each ESF atmosphere cleanup train should be operated at least 10 hours per month, with the heaters on (if so equipped), in order to reduce the buildup of moisture on the adsorbers and HEPA filters."

   ESF atmosphere cleanup units are not provided with heaters. The units are operated at least 15 minutes per month in accordance with the applicable Davis-Besse Technical Specifications Sections.

g. Paragraphs 5.c and 5.d of Section C of the Guide state: "The in-place DOP test for HEPA should be tested...at least once per 18 months…to confirm a penetration of less than 0.05% at rated flow". "The activated carbon adsorber...to ensure that bypass leakage through the adsorber section is less than 0.05%...at least once per 18 months."

To conform with an extended fuel cycle, these tests are performed at 24-month intervals. During each refueling interval, "in-place" leakage tests using DOP on HEPA units and freon 112 (or equivalent) on charcoal units shall be performed at design flow on each filter train. Less than 1% penetration and bypass leakage DOP by each entire HEPA filter unit and less than 1% penetration and bypass leakage of the freon 112 (or equivalent) by each entire charcoal adsorber unit shall constitute acceptable performance. These tests must also be performed after any maintenance which may affect the structural integrity of the filtration system units or of the housing.

h. Paragraph 6.a of Section C of the Guide states: "New activated carbon meets the physical property specifications given in Table 5-1 of ANSI N509-1976, "Nuclear Power Plant Air Cleaning Units and Components". ANSI N509-1976, Table 5-1, specifies that an acceptable test method is RDT M16-1T, "Gas-Phase Adsorbents for Trapping Radioactive Iodine and Iodine Compounds," and that the test be performed at 80°C and 95% relative humidity and the pre-loading and post-loading sweep medium is 25°C.

   In lieu of these requirements, testing is performed in accordance with ASTM D3803-1989, "Standard Test Methods for Radioiodine Testing of Nuclear-Grade Gas-Phase Adsorbents." New activated carbon is to meet the physical property specifications given in Table 5-1 ANSI N509-1980.

6.2.3.2 System Design

The systems are shown on Figures 9.4-11 and 9.4-12.

The emergency ventilation equipment consists of two-full-capacity redundant fan-filter assemblies. Each filter system is made up of two units - the first contains roughing, HEPA filters, and charcoal adsorbers in series; and the second, a duplicate set of charcoal adsorbers. The arrangement allows each set of charcoal adsorbers to be tested separately. Each fan-filter assembly and associated ductwork are designed for a flow rate of 8,000 cfm.

Following a loss-of-coolant accident, a safety features actuation signal starts the Emergency Ventilation System fans and open the dampers located in the penetration room's outlet ductwork. The safety features actuation signal closes all containment isolation valves, the mechanical penetration room, purge system valves and the connection between the emergency ventilation system and the fuel handling area. The purge system fans, if running, are shut down automatically. A differential pressure controller regulates the exhaust air modulating damper and the recirculation air modulating damper to maintain the set point negative pressure within the negative pressure areas. Layout of system equipment and air flow guidance ducts are shown on Figures 6.2-34, 6.2-35 and 6.2-36.

Immediately after the loss-of-coolant accident, the temperature and pressure within the Containment Vessel rises rapidly, causing an increase in the air temperature in the annulus. This increase in air temperature causes an increase in the annulus air pressure. The air pressure in the annulus rises and then decreases to a pressure below the outside barometric pressure within a short time after the Emergency Ventilation System is started. The fan starts with one hundred percent exhaust mode, and partial recirculation of the annulus air begins as the recirculation damper begins to open in order to maintain the set-point negative pressure. The flow split between the recirculation and exhaust ducts are determined by the rate of thermal expansion of air into the annulus and Containment Vessel and Shield Building in-leakage rates. As the rate of thermal expansion decreases, the rate of recirculation increases. When the annulus temperature has ceased to rise, the exhaust to the station vent is just that required to offset the air mass addition due to Containment Vessel and Shield Building in-leakages.

The recirculated discharge from the fan is returned to the penetration rooms to enhance the mixing of Containment Vessel leakage, thereby avoiding direct streaming of the radioisotopes to the filter system and increasing holdup within the annulus. The entire system is designed to operate under negative pressure up to the fan discharge.

Differential pressure indicators are provided across the filters to indicate filter dust loading.

The assumptions, the mathematical model used in the annular space pressure response analysis, and the plotted results are presented below:

Assumptions:

a. Various effective leakage areas in the negative pressure boundary are considered: 0 ft^2 - 3.2 ft^2.

b. The annular space, mechanical penetration rooms, and the ESF pump rooms are at the same pressure throughout the transient.

c. Heat transfer contributions from the Containment Vessel cylinder and dome are treated separately as a consequence of their differing thicknesses.

d. Both convective and radiant heat transfer to and from the steel containment wall and the concrete Shield Building wall are taken into account.

e. Radiant heat emission from and absorption to both walls as well as the air are all included.

f. Homogeneous thermodynamic equilibrium within the annulus vapor space is assumed.

g. The annulus air is assumed to be an ideal gas.

h. One-dimensional transient heat conduction is modeled for the concrete walls, with the outer boundary assumed insulated during the transient.

The analysis of the annulus space is based on the conservation of mass and energy and the equation of state. These equations are summarized below:

a. Conservation of mass

$$\frac{dm}{dt} = \dot{M}_{in} - \dot{M}_{out}$$

where

m = mass of air in the system (lbm)

$\dot{M}_{in}$ = inleakage mass flow rate (lbm/hr)

$\dot{M}_{out}$ = mass flow rate discharged by fan (lbm/hr)

b. Conservation of Energy

$$mCv\frac{dT}{dt} + u\frac{dm}{dt} = Q_{c1} + Q_R - Q_{c2} + \dot{M}_{in} h_{in} - \dot{M}_{out} h_{out}$$

where

Cv = specific heat of air at constant volume

= 0.171 Btu/lbm-°R

T = temperature of the annulus air (°R)

h,u = specific enthalpy, internal energy of the air (Btu/lbm).

Q_{c1} = convective heat transfer from the steel containment wall to the annulus air (Btu/hr).

Q_{c2} = convective heat transfer from the annulus air to the concrete wall (Btu/hr).

Q_R = the net radiant heat to the air from the two gray walls (Btu/hr)

$Q_{c1} - Q_{c2}$ = the net convective heat to the annulus air.

c. Equation of state for ideal gas

$$P = \frac{m}{V} RT$$

P = annulus air pressure (lbf/ft^2)

V = total system volume (ft^3)

R = gas constant = 53.34 (ft-lbf/lbm-°R)

m = mass of air (lbm)

The convective heat transfer coefficients for the cylindrical and dome portions of the steel Containment Vessel and the Shield Building are treated separately.

For the cylindrical part [1]

$$\overline{N}u = 0.021\ (Pr\ Gr)^{2/5}$$

For dome part[1]

$$\overline{N}u = 0.14\ (Pr\ Gr)^{1/3}$$

where

$\overline{N}u$ = average value of Nusselt Number over surface

Pr = Prandtl Number

Gr = Grashof Number

The radiant heat is calculated by Kirchhoff's law of gray body radiation

$$E_g = \varepsilon_g \sigma T^4 \quad Btu/ft^2$$

Where

E_g = radiated energy from a gray body (Btu/hr-ft^2)

ε_g = gray wall emissivity

σ = Stefan-Boltzmann constant
17.3 x 10^{-10} Btu/hr - ft^2 —°R^4

T = Temperature of the gray body (°R)

The phenomena of emission and reflection from the two gray walls and absorption and emission of the air are taken into account, yielding a net radiant heat input rate to the annulus air as shown in Figure 6.2-41.

[1]Rohsenow and Hartnett, "Handbook of Heat Transfer"

One-Dimensional transient heat conduction to the concrete wall is described as follows:

$$\lfloor \rho c_p \rfloor \frac{\partial T}{\partial t} = \frac{\partial}{\partial x} \left[k \frac{\partial T}{\partial x} \right] + S\,(x,t)$$

Where

T = temperature of the concrete wall (°F)

$\lfloor \rho c_p \rfloor$ - volumetric heat capacity of the concrete (Btu/ft^3 - °F)

k = thermal conductivity of the concrete (Btu/hr-ft-°F)

S = heat source (Btu/hr-ft^3)

The thermal expansion and pressure of the steel containment is treated in the analysis and results in a reduction of the annular volume and an increase of the heat transfer area. Heat loads from ECCS equipment within the negative pressure boundary are added to the total heat load until the ECCS rooms coolers offset this heat load.

Results:

The analysis was run assuming different effective leakage areas in the negative pressure boundary ranging from 0 ft^2 to 3.2 ft^2. The results are that a negative pressure of 0.25 inch w.g. is established as shown in Figure 6.2-42 for varying C_{out}. The worst case (C_{out}= C_{IN}) indicates that an allowable effective leakage area of 2.4 ft^2 will ensure that the 13-minute drawdown time, assumed in Chapter 15, Accident Analysis, is met. The 2.4 ft^2 effective leakage area is the total assumed boundary leakage, which includes non-designated, intrinsic leakage paths such as door seal gaps and crevices in buildings, structures, as well as any tracked leakage areas.

The drawdown time in the negative pressure boundary increases markedly as a function of leakage area when size is greater than 2.4 ft^2. Therefore, the negative pressure boundary is periodically tested to verify that this value of 2.4 ft^2 is not exceeded.

The case using an effective leakage area of 0 ft^2 is illustrated in Figures 6.2-37 through 6.2-41.

As stated below, the negative pressure boundary pressure response given in Figure 6.2-37 is based on a constant EVS fan capacity of 8000 cfm and no outleakage through the negative pressure boundary. Since the calculated pressures during the transient are significantly positive, based on the fan curve it is determined that the flow through EVS will be significantly higher than that assumed in the analyses on which Figure 6.2-37 is based.

The negative pressure boundary pressure response following a postulated LOCA was re-evaluated by taking credit for additional flow through the EVS based on the fan curve. These calculations show that the peak pressure in the negative pressure boundary will be less than 0.64 psig assuming that the system flow resistance corresponds to the minimum EVS flow, 7200 cfm, allowed by DBNPS Technical Specifications. The calculated negative pressure boundary pressure will be approximately 0.55 psig for system resistance corresponding to a nominal flow of 8000 cfm. These calculations also assumed that there is no outleakage through the negative pressure boundary. Sensitivity studies have shown that very small amounts of outleakage are highly effective in limiting the positive pressure in the negative pressure boundary following a LOCA. The revaluated pressure response is given in Figure 6.2-37A. Based on conservatism in these evaluations it is concluded that a nominal blowout panel pressure setpoint of 0.65 psid will assure that the blowout panels on penetration rooms will not prematurely lift during a postulated LOCA and will also limit the containment external differential pressure to approximately 0.8 psid following a main feedwater line break in the penetration room. Although, the calculated draw down time using new analysis would be smaller than the draw down time calculated using a constant EVS flow, Chapter 15 Accident Analysis basis was not changed.

The inputs for the analyses are as follows:

a. Containment Vessel wall temperature transient Table 6.2-22 (obtained using the Bechtel COPATTA code for a 14.14 ft^2 reactor coolant system pipe rupture, initial containment temperature of 120°F, annular space temperature 85°F, heat transfer coefficient to the containment inner surface of four times Tagami coefficient).

b.	Initial Annular space temperature, °F	85
c.	Fan capacity, cfm	8000
d.	Fan starting time, sec	25
e.	Emissivity	
	Containment wall .	.96
	Concrete wall	.9
	Annulus air (with 70% relative humidity)	.17
f.	Effective leakage areas	0 ft^2 — 3.2 ft^2

g. Thermal and pressure expansion of the steel containment wall

1. Thermal expansion

Based on the linear thermal expansion coefficient

$\alpha = 6.7 \times 10^{-6}$ in/in/°F,

the thermal expansion is calculated as below:

$V - V_o – 3\alpha V_{cont} (T - T_o)$

$A = A_o [1 + \beta(T - T_o)]$

$\beta = 13.4 \times 10^{-6}$ $ft^2/ft^2/°F$

where

V_O, V	are the initial and reduced annulus volume
A_O, A	are the initial and increased Containment Vessel shell surface area
T_o, T	are the initial and increased containment wall temperature
V_{cont}	is the Containment vessel volume
β is	the surface thermal expansion coefficient

2. Pressure induced stress deformation

Based on the 37 psid maximum pressure differential across the Containment vessel wall, the maximum pressure stress induced deformations are annulus volume reduction of 4265.55 ft^3 and 89.08 ft^2 increase in Containment Vessel shell heat transfer area.

h. Pump heat

720,000 Btu/hr for 83.9 seconds (time at which ECCS room coolers offset the pump heat loads).

The effect of possible differences in the flow coefficient between the inleakage flow direction and the outleakage flow direction was investigated. Most leakage paths are characterized by a contraction, a fraction loss (which is independent of flow direction) and an expansion to a large volume. The directional dependence of the flow coefficient, as determined from the head loss coefficient, can be enveloped by minimizing the outleakage flow coefficient while maximizing the inleakage flow coefficient as shown in the table below.

Differences in Flow Coefficients Between Inleakage Flow Direction and Outleakage Flow Direction

Flow Direction	Head Loss Coefficient Contraction (1)	Head Loss Coefficient Expansion (1)	∑Ki	Flow Coefficient $C = \frac{1}{\sqrt{K}}$
Inleakage	0.0	1.0*	1.0	1.0
Outleakage	0.5	1.0*	1.5	0.816

* Since the final expansion for any leakage path is to a virtually infinite flow area, the head loss coefficient is 1.0 for this expansion regardless of direction.

(1) "Flow of Fluids Through Valves, Fittings, and Pipes," Crane Technical Paper 410, Page A-26, 1972.

From the above table, one would expect that the flow coefficient for outleakage will be no less than 0.816 times the inleakage flow coefficient.

Figure 6.2-42 shows the parametric dependence of annulus drawdown time on the degree of inequality between inleakage and outleakage flow coefficients. Note that the extremely conservative assumptions of a two-to one ratio ($C_{out} = 0.5C_{in}$) of inleakage to outleakage flow coefficients produces virtually no effect on the drawdown time vs. leakage area curve and even a ten-to-one ratio ($C_{out} = 0.1C_{in}$) produces results for all leakage areas below the design value of 2.4 ft^2 that are well below the 780-second value used in the dose consequence model (see Subsection 15.4.6.4).

Under normal operating conditions, the temperature of the air in the annular space will be above outside ambient temperature, varying from about 85°F with outside temperature at (-)10°F, to about 114°F with outside temperature at 94°F. The heating of the air within the annular space and penetration rooms reduces the relative humidity below that of the outside air introduced whenever the penetration and annular spaces are purged.

Regardless of ambient conditions, the relative humidity within the space is less than 70 percent. In the event of a LOCA, the temperature within the annular space rises rapidly, further reducing the relative humidity.

A discussion of the consequences of accidents under which the containment function becomes essential are included in Chapter 15.

The prefilters are provided to remove coarse airborne particles to prolong HEPA filter life. The HEPA filters are provided to remove fine airborne particles that penetrate the prefilters. The activated coconut shell charcoal adsorbers are impregnated to remove methyl iodine as well as elemental iodine contaminants resulting from a loss-of-coolant accident.

The following describes the Emergency Ventilation System fan and filter assembly components:

Fans:

Type	Centrifugal
Capacity, cfm	8000 each
Static pressure, in. w.g.	6.5
Brake horsepower, bhp	11.9
RPM	1997

Motors:

Type	Standard induction
Horsepower rating, hp	15
Voltage (volts)	480
Enclosure	Drip proof
Insulation class	B

Prefilters:

Quantity per unit	8
Rated flow per filter unit, cfm	1000
Type	Replaceable
Media	Reinforced nonwoven fire-retardant cotton fabric
Average efficiency, %	70
Rating basis	NIST (Formerly NBS) dust-spot method
Rated pressure drop unloaded (in. w.g.)	0.20

HEPA Filters:

Quantity per unit	8
Rated flow per filter unit, cfm	1000
Type	High efficiency, dry
Media	Glass fiber (waterproof, fire retardant)
Cell side material	Stainless steel
Face guards	4-mesh galvanized hardware cloth
Seal	High viscosity fluid seal.
Efficiency, %	99.97 with 0.3 micron
(Shop tested)	diameter dioctyl phthalate (DOP)
Rating basis	MIL-STD-282

Rated pressure drop unloaded, in. w.g.	1.0
Codes:	ASME AG-1-1997 (with exception of casing size) MIL-STD 282 UL-586

Charcoal:

Quantity per unit	24
Rated flow per charcoal element, cfm	333.3
Type	Activated coconut shell, impregnated.
Particle size	#6 - #18 sieve
Ignition temperature, °C	330 minimum
Charcoal per element, lb,	43
Maximum moisture content, %	3
Gasketing material	ASTMD-1056, Gr. SCE-43
Casing	Type 304 stainless steel.
Penetration, % (Unused Activated Carbon)	0.1% maximum molecular iodine, at 40 fpm, 30°C and 95 percent relative humidity. 3 percent maximum methyl iodide, 30°C, 95% relative humidity
Retentivity, seconds	0.25 minimum
Rated pressure drop, in. w.g.	1.00 maximum per bank
Air face velocity, fpm	42 approximate
References	ANSI N509-1980

The Emergency Ventilation System (EVS) fans are fully redundant and are powered from separate essential buses. The EVS fans are connected on the suction side by cross-tie ductwork which is provided with a parallel arrangement of electric motor-operated dampers. These dampers are normally closed and are opened automatically when the charcoal bed temperature reaches a preset-level following a fan failure. The design of the EVS is such that it renders a loss of cooling air to the filters due to fan failure incredible.

The assumptions and results of the analysis to find out the minimum air flow required to prevent desorption of radionuclides are summarized as follows:

a. Assumptions:

1. All radioiodine and methyl iodide are assumed to be adsorbed in one EVS filter unit.

2. The Containment Vessel leak rate is assumed to be 0.5 percent per day of the contained air weight.

3. The ambient air temperature is assumed to be 120°F.

4. Desorption of the radionuclides is assumed to begin at 302°F

5. Heat transfer from the charcoal adsorbers to the surroundings is neglected.

b. Results:

1. The peak heating rate of the charcoal filters is calculated to be 1400 Btu/hr.

2. The minimum required air flow to maintain the charcoal filters below the desorption temperature is approximately 20cfm. In addition to the conservative assumptions used in the computation, the air flow used for cooling is 150cfm.

Temperature switches are provided in the air space between the random charcoal elements to indicate excessive bed heating, Temperature switches are set to alarm in the control room at less than 200°F to provide sufficient time for remedial action.

The Purge System is designed to provide clean fresh air to the Containment Vessel or to the Shield Building and penetration rooms.

Normally, the Purge System is not in operation in the containment purge mode and the associated purge system isolation valves are closed. In Modes 5 and 6, when access to the containment is desired, the containment isolation valves are opened, the isolation valves on the supply and discharge lines to the Shield Building remain closed, and the purge fans are started. When purging of the Shield Building and penetration rooms is desired, the isolation valves are opened in the supply and discharge lines to the penetration rooms, the Containment Vessel isolation valves remain closed, and the purge fans are started.

Supply air is taken through an outside air intake, roughing filter, heating coil and purge supply fan and discharged into the containment or penetration rooms to provide adequate distribution. The purge air is exhausted by the purge exhaust fan through a roughing filter, a high efficiency particulate filter (HEPA) and a charcoal adsorber.

The Purge System is connected to the Emergency Ventilation System (EVS) by means of ductwork bypasses and dampers. In the event of a fuel handling accident which results in the release of radioactivity, during fuel handling operations, the EVS filters can be used for containment cleanup.

A detection system monitors the containment purge exhaust for particulate activity and isotopes I-131 and Xe-133. If containment purge exhaust radiation levels reach predetermined values, the following is automatically initiated: An alarm received in the control room; the shutdown of the purge system supply and exhaust fans; the closure of the outside air intake damper, the damper on the upstream side of the containment purge air exhaust filter and the fan discharge dampers; the damper in the bypass duct to the EVS is opened. If desired, the EVS fans are started and the dampers on the upstream side of the EVS filters if not in the open position are opened by operator action in the control room. The EVS filters air from the containment and exhausts through the station vent. Makeup air is induced into the containment by infiltration through the relief damper on the Purge System supply duct.

During the movement of irradiated fuel assemblies in Containment, containment purge and exhaust penetrations will be isolated by operator action after the purge system has

automatically been shut down by the purge exhaust noble gas monitor. As noted, operators may then choose to utilize the EVS cross-connection to assist in containment cleanup.

Vent Areas Between the Rooms Served by EVS:

The vent areas between the rooms served by the EVS including the Shield Building annulus, are shown in Figure 6.2-43.

6.2.3.3 Design Evaluation

The reliability of the EVS is ensured by providing two independent full capacity subsystems. Each subsystem is capable of maintaining the design negative pressure within the annulus and penetration rooms.

The equipment is located external to the containment, in the Auxiliary Building. The equipment is designed to operate in maximum ambient conditions of 110°F to 120°F and 90 percent relative humidity.

A duplicate set of charcoal adsorbers are provided in each filter assembly to improve the system performance.

Piping, cable tray and ductwork penetrations through the Emergency Ventilation System boundary are sealed to decrease leakage.

The system requires no additional decay heat removal system. Cooling air is always available to prevent excessive heat production due to iodine decay resulting in ignition of charcoal elements. The Emergency Ventilation System in conjunction with the Purge System can be used for containment cleanup as required during normal operation.

The components of the Emergency Ventilation System are designed as Seismic Class I.

Component evaluation for the Containment Spray System is given in Subsection 6.2.2.3. As discussed previously the initial pH of the spray water is approximately 5. Capabilities are provided, however, to raise the pH of spray water to 7 by baskets of Na_3PO_4 in containment to minimize any stress corrosion cracking of stainless steel. Raising the pH of the spray solution in the Containment Vessel Emergency Sump helps the hydrolysis reaction and improves the retention of iodine. Impurities in the spray water also aid the iodine removal and retention when the water inside the Containment Vessel is recirculated as the spray solution. The effective iodine removal rate is reduced to the reduced partition coefficient of the recirculated spray since this water has already absorbed much iodine.

Very little credit has been taken for the removal of airborne iodine by the borated containment spray although this system is expected to reduce it significantly. A more complete description is given in Subsection 15.4.6.4.

The assumptions on post-accident chemical composition of iodines in the Containment Vessel are consistent with Regulatory Guide 1.4.

The removal of elemental iodine by the Containment Spray System was calculated using the following assumptions:

a. The spray solution is borated water at a pH of approximately 5.

b. The spray flow rate is 1300gpm (the capacity of one spray pump).

c. The mass median diameter of the spray droplet assumed is 780 microns (Table 6.2-20)

d. The iodine removal theory of L. F. Parsly), (ORNL-TM-2412 Part VII).

e. The partition coefficients from L. F. Parsly (ORNL-TM-2412 Part VI).

f. No methyl iodide is removed by the sprays.

The assumption was made that only the minimum engineering safety features were available, i.e., only one containment spray pump is assumed to operate.

These assumptions and the iodine removal rates by the containment spray are given in Subsection 15.4.6.4.

Since the spray ring headers are located at the top of the Containment Vessel, essentially all of the free volume of the Containment Vessel is swept clean by the sprays.

The Containment Purge System is designed to provide fresh air for the containment or penetration rooms at a rate of approximately one air volume change per hour. All components of the Purge System are designed to operate in their respective environments. Components inside the containment are designed to operate in an air environment of 120°F, atmospheric pressure, and 100 percent relative humidity. All components located outside the containment are designed for operation in an environment of 110°F, atmospheric pressure and 100 percent relative humidity.

6.2.3.4 Tests and Inspections

EVS Testing:

Prior to initial station operation, the Emergency Ventilation System underwent a preoperational test to assure that the system performs its intended function. One of the acceptance criteria was that the Emergency Ventilation System maintain the Shield Building annulus under a minimum negative pressure of 0.25 inch w.g. with one fan operating. Differential pressure was measured at various locations within the annulus and penetration rooms. The test results indicated that the minimum negative pressure was maintained.

The acceptability of the EVS was demonstrated by the following tests:

a. Fan and motor vibrations were within acceptable limits.

b. The running current of the electric motors was within the nameplate rating.

c. Visual and audible alarms operated at their set point.

d. Controllers for modulating dampers CV5000A, CV5000B, CV5014A, and CV5014B respond to a change in the Shield Building annulus differential pressure and vary their output signal as required to maintain the set point negative pressure. Dampers respond properly to signals from their associated controller.

e. All computer input points are recorded by the computer.

f. Fan performance was verified to assure that each emergency ventilation fan delivers a minimum flow of 8,000 CFM at a total static pressure of 6.5 inch w.g.

g. Proper operation of all dampers was verified by observation.

h. All SFAS actions were verified by simulated signals.

i. Differential pressures across all filter banks were within acceptable limits.

Periodic testing to verify that the Emergency Ventilation System is operable is described in the Technical Specifications.

Taylor Pitot-Venturi flow element and ALNOR Series 6000 Velometer and static pressure probe type 6080 or equal were used to verify the fan flow rate and static pressure. A Taylor Pitot-Venturi flow element was mounted in the wall a duct on the discharge side of the fan.

A predetermined pressure drop across the EVS filter bank was used to determine if the fan had achieved its design flow within the tolerances. The time required to establish the desired negative pressure was recorded.

A pressure differential measuring instrument accurate to .001 inch water column was used to verify negative pressure in different areas served by the EVS. Dwyer's MICROTECTOR portable electronic gauge and others are available for fast, precise measurements and meet TED requirements. One end of the instrument was opened to the atmosphere just outside the Auxiliary Building, and the other end was opened to the following points (in the negative pressure boundary) to measure differential pressure:

a. Top and bottom in each penetration room.

b. Top and bottom inside each ECCS room.

The ability of the Emergency Ventilation System to maintain negative pressure was confirmed when the measured time to establish 1/4-inch w.g. negative pressure was within the prescribed time.

HEPA Filters Testing:

The prefilters are of the type which exhibit an average efficiency of 70 percent dust holding capacity when tested with the National Institute of Standards and Technology (formerly National Bureau of Standards) test using a mixture of Cottrell precipitate and lint.

The HEPA filters are of the type which meet the requirements of Appendix FC-I of AG-1 – 1997.

The HEPA filters are shop and acceptance tested for the efficiency-penetration test with homogenous particles of dioctyl-phthalate (DOP) in accordance with MIL-STD-282.

The HEPA filters are shop tested to measure the pressure drops across filter at rated flow.

The charcoal adsorbers are shop performance tested with methyl iodide tracers for efficiency and Freon for leakage. The fans are statically and dynamically balanced. Fan ratings are in

accordance with AMCA Standard Test Code 211-A. The emergency ventilation system and the purge system were given a preoperational test prior to startup as follows:

a. Fans were operated continuously for a least one hour, and all related louvers and other mechanical components were proven operable.

b. The HEPA filter banks were in-place tested at design flow for efficiency penetration with homogenous particles of dioctyl phthalate (DOP).

c. The charcoal adsorber banks were tested in place at design flow with Freon 112 (or equivalent) to ensure that there is no leakage across the filter banks and that the charcoal filter elements are not damaged. A Sample specimen of the charcoal used in the filter elements are analyzed annually by an independent laboratory to determine remaining charcoal filter life and replacement requirements.

d. The ductwork was tested for leakage.

e. The systems were balanced, adjusted, and tested for performance.

An in-service surveillance program to assure that the emergency ventilation system and the Containment Vessel purge isolation valves perform their design function is described in the Technical Specifications.

Operability testing of the isolation valves is accomplished each time the Purge System is put into operation.

The system equipment outside containment is fully accessible during all normal operation for maintenance and performance testing, including replacement of filter elements.

Performance test data for the Emergency Ventilation System and the Purge System are supplied. Testing and inspection of the Containment Spray System are discussed in Subsection 6.2.2.4.

6.2.3.5 Instrumentation Application

Two detection systems monitor the containment vessel atmosphere for particulate activity, I-131, and gross gaseous radioactivity. Each system continuously draws a sample from either one of two sample lines originating from different points within the containment vessel. Each system contains a normal and an accident range monitor. Each normal range monitor consists of two filter-detectors and a beta sensitive noble gas channel. Each accident range monitor consists of three paper-charcoal type filters, and a noble gas monitor with two gamma sensitive detectors.

Instrumentation is provided to measure the containment and the reactor coolant pressure. The safety features actuation system continuously monitors this instrumentation and take appropriate action to initiate the containment air purification and cleanup systems upon detecting containment pressure or reactor coolant pressure indicative of a loss-of-coolant type accident. The status of these systems is continuously displayed in the main control room. Failure of any of these components to go to its proper safety features position when required would result in an incorrect status indication.

Design details and logic of the instrumentation are discussed in Chapter 7.

Instrumentation application for the Containment Spray System is discussed in Subsection 6.2.2.5.

6.2.4 Containment Vessel Isolation Systems

6.2.4.1 Design Bases

The general design bases governing isolation valve requirements for containment piping penetrations are as indicated in the following paragraphs.

Leakage through all penetrations not serving accident-consequence-limiting systems is minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation. The installed double barriers take the form of closed piping systems, both inside and outside the containment, and various types of isolation valves or flanges.

Containment Vessel isolation valves are provided in lines penetrating the Containment Vessel to ensure that no uncontrolled release of radioactivity from the containment can occur, particularly following a radiation release type accident.

Containment Vessel isolation occurs on a safety features actuation signal. Development of the instrumentation circuits and signals is presented in the Technical Specifications.

The isolation system closes all penetrations not required for operation of the engineered safety features system, RCS makeup, or special exceptions as noted in subsequent sections. In addition, all pneumatically operated isolation valves, with the exception of those that are part of the engineered safety features, will fail closed. All motor-operated isolation valves, upon loss of normal and reserve electric power, are supplied with power from the emergency power system. Motor-operated isolation valves also have a manual override to be used in case of motor operator failure.

Isolation valves located outside the Containment Vessel are located as close to the Containment Vessel as practical. Upon loss of actuating power, the isolation valves are designed to maintain their present position or to take the position that provides the greater safety.

All control room operated containment isolation valves are provided with position indicating lights in the control room and either control switches or control and safety features actuation block switches in the control room.

To ensure the added reliability of containment integrity, the following penetration systems are designed in accordance with the ASME code, Section III, Class 2, designed and analyzed as Seismic Class I, protected against missiles and all high energy piping, suitably restrained so that passive failure of one component does not damage adjacent components, and subjected to a strict quality assurance program to ensure that material and workmanship meet specifications:

a. All piping between the inside and outside isolation valves up to and including the valves.

b. In a closed system having only one isolation valve outside the containment, the entire system inside the containment to and including the isolation valve.

The design of the containment isolation system conforms to NRC General Design Criteria Nos. 54, 55, 56 and 57 and AEC Safety Guide No. 11 with the exceptions indicated in Subsection 6.2.4.2.

6.2.4.2 System Design

Piping penetrations which require isolation after an accident are classified as follows:

Type I.:

Each line that is part of the reactor coolant pressure boundary and that penetrates the Containment Vessel is provided with containment isolation valves as follows:

a. One locked closed isolation valve inside and one locked closed isolation valve outside the containment; or

b. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

c. One locked closed isolation valve inside and automatic isolation valve outside the containment (check valves are not used outside containment as isolation valves); or

d. One automatic isolation valve inside and one automatic isolation valve outside containment (check valves are not used outside containment as isolation valves).

All welds in this type of penetration are subject to periodic inservice inspection in accordance with the requirements of the ASME Code, Section XI.

Type II.:

Each line that connect directly to the Containment Vessel atmosphere is provided with isolation valves as follows:

a. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

b. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

c. One locked closed isolation valve inside and one automatic isolation valve outside containment (check valves are not used outside containment as isolation valves): or

d. One automatic isolation valve inside and one automatic isolation valve outside containment (check valves are not used outside containment as isolation valves).

e. One blind flange inside the containment and one blind flange outside.

Those lines which do not normally connect directly to the containment atmosphere, but may fail following a seismic event are considered to be Type II. This consideration is applied to Penetrations 3, 4, 12, 16, 21, 41, 42-A, 43-A, 44-B, 48, and 68-A.

The following penetrations are exceptions to NRC Criterion 56 as described above;

1. Containment Vessel vacuum breakers.

2. Containment Vessel leak test inlet line.

3. Fuel transfer tubes.

4. Containment Vessel differential pressure sensors.

5. Containment Vessel hydrogen purge outlet lines.

6. Chemical cleaning line.

The above exceptions do not present a hazard to the public or safe operation for the following reasons:

1. Each containment vacuum breaker has one motor-operated isolation valve and one check valve attached outside the Containment Vessel between the vessel and the Shield Building. These two valves provide a double barrier complying essentially to NRC Criterion 56. The outside installation of the vacuum breaker facilitates periodic inspection, leak testing, and setting of the vacuum breakers while the station is in operation.

2. The containment leak test inlet line is locked closed during station operation and is only open at station shutdown when containment leak testing is performed or when the piping is being used to provide service air to containment for outage activities during Modes 5, 6 or when the core is off loaded. There is one locked closed isolation valve outside the containment and the line inside containment is fitted with a blind flange. This provides a double barrier. When the piping is used to provide service air during outage activities, the attached piping shall have isolation valve(s) and spring loaded check valves(s) inside containment to provide containment isolation for all branches, and outside containment isolation valve(s) to isolate all branches, on any attached manifold(s). Attachments for this purpose will be reviewed by Engineering for other considerations before installation.

3. Each fuel transfer tube has one blind flange with a double o-ring seal installed on the inside of the Containment Vessel. This provides a double barrier. The outboard valve is not considered part of the containment boundary.

4. Each Containment Vessel differential pressure sensor has one normally open remote manually operated valve outside of the containment. Beyond this valve, 3/8 inch dia. tubing is run to the pressure transmitter which provides a barrier to the containment. All components of this system are designed in accordance with the requirements of the ASME Code, Section III, Class 2, designed as Seismic Class I, protected against missiles, and are under a strict quality assurance program to ensure that material and workmanship meet specifications. These sensor systems satisfy the requirements of AEC Safety Guide No. 11. Because the design of

tubing and instrumentation downstream of the valve exceeds the design requirements of the Safety Guide, valve status or line condition information for purposes of valve isolation during design basis events is not provided.

5. The Containment Vessel hydrogen purge outlet line has double isolation valves provided outside containment for redundant isolation of the flow path. The maximum operating conditions (LOCA) and seismic loading cause stresses much below the allowable stresses of the penetration system. In addition, operation of this system is normally required only after the pressure-temperature conditions of a LOCA have been substantially reduced.

   These valves have been located outside to make the system more reliable. These are not required to be open until six to eight weeks (if at all required then) after LOCA. Although the valves are designed to be operable under LOCA conditions, one hundred percent assurance cannot be given that a valve, if installed in the Containment Vessel, would open when required after such a prolonged closure under post-LOCA environment. By bringing the valve outside containment it can be manually opened if it fails to open automatically.

   The Hydrogen Purge outlet line can also be used to vent containment. During plant heatup, power ascension and power operation valves CV5037 and CV5038 can be opened to reduce the containment internal pressure. The effluent flowpath will be through 4 inch containment penetration piping and motor-operated valves to the 2 ½ inch piping which is used for the hydrogen recombiner. The normal hydrogen purge HEPA and charcoal filter assembly is protected during containment pressure release by closing the normally open manual valve, CV60, upstream of the filter assembly prior to opening the hydrogen purge containment isolation valves. This ensures the filter will be undamaged and available for post-LOCA use.

   The flow is then directed through a HEPA and charcoal filter assembly to the fuel handling area atmosphere or the auxiliary building radwaste area ventilation system. A SFAS incident level 2 signal will close the containment isolation valves on a high containment pressure or a reactor coolant system low pressure condition. This feature will not be blocked or overridden for use in containment pressure control. An evaluation of this system lineup was conducted to ensure compliance with Branch Technical Position (BTP) CSB 6-4, Containment Purging During Normal Plant Operations. The evaluation concluded that Containment Hydrogen Purge system is in compliance with the intent of the BTP CSB 6-4.

6. The chemical cleaning line is required for steam generator secondary side cleaning. One blind flange is installed inside and one outside the Containment Vessel to provide a double barrier. This penetration will be open only during station shutdown.

Type III.:

Each line that penetrates the reactor Containment Vessel and is neither part of the reactor coolant pressure boundary nor connected directly to the Containment Vessel atmosphere has at least one containment isolation valve, which is either automatic, locked closed or capable of remote manual operation. Check valves are not used as automatic isolation valves outside the containment.

Steam traps are provided inside the containment isolation boundary on some Type III penetrations where they are needed for proper operation of the associated system. The steam trap isolation and bypass valves are considered to be small, special case valves which do not require automatic or remote isolation capability. These valves are administratively controlled in order to ensure that they are appropriately isolated when necessary to prevent release of radioactivity to the environment.

Type IV.:

Each line that serves the engineered safety features systems and penetrates the Containment Vessel is provided with isolation valves as follows:

a. One automatic isolation valve inside and one automatic isolation valve outside containment (check valves are not used as isolation valves outside containment); or

b. One automatic isolation valve outside containment (check valves are not used as isolation valves outside containment).

These isolation valves are automatically operated by the safety features actuation signal or remotely from the control room.

Depending on function, all components of the systems outside the containment and beyond the outside containment isolation valve, up to and including the normally closed system block valves, are designed in accordance with the requirements of the ASME Code, Section III, Class 2 or Class 3, designed and analyzed as Seismic Class I and protected against missiles. All high energy piping is suitably restrained, so that passive failure of one component does not damage adjacent components. A strict quality assurance program is applied to ensure that material and workmanship meet specifications.

The following penetrations are exceptions to this category:

a. The Containment Vessel Emergency Sump recirculation lines are opened during emergencies when the BWST is emptied into containment. Although they are open to the Containment Vessel atmosphere, outside of the containment they form a closed loop system terminating inside the Containment Vessel. All components of the closed loop system are in accordance with the ASME Code, as per Table 3.2-2, designed and analyzed as Seismic Class I, protected from damage by missiles, and under a strict quality assurance program to ensure that material and workmanship meet specifications.

b. The decay heat pump suction line is normally closed, but is used post-LOCA for a boron dilution flow path, thereby providing an engineered safety feature function. The containment penetration isolation valves inside containment are remotely operated valve DH-11, locked closed valve DH-23, and relief valve PSV-4849. This line forms a closed loop outside the containment and terminates inside the Containment Vessel. All components of the closed loop system are Class 2, designed and analyzed as seismic Class I, protected from missiles and under a strict quality assurance program. The design temperature and pressure rating exceed that of the containment. The relief valve set point is greater than 5 times the containment design pressure. At all times, after a LOCA, there is a water seal

from either the BWST or (upon recirculation) the emergency sump to ensure that there is no path for leakage from the containment atmosphere backwards through the relief valve.

c. The containment pressure sensors penetration design is as indicated for Item 4 under Type II penetrations.

d. The Containment Vessel Spray lines are opened by the SFAS following a LOCA. Although they open to the Containment Vessel atmosphere, external to the Containment Vessel, this system forms a closed loop terminating at the lines from the Containment Vessel Emergency Sump which in turn ends within the Containment Vessel. All components of the closed loop system are in accordance with the ASME Code, as shown in Table 3.2-2, designed and analyzed as Seismic Class I, protected from damage by missiles, and manufactured and installed under a strict quality assurance program. By the function of this system, the penetrations are classed as Type IV.

e. The RC Make up Valves MU 6421 and MU 6422 are remote manual isolation valves. MU 6422 is normally open to provide a make up flow path during power operation. Valves MU 6421 and MU 6422 will also be used to provide a flow path for core cooling following a loss of all secondary side cooling (Feed and Bleed mode of operation). The nature of the Makeup and Purification System during feed and bleed cooling is similar to that of an engineered safety feature (ESF) system since it provides the only means of maintaining adequate core water inventory to preclude offsite radiation in excess of 10CFR100 limits. If these valves (MU 6421 and MU 6422) were to close on the automatic SFAS signal, it would interrupt the feed and bleed process and could cause damage to the make up pumps due to dead heading. In order to support a reliable feed and bleed process the automatic SFAS isolation is not provided. Remote manual isolation valves are permitted in this situation provided possible leakage outside containment can be detected. The make up system is in continuous operation during normal plant operation. As such, any leakage would be observed and appropriate corrective measures implemented to ensure a leak-tight system is maintained. Thus, the Make up and Purification system meets the above criterion.

6.2.4.2.1 Additional Design Information

Additionally, there are various arrangements in each of these major groups. The individual system functional drawings show the manner in which each Containment Vessel isolation valve arrangement fits into its respective system. For convenience, each different valve arrangement is shown in Table 6.2-23.

Table 6.2-23 tabulates specific information for isolation valves at each penetration. Listed are the modes of actuation, the types of valves, their normal and emergency positions, and closing times. The specific system penetrations to which each of these arrangements is applied are also presented.

Containment isolation valves that are normally open and are required to be closed following an accident are designed with closure times of 60 seconds or less as shown in Table 6.2-23. The normally closed valves will receive a closure signal to close them if they are open, otherwise the signal serves to ensure valve closure.

Vents, drains, and/or test connections are installed between containment isolation valves or between the containment vessel and the containment isolation valve where they are needed in order to perform a necessary activity. Because these lines are normally maintained closed and capped, locking these valves in a closed position is not required. Leakage through these lines is precluded by means of an installed cap which is governed by administrative controls.

The following containment isolation system valves actuate to isolate containment upon receipt of an SFAS signal:

> Isolation valves of Containment Vessel isolation system number 1 (SFAS incident level 2) are tabulated in Table 6.2-25. Isolation valves of Containment Vessel isolation system number 2 (SFAS incident level 3) are tabulated in Table 6.2-26. Isolation valves of Containment Vessel isolation system number 3 (SFAS incident level 4) are tabulated in Table 6.2-27. Containment vessel isolation valves CV5010A through CV5010E and CV5011A through CV5011E (containment air sample valves) also receive an SFAS signal for containment isolation. Technical Specification Amendment 221 specifies containment vessel isolation valves CV5005, 5006, 5007 and 5008 (containment purge supply and exhaust valves) be closed with control power removed in modes 1 through 4 for containment isolation. Closure of the purge valves from the control room is required when necessitated by Technical Specifications.

The containment isolation system and all of its components, including piping, valves, supports, etc., are designed in such a manner that dynamic forces resulting from inadvertent sudden opening or closure of a valve under operating conditions would not result in loss of containment integrity. In addition, automatic controls are provided on the double isolation valves on the normal Decay Heat Removal System to prevent an inadvertent opening of the valves. A detailed description of this interlock system is in Subsection 7.6.1.1.

If a main steam isolation valve or turbine intercept valve closes suddenly during normal station operation, increased steam system pressure will cause the code safety valves to open.

All isolation valves and drives, including position indicators, motors, cables, sensing elements, etc., that are located in the containment are designed, fabricated, and tested in such a manner that they are capable of correct functioning under the environmental conditions of a loss of coolant accident or steam line break accident. These design conditions are tabulated below:

Environmental Conditions Inside Containment Vessel

Condition	Temperature, °F	Pressure, psig*	Normal Humidity Range (%)
Normal Operating	120	-14* to 25* w.g.	10-80**
LOCA	264	40	100
Testing	68	45	10-80**
Station Shutdown	50	0	10-80**

Environmental Conditions in Shield Building Outside Containment Vessel

Condition	Temperature, °F	Pressure, psig	Relative Humidity, (%)
Penetration Rooms	120	0	70
Main Steam (Normal)	120	0	70
Main Steam (Line Rupture)	222	3	100

The main steam and main feedwater pipe penetrations have guard pipes installed around the penetrating process pipes to protect the Containment Vessel against jet effects in case of pipe failure.

The Containment Vessel air sample inlet and outlet lines are provided with automatic isolation valves inside and outside of the Containment Vessel, which comply with General Design Criterion 56. The Hydrogen Dilution System inlet lines are provided with check valves inside the Containment Vessel which comply with the criterion. Refer to Subsection 6.2.4.2 for the rationale used to exempt the hydrogen purge outlet line from the criterion.

Core Flooding Tank Sample and Vent Lines:

The Core Flooding Tank sample line, penetration 47A, and the Core Flooding Tank vent line, penetration 47B, have been provided with safety actuated isolation valves outside the Containment Vessel. Inside the Containment Vessel are two remote motor-operated isolation valves on each line. During normal station operation, these valves would be open only when taking a sample or venting the tanks. When these valves are not open they would be closed and under administrative control. Should these valves be open, the check valves on the Core Flooding Tank outlets into the Low Pressure Injection lines would act as the Containment

*From Shield Building
**The design value is 100%

Vessel interior isolation valves. Therefore these lines do not fall into General Design Criterion 57.

Containment Vessel leakage could only occur as a result of all of the following:

a. Failure of operation of one of the low pressure injection trains.

b. The outside isolation valve (safety actuated) fails to close.

c. Two check valves leak.

d. One of the remote manual interior isolation valves is open, or the bypass check valve around one of the inside containment isolation valve leaks (CF2B).

6.2.4.3 Design Evaluation

The containment isolation system and all of its components are designed, fabricated, and installed to ensure their correct functioning at all times. To achieve this goal, the following have been incorporated into the system design:

a. All pressure retaining components of the system are designed in accordance with the ASME Code, Section III, Class 2 requirements;

   Check valves CC1568, CC1407C, CC1411C, CF2C and RC229C, were added as inside containment isolation valves for penetration 12, 4, 3, 47A and 48 under modification 97-0009. Relief valve MU60D was added as an inside containment isolation valve for penetration 56 under modification EC 19-0011-004. These valves were evaluated as meeting ASME Section III Class 2 requirements under the provisions of Generic Letter 89-09.

b. All components of the system are designed as Seismic Class I;

c. Reliability of material and workmanship of all components of the system is ensured by the application of a strict quality assurance program;

d. All components of the system are protected against missiles. All high energy piping (275 psig and higher and one inch diameter and larger) is restrained in such a manner that under conditions of a pipe rupture, no pipe or piping component can become a missile and damage adjacent systems.

e. Solenoid-actuated air cylinder exhaust valves used for the main steam line isolation valves are of the totally enclosed type. They have a direct acting solenoid, and therefore eliminate any failure due to external linkage malfunction which have occurred on other PWRs.

6.2.4.3.1 Leakage Paths That Could Bypass EVS

Following a LOCA and a concurrent seismic event, systems which are designed as Seismic Class 1 can be assumed to remain intact. Systems which penetrate the Containment Vessel and are Seismic Class land are not open inside the Containment Vessel, will not become leakage paths for radioactive liquid or gas from the Containment Vessel. Table 6.2-28 provides a listing of containment penetrations and their termination points.

Penetrations for systems which are normally open inside the Containment Vessel (or are postulated to be open following a LOCA and concurrent seismic event) and do not terminate in the area served by the EVS are the only possible in-line leakage paths through which radioactive liquid or gas could leak out of the Containment Vessel, escape the Emergency Ventilation System, and become a ground release. This includes the following:

Penetration Number	Service
1	Pressurizer sample line
4	Component cooling water outlet line
13	Containment Vessel normal sump drain line
14	Letdown line to purification demineralizers
16	Containment Vessel equipment vent header
17	Containment Vessel Leak Test Line
21	Demineralized water supply line
23, 24	Fuel Transfer Tubes
32	Reactor Coolant system drain line to R.C. drain tank
41	Pressurizer Quench Tank Circulating inlet line
42A	Service air supply line
42B	Containment Vessel air sample return line
43A	Instrument air supply line
44B	Pressurizer quench tank N_2 supply line
48	Pressurize Quench tank circulating outlet line
49	Refueling canal fill line
52, 53, 54, 55	Reactor coolant pump seal water supply
56	Reactor coolant pump seal water return
67	Hydrogen dilution supply line
68A	Pressurizer quench tank sample line
69	Hydrogen dilution supply line
71B	Containment air sample line
73B	Containment air sample line
74C	Pressurizer auxiliary spray line
80	Emergency lock
81	Personnel lock
82	Equipment hatch
101, 102	Electrical penetrations

The total leakage for the above listed penetrations will be less than 3.0 percent of the design containment leakage.

Estimation of the leakages through these lines would depend on several factors. These include the following:

a. Line size

b. Isolation valve type (ball, check, gate, etc.)

c. Isolation valve manufacturer

d. Usage during station operation

e. Process fluid contained during normal station operation.

Since there are many complex factors involved, it would be an academic exercise to use all these factors in estimating the leakage. This leakage would then, at best, be an estimate. The estimate used was based on the line size only.

The leakage from each line can be expressed as follows:

$$\text{Leakage from the line} = \frac{\text{line size (inches) X 0.0075\% / day}}{\text{summation of the line sizes}}$$

In the calculation of the accident doses all leakage from the containment was assumed to be released directly to the atmosphere (unfiltered) until negative pressure was established in the annulus.

6.2.4.4 Tests and Inspections

The containment isolation system is designed so that each of its components requiring testing can be tested periodically. Pressure retaining components of the containment isolation system, including piping and valves, undergo periodic leak testing. Leakage rate testing of the Containment Vessel (Type A) and of components requiring a local leakage rate test (Type B or C) is done in accordance with 10 CFR 50 Appendix J, Option B, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.

Containment isolation valves are tested in accordance with the Inservice Testing Program. Each automatic isolation valve actuated by SFAS has manual override capability. These valves and SFAS are periodically tested in accordance with the Technical Specifications.

All welds of pressure retaining parts of the containment isolation system undergo examination after installation to meet the requirements of ASME Code, Section III, Class 2 components.

All piping components of the containment isolation system that are directly connected to the primary reactor coolant system undergo periodic in-service inspection of welds in accordance with ASME Code, Section XI.

All mechanical and electrical components of the containment isolation system including valves, valve actuators, cables, motors, sensing elements, position indicators, etc., are periodically tested for operational functions.

6.2.4.5 Containment Isolation Valves Qualification Testing

Qualification testing is covered under Subsection 3.11.2.

6.2.5 Combustible Gas Control in Containment Vessel

6.2.5.1 Design Bases

The Combustible Gas Control System is designed to control the concentration of hydrogen which may be released within the Containment Vessel atmosphere following a LOCA. The system is composed of the Containment Hydrogen Dilution (containment atmosphere dilution) System and the Hydrogen Purge System. The Containment Hydrogen Dilution System is designed to add air to the Containment Vessel to effectively maintain hydrogen concentrations within acceptable limits.

The Hydrogen Purge System is designed to release air from the Containment Vessel atmosphere through HEPA and charcoal filters to the station vent. Post-accident hydrogen mixing is adequately accomplished by natural convective currents along with the turbulence created by the combined action of the containment spray and the containment air cooler fans. The Containment Recirculation System is another available means of providing post-accident hydrogen mixing.

Following a loss of coolant accident, hydrogen gas may accumulate within the Containment Vessel from various sources. If a sufficient amount of hydrogen is generated, it may react with oxygen present in the Containment Vessel atmosphere at rates rapid enough to lead to high temperature and significant overpressurization of the Containment Vessel. As stated in AEC Safety Guide No. 7, the lower flammability limit for hydrogen in air saturated with water vapor at room temperature and atmosphere pressure is assumed to be four volume percent.

The Combustible Gas Control System components are designed to be operated as necessary to maintain the maximum hydrogen concentration in the Containment Vessel at or below three volume percent following a LOCA. The limit of three volume percent was determined arbitrarily to reflect a reasonable margin to alleviate problems such as nonhomogeneous mixing, etc. Using the conservative assumptions of AEC Safety Guide No. 7, a concentration of three volume percent can be reached at approximately 17 days after the LOCA.

Note that 10 CFR 50.44 (Final Rule 68FR54123, effective October 16, 2003) no longer requires combustible gas control systems.

6.2.5.2 System Design

The systems are shown on Figures 9.4-11A and 9.4-12.

6.2.5.2.1 Containment Hydrogen Dilution System (CHD)

The Containment Hydrogen Dilution System consists of two full capacity, redundant, rotary, positive displacement type blowers to supply air to the containment. The CHD system controls the hydrogen concentration by the addition of air to the Containment Vessel, resulting in a pressurization of the containment and suppression of the hydrogen volume fraction.

Remotely-operated valves are located on the discharge side of each CHD system blower and on the hydrogen purge system inlet to provide containment isolation. The valves are located outside the Containment Vessel and are controlled from the control room. All components of the system are designed as Seismic Class I.

The curves of hydrogen concentration in the containment as a function of time are provided in Figures 6.2-45 and 6.2-48. The Containment Hydrogen Dilution System blowers are each capable of developing 25 psig, (25 psig is the setpoint of each blower's relief valve) but, the containment internal pressure is administratively controlled to a maximum of 18 psig with the blower(s) in operation. Containment design pressure is 36 psig.

With the maximum permissible containment inventory of aluminum and zinc, the hydrogen control limit of 3 percent can be reached in approximately 17 days. At that time, the Containment Vessel pressure is approximately 0.5 psig. CHD system operation will then be initiated, and the Containment Vessel pressure as a function of time is shown in Figure 6.2-46.

6.2.5.2.2 Hydrogen Purge System

The Hydrogen Purge System is available to release air from the Containment Vessel atmosphere through HEPA and charcoal filters to the station vent. Operation of the Containment Hydrogen Purge System functions in conjunction with the redundant Containment Hydrogen Dilution trains to relieve containment gases after the containment pressure limit is reached if operation of the Containment Hydrogen Dilution blowers is still required. The System, including the purge line and purge system filter unit (F60) is designed as Seismic Class I. Hence the design of these systems is considered to be in compliance with the criteria for an engineered safety feature.

The Hydrogen Purge System would be used only when the concentration of hydrogen reached the 3 percent control limit (at 18 psig, the upper limit of pressurization of the CHD system blowers). This limit will be reached in excess of 60 days, assuming hydrogen generation still occurs over this time period. When the Hydrogen Purge System is lined up to the station vent, only one Hydrogen Dilution Blower is left running to maintain pressure in the containment vessel. The capacity of the Hydrogen Purge System is equivalent to the capacity of one Hydrogen Dilution Blower. Therefore, when operating the Hydrogen Purge System and one Hydrogen Dilution Blower simultaneously the containment should not increase in pressure beyond the 18 psig containment pressure limit.

The Purge system has been designed for 100 scfm.

The Hydrogen Purge outlet lines can also be used to vent containment as described in Section 6.2.4.2.

6.2.5.2.3 Hydrogen Recombination System

The Hydrogen Recombination System functions as a means of reducing any hydrogen concentration in the Containment Building. The air containing hydrogen is pumped from containment through the recombiner. The air is heated electrically within the recombiner until recombination occurs between the hydrogen and oxygen to form water vapor. The hydrogen free effluent is then returned to containment. The installed system includes piping, manual remote operated valves, and electrical hookups for the self-contained Hydrogen Recombiner that would be brought on site if the need should arise. Note that the capability to install an external hydrogen recombination system is no longer a requirement of 10 CFR 50.44 (final rule 68FR54123, effective October 16, 2003).

6.2.5.3 Design Evaluation

The significant sources of hydrogen following the design basis loss-of-coolant accident are:

1. A metal-water reaction involving the zirconium fuel cladding and the reactor coolant.

2. A radiolytic decomposition of the post-LOCA emergency cooling solutions.

3. The corrosion of metals and paints by solutions used for emergency cooling or containment spray.

Each of these potential sources has been considered in detail, and the hydrogen generated from each has been calculated. The calculations in all respects conformed to the conservative assumptions outlined in AEC Safety Guide No. 7.

The following assumptions were made for the calculation of hydrogen generation:

a. All AEC Safety Guide No. 7 assumptions were used in the analysis.

b. The fission product source terms are derived from the energy release rates from halogens which were calculated on an individual isotope basis. The core inventory of each isotope was calculated by the method outlined in TID-14844.

c. The zirconium - water reaction occurs essentially instantaneously.

d. An insignificant quantity of H_2 is present due to noble gases in post-LOCA containment atmosphere.

e. An insignificant quantity of H_2 is dissolved in the coolant or trapped in the pressurizer stem space.

f. All evolved gases are mixed uniformly throughout the containment atmosphere.

g. No recombination of H_2 and O_2 occurs.

h. The LOCA considered is a double-ended break of a hot leg reactor coolant pipe.

i. An insignificant quantity of H_2 is present in the containment atmosphere before LOCA.

j. The average fuel exposure was assumed to be 600 FPD at 2,772MWt.

k. Transuranium isotopes were ignored in this calculation.

l. Pre - LOCA conditions assumed:

    1. T=120°F

    2. P=14.4 psia

    3. Relative Humidity=30 percent

m. Since galvanized ductwork may collapse upon rapid pressurization of the containment, 20% of the interior of the ductwork is considered to be exposed to the spray solution.

n. Thickness of the galvanized coating on the grating is 3.4 mils.

The post-accident fission product distribution outline in AEC Safety Guide No. 7 was used to calculate the hydrogen production due to radiolysis. All noble gases were considered to be released to the containment atmosphere; 50 percent of the halogens and 1 percent of the solids were considered to be released to the coolant, and the remaining fission products were considered to remain in the core. The decay energy release rate for solids was derived from the decay energy curves published by Shure. Source inventories for all halogen and noble gas

isotopes were calculated by the method defined in TID-14844. After release, each isotope was decayed according to its decay constant.

Radiolysis of water was considered for the coolant adjacent to the core and in the sump. Gamma energy from in-core fission products, attenuated by a factor of 0.1, was used to calculate radiolysis of the coolant adjacent to the core. In addition, gamma and beta energy from the released halogens and solids was used to calculate radiolysis of the coolant in the core region and in the sump.

The hydrogen generation rate due to zirconium-water reaction was based on 2 lb. - moles of free H_2 being produced for each lb.-mole of zirconium which reacts. There are 44,815.9 pounds of zircaloy clad in the core. Of this quantity, five percent was assumed to react after the postulated accident. The reaction of up to 5% produced 49.13 lb.-moles of free hydrogen. This is conservative because one of the criterion of 10CFR 50.46 is that the hydrogen generation rate that is calculated for the LOCA analyses shall not exceed 1% of the total amount possible if all the metal in the cladding cylinders surrounding the active fuel were to react. The LOCA analyses documented in Reference 35 confirms that the H_2 generation rate is less than 1%.

Hydrogen produced in this manner was assumed to be released to the containment atmosphere in the first few minutes after the accident occurred.

Free hydrogen may be liberated due to the corrosive reaction between the chemicals in the containment spray solutions and corrodible metals and paints in conjunction with elevated containment temperatures. The metals whose corrosion contributes significantly to hydrogen production are aluminum and zinc. Zinc base paints also contribute to hydrogen production. Table 6.2-29 lists the quantity of each of the materials which may be exposed to the spray solutions. Table 6.2-30 lists the hydrogen generation rates and corrosion rates for galvanized steel and zinc based paints in relation to time and temperature.

Paints used inside the Containment Vessel are listed in Table 6.2-31.

The hydrogen production rates and corrosion rates listed in Table 6.2-30 are based on experimental data taken at the Franklin Institute under simulated post-LOCA conditions.

The volume fraction of hydrogen as a function of time was calculated from the known quantities of noncondensables and the water vapor present in the containment atmosphere. Volume fractions were considered to be equal to the corresponding mole fractions. The total quantity of hydrogen produced from each source is plotted in Figure 6.2-48 for sixty days after the postulated accident. The total hydrogen production rate is shown in Figure 6.2-49. Finally, the hydrogen volume fraction is plotted in Figure 6.2-50, assuming that no control measures are instituted.

The lower flammability limit of hydrogen is taken to be four volume percent. In order to ensure a conservative safety margin, the combustible gas control systems would be started no later than the time the hydrogen concentration reaches three percent. As is shown in Figure 6.2-50, this limit is reached approximately 17 days after the accident.

As shown in Figure 6.2-49, the hydrogen production rate at the time when the hydrogen concentration reaches three volume percent is approximately 0.31 lb-moles/hr. The minimum initial air dilution rate required for suppression of the hydrogen volume fraction in the containment is approximately 71 cfm which is less than the 100 cfm capacity of each of the CHD system blowers. Since the hydrogen production rate continuously decreases, the required dilution flow rate would reduce slightly with time.

In order to evaluate the impact of purging for H_2 control in the event that it is required, site boundary doses have been calculated based on the following assumptions:

a. In accordance with TID-14844, the core inventories released to the containment are:

   1. 100 percent of the noble gases

   2. 50 percent of the halogens

b. Per Regulatory Guide 1.4, the chemical forms of the released halogens are:

   1. 91 percent - elemental

   2. 5 percent - particulate

   3. 4 percent - organic

c. Core saturation inventories are calculated by the TID-14844 model.

d. In accordance with TID-14844 one-half of the released halogens plate out leaving 25 percent of the core's iodine inventory available for leakage from the containment building.

e. For the purposes of this calculation, credit is taken for spray removal of halogens as derived in Subsection 15.4.6.4.

f. The initial leak rate is assumed to be 0.5% the first day after LOCA and 50 percent of this rate thereafter.

g. The X/Q assumptions for the low population zone are as follows:

Time Period	Low Population Zone
0-8 hrs	9.9×10^{-6}
8-24 hrs	2.3×10^{-6}
1-4 days	1.5×10^{-6}
4-30 days	5.6×10^{-7}
> 30 days	5.6×10^{-7}

h. For the purpose of comparison, hydrogen purge was assumed to be initiated, for the first time, at three different times: 21 days, 90 days and 140 days. Calculations were performed assuming two separate cases for each initiation time:

   1. Filter removal efficiency of 95 percent for all forms of iodine.

   2. No iodine removal by the filters.

i. The flow rate during purging is 100cfm.

The incremental dose at the LPZ due to the purge are:

a. Purge initiated at 21 days and terminated at 30 days

Thyroid dose (with filtration credit)	1.9	rem
Whole body dose (with filtration credit)	0.011	rem
Thyroid dose (without filtration credit)	39.7	rem
Whole body dose (without filtration credit)	0.021	rem

b. Purge initiated at 90 days and terminated at 97 days

Thyroid dose (with filtration credit)	3.82×10^{-3}	rem
Whole body dose (with filtration credit)	7.33×10^{-5}	rem
Thyroid dose (without filtration credit)	7.71×10^{-2}	rem
Whole body dose (without filtration credit)	9.17×10^{-5}	rem

c. Purge initiated at 140 days and terminated at 147 days

Thyroid dose (with filtration credit)	4.9×10^{-5}	rem
Whole body dose (with filtration credit)	6.17×10^{-5}	rem
Thyroid dose (without filtration credit)	9.25×10^{-4}	rem
Whole body dose (without filtration credit)	6.19×10^{-5}	rem

The integrated activity released from the containment over a period of time Δt from t to t + Δt is given by:

$$IAR_i = \left[\left[a + b\,(1 - \eta)\right] \lambda_{ie} + (1 - \eta_p)\,\lambda_p \right] A_i(t) \; \frac{1}{\lambda_T^i} \left[1 - e^{-\lambda_T^i \Delta t} \right]$$

λ_{ie} = removal constant due to containment leakage

a = direct unfiltered function of λ_{ie}

b = direct filtered function of λ_{ie}

η =filter efficiency

η_p = purge filter efficiency

λ_p = removal constant due to purging

λ_d^i = radioactive decay constant

t = time at which H_2 purge is initiated

Δt = duration of H_2 purge

$\Delta_i(t)$ = activity inventory at time t

$\lambda_T^i = \lambda_{ie} = \lambda_d^i = \lambda_p$

The doses are then given by:

Skin dose: $D_{skin} = \sum_{all\ isotopes} D_{skin}^i = \sum_{all\ i} [IAR_i]\ [X/Q]\ [DCF_i]_{skin}$

Total body: $DTB = \sum_{all\ isotopes} DTB^i = \sum_{all\ i} [IAR_i]\ [X/Q]\ [DCF_i]_{TB}$

Thyroid dose: $DTHY = \sum_{all\ halogens} DTHY^i = \sum_{all\ k} [IARk]\ [X/Q]\ [DCFk]\ THY\ [BR]$

[DCFi] skin' [DCFi]TB'[DCFk] THY = dose conversion factors

(BR) = breathing rate (2.32 x 10-4 cm3/sec)

The incremental doses, due to H_2 purge, are obtained by taking the difference between the doses with and without H_2 purge.

The Hydrogen Purge System charcoal filters are in-place tested with Freon 112 (or equivalent) with less than 1% penetration and bypass leakage in accordance with ANSI N510-1980 at rated flow.

6.2.5.4 Instrumentation Application

The combustible gas control systems are designed for remote-manual operation. The systems are initiated from individual blower and valve control switches located in the control room when the containment H_2 content reaches the predetermined value after a LOCA.

The H_2 content is determined by two redundant gas analyzer systems external to the Containment Vessel. The analyzer systems will result in an alarm on excessive H_2 concentrations. The hydrogen analyzer equipment is not required to operate in a continuous mode. Startup on the system is required 30 minutes after containment spray has been initiated during accident conditions. The analyzer systems are discussed further in Subsection 7.13.3.4 and 9.3.2. Note that 10 CFR 50.44 (Final Rule 68FR54123, effective October 16, 2003) relaxes the requirements for the hydrogen analyzer equipment.

Indication of containment hydrogen concentration is available in the control room. The range of measurement capability is zero to 10% hydrogen concentration under both positive and negative ambient pressure.

TABLE 6.2-1

LOCA Analysis Tabulation Containment Vessel Response Parameters

(Initial Containment Vessel Temperature of 90°F)

Break Location	Break Size, ft^2	Peak Pressure, psig	Peak Temp., °F	Time of Peak[2], sec	Heat Transferred to Structures at Time of Peak Pressure, 10^6 Btu	Energy Released to Containment at Time of Peak Pressure, 10^6 Btu
DE[1] Hot Leg Guillotine Break at SG Inlet	14.14	37.9	256	14	16	303
Hot Leg Split	14.14	37.6	257	16	19	303
DE Hot Leg Guillotine Break at Reactor Vessel Outlet	14.14	37.8	256	14	16	304
Cold Leg Guillotine Break at Pump Discharge	8.55	34.9	251	18	21	295
Cold Leg Guillotine Break at Pump Suction	8.55	35.1	251	18	21	291
DE Hot Leg Guillotine Break at SG Inlet	10.0	37.7	256	16	19	303
DE Hot Leg Guillotine Break at SG Inlet	5.00	36.2	253	22	25	293
DE Cold Leg Guillotine Break at Pump Discharge	5.13	34.7	251	20	22	292
DE Cold Leg Guillotine Break at Pump Suction	5.13	34.8	251	22	24	290
DE Cold Leg Guillotine Break at Pump Discharge	2.00	33.5	248	38	29	286
DE Cold Leg Guillotine Break at Pump Suction	2.00	33.9	249	42	30	284

[1]Double-ended
[2]Time of peak pressure. Time of peak temperature is within 3 seconds of peak pressure for all breaks.

TABLE 6.2-2

Mass-Energy Release for 8.55 ft^2 Double Ended
Cold Leg Guillotine Break at Pump Suction

Time, sec	Mass Release Rate, 1E6 lbm/hr	Enthalpy, Btu/lbm	Energy Rate, 1E6 Btu/sec
1	222	559	34.5
2	199	578	32.0
4	140	629	34.8
6	101	663	18.6
8	81.7	674	15.3
10	61.7	714	12.2
12	47.8	717	9.5
14	37.6	652	6.8
16	15.3	696	3.0
18	13.5	503	1.9
20	7.9	449	0.99
22	2.1	371	0.22
26	0.0	0	0
30	0.4	1050	0.12
40	3.7	613	0.63
50	9.9	691	1.9
60	4.7	544	0.71
70	2.4	690	0.46
80	1.0	864	0.24
90	0.8	945	0.21
100	0.7	977	0.19
120	1.2	660	0.22
160	1.6	608	0.27
200	1.9	512	0.27
240	1.8	520	0.26
280	1.5	504	0.21

TABLE 6.2-3

Mass-Energy Release for 8.55 ft^2 Double Ended
Cold Leg Guillotine Break at Pump Discharge

Time, sec	Mass Release Rate, 1E6 lbm/hr	Enthalpy, Btu/lbm	Energy Rate, 1E6 Btu/sec
1	256	555	39.5
2	198	570	31.4
4	150	601	25.0
6	119	620	20.5
8	81.6	682	15.5
10	42.9	867	10.3
12	38.9	729	7.9
14	30.4	590	5.0
16	26.9	442	3.3
18	24.7	318	2.2
20	19.6	248	1.4
22	13.0	220	0.79
26	0.0	0.0	0.0
30	0.13	1284	0.41
40	3.5	670	0.65
50	3.2	675	0.60
60	2.7	601	0.45
70	2.4	593	0.40
80	2.1	576	0.34
90	2.2	612	0.37
100	2.1	588	0.34
120	1.9	589	0.31
160	1.5	523	0.22
200	1.4	519	0.20
240	1.8	530	0.27
280	1.3	515	0.19

TABLE 6.2-4

Mass-Energy Release for 14.14 ft^2 Double Ended
Hot Leg Guillotine Break at Steam Generator Inlet

Time, sec	Mass Release Rate, 1E6 lbm/hr	Enthalpy, Btu/lbm	Energy Rate, 1E6 Btu/sec
1	286	601	47.7
2	237	610	40.2
4	171	622	29.5
6	116	672	21.7
8	65.7	795	14.5
10	32.8	913	8.32
12	16.9	957	4.49
14	7.56	1020	2.14
16	4.27	990	1.17
18	3.05	926	0.78
20	2.01	856	0.48
22	2.01	856	0.48
26	2.36	687	0.45
30	2.36	687	0.45
40	8.86	319	0.786
50	2.96	401	0.330
60	1.37	473	0.18
70	1.37	473	0.18
80	1.17	462	0.15
90	1.17	462	0.15
100	1.17	462	0.15
120	1.66	434	0.20
160	1.77	447	0.22
200	1.76	430	0.21
240	1.70	424	0.20
280	1.48	414	0.17

TABLE 6.2-5a

Containment Vessel Heat Sinks

Heat Sinks #1 & #2 Containment Cylinder and Dome	
Containment cylinder	75,925 ft^2
Containment dome	26,533 ft^2
	102,458 ft^2
Heat Sink #3 Unlined Concrete	
Concrete	87,849 ft^2
Heat Sink #4 Galvanized Steel	
Grating	34,540 ft^2
Cable trays	12,222 ft^2
Ventilation ductwork	13,851 ft^2
	60,613 ft^2
Heat Sink #5 Painted Steel <0.12"	
Structural steel	2,879 ft^2
Polar crane	1,720 ft^2
Miscellaneous	173 ft^2
	4,772 ft^2
Heat Sink #6 Painted Steel 0.12" to <0.16"	
Structural steel	7,415.9 ft^2
Polar crane	424 ft^2
Miscellaneous	938.7 ft^2
	8,778.6 ft^2
Heat Sink #7 Painted Steel 0.16" to <0.24"	
Structural steel	4,313 ft^2
Polar crane	11,893 ft^2
Miscellaneous	3,996 ft^2
	20,202 ft^2
Heat Sink #8 Painted Steel 0.24" to <0.3"	
Structural steel	6,074 ft^2
Polar crane	6,796 ft^2
Miscellaneous	1,606 ft^2
	14,476 ft^2
Heat Sink #9 Painted Steel 0.3" to <0.4"	
Structural steel	7,694 ft^2
Polar crane	4,080 ft^2
Miscellaneous	504 ft^2
	12,278 ft^2
Heat Sink #10 Painted Steel 0.4" to <0.5"	
Structural steel	3,170 ft^2
Polar crane	2,604 ft^2
	5,774 ft^2

TABLE 6.2-5a (Continued)

Containment Vessel Heat Sinks

Heat Sink #11 Painted Steel 0.5" to <0.625"	
Structural steel	4,591 ft^2
Polar crane	18,988 ft^2
Miscellaneous	428 ft^2
	24,007 ft^2
Heat Sink #12 Painted Steel 0.625" to <0.75"	
Structural steel	1,987 ft^2
Polar crane	4,304 ft^2
Miscellaneous	650 ft^2
	6,941 ft^2
Heat Sink #13 Painted Steel 0.75" to <1.0"	
Structural steel	776 ft^2
Polar crane	4,602 ft^2
Miscellaneous	2,499 ft^2
	7,877 ft^2
Heat Sink #14 Painted Steel 1.0" to <1.5"	
Structural steel	1,953 ft^2
Polar crane	463 ft^2
Miscellaneous	928 ft^2
	3,344 ft^2
Heat Sink #15 Painted Steel ≥ 1.5"	
Structural steel	198 ft^2
Polar crane	1,055 ft^2
Miscellaneous	8,903 ft^2
	10,156 ft^2
Heat Sink #16 Refueling Pool Stainless Steel Liner (assume 1.0 ft concrete)	
	8,069 ft^2
Heat Sink #17 Stainless Steel Ductwork	
	5,609 ft^2

TABLE 6.2-5b

Heat Sink Material Properties

I.D.	Material	Thermal Conductivity (K) (BTU/hr-ft-°F)	Volumetric Heat Capacity (ρCp) (BTU/ft^3-°F)
1	Carbon steel	29.6	53.6
2	Stainless steel	8.6	60.1
3	Concrete	0.568	22.3
4	Inorganic zinc primer	0.633	21.7
5	Inorganic topcoat	0.5	31.2
6	Organic topcoat	0.19	47.1
7	Carboguard 890	0.399	34.9

Material I.D.	Node Spacing (in.)	Thickness (in.)
Heat Sink #1-Containment Dome		
7	1.2×10^{-3}	6.0×10^{-3}
4	5.0×10^{-4}	2.5×10^{-3}
1	8.25×10^{-2}	0.4125
1	8.0×10^{-2}	0.4
Heat Sink #2-Containment Walls		
6	1.2×10^{-3}	6.0×10^{-3}
4	6.0×10^{-4}	3.0×10^{-3}
1	5.0×10^{-2}	0.5
1	1.0×10^{-2}	1.0
Heat Sink #3-Unlined Concrete		
6	2.67×10^{-3}	8.0×10^{-3}
3	3.33×10^{-1}	1
3	0.16	4
3	1.3	13
Heat Sink #4- Galvanized Steel (Grating, HVAC, Cable Trays)		
6	2.4×10^{-3}	1.2×10^{-2}
4	1.4×10^{-4}	3.0×10^{-3}
1	1.3×10^{-3}	8.0×10^{-3}
1	1.7×10^{-2}	0.105

TABLE 6.2-5b (Continued)

Heat Sink Material Properties

Material I.D.	Node Spacing (in.)	Thickness (in.)
Heat Sink #5- Painted Steel <0.12"		
6	5.0×10^{-4}	5.0×10^{-3}
4	3.0×10^{-4}	3.0×10^{-3}
1	1.6×10^{-3}	8.0×10^{-3}
1	1.4×10^{-2}	7.0×10^{-2}
Heat Sink #6- Painted Steel 0.12" to <0.16"		
6	1.0×10^{-3}	5.0×10^{-3}
4	6.0×10^{-4}	3.0×10^{-3}
1	3.0×10^{-3}	3.0×10^{-2}
1	1.0×10^{-2}	0.1
Heat Sink #7- Painted Steel 0.16" to <0.24"		
6	1.0×10^{-3}	5.0×10^{-3}
4	6.0×10^{-4}	3.0×10^{-3}
1	1.0×10^{-3}	7.0×10^{-3}
1	2.5×10^{-2}	0.2
Heat Sink #8-Painted Steel 0.24" to <0.3"		
6	5.0×10^{-4}	5.0×10^{-3}
4	3.0×10^{-4}	3.0×10^{-3}
1	1.4×10^{-3}	7.0×10^{-3}
1	1.67×10^{-2}	0.25
Heat Sink #9-Painted Steel 0.3" to <0.4"		
6	1.0×10^{-3}	5.0×10^{-3}
4	6.0×10^{-4}	3.0×10^{-3}
1	5.25×10^{-3}	4.2×10^{-2}
1	3.75×10^{-2}	0.3
Heat Sink #10-Painted Steel 0.4" to <0.5"		
6	1.0×10^{-3}	5.0×10^{-3}
4	6.0×10^{-4}	3.0×10^{-3}
1	3.4×10^{-3}	5.1×10^{-2}
1	2.6×10^{-2}	0.4

TABLE 6.2-5b (Continued)

Heat Sink Material Properties

Material I.D.	Node Spacing (in.)	Thickness (in.)
Heat Sink #11-Painted Steel 0.5" to <0.625"		
6	1.0×10^{-3}	5.0×10^{-3}
4	6.0×10^{-4}	3.0×10^{-3}
1	7.5×10^{-4}	6.0×10^{-3}
1	7.14×10^{-2}	0.5
Heat Sink #12-Painted Steel 0.625" to <0.75"		
6	1.0×10^{-3}	5.0×10^{-3}
4	6.0×10^{-4}	3.0×10^{-3}
1	2.7×10^{-3}	4.1×10^{-2}
1	4.0×10^{-2}	0.6
Heat Sink #13-Painted Steel 0.75" to <1.0"		
6	1.0×10^{-3}	5.0×10^{-3}
4	6.0×10^{-4}	3.0×10^{-3}
1	6.3×10^{-3}	6.3×10^{-2}
1	4.67×10^{-2}	0.7
Heat Sink #14-Painted Steel 1.0" to <1.5"		
6	1.0×10^{-3}	5.0×10^{-3}
4	6.0×10^{-4}	3.0×10^{-3}
1	8.0×10^{-4}	6.4×10^{-2}
1	1.25×10^{-1}	1.0
Heat Sink #15-Painted Steel ≥1.5"		
6	1.0×10^{-3}	5.0×10^{-3}
4	6.0×10^{-4}	3.0×10^{-3}
1	5.4×10^{-2}	0.81
1	1.33×10^{-1}	2.0
Heat Sink #16-Refueling Pool		
2	5.2×10^{-3}	5.2×10^{-2}
2	1.3×10^{-2}	0.2
3	0.8	4
3	1.6	8
Heat Sink #17-Stainless Steel Ductwork		
2	2.4×10^{-3}	2.4×10^{-2}
2	3.2×10^{-3}	4.8×10^{-2}

TABLE 6.2-7

Containment Vessel Conditions

Parameters (units)	Prior to LOCA	At Peak Pressure
Time (sec)	0.0	14.3
Pressure:		
Air (psia)	14.7	19.3
Steam (psia)	0.6	33.0
Total (psia)	15.3	52.3
Total (psig)	0.9	37.9
Temperature		
Steam-Air (°F)	90.0	256.0
Sump Water (°F)	--	248.4
Mass:		
Air (lbm)	206,400	206,400
Steam (lbm)	4,923	224,800
Sump Water (lbm)	0	269,100
Total (lbm)	211,323	700,300

TABLE 6.2-8

Energy Distribution for 14.14 ft^2 DBA

Parameter	Energy Distribution Prior To LOCA, 10^6 Btu	Energy Distribution at 300 seconds After Break, 10^6 Btu
Energy Stored in RCS Metal	193	167
Energy Stored in SG Metal	79	75
Energy Stored in Core Metal	25	5
RCS Internal (fluid) Energy	300	30
CFT Coolant Internal Energy	10	0
SG Internal (fluid) Energy	70	82
Energy Content of Water Vapor in Containment Vessel	5	176
Energy Content of Air in Containment Vessel	19	24
Energy Content of Water in Sump	0	116
Total Energy Removed by CAC	0	0
Total Energy Removed by Containment Spray to Sump	0	3
Total Energy Removed by Structures	0	106

TABLE 6.2-9

Design Basis Accident Chronology Containment Vessel Analysis

Time, sec	Event
0	transient initation
10	CFT injection begins
14	peak pressure in containment vessel
15	end of primary system blowdown
40	CFTs empty
42	LPI/HPI flow begins
160	Containment Spray begins
300	CAC heat removal begins
4,500	Borated Water Storage Tank supply depleted, recirculation begins
6,500	secondary peak in Containment Vessel pressure
1,000,000	end of analysis, pressure reduced to about 4 psig

TABLE 6.2-10

Main Steam and Feedwater Line Breaks

Breaksize	Peak pressure (psig)	Time to peak (sec)
18.0" feedwater line	14.1	106.0
2.6 ft^2	19.478	29.5
3.1 ft^2	19.678	28.5
4.4 ft^2	20.435	28.0
5.4 ft^2	21.4	27
5.4 ft^2*	27.2	36

*Current analysis. The current analysis was only performed for the limiting break size of 5.4 ft^2. All other data is historical.

TABLE 6.2-11

Data for Main Steam Line Break
(Input to the Containment Vessel response analysis)

Time, sec	Mass Release Rate, Lbm/hr	Enthalpy, Btu/lbm
0.0	0.000E+00	0
0.0001	1.014E+06	1250.2
0.1	4.494E+07	1219.9
0.2	4.041E+07	1222.3
0.3	3.983E+07	1222.3
0.4	4.081E+07	1222.7
0.5	4.107E+07	1222.6
0.6	4.134E+07	1222.3
0.7	4.173E+07	1222.0
0.8	4.218E+07	1221.7
0.9	4.306E+07	1221.2
1	4.420E+07	1220.8
1.1	4.520E+07	1220.4
1.2	4.600E+07	1220.2
1.3	4.650E+07	1219.7
1.4	4.676E+07	1219.4
1.5	4.672E+07	1219.2
1.6	4.650E+07	1219.1
1.7	4.616E+07	1219.1
1.8	4.573E+07	1218.9
1.9	4.519E+07	1219.0
2	4.456E+07	1219.0
2.1	4.384E+07	1219.0
2.2	4.305E+07	1219.2
2.3	4.218E+07	1219.3
2.4	4.127E+07	1219.2
2.5	4.033E+07	1219.4
2.6	3.942E+07	1219.5
2.7	3.857E+07	1219.4
2.8	3.776E+07	1219.6

TABLE 6.2-11 (Continued)

Data for Main Steam Line Break
(Input to the Containment Vessel response analysis)

Time, sec	Mass Release Rate, Lbm/hr	Enthalpy, Btu/lbm
2.9	3.699E+07	1219.6
3	3.626E+07	1219.5
3.1	3.554E+07	1219.5
3.2	3.483E+07	1219.6
3.3	3.411E+07	1219.5
3.4	3.342E+07	1219.5
3.5	3.271E+07	1219.6
3.6	3.200E+07	1219.5
3.7	3.128E+07	1219.3
3.8	3.056E+07	1219.4
3.9	2.987E+07	1219.4
4	2.920E+07	1219.4
4.1	2.855E+07	1219.3
4.2	2.791E+07	1219.1
4.3	2.728E+07	1219.2
4.4	2.665E+07	1219.0
4.5	2.598E+07	1218.8
4.6	2.538E+07	1218.8
4.7	2.497E+07	1218.4
4.8	2.458E+07	1218.5
4.9	2.413E+07	1218.1
5	2.364E+07	1217.9
5.1	2.310E+07	1217.9
5.2	2.252E+07	1217.7
5.3	2.194E+07	1217.4
5.4	2.136E+07	1217.1
5.5	2.077E+07	1217.3
5.6	2.029E+07	1216.5
5.7	1.967E+07	1216.7
5.8	1.870E+07	1215.8

TABLE 6.2-11 (Continued)

Data for Main Steam Line Break
(Input to the Containment Vessel response analysis)

Time, sec	Mass Release Rate, Lbm/hr	Enthalpy, Btu/lbm
5.9	1.787E+07	1215.3
6	1.731E+07	1215.0
6.1	1.700E+07	1214.8
6.2	1.686E+07	1214.8
6.3	1.651E+07	1214.8
6.4	1.598E+07	1215.3
6.5	1.558E+07	1215.6
6.6	1.540E+07	1217.2
6.7	1.564E+07	1217.3
6.8	1.616E+07	1216.4
6.9	1.637E+07	1216.5
7	1.652E+07	1216.2
7.1	1.672E+07	1215.5
7.2	1.676E+07	1215.7
7.3	1.682E+07	1215.3
7.4	1.673E+07	1215.0
7.5	1.678E+07	1215.1
7.6	1.666E+07	1214.8
7.7	1.670E+07	1214.5
7.8	1.658E+07	1215.0
7.9	1.650E+07	1214.2
8	1.647E+07	1214.6
8.1	1.630E+07	1214.2
8.2	1.613E+07	1214.3
8.3	1.612E+07	1214.4
8.4	1.607E+07	1214.2
8.5	1.610E+07	1214.3
8.6	1.606E+07	1213.8
8.7	1.593E+07	1214.2
8.8	1.603E+07	1213.6

TABLE 6.2-11 (Continued)

Data for Main Steam Line Break
(Input to the Containment Vessel response analysis)

Time, sec	Mass Release Rate, Lbm/hr	Enthalpy, Btu/lbm
8.9	1.590E+07	1213.9
9	1.569E+07	1213.6
9.1	1.556E+07	1213.8
9.2	1.535E+07	1213.8
9.3	1.519E+07	1213.7
9.4	1.502E+07	1213.8
9.5	1.488E+07	1213.6
9.6	1.471E+07	1213.6
9.7	1.456E+07	1213.6
9.8	1.443E+07	1213.5
9.9	1.426E+07	1214.0
10	1.420E+07	1213.9
12	1.308E+07	1213.1
14	1.163E+07	1211.7
15	1.085E+07	1210.3
16	1.027E+07	1210.0
18	9.314E+06	1209.6
20	8.066E+06	1208.8
22	6.932E+06	1208.5
24	6.314E+06	1205.8
25	6.199E+06	1203.3
26	5.555E+06	1202.2
28	4.682E+06	1200.7
30	3.800E+06	1200.4
32	3.148E+06	1215.0
34	2.563E+06	1214.2
35	1.951E+06	1217.7
36	1.573E+06	1226.5
38	1.130E+06	1227.7
40	8.046E+05	1230.4

TABLE 6.2-11 (Continued)

Data for Main Steam Line Break
(Input to the Containment Vessel response analysis)

Time, sec	Mass Release Rate, Lbm/hr	Enthalpy, Btu/lbm
42	6.894E+05	1235.0
44	6.678E+05	1231.8
45	6.588E+05	1240.4
46	6.444E+05	1234.6
48	6.084E+05	1236.7
50	5.994E+05	1240.2
52	6.012E+05	1239.5
54	6.030E+05	1235.8
55	6.012E+05	1239.5
56	6.048E+05	1238.1
58	6.624E+05	1238.5
60	6.678E+05	1239.9
61	5.832E+05	1240.7
62	5.256E+05	1246.6
63	4.932E+05	1255.5
64	4.644E+05	1255.8
65	4.464E+05	1241.9
66	4.320E+05	1266.7
67	4.284E+05	1243.7
68	4.176E+05	1266.7
69	4.104E+05	1263.2
70	4.104E+05	1245.6
71	4.104E+05	1254.4
72	4.104E+05	1254.4
73	4.032E+05	1258.9
74	4.068E+05	1247.8
75	4.032E+05	1258.9
76	4.068E+05	1247.8
77	4.032E+05	1258.9

TABLE 6.2-11 (Continued)

Data for Main Steam Line Break
(Input to the Containment Vessel response analysis)

Time, sec	Mass Release Rate, Lbm/hr	Enthalpy, Btu/lbm
78	4.032E+05	1258.9
79	4.032E+05	1250.0
80	4.032E+05	1258.9
81	4.032E+05	1258.9
82	4.068E+05	1247.8
83	4.032E+05	1258.9
84	4.032E+05	1250.0
85	4.032E+05	1258.9
86	4.032E+05	1258.9
87	4.068E+05	1247.8
88	4.032E+05	1258.9
89	4.032E+05	1258.9
90	4.032E+05	1250.0
91	4.032E+05	1258.9
92	4.032E+05	1258.9
93	4.068E+05	1247.8
94	4.032E+05	1258.9
95	4.032E+05	1250.0
96	4.032E+05	1258.9
97	4.032E+05	1258.9
98	4.032E+05	1258.9
99	4.068E+05	1247.8
100	4.032E+05	1258.9
101	4.032E+05	1258.9
102	4.032E+05	1250.0
103	4.032E+05	1258.9
104	4.068E+05	1247.8
105	4.032E+05	1258.9
106	4.032E+05	1258.9

TABLE 6.2-11 (Continued)

Data for Main Steam Line Break
(Input to the Containment Vessel response analysis)

Time, sec	Mass Release Rate, Lbm/hr	Enthalpy, Btu/lbm
107	4.032E+05	1258.9
108	4.032E+05	1250.0
109	4.032E+05	1258.9
136	4.032E+05	1258.9
166	3.996E+05	1261.3
196	3.996E+05	1252.3
227	3.978E+05	1257.9
258	3.960E+05	1254.5
290	4.050E+05	1253.3
324	4.284E+05	1256.3
356	4.230E+05	1255.3
505	3.989E+05	1254.5
555	3.960E+05	1250.9
595	3.989E+05	1249.1
600	3.996E+05	1246.8

TABLE 6.2-12

Reactor Cavity Pressurization Vent Areas,
Flow Coefficients, and Compartment Volumes

Vent Areas and Flow Coefficients

Path From	Path To	Vent Area (ft^2)	Flow Coefficient	Orifice (ft^2)
1	2	18.0	.78	
1	3	28.7	.92	
1	5	41.7	.99	
1	9	20.9	.94	
1	10	21.3	.94	
1	18	0.7	.78	
1	27	22.7	.87	
2	24	1002.6	.94	
2	26	175.8	.91	
2	27	85.0	.86	
3	2	10.8	.78	
3	4	21.9	.88	
3	9	21.3	.94	
3	27	11.2	.87	
4	2	7.3	.79	
4	6	41.7	.99	
4	11	20.1	.94	
4	27	9.9	.87	
5	7	21.9	.88	
5	10	20.1	.94	
5	18	0.7	.78	
5	27	17.2	.84	
6	8	21.9	.88	
6	11	21.3	.94	
6	27	18.9	.84	
7	8	28.7	.92	
7	12	21.3	.94	
7	27	22.0	.83	
8	12	20.9	.94	
8	27	22.0	.83	
9	10	31.6	.97	
9	11	31.6	.97	
9	13	53.6	.98	
10	12	31.6	.97	
10	13	53.6	.98	
11	12	31.6	.97	
11	13	53.6	.98	
12	13	53.6	.98	
13	14	21.0	.62	
13	19	85.5	.83	
14	15	60.0	.86	
15	16	60.0	.86	

TABLE 6.2-12 (Continued)

Reactor Cavity Pressurization Vent Areas,
Flow Coefficients, and Compartment Volumes

Path From	Path To	Vent Area (ft^2)	Flow Coefficient	Orifice (ft^2)
16	17	60.0	.86	
17	18	29.2	.88	
18	27	24.5	.85	
19	20	24.0	.86	27.5
20	21	29.3	.88	27.5
21	22	27.0	.88	27.5
22	23	41.0	.89	34.0
23	27	60.0	.73	34.0
24	25	385.0	.90	
24	26	92.5	.85	
24	27	8.0	.83	
25	27	690.0	.89	
26	27	277.0	.97	

TABLE 6.2-12 (Continued)

Reactor Cavity Pressurization Vent Areas,
Flow Coefficients, and Compartment Volumes

Compartment Volumes

	Compartment	Volume (ft^3)
1	Upper Cavity	590
2	Steam Generator	12069
3	Upper Cavity	284
4	Upper Cavity	306
5	Upper Cavity	306
6	Upper Cavity	299
7	Upper Cavity	284
8	Upper Cavity	267
9	Middle Cavity	562
10	Middle Cavity	556
11	Middle Cavity	556
12	Middle Cavity	528
13	Bottom of R. Cavity	5688
14	Access Tunnel	1091
15	Access Tunnel	300
16	Access Tunnel	322
17	Access Tunnel	646
18	Access Tunnel	1138
19	Instrumentation Tunnel	1747
20	Instrumentation Tunnel	820
21	Instrumentation Tunnel	404
22	Instrumentation Tunnel	1618
23	Instrumentation Tunnel	651
24	Steam Generator	9765
25	Steam Generator	66970
26	Steam Generator	5338
27	Containment Free Volume	2.7E6

TABLE 6.2-13

Steam Generator Cavity Pressurization
Vent Areas and Flow Coefficients

Flow Path	Area (ft^2)	Flow Coefficient
1-2	50	0.97
1-3	107	0.93
1-4	143.6	0.95
1-8	19.59	0.76
2-3	90	0.86
2-5	414	0.97
2-8	19.59	0.76
2-11	85	0.86
3-5	445	0.90
3-7	175.75	0.91
4-5	185	0.88
4-6	99	0.90
5-6	286	0.90
5-7	92.5	0.85
5-11	8	0.83
6-11	690	0.89
7-11	277	0.97
8-9	21	0.62
8-10	55	0.776
8-11	72.9	0.86
9-11	29	0.87
10-11	57.5	0.767

TABLE 6.2-13 (Continued)

Steam Generator Cavity Pressurization
Vent Areas and Flow Coefficients

Compartment Volumes

Compartment	Volume (ft^3)
1	2,145
2	5,319
3	4,502
4	2,276
5	7,489
6	66,970
7	5,338
8	10,946
9	1,573
10	6,053
11	2,740,000

TABLE 6.2-14

Components of Flow Coefficients

A. Reactor Cavity

Vent	F1/D	Contraction	Bend	Expansion	ΣK	C
1-2		.50		1.10	1.60	
1-2		.72		1.00	1.72	.78*
1-3		.18		1.00	1.18	.92
1-5		.02		1.00	1.02	.99
1-9	.04	.08		1.00	1.12	.94
1-10	.04	.08		1.00	1.12	.94
1-18	.20	.43		1.00	1.63	.78
1-27	.02	.30		1.00	1.32	.87
2-24	0	.13		1.00	1.13	.94
2-26	0	.21		1.00	1.21	.91
2-27	0	.25		1.00	1.35	.86
3-2		.72		1.00	1.72	
3-2		.50		1.10	1.6	.78*
3-4		.28		1.00	1.28	.88
3-9	.04	.08		1.00	1.12	.94
3-27	.02	.30		1.00	1.32	.87
4-2		.5		1.1	1.6	.79
4-6		.02		1.0	1.02	.99
4-11	.04	.08		1.00	1.12	.94
4-27	.02	.30		1.00	1.32	.87
5-7		.28		1.00	1.28	.88
5-10	.04	.08		1.00	1.12	.94
5-18	.20	.43		1.00	1.63	.78
5-27	.02	.30		1.00	1.32	.87
6-8		.28		1.00	1.28	.88
6-11	.04	.08		1.00	1.12	.94
6-27	.02	.30		1.00	1.32	.87
7-8		.18		1.00	1.18	.92
7-12	.04	.08		1.00	1.12	.94
7-27	.02	.30		1.00	1.32	.87
8-12	.04	.08		1.00	1.12	.94
8-27	.02	.30		1.00	1.32	.87
9-10		.07		1.00	1.07	.97
9-11		.07		1.00	1.07	.97
9-13	.04			1.00	1.04	.98
10-12		.07		1.00	1.07	.97
10-13	.04			1.00	1.04	.98
11-12		.07		1.00	1.07	.97
11-13	.04			1.00	1.04	.98

*Area weighted flow coefficients

TABLE 6.2-14 (Continued)

Components of Flow Coefficients

Vent	F1/D	Contraction	Bend	Expansion	ΣK	C
12-13	.04			1.00	1.04	.98
13-14	.08	.44	1.4	0.665	2.585	.62
13-19	0	.44		1.00	1.44	.83
14-15	0	.33		1.00	1.33	.86
15-16	0	.33		1.00	1.33	.86
16-17	0	.33		1.00	1.33	.86
17-18	0	.29		1.00	1.29	.88
18-27	0	.40		1.00	1.40	.85
19-20	0	.31	.052	1.00	1.36	.86
20-21	0	.29		1.00	1.29	.88
21-22	0	.29		1.00	1.29	.88
22-23	0	.26		1.00	1.26	.89
23-27	0	.88		1.00	1.88	.73
24-25	0	.23		1.00	1.23	.94
24-26	0	.40		1.00	1.40	.85
24-27	0	.44		1.00	1.44	.83
25-27	0	.26		1.00	1.26	.89
26-27	0	.07		1.00	1.07	.97

B. Steam Generator

Vent	F1/D	Contraction	Bend	Expansion	ΣK	C
1-2	0	0.24	0	0.82	1.06	0.97
1-3	0	0.15	0	1.0	1.15	0.93
1-4	0	0.11	0	1.0	1.11	0.95
1-8	0	0.55	0	1.18	1.73	0.76
2-3	0	0.35	0	1.0	1.35	0.86
2-5	0	0.06	0	1.0	1.06	0.97
2-8	0	0.55	0	1.18	1.73	0.76
2-11	0	0.35	0	1.0	1.35	0.86
3-5	0	0.23	0	1.0	1.23	0.90
3-7	0	0.21	0	1.0	1.21	0.91
4-5	0	0.30	0	1.0	1.30	0.88
4-6	0	0.23	0	1.0	1.23	0.90
5-6	0	0.23	0	1.0	1.23	0.90
5-7	0	0.40	0	1.0	1.40	0.85
5-11	0	0.44	0	1.0	1.44	0.83
6-11	0	0.26	0	1.0	1.26	0.89
7-11	0	0.07	0	1.0	1.07	0.97
8-9	0.08	0.44	1.4	0.67	2.59	0.62
8-10	0	0.55	0	1.11	1.66	0.776
8-11	0.19	0.55	0	0.61	1.35	0.86
9-11	0	0.31	0	1.0	1.31	0.87
10-11	0	0.06	0.58	1.06	1.70	0.767

TABLE 6.2-15

7.07 ft^2 (1A) Hot Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
.0000	8.0240+04	5.0110+07	.0000	.0000
.0000	8.0240+04	5.0110+07	8.4252+02	5.2615+05
1.0500-02	8.0240+04	5.0110+07	8.4252+02	5.2615+05
1.0500-02	9.3100+04	5.8120+07	1.7549+03	1.0957+06
2.0300-02	9.3100+04	5.8120+07	1.7549+03	1.0957+06
2.0300-02	8.0800+04	5.0430+07	2.5467+03	1.5899+06
3.0100-02	8.0800+04	5.0430+07	2.5467+03	1.5899+06
3.0100-02	6.7060+04	4.1850+07	3.2375+03	2.0210+06
4.0400-02	6.7060+04	4.1850+07	3.2375+03	2.0210+06
4.0400-02	6.4670+04	4.0350+07	3.8842+03	2.4245+06
5.0400-02	6.4670+04	4.0350+07	3.8842+03	2.4245+06
5.0400-02	6.4480+04	4.0230+07	4.5161+03	2.8188+06
6.0200-02	6.4480+04	4.0230+07	4.5161+03	2.8188+06
6.0200-02	6.4510+04	4.0240+07	5.1934+03	3.2413+06
7.0700-02	6.4510+04	4.0240+07	5.1934+03	3.2413+06
7.0700-02	6.4670+04	4.0340+07	5.8272+03	3.6366+06
8.0500-02	6.4670+04	4.0340+07	5.8272+03	3.6366+06
8.0500-02	6.4920+04	4.0490+07	6.4634+03	4.0334+06
9.0300-02	6.4920+04	4.0490+07	6.4634+03	4.0334+06
9.0300-02	6.5220+04	4.0660+07	7.1026+03	4.4319+06
1.0010-01	6.5220+04	4.0660+07	7.1026+03	4.4319+06
1.0010-01	6.5470+04	4.0810+07	7.7900+03	4.8604+06
1.1060-01	6.5470+04	4.0810+07	7.7900+03	4.8604+06
1.1060-01	6.5660+04	4.0920+07	8.4335+03	5.2614+06
1.2040-01	6.5660+04	4.0920+07	8.4335+03	5.2614+06
1.2040-01	6.5770+04	4.0980+07	9.0780+03	5.6630+06
1.3020-01	6.5770+04	4.0980+07	9.0780+03	5.6630+06
1.3020-01	6.5810+04	4.0990+07	9.7690+03	6.0934+06
1.4070-01	6.5810+04	4.0990+07	9.7690+03	6.0934+06
1.4070-01	6.5790+04	4.0970+07	1.0414+04	6.4949+06
1.5050-01	6.5790+04	4.0970+07	1.0414+04	6.4949+06
1.5050-01	6.5780+04	4.0950+07	1.1058+04	6.8962+06
1.6030-01	6.5780+04	4.0950+07	1.1058+04	6.8962+06
1.6030-01	6.5690+04	4.0880+07	1.1702+04	7.2968+06
1.7010-01	6.5690+04	4.0880+07	1.1702+04	7.2968+06
1.7010-01	6.5540+04	4.0780+07	1.2390+04	7.7250+06
1.8060-01	6.5540+04	4.0780+07	1.2390+04	7.7250+06
1.8060-01	6.5410+04	4.0690+07	1.3031+04	8.1238+06
1.9040-01	6.5410+04	4.0690+07	1.3031+04	8.1238+06
1.9040-01	6.5240+04	4.0580+07	1.3671+04	8.5215+06

TABLE 6.2-15 (Continued)

7.07 ft^2 (1A) Hot Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
2.0020-01	6.5240+04	4.0580+07	1.3671+04	8.5215+06
2.0020-01	6.5040+04	4.0460+07	1.4354+04	8.9463+06
2.1070-01	6.5040+04	4.0460+07	1.4354+04	8.9463+06
2.1070-01	6.4980+04	4.0390+07	1.4990+04	9.3421+06
2.2050-01	6.4980+04	4.0390+07	1.4990+04	9.3421+06
2.2050-01	6.4910+04	4.0330+07	1.5627+04	9.7374+06
2.3030-01	6.4910+04	4.0330+07	1.5627+04	9.7374+06
2.3030-01	6.4830+04	4.0270+07	1.6262+04	1.0132+07
2.4010-01	6.4830+04	4.0270+07	1.6262+04	1.0132+07
2.4010-01	6.4800+04	4.0240+07	1.6942+04	1.0555+07
2.5060-01	6.4800+04	4.0240+07	1.6942+04	1.0555+07
2.5060-01	6.4810+04	4.0230+07	1.7577+04	1.0949+07
2.6040-01	6.4810+04	4.0230+07	1.7577+04	1.0949+07
2.6040-01	6.4810+04	4.0210+07	1.8213+04	1.1343+07
2.7020-01	6.4810+04	4.0210+07	1.8213+04	1.1343+07
2.7020-01	6.4760+04	4.0170+07	1.8893+04	1.1765+07
2.8070-01	6.4760+04	4.0170+07	1.8893+04	1.1765+07
2.8070-01	6.4740+04	4.0150+07	1.9527+04	1.2158+07
2.9050-01	6.4740+04	4.0150+07	1.9527+04	1.2158+07
2.9050-01	6.4780+04	4.0160+07	2.0162+04	1.2552+07
3.0030-01	6.4780+04	4.0160+07	2.0162+04	1.2552+07
3.0030-01	6.4790+04	4.0160+07	2.0797+04	1.2945+07
3.1010-01	6.4790+04	4.0160+07	2.0797+04	1.2945+07
3.1010-01	6.4800+04	4.0150+07	2.1477+04	1.3367+07
3.2060-01	6.4800+04	4.0150+07	2.1477+04	1.3367+07
3.2060-01	6.4890+04	4.0190+07	2.2113+04	1.3761+07
3.3040-01	6.4890+04	4.0190+07	2.2113+04	1.3761+07
3.3040-01	6.4990+04	4.0250+07	2.2750+04	1.4155+07
3.4020-01	6.4990+04	4.0250+07	2.2750+04	1.4155+07
3.4020-01	6.5050+04	4.0270+07	2.3433+04	1.4578+07
3.5070-01	6.5050+04	4.0270+07	2.3433+04	1.4578+07
3.5070-01	6.5090+04	4.0290+07	2.4071+04	1.4973+07
3.6050-01	6.5090+04	4.0290+07	2;4071+04	1.4973+07
3.6050-01	6.5160+04	4.0320+07	2.4709+04	1.5368+07
3.7030-01	6.5160+04	4.0320+07	2.4709+04	1.5368+07
3:7030-01	6.5180+04	4.0320+07	2.5348+04	1.5763+07
3.8010-01	6.5180+04	4.0320+07	2.5348+04	1.5763+07
3.8010-01	6.5090+04	4.0260+07	2.6032+04	1.6186+07
3.9060-01	6.5090+04	4.0260+07	2.6032+04	1.6186+07
3.9060-01	6.4970+04	4.0180+07	2.6668+04	1.6580+07

TABLE 6.2-15 (Continued)

7.07 ft^2 (1A) Hot Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
4.0040-01	6.4970+04	4.0180+07	2.6668+04	1.6580+07
4.0040-01	6.4870+04	4.0110+07	2.7304+04	1.6973+07
4.1020-01	6.4870+04	4.0110+07	2.7304+04	1.6973+07
4.1020-01	6.4690+04	3.9980+07	2.7983+04	1.7392+07
4.2070-01	6.4690+04	3.9980+07	2.7983+04	1.7392+07
4.2070-01	6.4440+04	3.9810+07	2.8615+04	1.7783+07
4.3050-01	6.4440+04	3.9810+07	2.8615+04	1.7783+07
4.3050-01	6.4220+04	3.9670+07	2.9244+04	1.8171+07
4.4030-01	6.4220+04	3.9670+07	2.9244+04	1.8171+07
4.4030-01	6.4050+04	3.9550+07	2.9872+04	1.8559+07
4.5010-01	6.4050+04	3.9550+07	2.9872+04	1.8559+07
4.5010-01	6.3840+04	3.9410+07	3.0542+04	1.8973+07
4.6060-01	6.3840+04	3.9410+07	3.0542+04	1.8973+07
4.6060-01	6.3620+04	3.9270+07	3.1166+04	1.9358+07
4.7040-01	6.3620+04	3.9270+07	3.1166+04	1.9358+07
4.7040-01	6.3520+04	3.9200+07	3.1788+04	1.9742+07
4.8020-01	6.3520+04	3.9200+07	3.1788+04	1.9742+07
4.8020-01	6.3480+04	3.9170+07	3.2455+04	2.0153+07
4.9070-01	6.3480+04	3.9170+07	3.2455+04	2.0153+07
4.9070-01	6.3440+04	3.9130+07	3.3076+04	2.0536+07
5.0050-01	6.3440+04	3.9130+07	3.3076+04	2.0536+07
5.0050-01	6.3410+04	3.9110+07	3.3698+04	2.0920+07
5.1030-01	6.3410+04	3.9110+07	3.3698+04	2.0920+07
5.1030-01	6.3460+04	3.9130+07	3.4320+04	2.1303+07
5.2010-01	6.3460+04	3.9130+07	3.4320+04	2.1303+07
5.2010-01	6.3510+04	3.9160+07	3.4987+04	2.1714+07
5.3060-01	6.3510+04	3.9160+07	3.4987+04	2.1714+07
5.3060-01	6.3490+04	3.9140+07	3.5609+04	2.2098+07
5.4040-01	6.3490+04	3.9140+07	3.5609+04	2.2098+07
5.4040-01	6.3440+04	3.9100+07	3.6231+04	2.2481+07
5.5020-01	6.3440+04	3.9100+07	3.6231+04	2.2481+07
5.5020-01	6.3400+04	3.9070+07	3.6896+04	2.2891+07
5.6070-01	6.3400+04	3.9070+07	3.6896+04	2.2891+07
5.6070-01	6.3310+04	3.9010+07	3.7517+04	2.3274+07
5.7050-01	6.3310+04	3.9010+07	3.7517+04	2.3274+07
5.7050-01	6.3150+04	3.8900+07	3.8136+04	2.3655+07
5.8030-01	6.3150+04	3.8900+07	3.8136+04	2.3655+07
5.8030-01	6.2980+04	3.8790+07	3.8753+04	2.4035+07
5.9010-01	6.2980+04	3.8790+07	3.8753+04	2.4035+07
5.9010-01	6.2820+04	3.8680+07	3.9412+04	2.4441+07

TABLE 6.2-15 (Continued)

7.07 ft^2 (1A) Hot Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
6.0060-01	6.2820+04	3.8680+07	3.9412+04	2.4441+07
6.0060-01	6.2640+04	3.8560+07	4.0026+04	2.4819+07
6.1040-01	6.2640+04	3.8560+07	4.0026+04	2.4819+07
6.1040-01	6.2450+04	3.8440+07	4.0638+04	2.5196+07
6.2020-01	6.2450+04	3.8440+07	4.0638+04	2.5196+07
6.2020-01	6.2280+04	3.8330+07	4.1292+04	2.5598+07
6.3070-01	6.2280+04	3.8330+07	4.1292+04	2.5598+07
6.3070-01	6.2130+04	3.8230+07	4.1901+04	2.5973+07
6.4050-01	6.2130+04	3.8230+07	4.1901+04	2.5973+07
6.4050-01	6.2000+04	3.8140+07	4.2509+04	2.6347+07
6.5030-01	6.2000+04	3.8140+07	4.2509+04	2.6347+07
6.5030-01	6.1880+04	3.8060+07	4.3115+04	2.6720+07
6.6010-01	6.1880+04	3.8060+07	4.3115+04	2.6720+07
6.6010-01	6.1770+04	3.7990+07	4.3764+04	2.7119+07
6.7060-01	6.1770+04	3.7990+07	4.3764+04	2.7119+07
6.7060-01	6.1690+04	3.7930+07	4.4368+04	2.7490+07
6.8040-01	6.1690+04	3.7930+07	4.4368+04	2.7490+07
6.8040-01	6.1620+04	3.7880+07	4.4972+04	2.7862+07
6.9020-01	6.1620+04	3.7880+07	4.4972+04	2.7862+07
6.9020-01	6.1570+04	3.7840+07	4.5619+04	2.8259+07
7.0070-01	6.1570+04	3.7840+07	4.5619+04	2.8259+07
7.0070-01	6.1520+04	3.7810+07	4.6221+04	2.8629+07
7.1050-01	6.1520+04	3.7810+07	4.6221+04	2.8629+07
7.1050-01	6.1480+04	3.7780+07	4.6824+04	2.9000+07
7.2030-01	6.1480+04	3.7780+07	4.6824+04	2.9000+07
7.2030-01	6.1440+04	3.7750+07	4.7426+04	2.9370+07
7.3010-01	6.1440+04	3.7750+07	4.7426+04	2.9370+07
7.3010-01	6.1400+04	3.7710+07	4.8071+04	2.9766+07
7.4060-01	6.1400+04	3.7710+07	4.8071+04	2.9766+07
7.4060-01	6.1350+04	3.7670+07	4.8672+04	3.0135+07
7.5040-01	6.1350+04	3.7670+07	4.8672+04	3.0135+07
7.5040-01	6.1290+04	3.7640+07	4.9273+04	3.0504+07
7.6020-01	6.1290+04	3.7640+07	4.9273+04	3.0504+07
7.6020-01	6.1240+04	3.7590+07	4.9916+04	3.0898+07
7.7070-01	6.1240+04	3.7590+07	4.9916+04	3.0898+07
7.7070-01	6.1170+04	3.7550+07	5.0515+04	3.1266+07
7.8050-01	6.1170+04	3.7550+07	5.0515+04	3.1266+07
7.8050-01	6.1110+04	3.7500+07	5.1114+04	3.1634+07
7.9030-01	6.1110+04	3.7500+07	5.1114+04	3.1634+07
7.9030-01	6.1050+04	3.7460+07	5.1712+04	3.2001+07

TABLE 6.2-15 (Continued)

7.07 ft^2 (1A) Hot Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
8.0010-01	6.1050+04	3.7460+07	5.1712+04	3.2001+07
8.0010-01	6.0990+04	3.7420+07	5.2353+04	3.2394+07
8.1060-01	6.0990+04	3.7420+07	5.2353+04	3.2394+07
8.1060-01	6.0940+04	3.7370+07	5.2950+04	3.2760+07
8.2040-01	6.0940+04	3.7370+07	S.2950+04	3.2760+07
8.2040-01	6.0890+04	3.7340+07	5.3547+04	3.3126+07
8.3020-01	6.0890+04	3.7340+07	5.3547+04	3.3126+07
8.3020-01	6.0850+04	3.7310+07	5.4186+04	3.3518+07
8.4070-01	6.0850+04	3.7310+07	5.4186+04	3.3518+07
8.4070-01	6.0820+04	3.7280+07	5.4782+04	3.3883+07
8.5050-01	6.0820+04	3.7280+07	5.4782+04	3.3883+07
8.5050-01.	6.0800+04	3.7260+07	5.5377+04	3.4248+07
8.6030-01	6.0800+04	3.7260+07	5.5377+04	3.4248+07
8.6030-01	6.0780+04	3.7250+07	5.5973+04	3.4613+07
8.7010-01.	6.0780+04	2.7250+07	5.5973+04	3.4613+07
8.7010-01	6.0770+04	3.7230+07	5.6611+04	3.5004+07
8.8060-01	6.0770+04	3.7230+07	S.6611+04	3.5004+07
8.8060-01	6.0760+04	3.7220+07	5.7207+04	3.5369+07
8.9040-01	6.0760+04	3.7220+07	5.7207+04	3.5369+07
8.9040-01	6.0740+04	3.7210+07	5.7802+04	3.5734+07
9.0020-01	6.0740+04	3.7210+07	5.7802+04	3.5734+07
9.0020-01	6.0730+04	3.7190+07	5.8440+04	3.6124+07
9.1070-01	6.0730+04	3.7190+07	5.8440+04	3.6124+07
9.1070-01	6.0710+04	3.7180+07	5.9035+04	3.6488+07
9.2050-01	6.0710+04	3.7180+07	5.9035+04	3.6488+07
9.2050-01	6.0690+04	3.7160+07	5.9629+04	3.6853+07
9.3030-01	6.0690+04	3.7160+07	5.9629+04	3.6853+07
9.3030-01	6.0660+04	3.1140+07	6.0224+04	3.7217+07
9.4010-01	6.0660+04	3.7140+07	6.0224+04	3.7217+07
9.4010-01	6.0620+04	3.7120+07	6.0860+04	3.7606+07
9.5060-01	6.0620+04	3.7120+07	6.0860+04	3.7606+07
9.5060-01	6.0500+04	3.7010+07	6.1453+04	3.7969+07
9.6040-01	6.0500+04	3.7010+07	6.1453+04	3.7960+07
9.6040-01	6.0540+04	3.7060+07	6.2046+04	3.8332+07
9.7020-01	6.0540+04	3.7060+07	6.2046+04	3.8332+07
9.7020-01	6.0490+04	3.7020+07	6.2682+04	3.8721+07
9.8070-01	6.0490+04	3.7020+07	6.2682+04	3.8721+07
9.8070-01	6.0430+04	3.6990+07	6.3298+04	3.9098+07

TABLE 6.2-15 (Continued)

7.07 ft^2 (1A) Hot Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
9.9090-01	6.0430+04	3.6990+07	6.3298+04	3.9098+07
9.9090-01	6.0370+04	3.6950+07	6.3847+04	3.9434+07
1.0000+00	6.0370+04	3.6950+07	6.3847+04	3.9434+07
1.0000+00	6.0300+04	3.6900+07	6.4450+04	3.9803+07
1.0100+00	6.0300+04	3.6900+07	6.4450+04	3.9803+07
1.0100+00	6.0230+04	3.6840+07	6.5113+04	4.0209+07
1.0210+00	6.0230+04	3.6840+07	6.5113+04	4.0209+07
1.0210+00	6.0160+04	3.6820+07	6.5654+04	4.0540+07
1.0300+00	6.0160+04	3.6820+07	6.5654+04	4.0540+07
1.0300+00	6.0080+04	3.6770+07	6.6255+04	4.0908+07
1.0400+00	6.0080+04	3.6770+07	6.6255+04	4.0908+07
1.0400+00	6.0010+04	3.6720+07	6.6915+04	4.1312+07
1.0510+00	6.0010+04	3.6720+07	6.6915+04	4.1312+07
1.0510+00	5.9940+04	3.6680+07	6.7455+04	4.1642+07
1.0600+00	5.9940+04	3.6680+07	6.7455+04	4.1642+07
1.0600+00	5.9870+04	3.6640+07	6.8053+04	4.2008+07
1.0700+00	5.9870+04	3.6640+07	6.8053+04	4.2008+07
1.0700+00	5.9810+04	3.6600+07	6.8651+04	4.2374+07
1.0800+00	5.9810+04	3.6600+07	6.8651+04	4.2374+07
1.0800+00	5.9750+04	3.6570+07	6.9309+04	4.2776+04
1.0910+00	5.9750+04	3.6570+07	6.9309+04	4.2776+07
1.0910+00	5.9700+04	3.6540+07	6.9846+04	4.3105+07
1.1000+00	5.9700+04	3.6540+07	6.9846+04	4.3105+07
1.1000+00	5.9660+04	3.6520+07	7.0443+04	4.3471+07
1.1100+00	5.9660+04	3.6520+07	7.0443+04	4.3471+07
1.1100+00	5.9630+00	3.6510+07	7.1099+04	4.3872+07
1.1210+00	5.9630+04	3.6510+07	7.1099+04	4.3872+07
1.1210+00	5.9610+04	3.6500+07	7.1635+04	4.4201+07
1.1300+00	5.9610+04	3.6500+07	7.1635+04	4.4201+07
1.1300+00	5.9600+04	3.6490+07	7.2231+04	4.4566+07
1.1400+00	5.9600+04	3.6490+07	7.2231+04	4.4566+07
1.1400+00	5.9580+04	3.6490+07	7.2827+04	4.4930+07
1.1500+00	5.9580+04	3.6490+07	7.2827+04	4.4930+07
1.1500+00	5.9570+04	3.6490+07	7.3482.04	4.5332+07
1.1610+00	5.9570+04	3.6490+07	7.3482+04	4.5332+07
1.1610+00	5.9560+04	3.6490+07	7.4018+04	4.5660+07
1.1700+00	5.9560+04	3.6490+07	7.4018+04	4.5660+07
1.1700+00	5.9550+04	3.6480+07	7.4614+04	4.6025+07
1.1800+00	5.9550+04	3.6480+07	7.4614+04	4.6025+07

TABLE 6.2-15 (Continued)

7.07 ft^2 (1A) Hot Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
1.1800+00	5.9530+04	3.6480+07	7.5268+04	4.6426+07
1.1800+00	5.9530+04	3.6480+07	7.5268+04	4.6426+07
1.1910+00	5.9530+04	3.6480+07	7.5268+04	4.6426+07
1.1910+00	5.9500+04	3.6460+07	7.5804+04	4.6754+07
1.2000+00	5.9500+04	3.6460+07	7.5804+04	4.6754+07
1.2000+00	5.9460+04	3.6450+07	7.6399+04	4.7119+07
1.2100+00	5.9460+04	3.6450+07	7.6399+04	4.7119+07
1.2100+00	5.9420+04	3.6420+07	7.6993+04	4.7483+07
1.2200+00	5.9420+04	3.6420+07	7.6993+04	4.7483+07
1.2200+00	5.9360+04	3.6390+07	7.7646+04	4.7883+07
1.2310+00	5.9360+04	3.6390+07	7.7646+04	4.7883+07
1.2310+00	5.9310+04	3.6370+07	7.8180+04	4.8211+07
1.2400+00	5.9310+04	3.6370+07	7.8180+04	4.8211+07
1.2400+00	5.9250+04	3.6340+07	7.8772+04	4.8574+07
1.2500+00	5.9250+04	3.6340+07	7.8772+04	4.8574+07
1.2500+00	5.9190+04	3.6310+07	7.9423+04	4.8974+07
1.2610+00	5.9190+04	3.6310+07	7.9423+04	4.8974+07
1.2610+00	5.9140+04	3.6280+07	7.9955+04	4.9300+07
1.2700+00	5.9140+04	3.6280+07	7.9955+04	4.9300+07
1.2700+00	5.9090+04	3.6260+07	8.0546+04	4.9663+07
1.2800+00	5.9090+04	3.6260+07	8.0546+04	4.9663+07
1.2800+00	5.9050+04	3.6240+07	8.1137+04	5.0025+07
1.2900+00	5.9050+04	3.6240+07	8.1137+04	5.0025+07
1.2900+00	5.9010+04	3.6220+07	8.1786+04	5.0424+07

TABLE 6.2-16

8.55 ft^2 (2A) Cold Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
.0000	.0000	.0000	.0000	.0000
.0000	.0000	.0000	5.2786+02	2.9050+05
1.0500-02	1.0054+05	5.5334+07	1.5132+03	8.3278+05
2.0300-02	1.0054+05	5.5334+07	1.5132+03	8.3278+05
2.0300-02	1.0087+05	5.5534+07	2.5017+03	1.3770+06
3.0100-02	1.0087+05	5.5534+07	2.5017+03	1.3770+06
3.0100-02	9.6934+04	5.3362+07	3.5195+03	1.9373+06
4.0600-02	9.6934+04	5.3362+07	3.5195+03	1.9373+06
4.0600-02	9.0696+04	4.9910+07	4.4084+03	2.4264+06
5.0400-02	9.0696+04	4.9910+07	4.4084+03	2.4264+06
5.0400-02	9.1334+04	5.0281+07	5.3034+03	2.9192+06
6.0200-02	9.1334+04	5.0281+07	5.3034+03	2.9192+06
6.0200-02	9.4487+04	5.2050+07	6.2956+03	3.4657+06
7.0700-02	9.4487+04	5.2050+07	6.2956+03	3.4657+06
7.0700-02	9.1268+04	5.0273+07	7.1900+03	3.9584+06
8.0500-02	9.1268+04	5.0273+07	7.1900+03	3.9584+06
8.0500-02	9.9590+04	5.4888+07	8.1660+03	4.4963+06
9.0300.02	9.9590+04	5.4888+07	8.1660+03	4.4963+06
9.0300-02	1.0562+05	5.8240+07	9.2011+03	5.0670+06
1.0010-01	1.0562+05	5.8240+07	9.2011+03	5.0670+06
1.0010-01	1.0741+05	5.9247+07	1.0329+04	5.6891+06
1.1060-01	1.0741+05	5.9247+07	1.0329+04	5.6891+06
1.1060-01	1.0480+05	5.7807+07	1.1356+04	6.2556+06
1.2040-01	1.0480+05	5.7807+07	1.1356+04	6.2556+06
1.2040-01	1.0712+05	5.9109+07	1.2406+04	6.8349+06
1.3020-01	1.0712+05	5.9109+07	1.2406+04	6.8349+06
1.3020-01	1.0943+05	6.0406+07	1.3555+04	7.4692+06
1.4070-01	1.0943+05	6.0406+07	1.3555+04	7.4692+06
1.4070-01	1.0796+05	5.9601+07	1.4613+04	8.0533+06
1.5050.01	1.0796+05	5.9601+07	1.4613+04	8.0533+06
1.5050-01	1.0631+05	5.8693+07	1.5655+04	8.6284+06
1.6030-01	1.0631+05	5.8693+07	1.5655+04	8.6284+06
1.6030-01	1.0340+05	5.7077+07	1.6668+04	9.1878+06
1.7010-01	1.0340+05	5.7077+07	1.6668+04	9.1878+06
1.7010-01	1.0326+05	5.7003+07	1.7752+04	9.7868+06
1.8060-01	1.0326+05	5.7003+07	1.7752+04	9.7863+06
1.8060-01	1.0349+05	5.7145+07	1.8766+04	1.0346+07
1.9040-01	1.0349+05	5.7145+07	1.8766+04	1.0346+07

TABLE 6.2-16 (Continued)

8.55 ft^2 (2A) Cold Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
1.9040-01	1.0347+05	5.7141+07	1.9780+04	1.0906+07
2.0020-01	1.0347+05	5.7141+07	1.9780+04	1.0906+07
2.0020-01	1.0375+05	5.7306+07	2.0870+04	1.1508+07
2.1070-01	1.0375+05	5.7306+07	2.0870+04	1.1508+07
2.1070-01	1.0463+05	5.7805+07	2.1895+04	1.2075+07
2.2050-01	1.0463+05	5.7805+07	2.1895+04	1.2075+07
2.2050-01	1.0579+05	5.8458+07	2.2932+04	1.2647+07
2.3030-01	1.0579+05	5.8450+07	2.2932+04	1.2647+07
2.3030-01	1.0403+05	5.7487+07	2.3951+04	2.3211+07
2.4010-01	1.0403+05	5.7487+07	2.3951+04	1.3211+07
2.4010-01	1.0149+05	5.6073+07	2.5017+04	1.3800+07
2.5060-01	1.0149+05	5.6073+07	2.5017+04	1.3800+07
2.5060-01	1.0286+05	5.6850+07	2.6025+04	1.4357+07
2;6040-01	1.0287+05	5.6850+07	2.6025+04	1.4357+07
2.6040-01	1.0416+05	5.7578+07	2.7046+04	1.4921+07
2.7020-01	1.0416+05	5.7578+07	2.7046+04	1.4921+07
2.7020-01	1.0182+05	5.6283+07	2.8115+04	1.5512+07
2.8070-01	1.0182+05	5.6283+07	2.8115+04	1.5512+07
2.8070-01	1.0157+05	5.6149+07	2.9110+04	1.6062+07
2.9050-01	1.0157+05	5.6149+07	2.9110+04	1.6062+07
2.9050-01	1.0526+05	5.8215+07	3.0142+04	1.6633+07
3.0030-01	1.0526+05	5.8215+07.	3.0142+04	1.6633+07
3.0030-01	1.0600+05	5.8635+07	3.1181+04	1.7207+07
3.1010-01	1.0600+05	5.8635+07	3.1181+04	1.7207+07
3.1010-01	1.0357+05	5.7283+07	3.2268+04	1.7809+07
3.2060-01	2.0357+05	5.7283+07	3.2268+04	1.7809+07
3.2060-01	1.0254+05	5.6715+07	3.3273+04	1.8365+07
3.3040-01	1.0254+05	5.6715+07	3.3273+04	1.8365+07
3.3040-01	1.0350+05	5.7259+07	3.4286+04	1.8926+07
3.4020-01	1.0350+05	5.7259+07	3.4286+04	1.8926+07
3.4020-01	1.0325+05	5.7123+07	3.5372+04	1.9526+07
3.5070-01	1.0325+05	5.7123+07	3.5372+04	1.9526+07
3.5070-01	1.0243+05	5.6670+07	3.6375+04	2.0081+07
3.6050-01	1.0243+05	5.6670+07	3.6375+04	2.0081+07
3.6050-01	1.0216+05	5.6528+07	3.7277+04	2.0635+07
3.7030-01	1.0216+05	5.6528+07	3.7277+04	2.0635+07
3.7030-01	1.0263+05	5.6797+07	3.8382+04	2.1191+07
3.8010-01	1.0263+05	5.6797+07	3.8382+04	2.1191+07

TABLE 6.2-16 (Continued)

8.55 ft^2 (2A) Cold Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
3.8010-01	1.0280+05	5.6901+07	3.9462+04	2.1789+07
3.9060-01	1.0280+05	5.6901+07	3.9462+04	2.1789+07
3.9060-01	1.0304+05	5.7047+07	4.0472+04	2.2348+07
4.0040-01	1.0304+05	5.7047+07	4.0472+04	2.2348+07
4.0040-01	1.0290+05	5.6979+07	4.1480+04	2.2906+07
4.1020-01	1.0290+05	5.6979+07	4.1480+04	2.2906+07
4.1020-01	1.0128+05	5.6079+07	4.2543+04	2.3495+07
4.2070-01	1.0128+05	5.6079+07	4.2543+04	2.3495+07
4.2070-01	1.0056+05	5.5689+07	4.3529+04	2.4041+07
4.3050-01	1.0056+05	5.5689+07	4.3529+04	2.4041+07
4.3050-01	1.0131+05	5.6120+07	4.4522+04	2.4591+07
4.4030-01	1.0131+05	5.6120+07	4.4522+04	2.4591+07
4.4030-01	1.0164+05	5.6316+07	4.5518+04	2.5143+07
4.5010-01	1.0164+05	5.6316+07	4.5518+04	2.5143+07
4.5010-01	1.0082+05	5.5869+07	4.6576+04	2.5729+07
4.6060-01	1.0082+05	5.5869+07	4.6576+04	2.5729+07
4.6060-01	1.0073+05	5.5830+07	4.7564+04	2.6277+07
4.7040-01	1.0073+05	5.5830+07	4.7564+04	2.6277+07
4.7040-01	1.0171+05	5.6397+07	4.8560+04	2.6829+07
4.8020-01	1.0171+05	5.6397+07	4.8560+04	2.6829+07
4.8020-01	1.0114+05	5.6086+07	4.9622+04	2.7418+07
4.9070-01	1.0114+05	5.6086+07	4.9622+04	2.7418+07
4.9070-01	1.0011+05	5.5523+07	5.0603+04	2.7962+07
5.0050-01	1.0011+05	5.5523+07	5.0603+04	2.7962+07
5.0050-01	1.0045+05	5.5727+07	5.1588+04	2.8508+07
5.1030-01	1.0045+05	5.5727+07	5.1588+04	2.8508+07
5.1030-01	1.0020+05	5.5599+07	5.2570+04	2.9053+07
5.2010-01	1.0020+05	5.5599+07	5.2570+04	2.9053+07
5.2010-01	9.8435+04	5.4824+07	5.3603+04	2.9627+07
5.3060-01	9.8435+04	5.4824+07	5.3603+04	2.9627+07
5.3060-01	9.7825+04	5.4296+07	5.4562+04	3.0159+07
5.4040-01	9.7825+04	5.4296+07	5.4562+04	3.0159+07
5.4040-01	9.9811+04	5.5426+07	5.5540+04	3.0702+07
5.5020-01	9.9811+04	5.5426+07	5.5540+04	3.0702+07
5.5020-01	1.0101+05	5.6111+07	5.6601+04	3.1291+07
5.6070-01.	1.0101+05	5.6111+07	5.6601+04	3.1291+07
5.6070-01	9.8734+04	5.4849+07	5.7568+04	3.1829+07
5.7050-01	9.8734+04	5.4849+07	5.7568+04	3.1829+07

TABLE 6.2-16 (Continued)

8.55 ft^2 (2A) Cold Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
5.7050-01	9.6657+04	5.3699+07	5.8516+04	3.3255+07
5.8030-01	9.6657+04	5.3699+07	5.8516+04	3.3255+07
5.8030-01	9.7821+04	5.4369+07	5.9474+04	3.2888+07
5.9010-01	9.7821+04	5.4369+07	5.9474+04	3.2888+07
5.9010-01	9.9112+04	5.5111+07	6.0515+04	3.3467+07
6.0060-01	9.9112+04	5.5111+07	6.0515+04	3.3467+07
6.0060-01	9.7707+04	5.4335+07	6.1472+04	3.3999+07
6.1040-01	9.7707+04	5.4335+07	6.1472+04	3.3999+07
6.1040-01	9.6220+04	5.3516+07	6.2415+04	3.4524+07
6.2020-01	9.6220+04	5.3516+07	6.2415+04	3.4524+07
6.2020-01	9.7558+04	5.4286+07	6.3440+04	3.5094+07
6.3070-01	9.7558+04	5.4286+07	6.3440+04	3.5094+07
6.3070-01	9.8996+04	5.5114+07	6.4410+04	3.5624+07
6.4050-01	9.8996+04	5.5114+07	6.4410+04	3.5624+07
6.4050-01	9.7449+04	5.4261+07	6.5365+04	3.6165+07
6.5030-01	9.7449+04	5.4261+07	6.5365+04	3.6165+07
6.5030-01	9.6412+04	5.3693+07	6.6310+04	3.6692+07
6.6010-01	9.6412+04	5.3693+07	6.6310+04	3.6692+07
6.6010-01	9.7838+04	5.4513+07	6.7337+04	3.7264+07
6.7060-01	9.7838+04	5.4513+07	6.7337+04	3.7264+07
6.7060-01	9.9198+04	5.5296+07	6.8309+04	3.7806+07
6.8040-01	9.9198+04	5.5296+07	6.8309+04	3.7806+07
6.8040-01	9.8094+04	5.4691+07	6.9270+04	3.8342+07
6.9020-01	9.8094+04	5.4691+07	6.9270+04	3.8342+07
6.9020-01	9.6419+04	5.3766+07	7.0283+04	3.8906+07
7.0070-01	9.6419+04	5.3766+07	7.0283+04	3.8906+07
7.0070-01	9.6440+04	5.3800+07	7.1228+04	3.9434+07
7.1050-01	9.6440+04	5.3800+07	7.1228+04	3.9434+07
7.1050-01	9.7197+04	5.4248+07	7.2190+04	3.9965+07
7.2030-01	9.7197+04	5.4248+07	7.2190+04	3.9965+07
7.2030-01	9.7075+04	5.4195+07.	7.3132+04	4.0496+07
7.3010-01	9.7075+04	5.4195+07	7.3132+04	4.0496+07
7.3010-01	9.6553+04	5.3915+07	7.4166+04	4.1063+07
7.4060-01	9.6553+04	5.3915+07	7.4166+04	4.1063+07
7.4060-01	9.6893+04	5.4126+07	7.5095+04	4.1593+07
7.5040-01	9.6893+04	5.4126+07	7.5095+04	4.1593+07
7.5040-01	9.7018+04	5.4218+07	7.6046+04	4.2124+07
7.6020-01	9.7018+04	5.4218+07	7.6046+04	4.2124+07

TABLE 6.2-16 (Continued)

8.55 ft^2 (2A) Cold Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
7.6020-01	9.5348+04	5.3295+07	7.7047+04	4.2684+07
7.7070-01	9.5348+04	5.3295+07	7.7047+04	4.2684+07
7.7070-01	9.3278+04	5.2147+07	7.7961+04	4.3195+07
7.8050-01	9.3278+04	5.2147+07	7.7961+04	4.3195+07
7.8050-01	9.3366+04	5.2217+07	7.8876+04	4.3707+07
7.9030-01	9.3366+04	5.2217+07	7.8876+04	4.3707+07
7.9030-01	9.5065+04	5.3195+07	7.9808+04	4.4228+07
8.0010-01	9.5065+04	5.3195+07	7.9808+04	4.4228+07
8.0010-01	9.6896+04	5.4250+07	8.0825+04	4.4798+07
8.1060-01	9.6896+04	5.4250+07	8.0825+04	4.4798+07
8.1060-01	9.6238+04	5.3898+07	8.1768+04	4.5326+07
8.2040-01	9.6238+04	5.3898+07	8.1768+04	4.5326+07
8.2040-01	9.2210+04	5.1641+07	8.2672+04	4.5832+07
8.3020-01	9.2210+04	5.1641+07	8.2672+04	4.5832+07
8.3020-01	9.2847+04	5.2027+07	8.3647+04	4.6378+07
8.4070-01	9.2847+04	5.2027+07	8.3647+04	4.6378+07
8.4070-01	9.3246+04	5.2278+07	8.4561+04	4.6890+07
8.5050-01	9.3246+04	5.2278+07	8.4561+04	4.6890+07
8.5050-01	9.0797+04	5.0902+07	8.5451+04	4.7369+07
8.6030-01	9.0797+04	5.0902+07	8.5451+04	4.7369+07
8.6030-01	9.5389+04	5.3528+07	8.6385+04	4.7914+07
8.7010-01	9.5389+04	5.3528+07	8.6385+04	4.7914+07
8.7010-01	9.1250+04	5.1195+07	8.7344+04	4.8451+07
8.8060-01	9.1250+04	5.1195+07	8.7344+04	4.8451+07
8.8060-01	9.4081+04	5.2833+07	8.8265+04	4.8969+07
8.9040-01	9.4081+04	5.2833+07	8.8265+04	4.8969+07
8.9040-01	8.7810+04	4.9293+07	8.9126+04	4.9452+07
9.0020-01	8.7810+04	4.9293+07	8.9126+04	4.9452+07
9.0020-01	8.9903+04	5.0509+07	9.0070+04	4.9983+07
9.1070-01	8.9903+04	5.0509+07	9.0070+04	4.9983+07
9.1070-01	8.8927+04	4.9969+07	9.0942+04	5.0472+07
9.2050-01	8.8927+04	4.9969+07	9.0942+04	5.0472+07
9.2050-01	9.0073+04	5.1034+07	9.1831+04	5.0972+07
9.3030-01	9.0073+04	5.1034+07	9.1831+04	5.0972+07
9.3030-01	9.2147+04	5.1834+07	9.2734+04	5.1480+07
9.4010-01	9.2147+04	5.1834+07	9.2734+04	5.1480+07
9.4010-01	8.9906+04	5.0581+07	9.3678+04	5.2012+07
9.5060-01	8.9906+04	5.0581+07	9.3678+04	5.2012+07

TABLE 6.2-16 (Continued)

8.55 ft^2 (2A) Cold Leg Break Data
For Reactor Cavity Pressurization

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Total Mass (lb)	Total Energy (Btu)
9.5060-01	8.9012+04	5.0101+07	9.4550+04	5.2503+07
9.6040-01	8.9012+04	5.0101+07	9.4550+04	5.2503+07
9.6040-01	8.8085+04	4.9599+07	9.5414+04	5.2989+07
9.7020-01	8.8085+04	4.9599+07	9.5414+04	5.2989+07
9.7020-01	8.8492+04	4.9848+07	9.6343+04	5.3512+07
9.8070-01	8.8492+04	4.9848+07	9.6343+04	5.3512+07
9.8070-01	9.0192+o4	5.0828+07	9.7227+04	5.4010+07
9.9050-01	9.0192+04	5.0828+07	9.7227+04	5.4010+07
9.9050-01	9.1244+04	5.1442+07	9.8094+04	5.4499+07
1.0000+00	9.1244+04	5.1442+07	9.8094+04	5.4499+07
1.0000+00	9.0594+04	5.1094+07	9.8999+04	5.5010+07

TABLE 6.2-17

14.14 ft^2 (2A) Hot Leg Break Data
For Steam Generator Compartment Pressurization

Time Interval (sec)	Mass Release Rate (lbm/sec)	Energy Release Rate (Btu/sec)	Enthalpy (Btu/lbm)
0.00 – 0.01	131598.40	.8216284E+08	624.35
0.01 – 0.02	117632.37	.7334166E+08	623.48
0.02 – 0.03	103616.38	.6449550E+08	622.45
0.03 – 0.04	90563.38	.5628773E+08	621.53
0.04 – 0.05	81408.59	.5057942E+08	621.30
0.05 – 0.06	77832.17	.4844156E+08	622.38
0.06 – 0.07	79030.97	.4928072E+08	623.56
0.07 – 0.08	82497.50	.5151848E+08	624.49
0.08 – 0.09	86373.63	.5397602E+08	624.91
0.09 – 0.10	89660.34	.5603397E+08	624.96
0.10 – 0.11	92434.61	.5776660E+08	624.95
0.11 – 0.12	94605.39	.5910090E+08	624.71
0.12 – 0.13	96203.80	.6004995E+08	624.20
0.13 – 0.14	97102.90	.6060939E+08	624.18
0.14 – 0.15	97502.50	.6079920E+08	623.57
0.15 – 0.16	97302.70	.6064935E+08	623.31
0.16 – 0.17	96703.30	.6019930E+08	622.52
0.17 – 0.18	95774.65	.5955734E+08	621.85
0.18 – 0.19	94605.39	.5874126E+08	620.91
0.19 – 0.20	93506.49	.5804196E+08	620.73
0.20 – 0.21	92307.69	.5724276E+08	620.13
0.21 – 0.22	91508.49	.5664336E+08	619.00
0.22 – 0.23	90809.19	.5624376E+08	619.36
0.23 – 0.24	90409.59	.5584416E+08	617.68
0.24 – 0.25	90442.66	.5583501E+08	617.35
0.25 – 0.26	90509.49	.5584416E+08	617.00
0.26 – 0.27	90809.19	.5594406E+08	616.06
0.27 – 0.28	91308.69	.5624376E+08	615.97
0.28 – 0.29	91808.19	.5644356E+08	614.80
0.29 – 0.30	92207.79	.5664336E+08	614.30
0.30 – 0.31	92707.29	.5694306E+08	614.22
0.31 – 0.32	92957.75	.5704225E+08	613.64
0.32 – 0.33	93206.79	.5714286E+08	613.08
0.33 – 0.34	93306.69	.5714286E+08	612.42
0.34 – 0.35	93106.89	.5714286E+08	613.73
0.35 – 0.36	92907.09	.5684316E+08	611.83
0.36 – 0.37	92507.49	.5664336E+08	612.31
0.37 – 0.38	92007.99	.5634366E+08	612.38
0.38 – 0.39	91448.69	.5593561E+08	611.66
0.39 – 0.40	90709.29	.5544456E+08	611.23

TABLE 6.2-17 (Continued)

14.14 ft^2 (2A) Hot Leg Break Data
For Steam Generator Compartment Pressurization

Time Interval (sec)	Mass Release Rate (lbm/sec)	Energy Release Rate (Btu/sec)	Enthalpy (Btu/lbm)
0.40 - 0.41	89910.09	.5504496E+08	612.22
0.41 - 0.42	89010.99	.5444555E+08	611.67
0.42 - 0.43	88111.89	.5384615E+08	611.11
0.43 - 0.44	87212.79	.5334665E+08	611.68
0.44 - 0,45	86413.59	.5274725E+08	610.40
0.45 - 0.46	85613.68	.5231388E+08	611.15
0.46 - 0.47	84915.08	.5184815E+08	610.59
0.47 - 0.48	84215.78	.5144855E+08	610.91
0.48 - 0.49	83616.38	.5104895E+08	610.51
0.49 - 0.50	83216.78	.5084915E+08	611.04
0.50 - 0.51	82817.18	.5044955E+08	609.17
0.51 - 0.52	82317.68	.5024975E+08	610.44
0.52 - 0.53	82092.56	.5000000E+08	609.07
0.53 - 0.54	81718.28	.4975025E+08	608.80
0.54 - 0.55	81418.58	.4965035E+08	609.82
0.55 - 0.56	81218.78	.4945055E+08	608.86
0.56 - 0.57	80919.08	.4915085E+08	607.41
0.57 - 0.58	80619.38	.4915085E+08	609.67
0.58 - 0.59	80319.68	.4885115E+08	608.21
0.59 - 0.60	80080.48	.4869215E+08	608.24
0.60 - 0.61	79820.18	.4845155E+08	607.01
0.61 - 0.62	79420.58	.4825175E+08	607.55
0.62 - 0.63	79220.78	.4815185E+08	607.82
0.63 - 0.64	78821.18	.4785215E+08	607.10
0.64 - 0.65	78521.48	.4765235E+08	606.87
0.65 - 0.66	78121.88	.4745255E+08	607.42
0.66 - 0.67	77766.60	.4718310E+08	606.73
0.67 - 0.68	77522.48	.4695305E+08	605.67
0.68 - 0.69	77122.88	.4685315E+08	607.51
0.69 - 0.70	76823.18	.4655345E+08	605.98
0.70 - 0.71	76523.48	.4635365E+08	605.74
0.71 - 0.72	76223.78	.4615385E+08	605.58
0.72 - 0.73	75924.08	.4605395E+08	606.58
0.73 - 0.74	75653.92	.4577465E+08	605.05
0.74 - 0.75	75324.88	.4565435E+08	606.10
0.75 - 0.76	75124.88	.4545455E+08	605.05
0.76 - 0.77	74825.17	.4525475E+08	604.81
0.77 - 0.78	74525.47	.4505495E+08	604.56
0.78 - 0.79	74325.67	.4495504E+08	604.84
0.79 - 0.80	74025.97	.4475524E+08	604.59

TABLE 6.2-17 (Continued)

14.14 ft^2 (2A) Hot Leg Break Data
For Steam Generator Compartment Pressurization

Time Interval (sec)	Mass Release Rate (lbm/sec)	Energy Release Rate (Btu/sec)	Enthalpy (Btu/lbm)
0.80 - 0.81	73843.06	.4466801E+08	604.90
0.81 - 0.82	73526.47	.4445554E+08	604.62
0.82 - 0.83	73326.67	.4435564E+08	604.90
0.83 - 0.84	73226.77	.4415584E+08	603.00
0.84 - 0.85	72927.07	.4405594E+08	604.11
0.85 - 0.86	72727.27	.4385614E+08	603.02
0.86 - 0.87	72527.47	.4385614E+08	604.68
0.87 - 0.88	72334.00	.4366197E+08	603.62
0.88 - 0.89	72227.77	.4355644E+08	603.04
0.89 - 0.90	71928.07	.4335664E+08	602.78
0.90 - 0.91	71928.07	.4335664E+08	602.78
0.91 - 0.92	71628.37	.4325674E+04	603.91
0.92 - 0.93	71628.37	.4305694E+08	601.12
0.93 - 0.94	71328.67	.4305694E+08	603.64
0.94 - 0.95	71327.97	.4295775E+08	602.26
0.95 - 0.96	71128.87	.4285714E+08	602.53
0.96 - 0.97	71028.97	.4275724E+08	601.97
0.97 - 0.98	70929.07	.4265734E+08	601.41
0.98 - 0.99	70829.17	.4265734E+08	602.26
0.99 - 1.00	70870.87	.4264264E+08	601.69
1.00 - 1.01	70700.00	.4250000E+08	601.13
1.01 - 1.02	70010.00	.4220000E+08	602.00
1.02 - 1.03	70500.00	.4240000E+08	601.42
1.03 - 1.04	70500.00	.4230000E+08	600.00
1.04 - 1.05	70400.00	.4230000E+08	600.85
1.05 - 1.06	70300.00	.4220000E+08	600.28
1.06 - 1.07	70200.00	.4220000E+08	601.14
1.07 - 1.08	70100.00	.4210000E+08	600.57
1.08 - 1.09	69600.00	.4180000E+08	600.57
1.09 - 1.10	70100.00	.4210000E+08	600.57
1.10 - 1.11	70000.00	.4200000E+08	600.00
1.11 - 1.12	69900.00	.4190000E+08	599.43
1.12 - 1.13	69900.00	.4190000E+08	599.43
1.13 - 1.14	69800.00	.4190000E+08	600.29
1.14 - 1.15	69800.00	.4180000E+08	598.85
1.15 - 1.16	69200.00	.4150000E+08	599.71
1.16 - 1.17	69800.00	.4180000E+08	598.85
1.17 - 1.18	69600.00	.4170000E+08	599.14
1.18 - 1.19	69700.00	.4180000E+08	599.71
1.19 - 1.20	69600.00	.4170000E+08	599.14

TABLE 6.2-17 (Continued)

14.14 ft^2 (2A) Hot Leg Break Data
For Steam Generator Compartment Pressurization

Time Interval (sec)	Mass Release Rate (lbm/sec)	Energy Release Rate (Btu/sec)	Enthalpy (Btu/lbm)
1.20 - 1.21	69600.00	.4160000E+08	597.70
1.21 - 1.22	69600.00	.4170000E+08	599.14
1.22 - 1.23	69100.00	.4130000E+08	597.68
1.23 - 1.24	69300.00	.4160000E+08	600.29
1.24 - 1.25	70000.00	.4160000E+08	594.29
1.25 - 1.26	69000.00	.4160000E+08	602.90
1.26 - 1.27	70000.00	.4160000E+08	594.29
1.27 - 1.28	69000.00	.4150000E+08	601.45
1.28 - 1.29	70000.00	.4160000E+08	594.29
1.29 - 1.30	69000.00	.4120000E+08	597.10
1.30 - 1.31	69000.00	.4150000E+08	601.45
1.31 - 1.32	70000.00	.4150000E+08	592.86
1.32 - 1.33	69000.00	.4150000E+08	601.45
1.33 - 1.34	70000.00	.4150000E+08	592.86
1.34 - 1.35	69000.00	.4140000E+08	600.00
1.35 - 1.36	69000.00	.4150000E+08	601.45
1.36 - 1.37	69000.00	.4110000E+08	595.65
1.37 - 1.38	70000.00	.4140000E+08	591.43
1.38 - 1.39	69000.00	.4140000E+08	600.00
1.39 - 1.40	69000.00	.4130000E+08	598.55
1.40 - 1.41	69000.00	.4140000E+08	600.00
1.41 - 1.42	70000.00	.4130000E+08	590.00
1.42 - 1.43	69000.00	.4130000E+08	598.55
1.43 - 1.44	69000.00	.4100000E+08	594.20
1.44 - 1.45	69000.00	.4130000E+08	598.55
1.45 - 1.46	69000.00	.4130000E+08	598.55
1.46 - 1.47	69000.00	.4120000E+08	597.10
1.47 - 1.48	69000.00	.4120000E+08	597.10
1.48 - 1.49	69000.00	.4120000E+08	597.10
1.49 - 1.50	69000.00	.4120000E+08	597.10
1.50 - 1.51	69000.00	.4080000E+08	591.30
1.51 - 1.52	69000.00	.4120000E+08	597.10
1.52 - 1.53	69000.00	.4110000E+08	595.65
1.53 - 1.54	68000.00	.4100000E+08	602.94
1.54 - 1.55	69000.00	.4110000E+08	595.65
1.55 - 1.56	69000.00	.4100000E+08	594.20
1.56 - 1.57	69000.00	.4100000E+08	594.20
1.57 - 1.58	68000.00	.4070000E+08	598.53
1.58 - 1.59	69000.00	.4100000E+08	594.20
1.59 - 1.60	69000.00	.4090000E+08	592.75

TABLE 6.2-17 (Continued)

14.14 ft^2 (2A) Hot Leg Break Data
For Steam Generator Compartment Pressurization

Time Interval (sec)	Mass Release Rate (lbm/sec)	Energy Release Rate (Btu/sec)	Enthalpy (Btu/lbm)
1.60 - 1.61	68000.00	.4090000E+08	601.47
1.61 - 1.62	69000.00	.4090000E+08	592.75
1.62 - 1.63	68000.00	.4080000E+08	600.00
1.63 - 1.64	69000.00	.4080000E+08	591.30
1.64 - 1.65	68000.00	.4060000E+08	597.06
1.65 - 1.66	68000.00	.4080000E+08	600.00
1.66 - 1.67	69000.00	.4070000E+08	589.86
1.67 - 1.68	68000.00	.4080000E+08	600.00
1.68 - 1.69	69000.00	.4080000E+08	591.30
1.69 - 1.70	68000.00	.4070000E+08	598.53
1.70 - 1.71	68000.00	.4070000E+08	598.53
1.71 - 1.72	68000.00	.4050000E+08	595.59
1.72 - 1.73	68000.00	.4060000E+08	597.06
1.73 - 1.74	68000.00	.4070000E+08	598.53
1.74 - 1.75	68000.00	.4060000E+08	597.06
1.75 - 1.76	68000.00	.4060000E+08	597.06
1.76 - 1.77	67000.00	.4060000E+08	605.97
1.77 - 1.78	68000.00	.4050000E+08	595.59
1.78 - 1.79	67000.00	.4020000E+08	600.00
1.79 - 1.80	67000.00	.4040000E+08	602.99
1.80 - 1.81	68000.00	.4050000E+08	595.59
1.81 - 1.82	67000.00	.4030000E+08	601.49
1.82 - 1.83	67000.00	.4030000E+08	601.49
1.83 - 1.84	67000.00	.4030000E+08	601.49
1.84 - 1.85	67000.00	.4030000E+08	601.49
1.85 - 1.86	67000.00	.3990000E+08	595.52
1.86 - 1.87	66000.00	.4020000E+08	609.09
1.87 - 1.88	67000.00	.4010000E+08	598.51
1.88 - 1.89	67000.00	.4000000E+08	597.01
1.89 - 1.90	66000.00	.4010000E+08	607.58
1.90 - 1.91	66000.00	.3990000E+08	604.55
1.91 - 1.92	67000.00	.4000000E+08	597.01
1.92 - 1.93	65000.00	.3960000E+08	609.23
1.93 - 1.94	66000.00	.3980000E+08	603.03
1.94 - 1.95	66000.00	.3980000E+08	603.03
1.95 - 1.96	66000.00	.3970000E+08	601.52
1.96 - 1.97	65000.00	.3970000E+08	610.77
1.97 - 1.98	66000.00	.3960000E+08	600.00
1.98 - 1.99	65000.00	.3960000E+08	609.23

TABLE 6.2-18

Steam Generator Compartment Pressure

Compartment	Peak Calculated (psi)	Design (psi)
1 - 11	86.54	114.06
2 - 11	23.44	127.5
3 - 11	23.64	51.0
4 - 11	38.84	124.1
5 - 11	23.55	51.0

TABLE 6.2-19

Containment Air Cooler Unit
Equipment Design Data
(Capacities are for single components and were used for original sizing)

Equipment Data	Duty: LOCA Operation	Duty: Normal Operation
Peak heat load, Btu/hr each	75×10^{6} (1)	1.80×10^{6}
Fan capacity, cfm (Total per unit)	58,000 (2)	117,000
Containment atmosphere inlet temperature, °F	265	120
Service water flow, gpm	1600 (1)	540
Fan speed	Half	Full

(1) Refer to Subsection 6.2.1.3.2 for analysis using reduced Service Water flow and resulting reduced heat removal rate of the Containment Air Coolers for LOCA operation.

(2) As noted in Section 6.2.2.2.1, the CAC slow speed fan flowrate has been reduced from 58,000 cfm to 45,000 cfm. The resulting decrease in heat removal capability has been analyzed and found acceptable.

TABLE 6. 2-20

Containment Spray System Design Parameters and Major Equipment Data
(Equipment capacities are for single components)

Containment Spray System Performance Data

System heat removal capacity, Btu/hr	150×10^6
System total flow capacity, gpm	2600
System design pressure, psig	200
Pump discharge piping design pressure, psig	300
System design temperature, °F	300

Containment Spray Pumps

Number	2
Type	Horizontal, Centrifugal
Rated capacity, gpm	1,300
Rated head, ft H_2O	400
Motor horsepower	200
Material	SS

Spray Headers and Nozzles

Number of headers	2
Nozzles on each header	90
Nozzle type	Full cone
Spray drop size, mass median diameter (micron)	780
Material	SS

TABLE 6.2-21

Single Failure Analysis-Containment Vessel Heat Removal Systems

	Component	Malfunction	Comments and Consequences
1.	Containment spray nozzles	Clogged	Large number of nozzles on each of two headers renders clogging of significant number of nozzles as incredible. Also, the nozzles are designed to pass up to 1/4" size particles.
2.	Containment spray header	Rupture	This is considered incredible due to the systems design to withstand the design basis temperature, pressure and seismic forces. However, there are two independent headers provided. With the loss of one header, the second header is still available.
3.	Check valve in spray header line	Sticks closed	This is considered incredible due to large opening force available at pump shutoff head. However, there are two independent spray headers. The second header is available to perform the function.
4.	Motor-operated valve in spray header line	Fails to open	Second header delivers 50 percent flow.
5.	Spray pump isolation valve	Left closed	Flow and cooling capacity reduced to 50 percent of design. In combination with containment air coolers, 150 percent of total design requirement is still provided.
6.	Containment spray pump	Fails to start	Flow and cooling capacity reduced to 50 percent of design. In combination with containment air coolers, 150 percent of total design requirement is still provided.

TABLE 6.2-21 (Continued)

Single Failure Analysis-Containment Vessel Hebat Removal Systems

	Component	Malfunction	Comments and Consequences
7.	Containment air cooling unit fan	Stops	The third standby containment air cooling fan unit is available and can be manually started. In combination with the two spray pumps, 200% of the total design requirements are still available.
8.	Containment air cooling unit	Rupture of cooling coil	The tubes are designed for 150 psi and 300°F which exceeds maximum operating conditions. Tubes are protected against credible missiles. Hence, rupture is not considered credible.
9.	Containment air cooling unit	Rupture of casing and/or ducts	The design basis temperature, pressure and seismic forces during a post-accident situation were utilized in the system design. The units are also inspectable and protected against credible missiles. Cooling with these units is supplemented by the sprays.
10.	Containment air cooling units	Rupture of system piping	The piping, including expansion bellows, are designed to withstand the design basis temperature, pressure and seismic forces during a post-accident situation and are inspectable and protected from missiles. Maximum actual internal pressure is less than 150 psi at temperatures below 300 degree°F.
11.	Motor-operated valve at inlet penetration	Sticks closed	Two of the air cooling units are in operation normally. Flow is periodically established through the idle line to check the operational capability of the standby unit. Such tests indicate if the valve is malfunctioning.

TABLE 6.2-21 (Continued)

Single Failure Analysis-Containment Vessel Hebat Removal Systems

	Component	Malfunction	Comments and Consequences
12.	Air-operated valve at outlet penetration	Fails to open	Comments for Item 11 apply.
13.	Manual valve at combined ECCS room cooler/train one CAC service water outlet header	Fails closed	This is considered incredible due to the fact that the valve is a manual valve.
14.	Motor-operated valve at inlet penetration	Fails to close on LOOP/LOCA	If valve fails to close on a LOOP/LOCA, the effects from a water hammer transient will be intensified, however, analysis has shown that code allowables for the piping will not be exceeded and the structural integrity of the piping and the CAC coils will be maintained.
15.	Air-operated valve at outlet penetration	Fails to open on LOOP/LOCA	If valve fails to open a LOOP/LOCA, the effects from a water hammer transient will be intensified, however, analysis has shown that code allowables for the piping will not be exceeded and the structural integrity of the piping and the CAC coils will be maintained.

TABLE 6.2-22

Containment Shell External Surface Temperature
(Annulus Air Temperature 85°F)

Time (sec)	Temperature (°F) Dome	Temperature (°F) Cylinder
0	102.3	102.2
10	112.8	102.6
20	148.3	109.1
30	175.7	120.9
40	194.5	133.1
50	207.3	144.1
70	222.7	161.7
90	230.1	174.1
110	233.4	182.1
140	236.6	189.6
170	238.9	195.8
200	240.7	201.3
250	242.5	209.0
300	243.4	215.2
350	243.8	220.3
400	243.9	224.4
450	243.9	227.7
500	243.9	230.4
700	243.9	237.1
900	243.8	240.2
1100	243.8	241.7
3100	238.5	239.1
5100	227.2	232.5
7100	222.4	229.0
9100	222.7	227.6
11100	225.7	227.2
31100	189.5	198.3
51100	172.3	177.7
71100	162.4	165.6

TABLE 6.2-23

Containment Vessel Isolation Valve Arrangements

Pene-tration Number	Service	Flow Direc ion	Number Of Iso. Valves	Type	Vlv. Number	Signal (Note 1)	Normal Valve Position	CIS Position	Close Time (Note 2)	Type of Valve	Valve Operator	Valve Op. Power Source (Note 3)	Valve Failure Position	Line Size (Note 4)	Vlv. Arr. Figure
1	Pressurizer Sample Line	Out	2	I	RC240A	SA	Closed	Closed	30 sec.	Gate	Motor	MCCE11B	As is	1"	5.1-2
					RC240B	SA	Closed	Closed	30 sec.	Gate	Motor	MCCF11A	As is	1"	5.1-2
2	SG Secondary Water Sample Line	Out	1	III	SS607 (Note 7)	SA	Open	Closed	10 sec.	Globe	Air	----	Closed	3/4"	10.3-1
3	Component Cooling Water Inlet Line	In	3	II	CC1411A	SA	Open	Closed	15 sec.	Butterfly	Motor	MCCE11B	As is	10"	9.2-2
					CC1411B	SA	Open	Closed	15 sec.	Butterfly	Motor	MCCF11B	As is	10"	9.2-2
					CC1411C	---	---	---	---	Check	---	---	---	3/8”	9.2-2
4	Component Cooling Water Outlet Line	Out	3	II	CC1407A	SA	Open	Closed	15 sec.	Butterfly	Motor	MCCE11B	As is	10"	9.2-2
					CC1407B	SA	Open	Closed	15 sec.	Butterfly	Motor	MCCF11B	As is	10"	9.2-2
					CC1407C	---	---	---	---	Check	---	---	---	3/8”	9.2-2
5	Containment Air Cooling Unit SW Inlet Line	In	1	IV	SW1366 (Note 7)	Remote Manual (Note 9)	Open	Open	----	Ball	Motor	MCCE11C	As is	8"	9.2-1
6	Containment Air Cooling Unit SW Inlet Line	In	1	IV	SW1368 (Note 7)	Remote Manual (Note 9)	Closed	Closed	----	Ball	Motor	MCCE12A/ MCCF12A	As is	8"	9.2-1
7	Containment Air Cooling Unit SW Inlet Line	In	1	IV	SW1367 (Note 7)	Remote Manual (Note 9)	Open	Open	----	Ball	Motor	MCCF12A	As is	8"	9.2-1
8 A-J	Containment Vessel Vacuum Breakers	In	2	II	CV5070 thru CV5079	SA	Open	Closed	15 sec.	Butterfly	Motor	MCCE11C MCCF11A	As is	8"	9.4-11A
					CV5080 thru CV5089	----	----	----	----	Check	----	----	----	8"	9.4-11A
9	Containment Air Cooling Unit SW Outlet Line	Out	1	IV	SW1356 (Note 7)	Remote Manual (Note 9)	Open	Open	----	Ball	Air	----	Open	8"	9.2-1
10	Containment Air Cooling Unit SW Outlet Line	Out	1	IV	SW1358 (Note 7)	Remote Manual (Note 9)	Open	Open	----	Ball	Air	----	Open	8"	9.2-1
11	Containment Air Cooling Unit SW Outlet Line	Out	1	IV	SW1357 (Note 7)	Remote Manual (Note 9)	Open	Open	----	Ball	Air	----	Open	8"	9.2-1
12	Component Cooling Supply To CRDM	In	3	II	CC1567A	SA	Open	Closed	15 sec.	Gate	Motor	MCCE11B	As is	3"	9.2-2
					CC1567B	SA	Open	Closed	15 sec.	Gate	Motor	MCCF11B	As is	3"	9.2-2
					CC1568	---	---	---	---	Check	---	---	---	3/8”	9.2-2

TABLE 6.2-23 (Continued)

Containment Vessel Isolation Valve Arrangements

Pene-tration Number	Service	Flow Direction	Number Of Iso. Valves	Type	Vlv. Number	Signal (Note 1)	Normal Valve Position	CIS Position	Close Time) (Note 2)	Type of Valve	Valve Operator	Valve Op. Power Source (Note 3)	Valve Failure Position	Line Size (Note 4)	Vlv. Arr. Figure
13	Containment Vessel Normal Sump Drain	Out	3	II	DR2012A	SA	Open	Closed	15 sec.	Gate	Motor	MCCE11B	As is	4"	9.3-4
					DR2012B	SA	Open	Closed	15 sec	Gate	Motor	MCCF11B	As is	4"	9.3-4
					DR2012	---	---	---	---	Relief	---	---	---	¾"	9.3-4
14	Letdown Line To Purification Demineralizers	Out	2	I	MU2A	SA	Open	Closed	15 sec.	Gate	Motor	MCCE11B	As is	2 1/2"	9.3-16
					MU3	SA	Open	Closed	10 sec.	Gate	Air	----	Closed	2 1/2"	9.3-16
15	Spare														
16	Containment Vessel Equipment Vent Header	Out	2	II	RC1719A	SA	Open	Closed	10 sec.	Diaphragm	Air	----	Closed	3"	5.1-2
					RC1719B	SA	Open	Closed	10 sec.	Diaphragm	Air	----	Closed	3"	5.1-2
17	Containment Vessel Leak Test Inlet Line	In	1	II	CV343	Manual	Locked Closed	Locked Closed	----	Gate	Manual	----	----	8"	9.4-11A
					Flange	----	Flanged	Flanged	----	Flange	----	----	----	8"	9.4-11A
18	SG Secondary Water Sample Line	Out	1	III	SS598 (Note 7)	SA	Open	Closed	10 sec.	Globe	Air	----	Closed	3/4"	10.3-1
19	High Pressure Injection Line	In	2	IV	HP2A (Note 7)	SA	Closed	Open	----	Globe	Motor	MCCF11C	As is	2 1/2"	6.3-2
					HP57 (Note 7)	Manual	Locked Open	Locked Open	----	Stop Check	Manual	----	----	2 1/2"	6.3-2
20	High Pressure Injection and Normal Makeup Line	In	3	IV	HP2B (Note 7)	SA	Closed	Open	----	Globe	Motor	MCCF11C	As is	2 1/2"	6.3-2
					HP56 (Note 7)	Manual	Locked Open	Locked Open	----	Stop Check	Manual	----	----	2 1/2"	6.3-2
					MU6422	Remote Manual	Open	Open	----	Gate	Motor	MCCF11A	As is	2 1/2"	9.3-16
21	Demineralized Water Supply Line	In	2	II	DW6831A	SA	Open	Closed	10 sec.	Globe	Air	----	Closed	4"	9.2-4A
					DW6831B	SA	Open	Closed	10 sec.	Globe	Air	----	Closed	4"	9.2-4A
22	High Pressure Injection Line	In	2	IV	HP2D (Note 7)	SA	Closed	Open	----	Globe	Motor	MCCE11A	As is	2 1/2"	6.3-2
					HP49 (Note 7)	Manual	Locked Open	Locked Open	----	Stop Check	Manual	----	----	2 1/2"	6.3-2
23	Fuel Transfer Tube	In/out	0	II	Flanged	----	Flanged	Flanged	----	Flanged	----	----	----	30"	9.1-5;
24	Fuel Transfer Tube	In/Out	0	II	Flange	----	Flanged	Flanged	----	Flanged	----	----	----	30"	9.1-5

TABLE 6.2-23 (Continued)

Containment Vessel Isolation Valve Arrangements

Pene-tra ion Number	Service	Flow Direction	Number Of Iso. Valves	Type	Vlv. Number	Signal (Note 1)	Normal Valve Position	CIS Position	Close Time (Note 2)	Type of Valve	Valve Operator	Valve Op. Power Source (Note 3)	Valve Failure Position	Line Size (Note 4)	Vlv. Arr. Figure
25	Containment Spray Line	In	5	IV	CS1531	SA	Closed	Open	----	Globe	Motor	MCCF11B	As is	8”	6.3-1
					CS17 (Note 8)	Manual	Closed	Closed	----	Globe	Manual	----	----	8”	6.3-1
					C833 (Note 8)	Manual	Locked Closed	Locked Closed	----	Gate	Manual	----	----	8”	6.3-1
					SA532	Manual	Closed	Closed	----	Globe	Manual	----	----	2”	6.3-1
					SA536	Manual	Locked Closed	Locked Closed	----	Gate	Manual	----	----	2”	6.3-1
26	Containment Spray Line	In	5	IV	CS1530	SA	Closed	Open	----	Globe	Motor	MCCE11C	As is	8”	6.3-1
					CS18 (Note 8)	Manual	Closed	Closed	----	Globe	Manual	----	----	8”	6.3-1
					C836 (Note 8)	Manual	Locked Closed	Locked Closed	----	Gate	Manual	----	----	8”	6.3-1
					SA533	Manual	Closed	Closed	----	Globe	Manual	----	----	2”	6.3-1
					SA535	Manual	Locked Closed	Locked Closed	----	Gate	Manual	----	----	2”	6.3-1
27	Low Pressure Injection Line	In	2	IV	DH1A (Note 7)	Remote Manual	Locked Open	Locked Open	---- ----	Gate	Motor	MCCF11C	As is	10”	6.3-2A
					DH76 (Note 7)	Manual	Locked Open	Locked Open	----	Stop Check	Manual	----	----	10”	6.3-2A
28	Low Pressure Injection Line	In	2	IV	DH1B (Note 7)	Remote Manual	Locked Open	Locked Open	---- ----	Gate	Motor	MCCE11A	As is	10”	6.3-2A
					DH77 (Note 7)	Manual	Locked Open	Locked Open	----	Stop Check	Manual	----	----	10”	6.3-2A
29	Low Pressure Injection/Decay Heat Pump Suction	Out	3	IV	DH11 (Notes 7 & 8)	Remote Manual	Closed	Closed	----	Gate	Motor	MCCF11A	As is	12”	6.3-2A
					DH23 (Notes 7 & 8)	Manual	Locked Closed	Locked Closed	----	Gate	Manual	----	----	8”	6.3-2A
					PBV4849 (Notes 7 & 11)	----	----	----	----	PBV	----	----	----	4”	6.3-2A
30	Containment Vessel Emergency Sump Recirculation Line	Out	1	IV	DH9A (Note 7)	SA	Locked Closed	Closed (Note 6)	71 sec.	Gate	Motor	MCCF11D	As is	18”	6.3-2A
31	Containment Vessel Emergency Sump Recirculation Line	Out	1	IV	DH9B (Note 7)	SA	Locked Closed	Closed (Note 6)	71 sec.	Gate	Motor	MCCE11A	As is	18”	6.3-2A
32	Reactor Cooling System Drain Line to RC Drain Tank	Out	2	I	RC1773A	SA	Closed	Closed	10 sec.	Diaphragm	Air	----	Closed	3”	5.1-2
					RC1773B	SA	Closed	Closed	10 sec.	Diaphragm	Air	----	Closed	3”	5.1-2

TABLE 6.2-23 (Continued)

Containment Vessel Isolation Valve Arrangements

Pene-tration Number	Service	Flow Direction	Number Of Iso. Valves	Type	Vlv. Number	Signal (Note 1)	Normal Valve Position	CIS Position	Close Time (Note 2)	Type of Valve	Valve Operator	Valve Op. Power Source (Note 3)	Valve Failure Position	Line Size (Note 4)	Vlv. Arr. Figure
33	Containment Vessel	In	2	II	CV5006	SA (Note 13)	Closed	Closed	-	Butterfly	Air	-	Closed	48"	9.4-12
	Purge Inlet Line				CV5005	SA (Note 13)	Closed	Closed	-	Butterfly	Air	-	Closed	48"	9.4-12
34	Containment Vessel	Out	2	II	CV5007	SA (Note 13)	Closed	Closed	-	Butterfly	Air	-	Closed	48"	9.4-12
	Purge Outlet Line				CV5008	SA (Note 13)	Closed	Closed	-	Butterfly	Air	-	Closed	48"	9.4-12
35	Auxiliary Feedwater Line	In	1	III	AF599 (Note 7)	Remote Manual	Locked Open	Locked Open	-	Gate	Motor	MCCF11A	As is	6"	10.4-12A
36	Auxiliary Feedwater Line	In	1	III	AF608 (Note 7)	Remote Manual	Locked Open	Locked Open	-	Gate	Motor	MCCF11E	As is	6"	10.4-12A
	Emergency Feedwater Line	In	1	III	EF3 (Note 7)	Manual	Locked Closed	Locked Closed	-	Ball	Manual	-	As-is	3”	9.2-7
37	Main Feedwater Line	In	1	III	FW601 (Note 7)	Remote Manual	Open	Open	-	Gate	Motor	MCCF11D	As is	18"	10.4-12
38	Main Feedwater Line	In	1	III	FW612 (Note 7)	Remote Manual	Open	Open	-	Gate	Motor	MCCE11C	As is	18"	10.4-12
39	Main Steam Line	Out	15	III	MS100 (Note 7)	Remote Manual	Open	Open	-	Stop Check	Air	-	Closed	36"	10.3-1
					MS100-1 (Note 7)	Remote Manual	Closed	Closed	-	Globe	Air	-	Closed	2"	10.3-1
					MS375 (Note 7)	Remote Manual	Open	Closed	-	Globe	Air	-	Closed	1 1/2"	10.3-1
					ICS11A (Note 7)	Remote Manual	Closed	Closed	-	Angle	Air	-	Closed	8"	10.3-1
					MS107 (Notes 7 & 8)	Remote Manual	Closed	Closed	-	Gate	Motor	MCCF11A	As is	6"	10.4-12A
					MS106A (Notes 7 & 8)	Remote Manual	Open	Open	-	Gate	Motor	MCCE12B	As is	6"	10.4-12A
					PSVSP17A1 thru A9 (9) (Notes 7 & 10)	-	-	-	-	PSV	-	-	-	6"	10.3-1

TABLE 6.2-23 (Continued)

Containment Vessel Isolation Valve Arrangements

Pene-tration Number	Service	Flow Direction	Number Of Iso. Valves	Type	Vlv. Number	Signal (Note 1)	Normal Valve Position	CIS Position	Close Time (Note 2)	Type of Valve	Valve Operator	Valve Op. Power Source (Note 3)	Valve Failure Position	Line Size (Note 4)	Vlv. Arr. Figure
40	Main Steam Line	Out	15	III	MS101 (Note 7)	Remote Manual	Open	Open	----	Stop Check	Air	----	Closed	36"	10.3-1
					MS101-1 (Note 7)	Remote Manual	Closed	Closed	----	Globe	Air	----	Closed	2"	10.3-1
					MS394 (Note 7)	Remote Manual	Open	Closed	----	Globe	Air	----	Closed	1 1/2"	10.3-1
					ICS11B (Note 7)	Remote Manual	Closed	Closed	----	Angle	Air	----	Closed	8"	10.3-1
					MS106 (Notes 7 & 8)	Remote Manual	Closed	Closed	----	Gate	Motor	DINA	As is	6"	10.4-12A
					MS107A (Notes 7 & 8)	Remote Manual	Open	Open	----	Gate	Motor	MCCF11B	As is	6"	10.4-12A
					PSVSP17B1 thru B9 (9) (Notes 7 & 10)	----	----	----	----	PSV	----	----	----	6"	10.3-1
41	Pressurizer Quench Tank circulating Inlet Line	In	2	II	RC232	SA	Open	Closed	10 sec.	Globe	Air	----	Closed	2"	5.1-2
					RC113	----	----	----	----	Check	----	----	----	2"	5.1-2
42A	Service Air Supply Line	In	2	II	SA2010	SA	Closed	Closed	10 sec.	Globe	Air	----	Closed	1 1/2"	9.3-1
					SA502	----	----	----	----	Check	----	----	----	1 1/2"	9.3-1
42B	CV Air Sample Return	In	2	II	CV5010E	SA	Open	Closed	15 sec.	Diaphragm	Motor	MCCYF2	As is	1 1/2"	9.4-11A
					CV124	----	----	----	----	Check	----	----	----	1 1/2"	9.4-11A
43A	Instrument Air Supply Line	In	2	II	IA2011	SA	Open	Closed	10 sec.	Globe	Air	----	Closed	1"	9 3-1
					IA501	----	----	----	----	Check	----	----	----	1"	9.3-1
43B	CV Air Sample Return	In	2	II	CV5011E	SA	Open	Closed	15 sec.	Diaphragm	Motor	MCCYE2	As is	1 1/2"	9.4-11A
					CV125	----	----	----	----	Check	----	----	----	1 1/2"	9.4-11A
44A	CFT Fill And Nitrogen Supply Line	In	2	I	CF1541	SA	Closed	Closed	10 sec.	Globe	Air	----	Closed	1"	6.3-1A
					CF15	Manual	Open	Open	----	Stop Check	Manual	----	----	1"	6 3-1A
44B	Pressurizer Quench Tank Nitrogen Supply Line	In	2	II	NN236	SA	Open	Closed	10 sec.	Globe	Air	----	Closed	1"	7.3-9
					NN58	----	----	----	----	Check	----	----	----	1"	7 3-9
45	Spare														
46	Spare														

TABLE 6.2-23 (Continued)

Containment Vessel Isolation Valve Arrangements

Pene-tra ion Number	Service	Flow Direction	Number Of Iso. Valves	Type	Vlv. Number	Signal (Note 1)	Normal Valve Position	CIS Position	Close Time (Note 2)	Type of Valve	Valve Operator	Valve Op. Power Source (Note 3)	Valve Failure Position	Line Size (Note 4)	Vlv. Arr. Figure
47A	CFT Sample Line	Out	4	I	CF1545	SA	Closed	Closed	10 sec.	Globe	Air	----	Closed	1"	6.3-1A
					CF2A (Note 8)	Remote Manual	Locked Closed	Locked Closed	----	Globe	Motor	MCCF11A	As is	1"	6.3-1A
					CF2B (Note 8)	Remote Manual	Locked Closed	Locked Closed	----	Globe	Motor	MCCE11B	As is	1"	6.3-1A
					CF2C	---	---	---	---	Check	---	---	---	3/8”	6.3-1A
47B	CFT Vent Line	Out	3	I	CF1542	SA	Closed	Closed	10 sec.	Globe	Air	----	Closed	1"	6.3-1A
					CF5A (Note 8)	Remote Manual	Locked Closed	Locked Closed	----	Globe	Motor	MCCF11A	As is	1"	6.3-1A
					CF5B (Note 8)	Remote Manual	Locked Closed	Locked Closed	----	Glove	Motor	MCCE11B	As is	1"	6.3-1A
48	Pressurizer Quench Tank Circulating Outlet Line	Out	3	II	RC229A	SA	Open	Closed	10 sec.	Globe	Air	----	Closed	3"	5.1-2
					RC229B	SA	Open	Closed	10 sec.	Globe	Air	----	Closed	3"	5.1-2
					RC229C	---	---	---	---	Check	---	---	---	3/8”	5.1-2
49	Refueling Canal Fill Line	In/Out	2	II	DH88	Manual	Locked Closed	Locked Closed	----	Gate	Manual	----	----	8"	6.3-2A
					DH87	Manual	Locked Closed	Locked Closed	----	Gate	Manual	----	----	8"	6.3-2A
50	High Pressure Injection And Alternate Makeup Line	In	3	IV	HP2C (Note 7)	SA	Closed	Open	----	Globe	Motor	MCCE11A	As is	2 1/2"	6.3-2
					MU6421	Remote Manual	Closed	Closed	----	Gate	Motor	MCCE11D	As is	2 1/2"	9.3-16
					HP48 (Note 7)	Manual	Locked Open	Locked Open	----	Stop Check	Manual	----	----	2 1/2"	6.3-2
51	Hydrogen Purge System Exhaust	Out	2	II	CV5037	SA	Closed	Closed	60 sec.	Butterfly	Motor	MCCF11C	As is	4"	9.4-11
					CV5038	SA	Closed	Closed	60 sec.	Butterfly	Motor	MCCE11A	As is	4"	9.4-11
52	Reactor Coolant Pump Seal Water Supply	In	2	I	MU66A	SA	Open	Closed	12 sec.	Globe	Air	----	Closed	1 1/2"	9.3-16
					MU242	Manual	Open	Open	----	Stop Check	Manual	----	----	1 1/2"	9.3-16
53	Reactor Coolant Pump Seal Water Supply	In	2	I	MU66B	SA	Open	Closed	12 sec.	Globe	Air	----	Closed	1 1/2"	9.3-16
					Mu243	Manual	Open	Open	----	Stop Check	Manual	----	----	1 1/2"	9.3-16
54	Reactor Coolant Pump Seal Water Supply	In	2	I	MU66C	SA	Open	Closed	12 sec.	Globe	Air	----	Closed	1 1/2"	9.3-16
					MU244	Manual	Open	Open	----	Stop Check	Manual	----	----	1 1/2"	9.3-16
55	Reactor Coolant Pump Seal Water Supply	In	2	I	MU66D	SA	Open	Closed	12 sec.	Globe	Air	----	Closed	1 1/2"	9.3-16
					MU245	Manual	Open	Open	----	Stop Check	Manual	----	----	1 1/2"	9.3-16

TABLE 6.2-23 (Continued)

Containment Vessel Isolation Valve Arrangements

Pene-tra ion Number	Service	Flow Direction	Number Of Iso. Valves	Type	Vlv. Number	Signal (Note 1)	Normal Valve Position	CIS Position	Close Time (Note 2)	Type of Valve	Valve Operator	Valve Op. Power Source (Note 3)	Valve Failure Position	Line Size (Note 4)	Vlv. Arr. Figure
56	Reactor Coolant	Out	6	I	MU59A	SA	Open	Closed	30 sec.	Globe	Motor	MCCE11B	As is	1"	9.3-16
	Pump Seal Water				MU59B	SA	Open	Closed	30 sec.	Globe	Motor	MCCE11B	As is	1"	9.3-16
	Return				MU59C	SA	Open	Closed	30 sec.	Globe	Motor	MCCE11B	As is	1"	9.3-16
					MU59D	SA	Open	Closed	30 sec.	Globe	Motor	MCCE11B	As is	1"	9.3-16
					MU38	SA	Open	Closed	12 sec.	Globe	Air	----	Closed	1"	9.3-16
					MU60D	----	----	----	----	Relief	----	----	----	¼”	9.3-16
57	Steam Generator	Out	1	III	MS603	Remote	Closed	Closed	----	Gate	Motor	MCCF11A	As is	4”	10.3-1
	Blowdown Line				(Note 7)	Manual	(Note 12)	(Note 12)							
58	Spare														
59	Secondary Side	In	0	II	Flange	----	Flanged	Flanged	----	Flange	----	----	----	8”	
	Chemical Cleaning				Flange	----	Flanged	Flanged	----	Flange	----	----	----	8”	----
															----
60	Steam Generator	Out	1	III	MS611	Remote	Closed	Closed	----	Gate	Motor	MCCE12E	As is	4”	10.3-1
	Blowdown Line				(Note 7)	Manual	(Note 12)	(Note 12)							
61	Spare														
62	Spare														
63	Spare														
64	Spare														
65	Spare														
66	Spare														
67	Hydrogen Dilution	In	2	II	CV5090	SA	Closed	Closed	60 sec.	Butterfly	Motor	MCCE11A	As is	4”	9.4-11
	System Supply				CV210	----	----	----	----	Check	----	----	----	4”	9.4-11
68A	Pressurizer Quench	Out	2	II	SS235A	SA	Closed	Closed	30 sec.	Globe	Air	----	Closed	1”	9.3-3A
	Tank Sample Line				SS235B	SA	Closed	Closed	30 sec.	Globe	Air	----	Closed	1”	9.3-3A
68B	Containment	Out	2	II	CV5010B	SA	Open	Closed	15 sec.	Ball	Motor	MCCYF2	As is	1”	9.4-11A
	Air Sample				CV5011B	SA	Open	Closed	15 sec.	Ball	Motor	MCCYE2	As is	1”	9.4-11A
69	Hydrogen Dilution	In	2	II	CV5065	SA		Closed	60 sec.	Butterfly	Motor	MCCF11A	As is	4”	9.4-11
	System Supply				CV209	----	----	----	----	Check	----	----	----	4”	9.4-11
70	Spare														
71A	Containment	Out	1	IV	CV2000B	Remote	Open	Open	----	Gate	Motor	MCCF11A	As is	3/4”	9.4-11A
	Pressure Sensor				(Note 7)	Manual									

TABLE 6.2-23 (Continued)

Containment Vessel Isolation Valve Arrangements

Pene-tra ion Number	Service	Flow Direction	Number Of Iso. Valves	Type	Vlv. Number	Signal (Note 1)	Normal Valve Position	CIS Position	Close Time (Note 2)	Type of Valve	Valve Operator	Valve Op. Power Source (Note 3)	Valve Failure Position	Line Size (Note 4)	Vlv. Arr. Figure
71B	Containment Air Sample	Out	2	II	CV5010A CV5011A	SA SA	Open Open	Closed Closed	15 sec. 15 sec.	Ball Ball	Motor Motor	MCCYF2 MCCYE2	As is As is	1” 1”	9.4-11A 9.4-11A
71C	CFT Fill and Nitrogen Supply Line	In	2	I	CV1544 CF16	SA Manual	Closed Open	Closed Open	10 sec. ----	Globe Stop Check	Air Manual	---- ----	Closed ----	1” 1”	6.3-1A 6.3-1A
72A	Containment Pressure Sensor	Out	1	IV	CV2001B (Note 7)	Remote Manual	Open	Open	----	Gate	Motor	MCCE11A	As is	3/4”	9.4-11A
72B	Spare														
72C	Containment Pressure Differential Transmitter	Out	1	II	CV624B (Note 7)	Remote Manual	Open	Open	----	Gate	Motor	MCCF11B	As is	3/4”	9.4-11A
73A	Containment Pressure Sensor	Out	1	IV	CV2002B (Note 7)	Remote Manual	Open	Open	----	Gate	Motor	MCCF11A	As is	3/4”	9.4-11A
73B	Containment Air Sample	Out	2	II	CV5010C CV5011C	SA SA	Open Open	Closed Closed	15 sec. 15 sec.	Ball Ball	Motor Motor	MCCYF2 MCCYE2	As is As is	1” 1”	9.4-11A 9.4-11A
73C	Containment Pressure Differential Transmitter	Out	1	II	CV645B (Note 7)	Remote Manual	Open	Open	----	Gate	Motor	MCCE11E	As is	3/4”	9.4-11A
74A	Containment Pressure Sensor	Out	1	IV	CV2003B (Note 7)	Remote Manual	Open	Open	----	Gate	Motor	MCCE11A	As is	3/4”	9.4-11A
74B	Containment Air Sample	Out	2	II	CV5010D CV5011D	SA SA	Open Open	Closed Closed	15 sec. 15 sec.	Ball Ball	Motor Motor	MCCYF2 MCCYE2	As is As is	1” 1”	9.4-11A 9.4-11A
74C	Pressurizer Auxiliary Spray	In	2	I	DH2735 (Note 8) DH2736 (Note 8)	Remote Manual Remote Manual	Locked Closed Locked Closed	Locked Closed Locked Closed	---- ----	Gate Globe	Motor Motor	MCCE11B MCCF11A	As is As is	1 1/2” 1 1/2”	6.3-2A 6.3-2A
75	Spare														
76	Spare														
77	Spare														
78	Spare														

TABLE 6.2-23 (Continued)

Containment Vessel Isolation Valve Arrangements

Pene-tration Number	Service	Flow Direction	Number Of Iso. Valves	Type	Vlv. Number	Signal (Note 1)	Normal Valve Position	CIS Posi ion	Close Time (Note 2)	Type of Valve	Valve Operator	Valve Op. Power Source (Note 3)	Valve Failure Position	Line Size (Note 4)	Viv. Arr. Figure
79	Spare														
80	Emergency Lock		See Chapter 3 for description and arrangement												
81	Personnel Lock														
82	Equipment Hatch														
101, 102	Electrical Penetrations		See Chapter 8 for description and arrangement												

Note 1: SA signal denotes Safety Features Actuation Signal.

Note 2: No diesel start and sequence delays, or SA signal response times included.

Note 3: All operators are 480 VAC except penetrations:

a. 42B, 43B, 68B, 71B, 73B, 74B which are 120 VAC.

b. 40 - MS106 is 125 VDC.

Note 4: Line size is for the containment vessel penetration and the branch line size for penetrations with multiple branches.

Note 5: Deleted

Note 6: Valve is normally closed and will stay closed on containment isolation signal. When level drops in borated water storage tank, an alarm alerts control room operator to open valve.

Note 7: Not subject to 10CFR50, Appendix J, Type C Leakage Test.

Note 8: May be opened on an intermittent basis under administrative control.

Note 9: For Containment Air Cooling Unit Service Water Inlet/Outlet Valves operation, see USAR Section 9.2.1.3.

Note 10: Valve is included for completeness. It is not a containment isolation valve but is part of a pressure boundary on a closed loop piping system.

Note 11: Valve is defined as a containment isolation valve in accordance with GDC 55, as defined by Note 55-3 in ANS56.2.

Note 12: During startup, shutdown and at low power levels, these valves may be left open, see USAR Section 10.4.8.2

Note 13: In Modes 1 through 4 SFAS is a confirmatory signal only (per Amendment 221 to the Technical Specifications, the containment purge isolation valves are administratively maintained closed and control power removed in Modes 1 through 4). Remote manual closure is required when necessitated by the Refueling Operations section of the Technical Specifications.

TABLE 6.2-25

Containment Vessel Isolation System Number 1

Valve Number	Valve Description	Trip Input* CV Pressure or RC Pressure
HV MU2A	RC Letdown Outlet Valve	
HV 2012A	Containment Normal Sump Isolation Valve	
HV 240A	RC Pressurizer Sample Valve	
HV 1399	Service Isolation Valve To Cooling Water	
HV 1773A	RC Drain Tank Header Isolation Valve	
HV 1719A	Containment Vent Header Isolation Valve	
HV 607	Steam Generator (SG) 1 Sample Isolation Valve	
HV 235A	Pressurizer Quench Tank Sample Isolation Valve	
HV 1544	Core Flooding Tank 1 H2&N2 Fill Isolation Valve	
HV MU3	RC Letdown High Temperature Valve	
HV 2012B	Containment Normal Sump Isolation Valve	
HV 240B	RC Pressurizer Vapor Sample Valve	
HV 1542	Core Flooding Tank Vent Isolation Valve	
HV 1395	Service Isolation Valve To Cooling Water	
HV 1773B	RC Drain Tank Header Isolation Valve	
HV 1719B	Containment Vent Header Isolation Valve	
HV 598	Steam Generator (SG) 2 Sample Isolation Valve	
HV 235B	Pressurizer Quench Tank Sample Isolation Valve	
HV 1541	Core Flooding Tank 2 H2&N2 Fill Isolation Valve	
HV DH9B	Containment Emergency Sump Valve	
HV DH7B	BWST Outlet Valve	
HV 236	N2 Containment Isolation Valve	
HV 229A	Pressurizer Quench Tank Outlet Isolation Valve	
HV 232	Pressurizer Quench Tank In Isolation Valve	
HV 229B	Pressurizer Quench Tank Outlet Isolation Valve	
HV 1545	Core Flooding Tank Sample Valve	
HV DH9A	Containment Emergency Sump Valve	

* See Table 7.3-3 for Setpoint

TABLE 6.2-25 (Continued)

Containment Vessel Isolation System Number 1

Valve Number	Valve Description	Trip Input* CV Pressure or RC Pressure
HV DH7A	BWST Outlet Valve	
HV 2011	Containment Instrument Air Isolation Valve	
HV 2010	Containment Service Air Isolation Valve	
HV 5065	Containment Hydrogen Dilution In Isolation Valve	
HV 6831A	Demin Water Isolation Valve	
HV 5038	Containment Hydrogen Dilution Outlet Isolation Valve	
HV 5090	Containment Hydrogen Dilution In Isolation Valve	
HV 6831B	Demin Water Isolation Valve	
HV 5037	Containment Hydrogen Dilution Outlet Isolation Valve	
HV 5070	Containment Vacuum Relief Isolation Valve	
HV 5071	Containment Vacuum Relief Isolation Valve	
HV 5072	Containment Vacuum Relief Isolation Valve	
HV 5073	Containment Vacuum Relief Isolation Valve	
HV 5074	Containment Vacuum Relief Isolation Valve	
HV 5075	Containment Vacuum Relief Isolation Valve	
HV 5076	Containment Vacuum Relief Isolation Valve	
HV 5077	Containment Vacuum Relief Isolation Valve	
HV 5078	Containment Vacuum Relief Isolation Valve	
HV 5079	Containment Vacuum Relief Isolation Valve	

* See Table 7.3-3 for Setpoint

TABLE 6.2-26

Containment Vessel Isolation System Number 2

Valve Number	Valve Description	Trip Input* CV Pressure or RC Pressure
HV 1460	Component Cooling (CC) to Makeup Pump Header Inlet Valve	
HV 1495	CC Auxiliary Equipment Inlet Valve	
HV MU59A	RCP 2-1 Seal Return Valve	
HV MU59B	RCP 2-2 Seal Return Valve	
HV MU59C	RCP 1-1 Seal Return Valve	
HV MU59D	RCP 1-2 Seal Return Valve	
HV MU66B	RCP 2-2 Seal In Isolation Valve	
HV MU66C	RCP 1-1 Seal In Isolation Valve	
HV MU66A	RCP 2-1 Seal In Isolation Valve	
HV MU38	RCP Seal Return Isolation Valve	
HV MU66D	RCP 1-2 Seal In Isolation Valve	

*See Table 7.3-3 for Setpoints

TABLE 6.2-27

Containment Vessel Isolation System Number 3

Valve Number	Valve Description	Trip Input* CV Pressure
HV 1411A	Component Cooling (CC) Inlet Isolation Valve To Containment	
HV 1407A	CC Outlet Isolation Valve From Containment	
HV 1567A	CC Inlet Isolation Valve To CRD	
HV 1328	CC CRD Booster Pump 1 Suction Valve	
HV 1411B	CC Inlet Isolation Valve To Containment	
HV 1407B	CC Outlet Isolation Valve From Containment	
HV 1567B	CC Inlet Isolation Valve To CRD	
HV 1338	CC CRD Booster Pump 2 Suction Valve	

*See Table 7.3-3 for Setpoint

TABLE 6.2-28

Containment Vessel Penetration Termination

Penetration Number	Service	Containment Vessel	Annulus	Penetration and ECCS Rooms	EVS Boundary	Filtration Bypass Postulated	Notes
1.	Pressurizer sample line	—	—	—	→	X	
2.	Steam generator secondary water sample line	—	—	—	→		(a)
3.	Component cooling water inlet line	—	—	—	→		(b)
4.	Component cooling water outlet line	—	—	—	→	X	(n)
5, 6, 7.	Containment air cooling units service water inlet lines	—	—	—	→		(c)
8A-J.	Containment Vessel vacuum breakers.	—	→				
9, 10, 11.	Containment air cooling units service water outlet lines	—	—	—	→		(c)
12.	Component cooling water to control rod drives	—	—	→			
13.	Containment Vessel normal sump drain	—	—	—	→	X	
14.	Letdown line to purification demineralizers	—	—	—	→	X	
15.	Spare	—	→				
16.	Containment Vessel equipment vent header	—	—	—	→	X	
17.	Containment Vessel leak test line	—	—	—	→	X	(k)

TABLE 6.2-28 (Continued)

Containment Vessel Penetration Termination

Penetration Number	Service	Containment Vessel	Annulus	Penetration and ECCS Rooms	EVS Boundary	Filtration Bypass Postulated	Notes
18.	Steam generator secondary water sample line	—	—	—	→		(a)
19, 20.	High-pressure injection lines	—	—	→			(d)
21.	Demineralized water supply line	—	—	—	→	X	
22.	High-pressure injection line	—	—	→			(d)
23, 24.	Fuel transfer tubes	—	—	—	→	X	(e)
25, 26.	Containment spray lines	—	—	→			(d)
27, 28.	Low-pressure injection lines	—	—	→			(d)
29.	Low pressure injection/Decay heat pump suction	—	—	→			(f)
30, 31.	Containment Vessel emergency sump recirculation lines	—	—	→			(d)
32.	Reactor coolant system drain line to R.C. drain tank	—	—	—	→	X	(l)
33, 34.	Containment Vessel purge inlet and outlet lines	—	—	—	→		(g)
35, 36.	Auxiliary feedwater lines	—	—	—	→		(a)
37, 38.	Main feedwater lines	—	—	—	→		(a)
39, 40.	Main steam lines	—	—	—	→		(a)

TABLE 6.2-28 (Continued)

Containment Vessel Penetration Termination

Penetration Number	Service	Containment Vessel	Annulus	Penetration and ECCS Rooms	EVS Boundary	Filtration Bypass Postulated	Notes
41.	Pressurizer quench tank circulating inlet line	—	—	—	→	X	(l)
42A.	Service air supply line	—	—	—	→	X	
42B.	Containment Vessel air sample returns	—	—	—	→	X	(m)
43A.	Instrument air supply line	—	—	—	→	X	
43B.	Containment Vessel air sample returns	—	—	→			
44A.	Core Flooding Tank fill and N_2 supply line	—	—	—	→		(j)
44B.	Pressurizer quench tank N_2 supply line	—	—	—	→	X	
45, 46.	Spares	—	→				
47A.	Core Flooding Tank sample line	—	—	—	→		(j)
47B.	Core Flooding Tank vent line	—	—	—	→		(j)
48.	Pressurizer quench tank circulating outlet line	—	—	—	→	X	(l)
49.	Refueling canal fill line	—	—	—	→	X	

TABLE 6.2-28 (Continued)

Containment Vessel Penetration Termination

Penetration Number	Service	Containment Vessel	Annulus	Penetration and ECCS Rooms	EVS Boundary	Filtration Bypass Postulated	Notes
50.	High-pressure injection line	—	—	→			(d)
51.	Hydrogen purge system exhaust	—	—	—	→		(h)
52,53,54, 55	Reactor coolant pump seal water supply	—	—	—	→	X	
56.	Reactor coolant pump seal water return	—	—	—	→	X	
57.	Steam generator blowdown line	—	—	—	→		(a)
58.	Spare	—	→				
59.	Secondary side chemical cleaning line	—	—	→			
60.	Steam generator blowdown line	—	—	—	→		(a)
61,62,63, 64,65,66	Spares	—	→				
67.	Hydrogen dilution system supply line	—	—	—	→	X	
68A.	Pressurizer quench tank sample line	—	—	—	→	X	
68B.	Containment air sample line	—	→				
69.	Hydrogen dilution supply line	—	—	—	→	X	
70.	Spare	—	→				
71A.	Containment pressure sensor line	—	—	—	→		(i)
71B.	Containment air sample line	—	—	—	→	X	(m)
71C.	Core Flooding Tank fill and N_2 supply line	—	—	—	→	X	(j)

TABLE 6.2-28 (Continued)

Containment Vessel Penetration Termination

Penetration Number	Service	Containment Vessel	Annulus	Penetration and ECCS Rooms	EVS Boundary	Filtration Bypass Postulated	Notes
72A.	Containment pressure sensor line	—	—	—	→		(i)
72B.	Spare	—	—	→			
72C.	Pressure differential transmitter line	—	—	—	→		(i)
73A.	Containment pressure sensor line	—	—	—	→		(i)
73B.	Containment air sample line	—	—	—	→	X	(m)
73C.	Pressure differential transmitter line	—	—	—	→		(i)
74A.	Containment pressure sensor line	—	—	—	→		(i)
74B.	Containment all sample line	—	—	→			
74C.	Pressurizer auxiliary spray line	—	—	→		X	
75,76,77, 78,79	Spares	—	→				
80.	Emergency lock	—	—	—	→	X	
81.	Personnel lock	—	—	—	→	X	
82.	Equipment hatch	—	—	—	→	X	
101, 102.	Electrical penetrations	—	—	—	→	X	

TABLE 6.2-28 (Continued)

Containment Vessel Penetration Termination

NOTES

a. Penetrations 2, 18, 35, 36, 37, 38, 39, 40, 57, and 60 are lines off the secondary side inside the containment. The secondary side is closed and Seismic Class I inside the containment.

b. Penetration 3 supplies component cooling water to the containment. The supply line is seismic class I outside of containment and the component cooling water system discharge pressure exceeds the analyzed post-LOCA containment pressure as well as the containment design pressure.

c. Penetrations 5, 6, 7, 9, 10, and 11 are lines associated with the service water for the containment air coolers. These lines are Seismic Class I and are used for post-LOCA cooling of the containment.

d. Penetrations 19, 20, 22, 25, 26, 27, 28, 30, 31, and 50 are the lines serving the emergency core cooling and containment spray systems. These lines are Seismic Class I outside the containment.

e. Penetrations 23 and 24 are the fuel transfer tubes and are fitted with a double gasketed blind flange inside the containment.

f. Penetration 29 is the Low Pressure Injection/Decay Heat Removal Pump suction line, and this line connects (through closed isolation valves) to the lines serving the ECCS.

g. Penetration 33 and 34 are provided with one 1/4-inch normally open gate valve (to the annulus) between two normally closed isolation valves. This gate valve vents the leakage to the annulus, thus eliminating the potential through-line leakage from bypassing the EVS.

h. Penetration 51 is the line serving the hydrogen dilution system exhaust. This line is Seismic Class I and passes through a charcoal filter, and therefore any leakage will be filtered.

i. Penetrations 71A, 72A, 72C, 73A, 73C, and 74A are lines serving the containment pressure sensors and the containment pressure differential sensors. These lines are Seismic Class I and terminate with a closed instrument.

j. Penetrations 44A, 47A, 47B, and 71C connect to the Core Flooding Tanks which connect to the Reactor Coolant System downstream of the low pressure injection discharge.

k. Penetration 17, Containment Vessel Leak Test Line, is Seismic Class I from CV343 to the blind flange located in containment. However, the piping upstream of CV343 to the blind flange located outside the Auxiliary Building, is not Seismic I and the piping is not contained within an area served by EVS. Therefore, this penetration shall be considered a Bypass Penetration.

TABLE 6.2-28 (Continued)

Containment Vessel Penetration Termination

l. Penetration 32, 41 and 48 are systems open inside the containment vessel atmosphere. The majority of the piping is Seismic Class I; however, upstream of RC232 and downstream of RC229A and RC1773B to the RCDT, it is neither seismic I nor is the piping contained within an area served by EVS. Therefore, these penetrations shall be considered Bypass Penetrations.

m. Penetrations 42B, 71B, and 73B are systems open inside the containment vessel atmosphere. The majority of the piping is Seismic Class I; however, upstream of CV5010E, and downstream of CV5011A and CV5011C to the Post Accident Gas sample system, located in the auxiliary building train bay, it is neither seismic I nor is the piping contained within an area served by EVS. Therefore, these penetrations shall be considered Bypass Penetrations.

n. Penetration 4, CCW containment return line, is considered a bypass penetration because peak containment post-LOCA pressure exceeds the CCW return header pressure. Therefore, leakage through the containment penetration isolation valves has the potential to migrate to areas not served by EVS.

TABLE 6.2-29

Significant Containment Metals and Paints Subject to Corrosion by Spray Solutions

Item	Component	Exposed Surface Area (ft^2)	Corrosion Rate (mils / year)
Zinc	Conduit Cable Trays, etc.	13,403	Variable
Zinc	Ductwork	19,276	Variable
Zinc	Galvanized Grating & Painted Surfaces	288,319	Variable
Zinc	Scaffolding and fittings	46,000	Variable
Zinc	Head Stand Access Tower	< 1,000	Variable
Aluminum	Control Rod Drive Mechanisms	400	200
Aluminum	Incore Instrument Handling Reel	134	200
Aluminum	Reactor Core Cover	690	200

NOTE: Storage of additional components in containment is controlled by revisions to reference 32 to ensure that the hydrogen generation rates as reflected by figures 6.2-45 through 6.2-50 are not exceeded.

TABLE 6.2-30

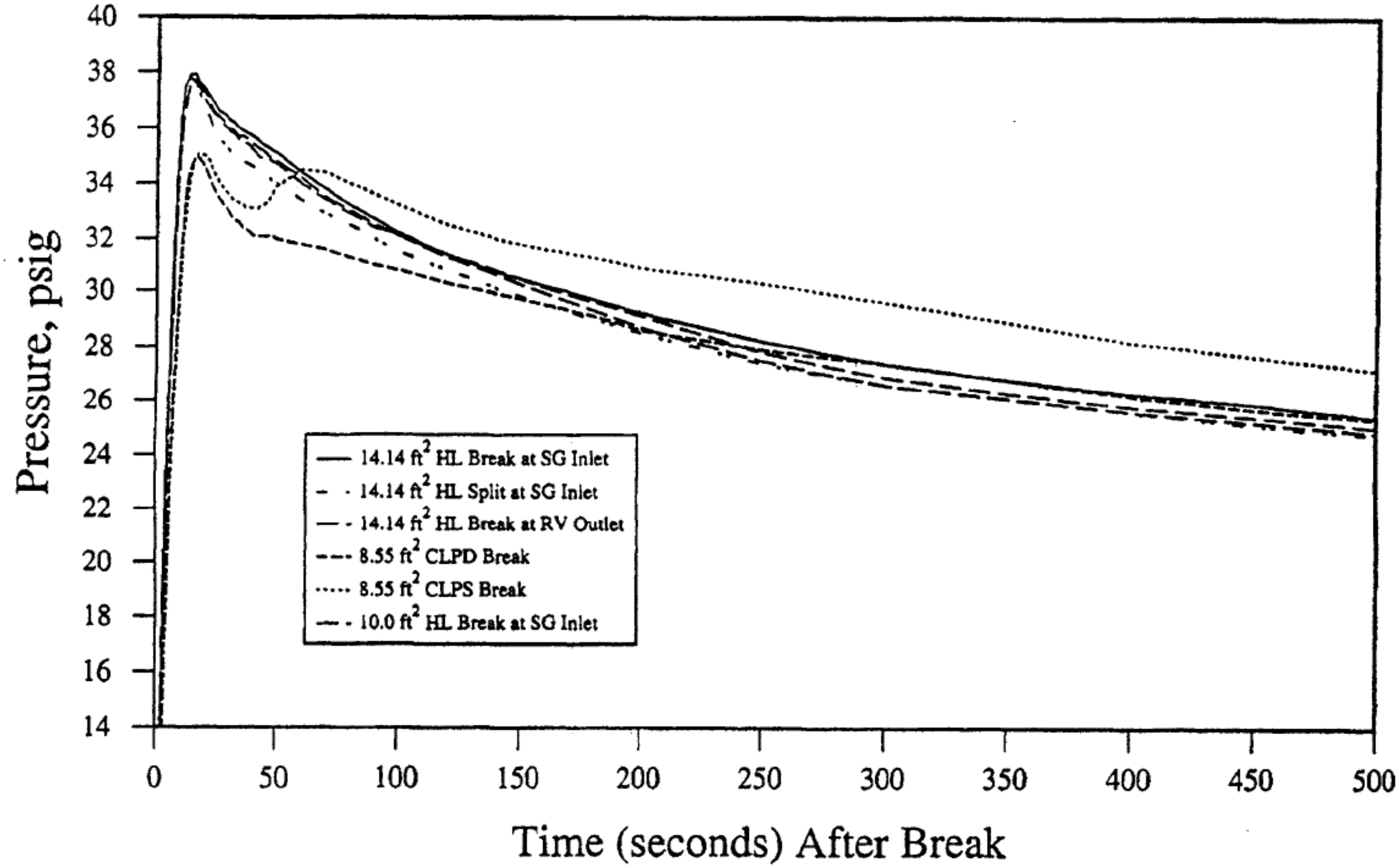
Hydrogen Generation Rates and Corrosion Rates for Galvanized Steel and Zinc Based Paints

Time After LOCA (days)	Average Temperature (°F)	H_2 Generation Galvanized Steel (lb-moles/ ft^2-hr)	H_2 Generation Zinc Based Paints (lb-moles/ ft^2-hr)	Zinc Corrosion Rates Galvanized Steel (mils/yr)	Zinc Corrosion Rates Zinc Base Paints (mils/yr)
0-1	170	7.938 x 10-7	7.938 x 10-7	12.24	12.24
1-96	150	2.54 x 10-7	2.54 x 10-7	3.92	3.92

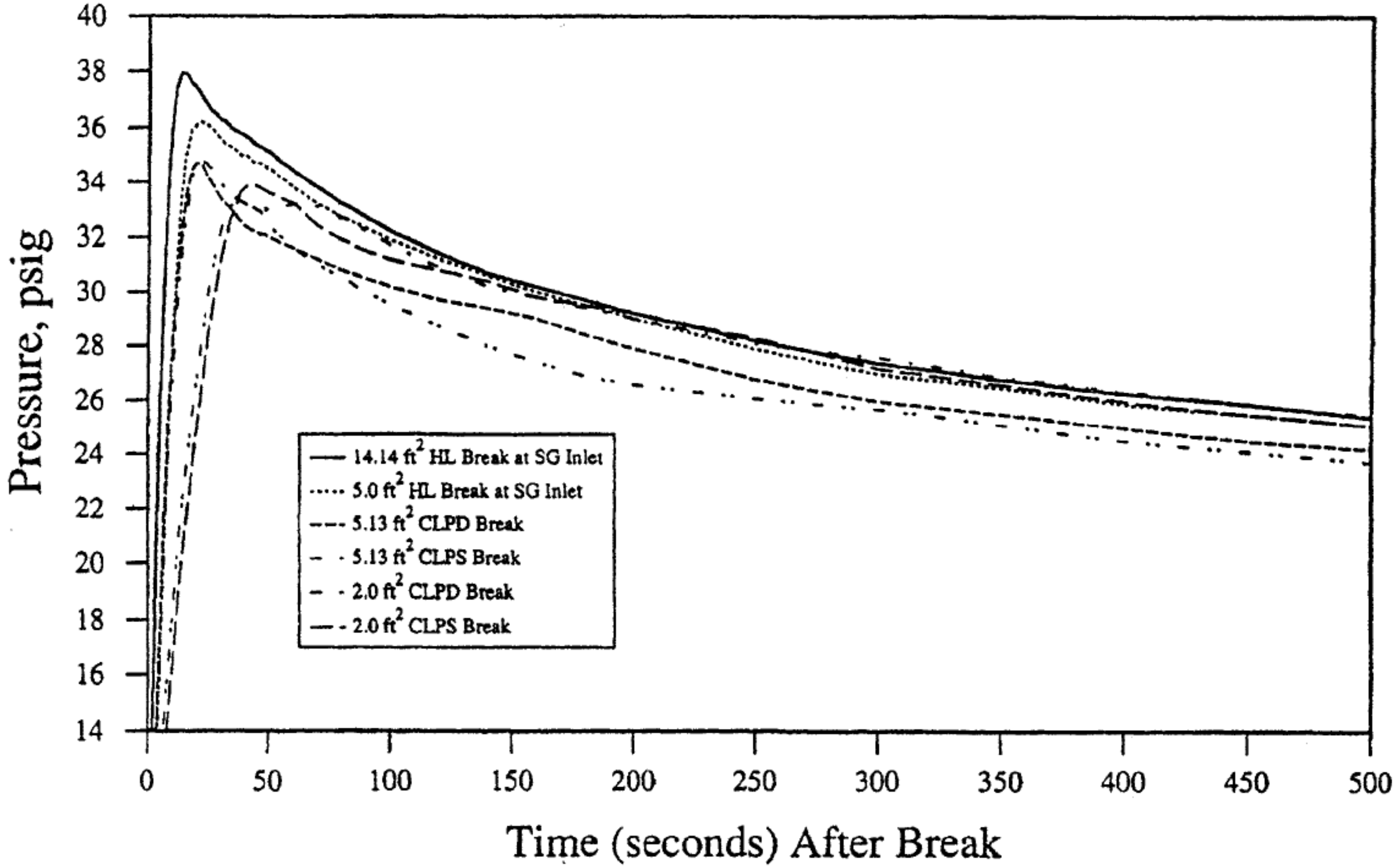
TABLE 6.2-31

Paints Used Inside Containment Vessel

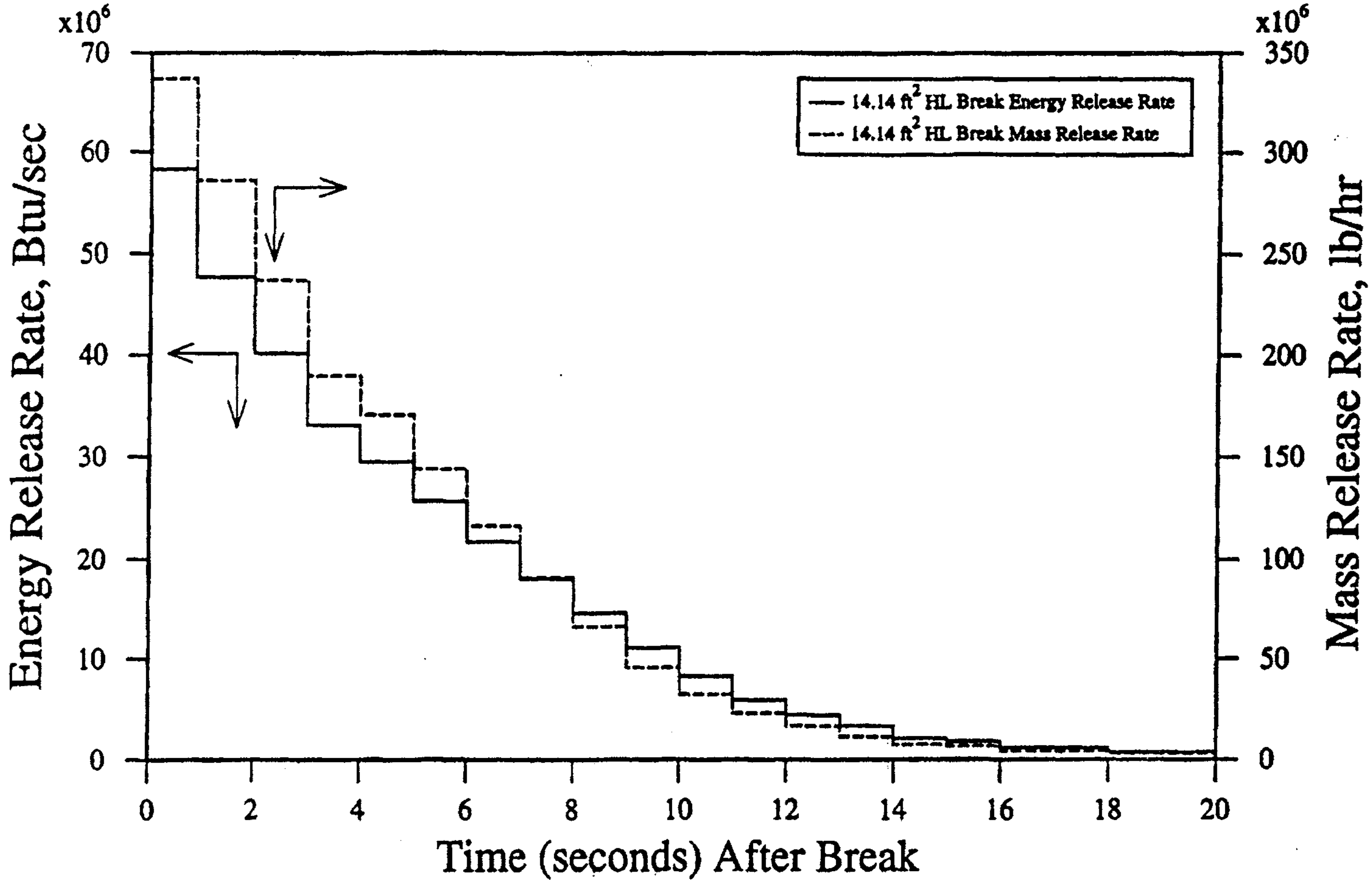
GENERIC TYPE	SUBSTRATE	MANUFACTURER'S DESIGNATION	DRY SP. GR.	AREA Ft^2	CURING
Epoxy Zinc Chromate Primer	Steel	Dupont Corlar 825-8031	1.5	1000	Polyamide Catalyzed Air Drying Self curing
Organic Zinc Primer	Steel	Durazinc K-815 or Z.R.C.	2.67 2.9	1810	Self Curing Air Drying
Inorganic Zinc Primer	Steel	Carbo-Zinc 11 or Dimetocote 6 or Mobilzinc 7	3.66	228,000	Self Curing Air Drying
Epoxy Surfacer	Concrete	Amercoat Naklad 110AA	2.24	108,000	Catalyzed Air Drying
Epoxy Topcoat	Concrete and/or Primed Steel	Phenoline 305 or Amercoat 66 or Mobil 89 Series	1.44 1.44	224,000 108,000	Catalyzed Air Drying
Modified Phenolic Coatings	Unprimed Steel	Phenoline 368	1.44	3,000	Catalyzed Air Drying
Aluminum Heat Resisting (1200°F)	Unprimed Steel	Red Spot-Federal Regulation TT-P-28d (April 26, 1967)	1.56	20,000	Air Drying Curing at 400°F Full Hardness



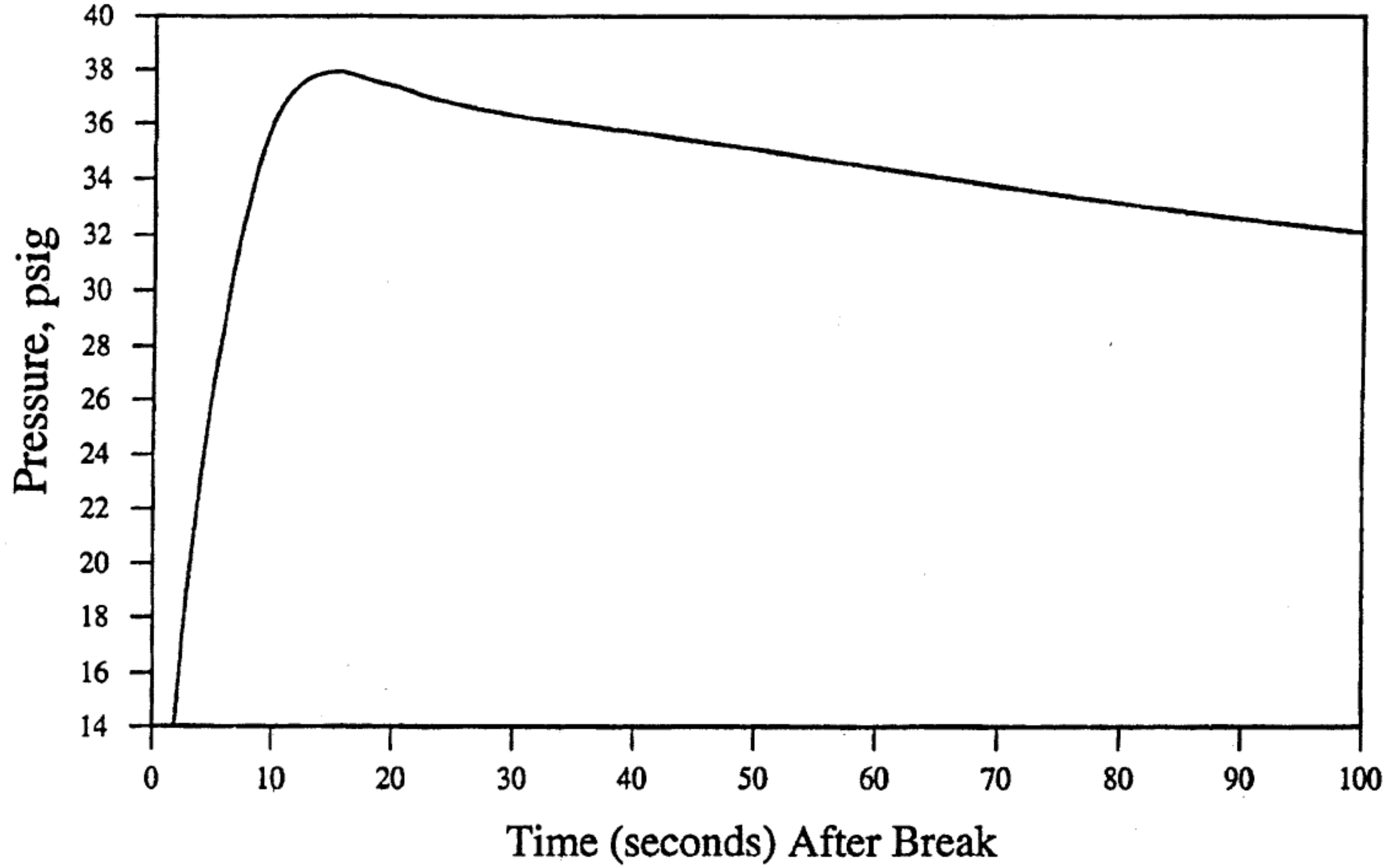
DAVIS-BESSE NUCLEAR POWER STATION
BREAK SPECTRUM PRESSURE RESPONSES
FIGURE 6.2-3a



DAVIS-BESSE NUCLEAR POWER STATION
BREAK SPECTRUM PRESSURE RESPONSES
FIGURE 6.2-3b

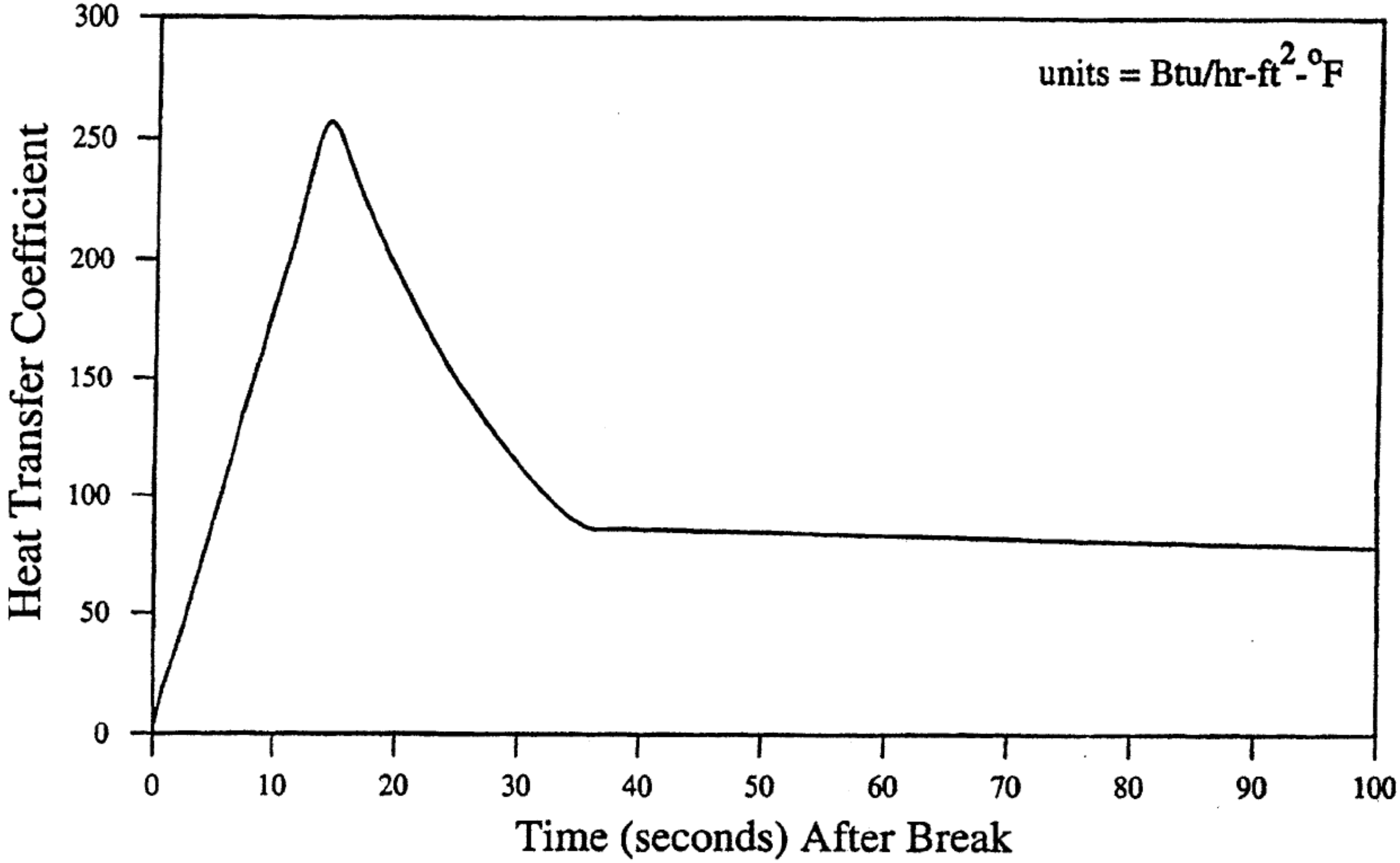


DAVIS-BESSE NUCLEAR POWER STATION
BLOWDOWN MASS AND ENERGY RELEASE RATES
14.14 FT^2 DBA BREAK
FIGURE 6.2-7



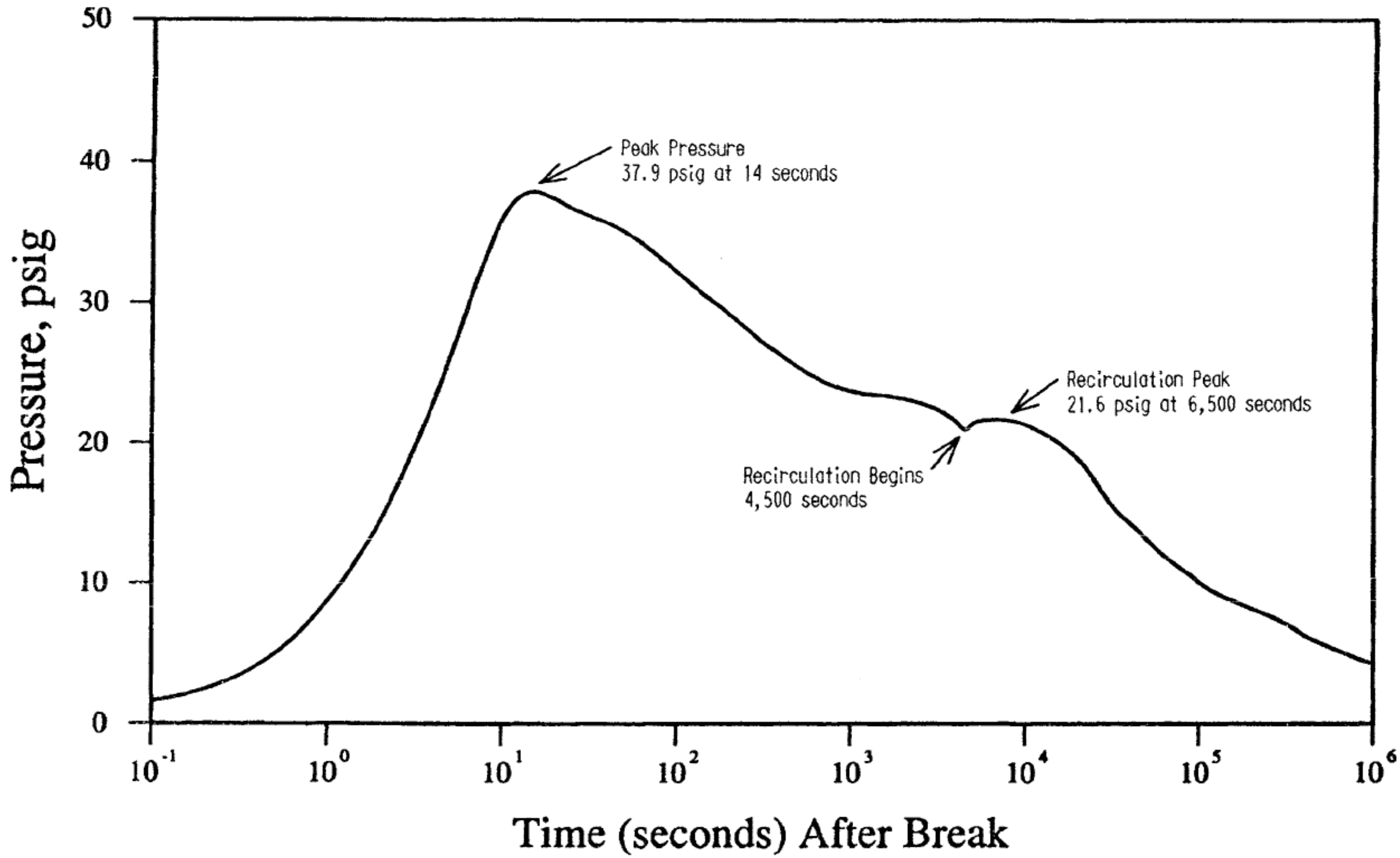
DAVIS-BESSE NUCLEAR POWER STATION
SHORT TERM CONTAINMENT PRESSURE RESPONSE
14.14 FT^2 DBA BREAK
FIGURE 6.2-8

REVISION 24
JUNE 2004



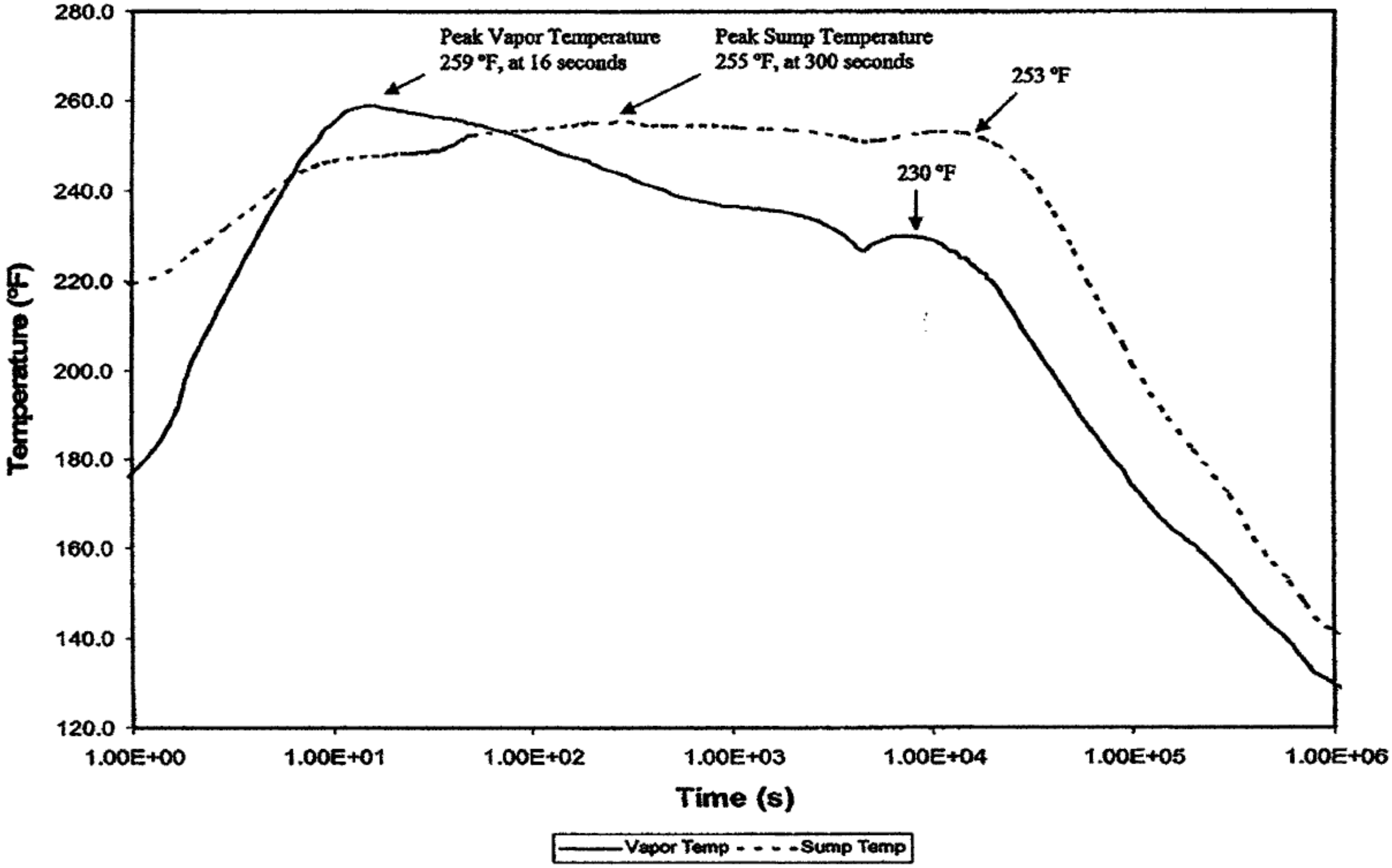
REVISION 24
JUNE 2004

DAVIS-BESSE NUCLEAR POWER STATION
HEAT TRANSFER COEFFICIENT BETWEEN CONTAINMENT
ATMOSPHERE AND HEAT SINKS
14.14 FT^2 DBA BREAK
FIGURE 6.2-9



DAVIS-BESSE NUCLEAR POWER STATION
LONG TERM CONTAINMENT PRESSURE RESPONSE
14.14 FT^2 DBA BREAK
FIGURE 6.2-10

REVISION 25
JUNE 2006

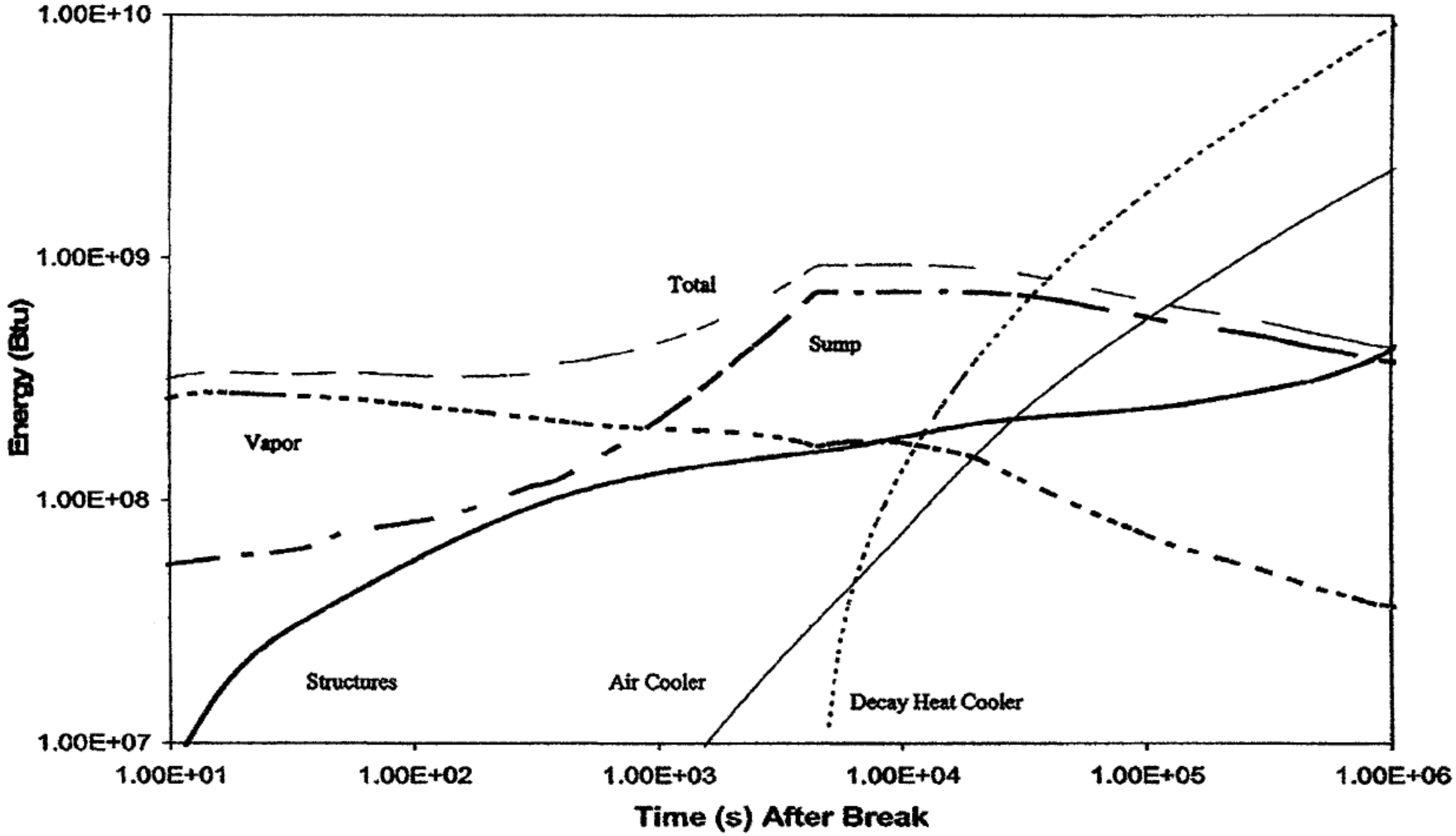


DAVIS-BESSE NUCLEAR POWER STATION
LONG TERM CONTAINMENT TEMPERATURE RESPONSE
14.14 FT^2 DBA BREAK
FIGURE 6.2-11

REVISION 26
JUNE 2008

DB 03-11-08

DFN: G/USAR/UFIG6211.DGN/TIF

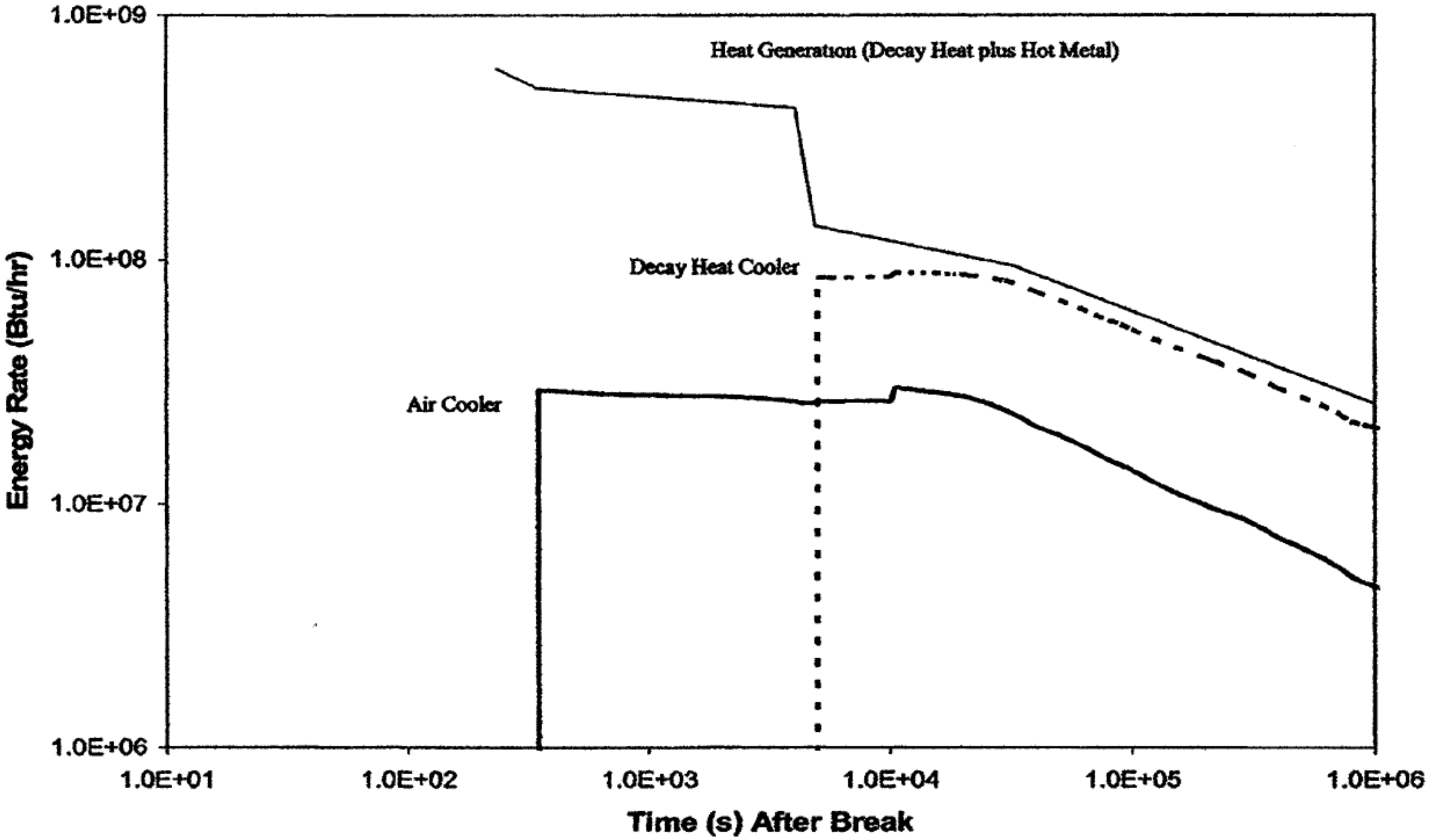


DAVIS-BESSE NUCLEAR POWER STATION
ENERGY DISTRIBUTION
14.14 FT ^{2} DBA BREAK
FIGURE 6.2-12

REVISION 26
JUNE 2008

DB 03-11-08

DFN: G/USAR/UFIG6212.DGN/TIF

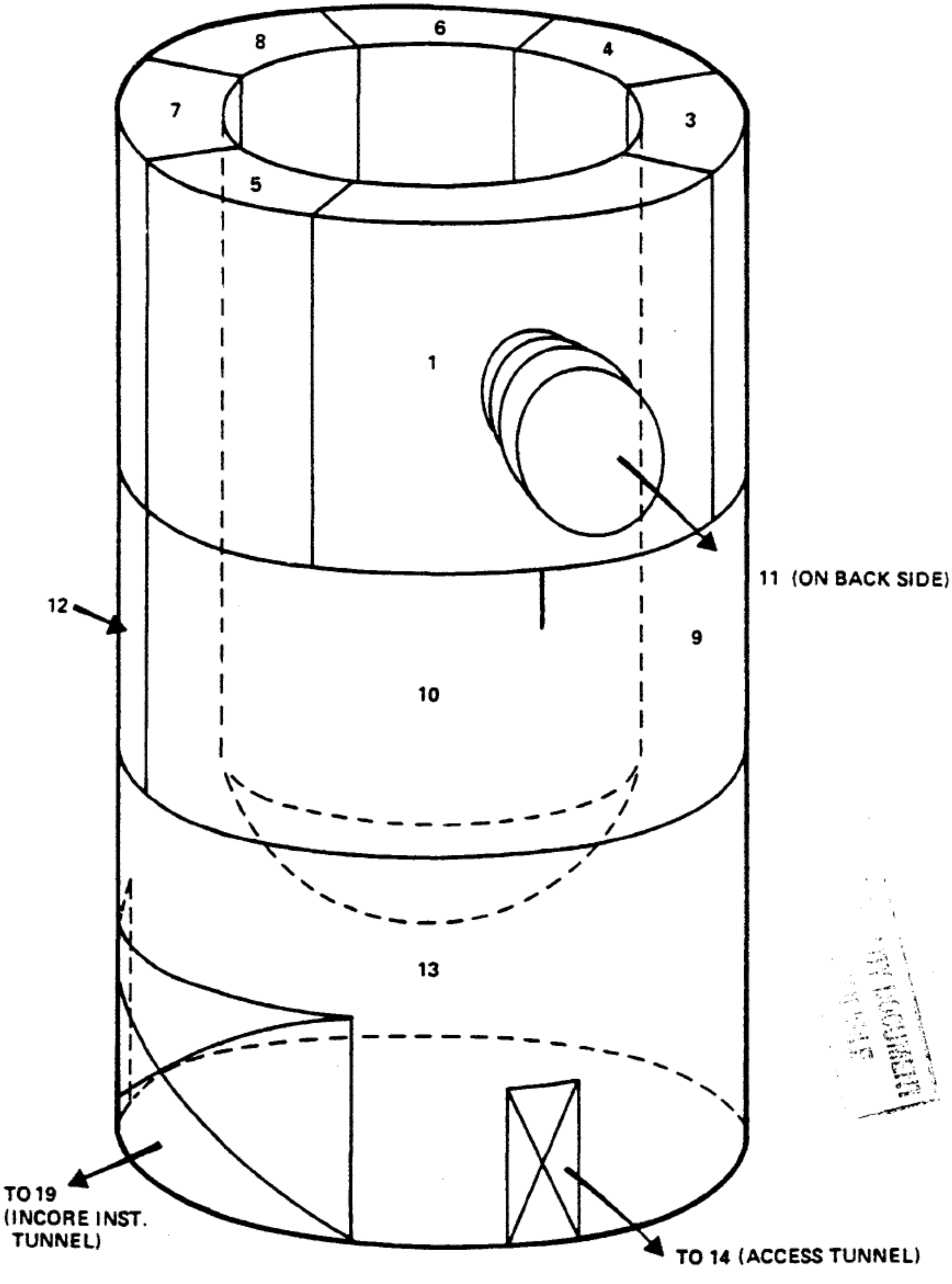


DAVIS-BESSE NUCLEAR POWER STATION
ENERGY GENERATION AND REMOVAL RATES
14.14 FT^2 DBA BREAK
FIGURE 6.2-13

REVISION 26
JUNE 2008

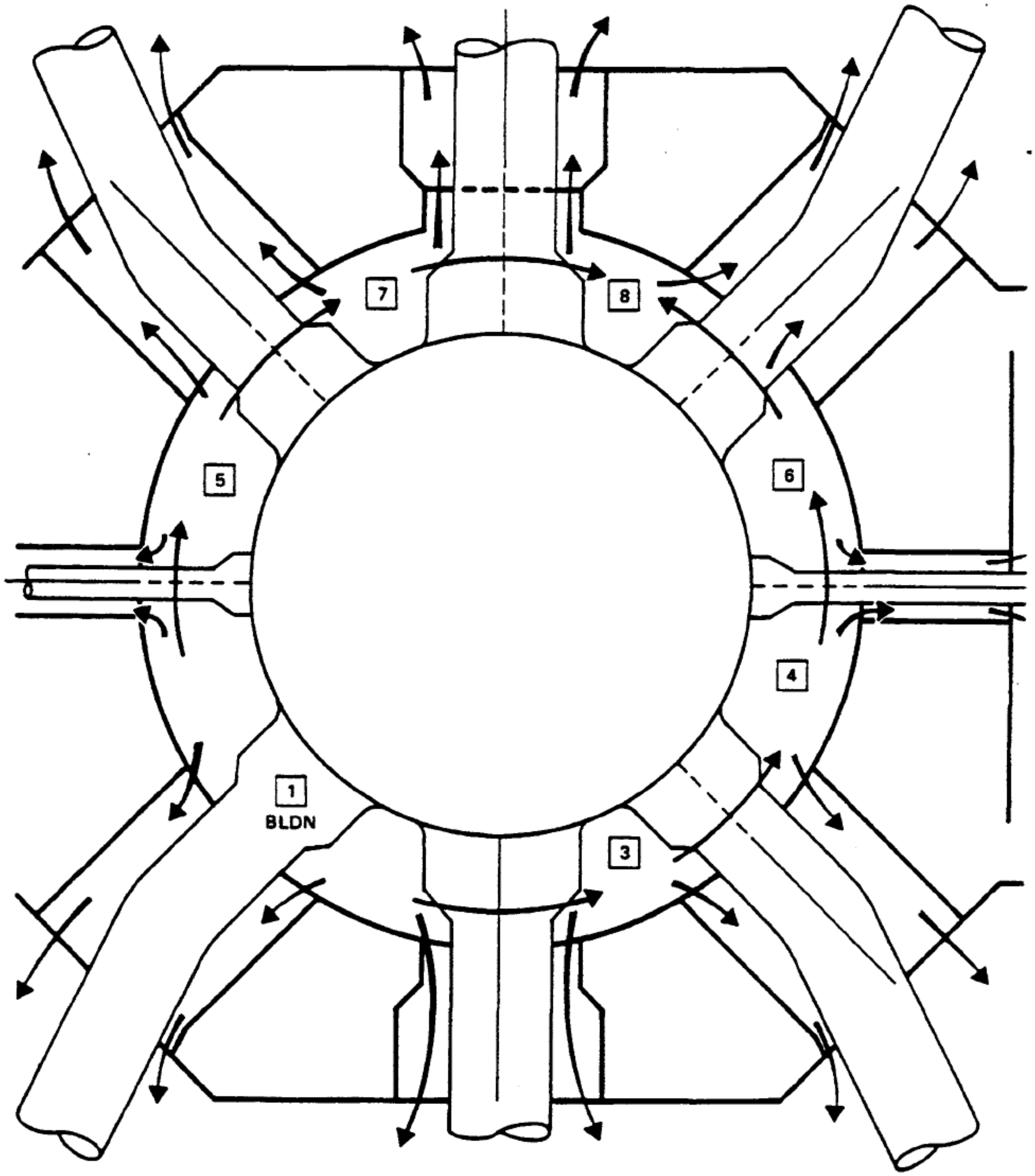
DB 03-11-08

DFN: G/USAR/UFIG6213.DGN/TIF



DAVIS-BESSE NUCLEAR POWER STATION
REACTOR CAVITY FLOW DIAGRAM
(SIDE VIEW)
FIGURE 6.2-14

REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
REACTOR CAVITY FLOW DIAGRAM
(TOP VIEW)
FIGURE 6.2-15

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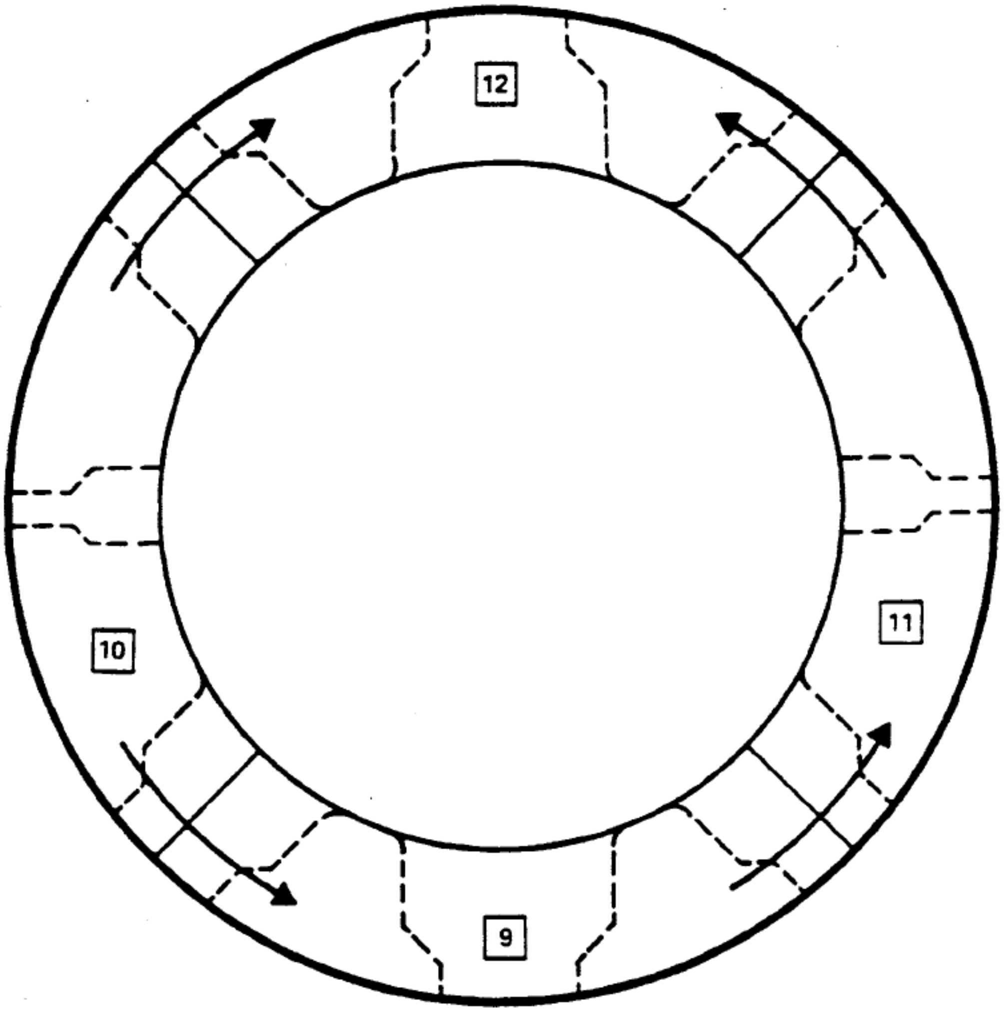

NOTE: PRIMARY PIPING ABOVE IS SHOWN IN PHANTOM.

DAVIS-BESSE NUCLEAR POWER STATION
REACTOR CAVITY FLOW DIAGRAM
(ELEVATION BELOW NOZZLES)
FIGURE 6.2-16

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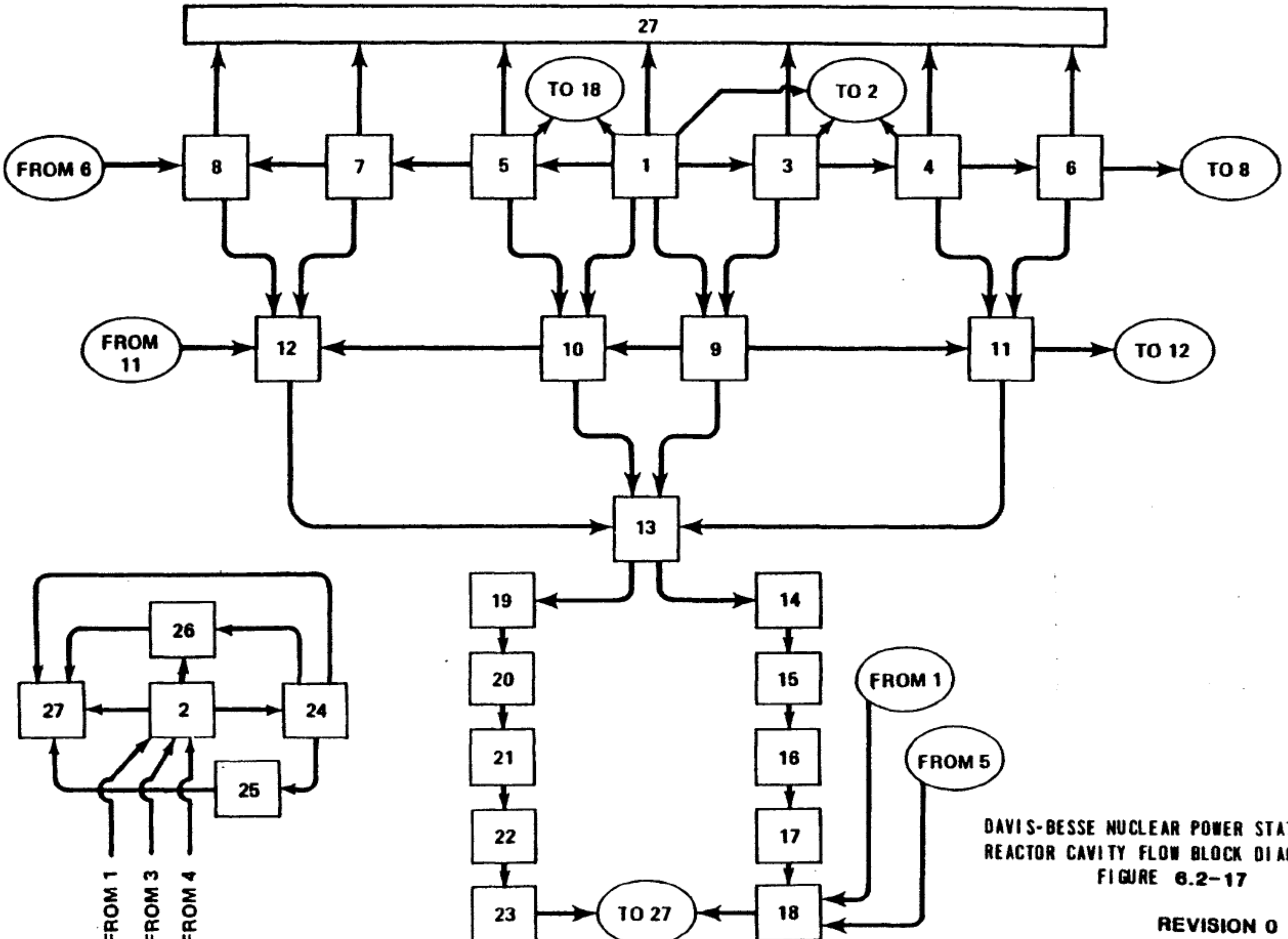

DAVIS-BESSE NUCLEAR POWER STATION
REACTOR CAVITY FLOW BLOCK DIAGRAM
FIGURE 6.2-17

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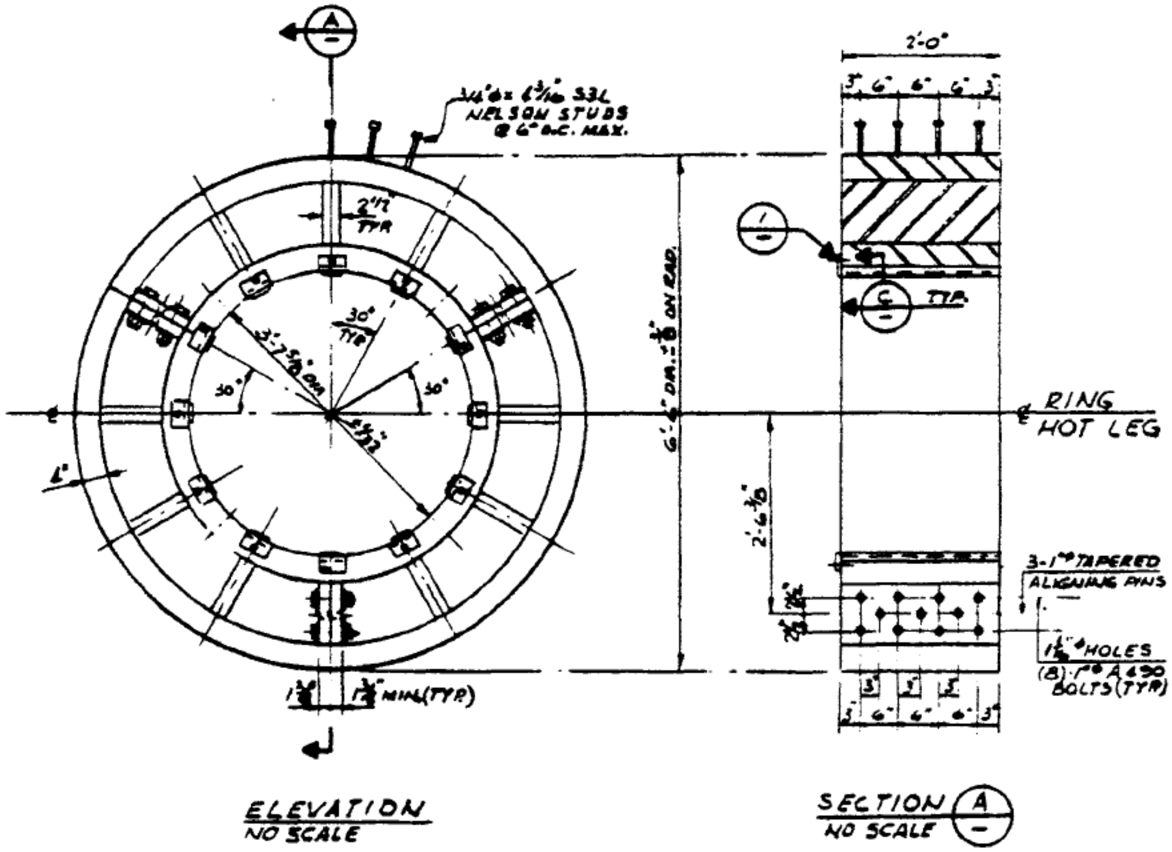

HOT LEG L.O.C.A. RING
HOT LEG REACTOR COOLANT SUPPORT

DAVIS-BESSE NUCLEAR POWER STATION
HOT LEG REACTOR COOLANT SUPPORT
FIGURE 6.2-18

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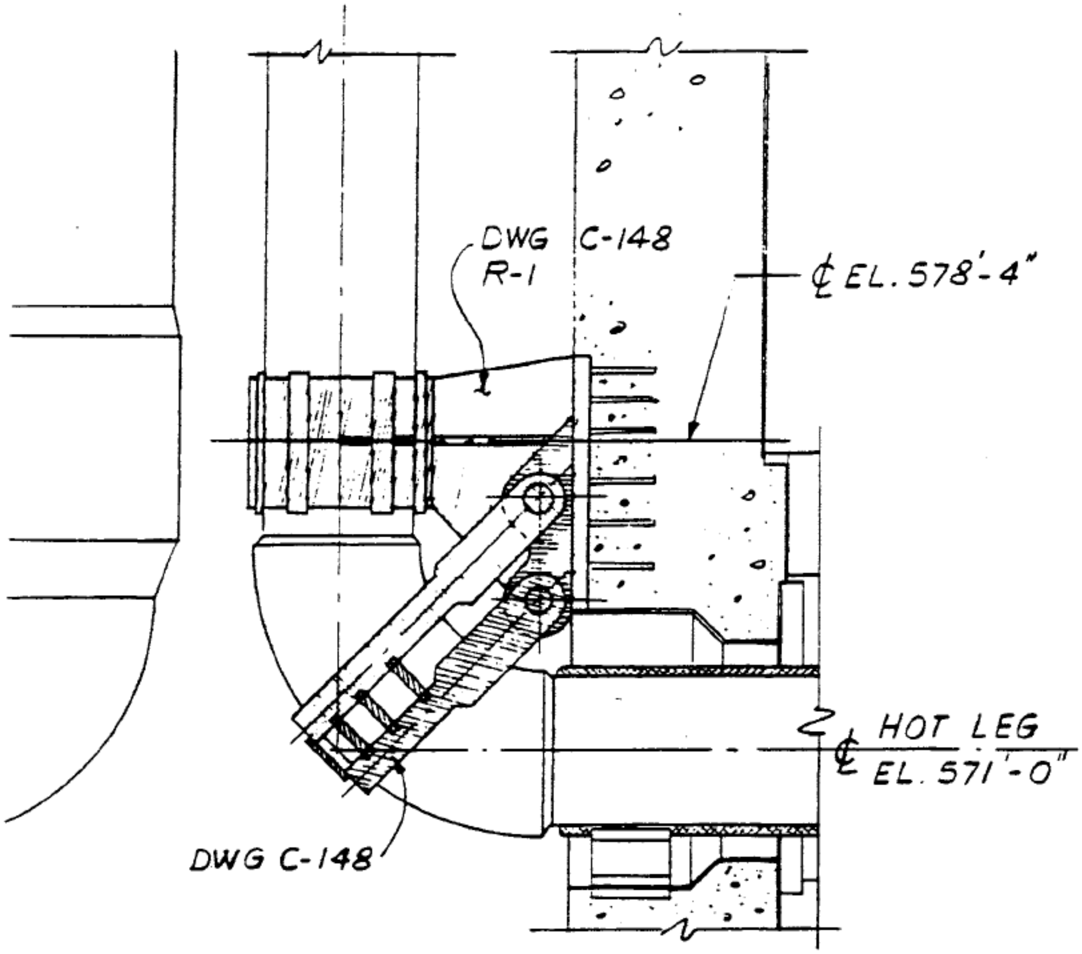

DAVIS-BESSE NUCLEAR POWER STATION
HOT LEG RESTRAINT
FIGURE 6.2-18A

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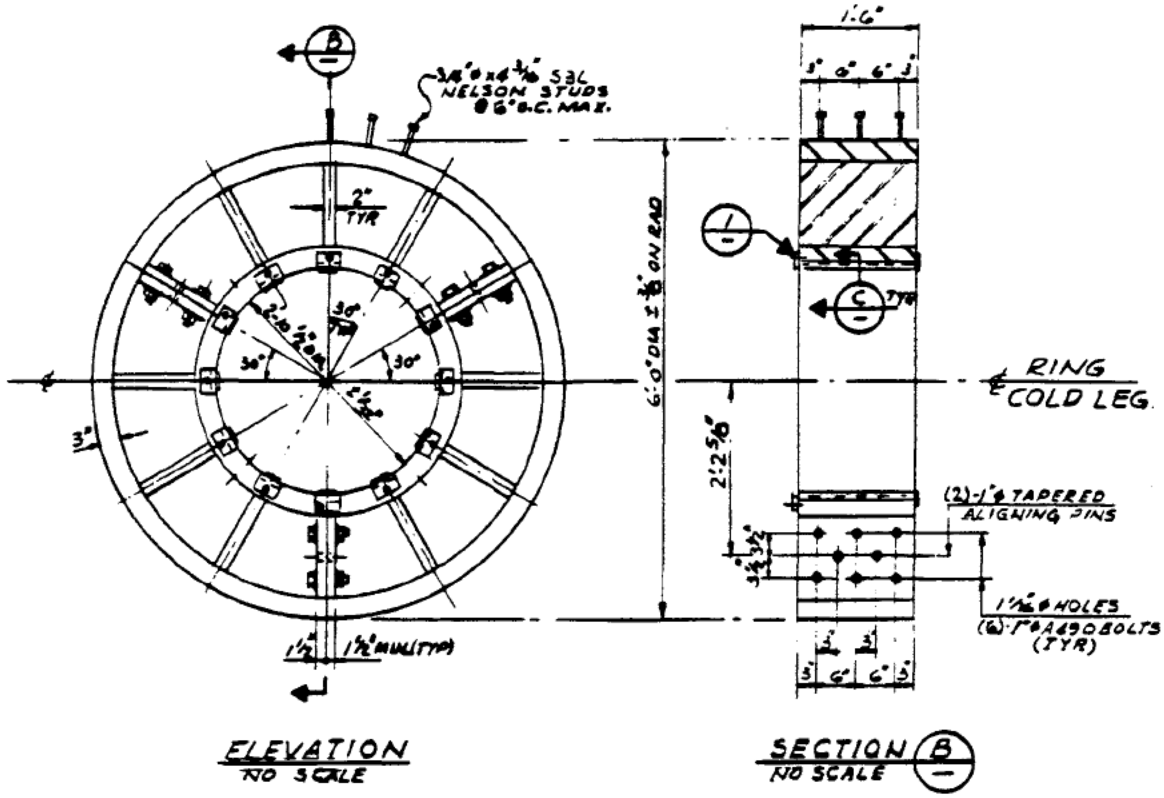

COLD LEG L.O.C.A. RING

COLD LEG REACTOR COOLANT SUPPORT

DAVIS-BESSE NUCLEAR POWER STATION
COLD LEG REACTOR COOLANT SUPPORT
FIGURE 6.2-19

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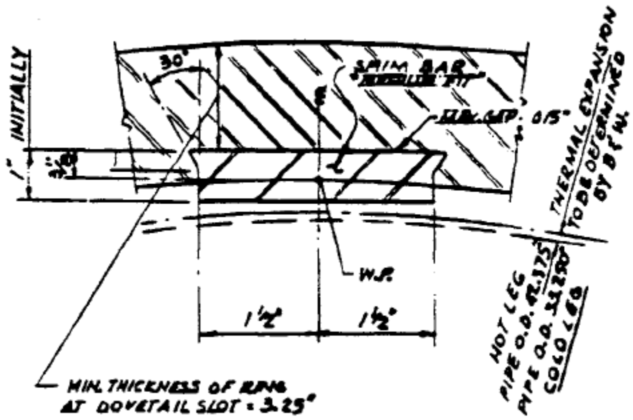

SECTION (C)
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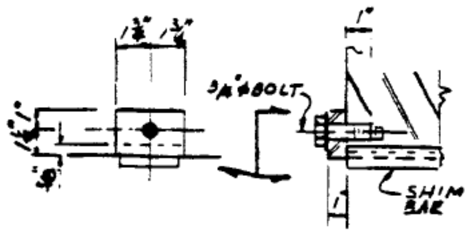

DETAIL (1)
NO SCALE

HOT AND COLD LEG SUPPORT RING DETAILS

DAVIS-BESSE NUCLEAR POWER STATION
HOT AND COLD LEG SUPPORT RING DETAILS
FIGURE 6.2-20

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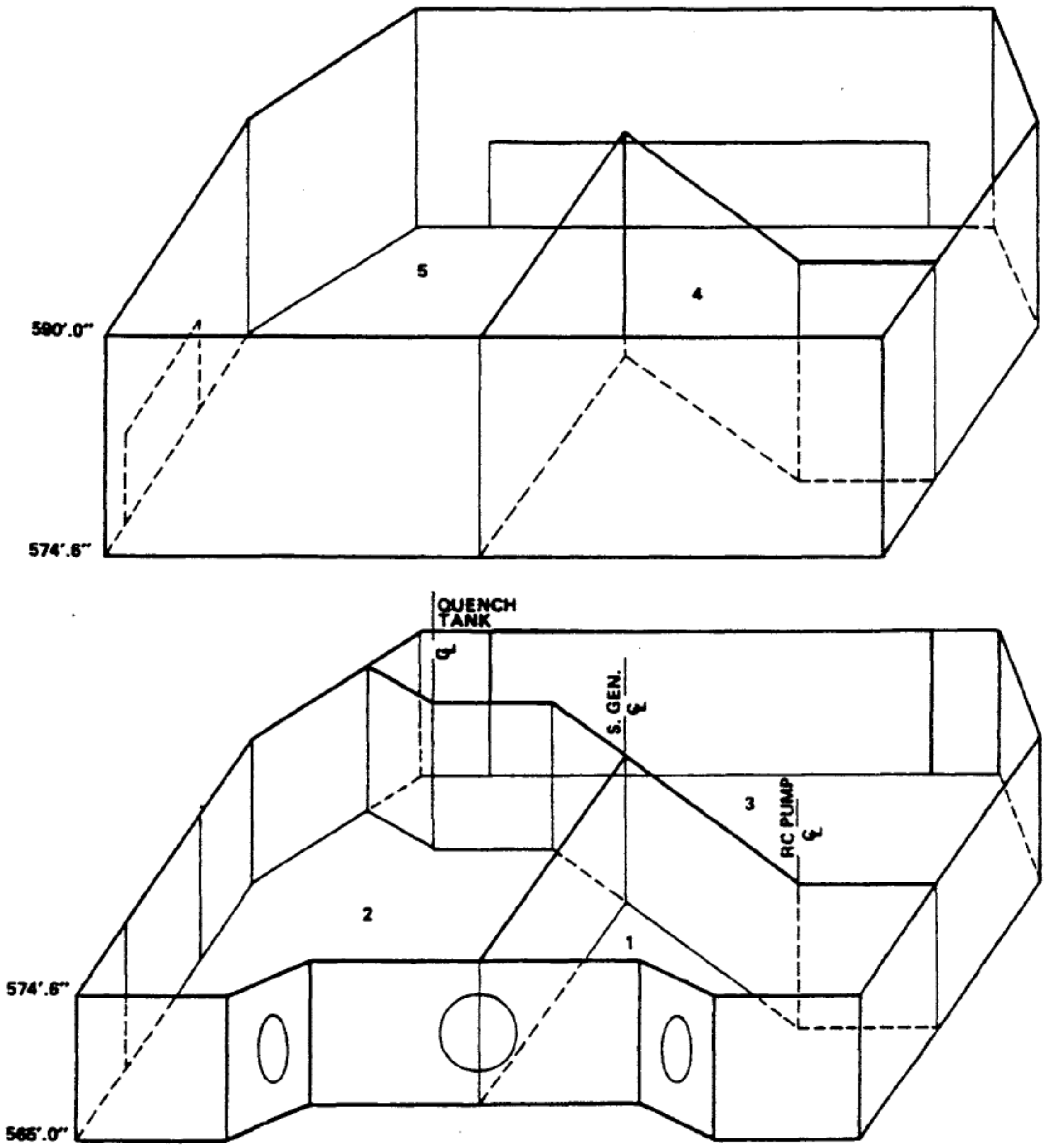

DAVIS-BESSE NUCLEAR POWER STATION
STEAM GENERATOR CAVITY FLOW DIAGRAM
(SIDE VIEW)
FIGURE 6.2-21

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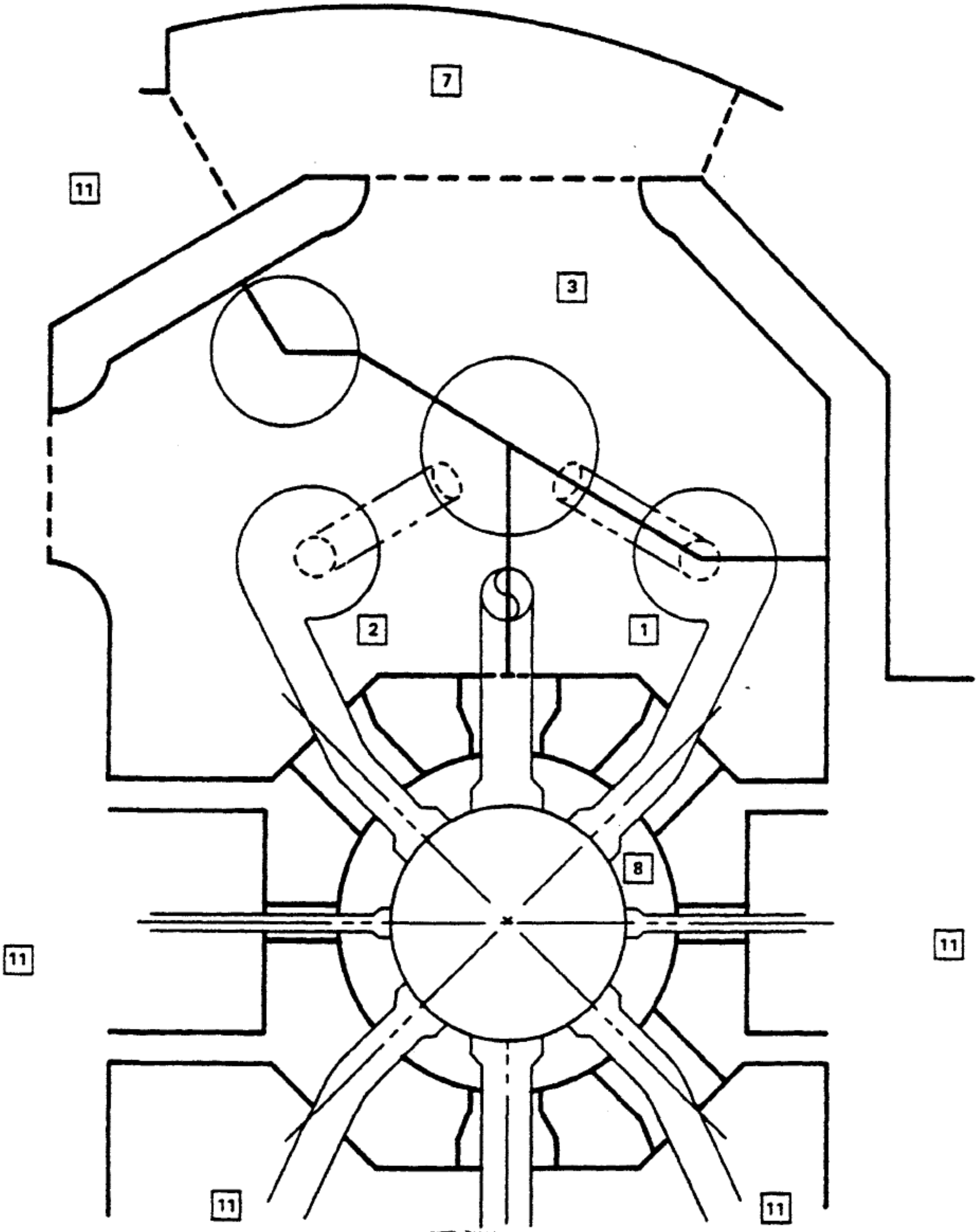
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
STEAM GENERATOR CAVITY FLOW DIAGRAM
(TOP VIEW) EL 565'TO 574'6''
FIGURE 6.2-22

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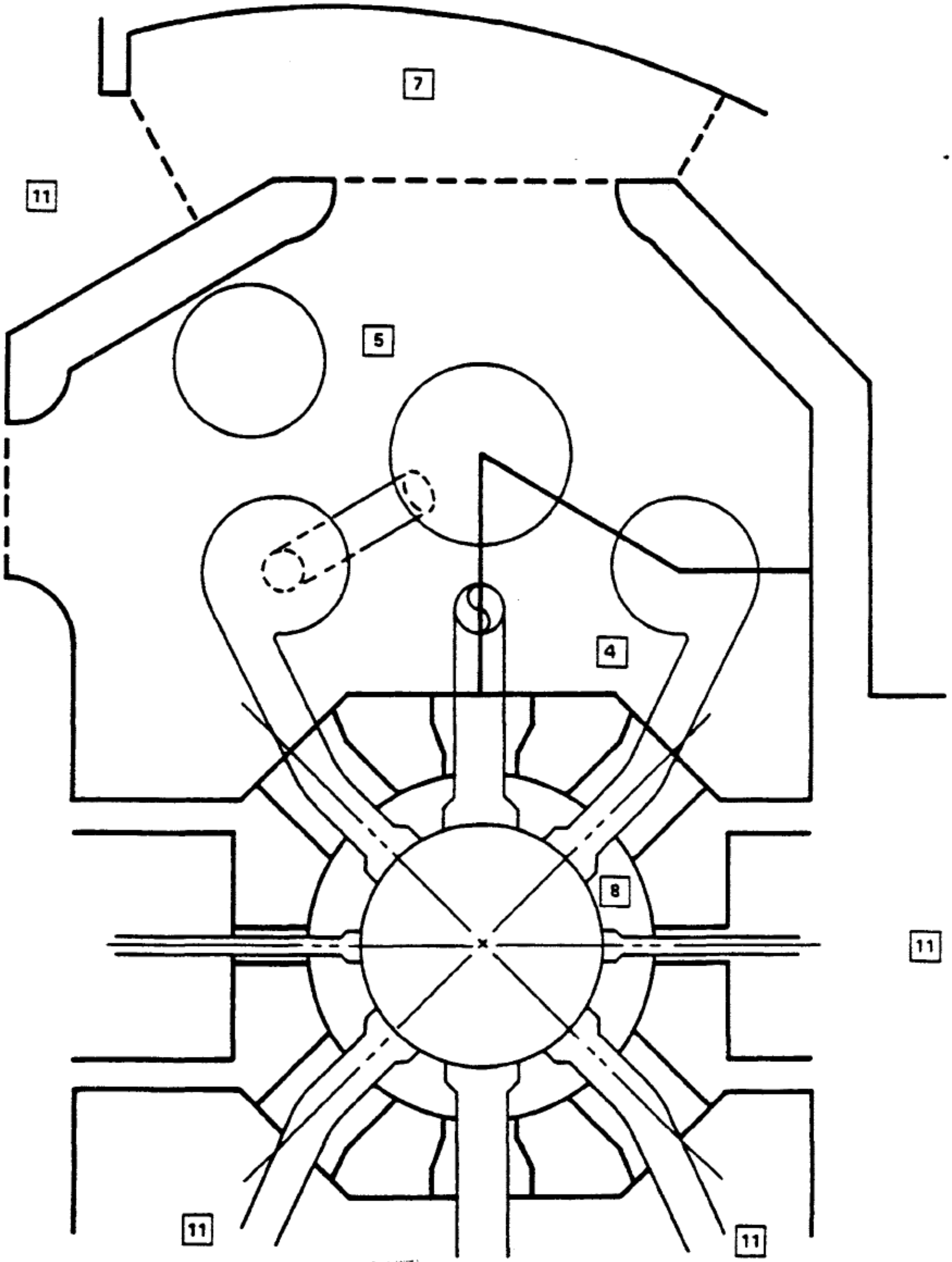

DAVIS-BESSE NUCLEAR POWER STATION
STEAM GENERATOR CAVITY FLOW
DIAGRAM (TOP VIEW) EL 574'6'' TO 590'
FIGURE 6.2-23

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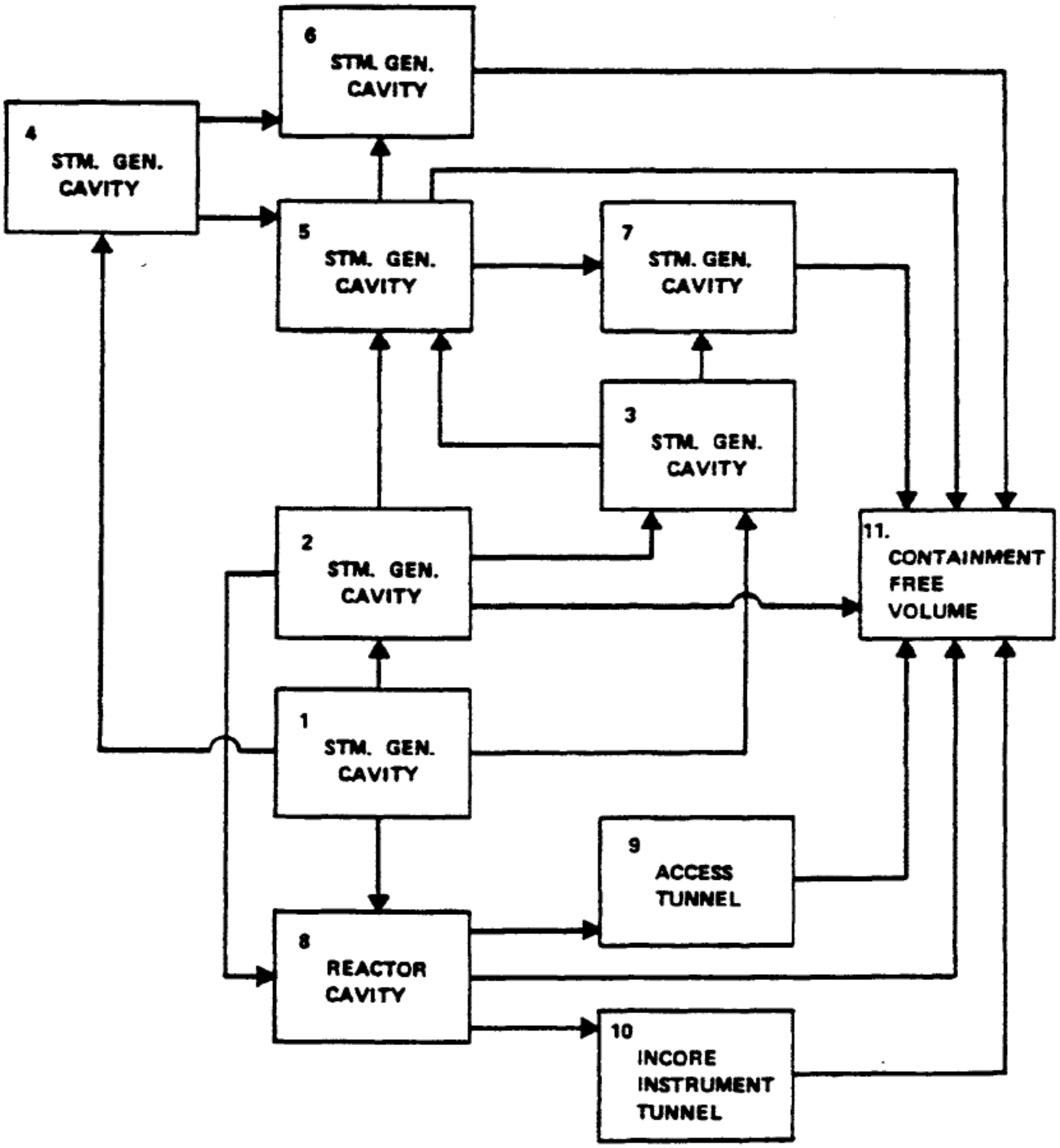

DAVIS-BESSE NUCLEAR POWER STATION
STEAM GENERATOR CAVITY
BLOCK DIAGRAM
FIGURE 6.2-24

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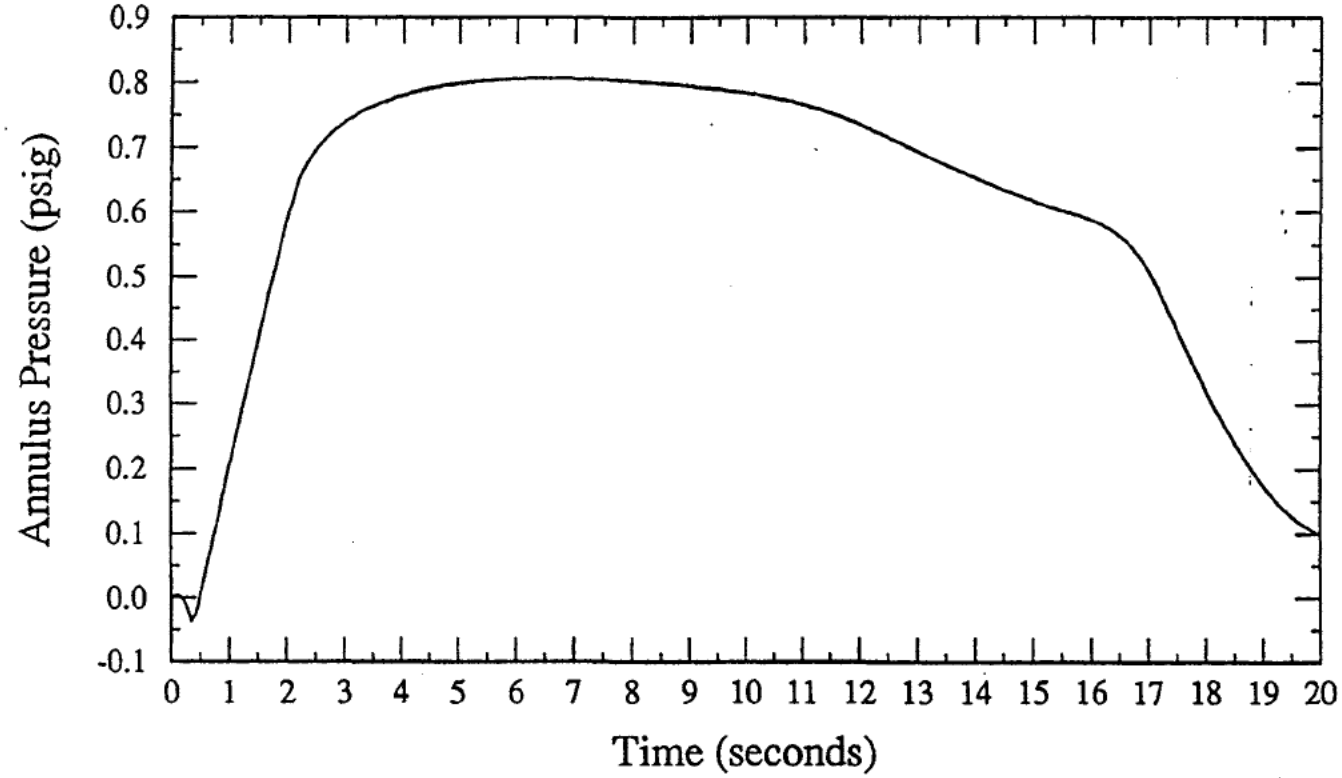
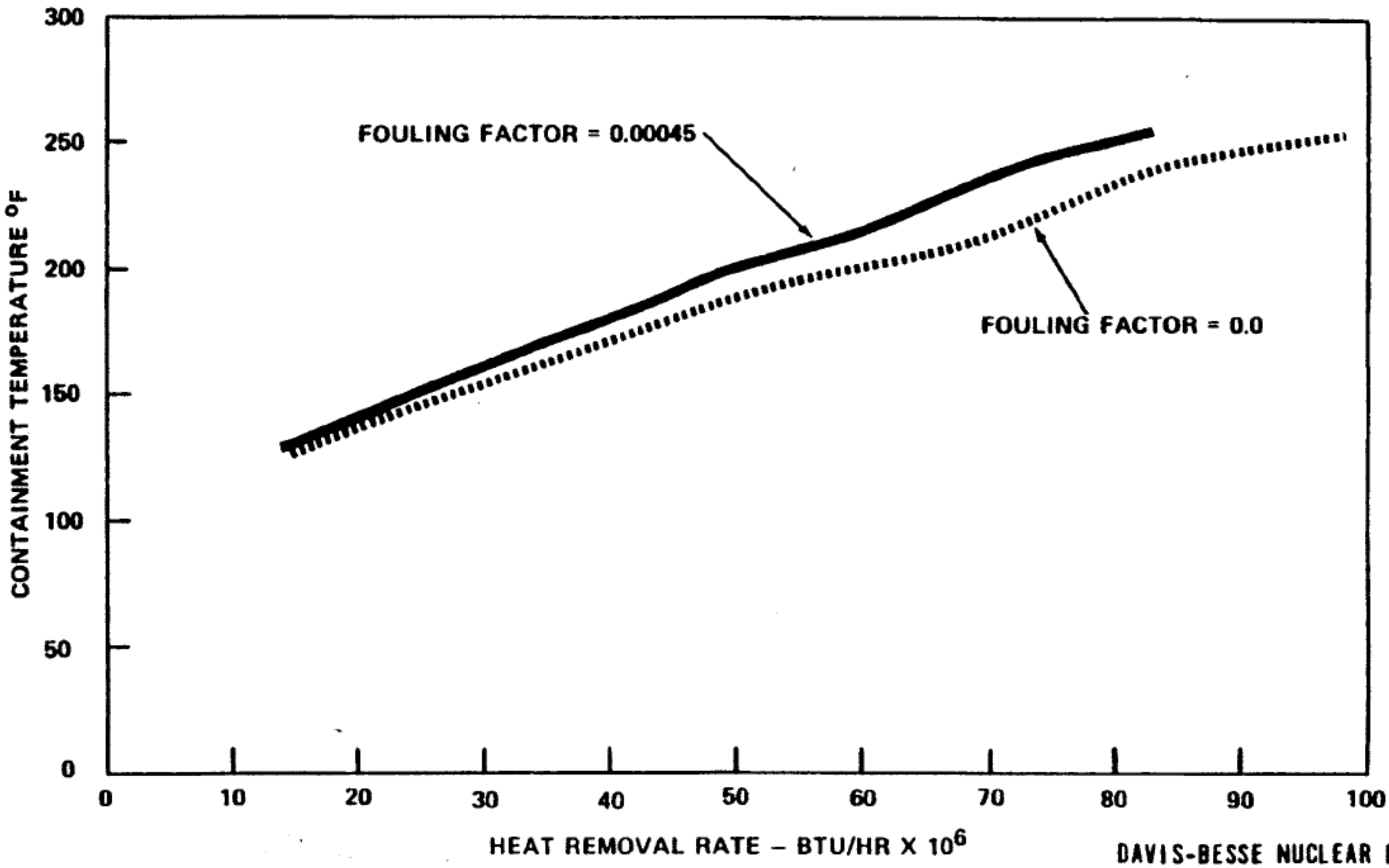


* Graph depicts a break in Room 314

Davis-Besse Nuclear Power Station
Annulus Pressure for an 18 inch
Main Feedwater Line Break
Figure 6.2-25

REVISION 20
DECEMBER 1996



DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT AIR COOLER HEAT
REMOVAL RATE VERSUS TEMPERATURE
FIGURE 6.2-26

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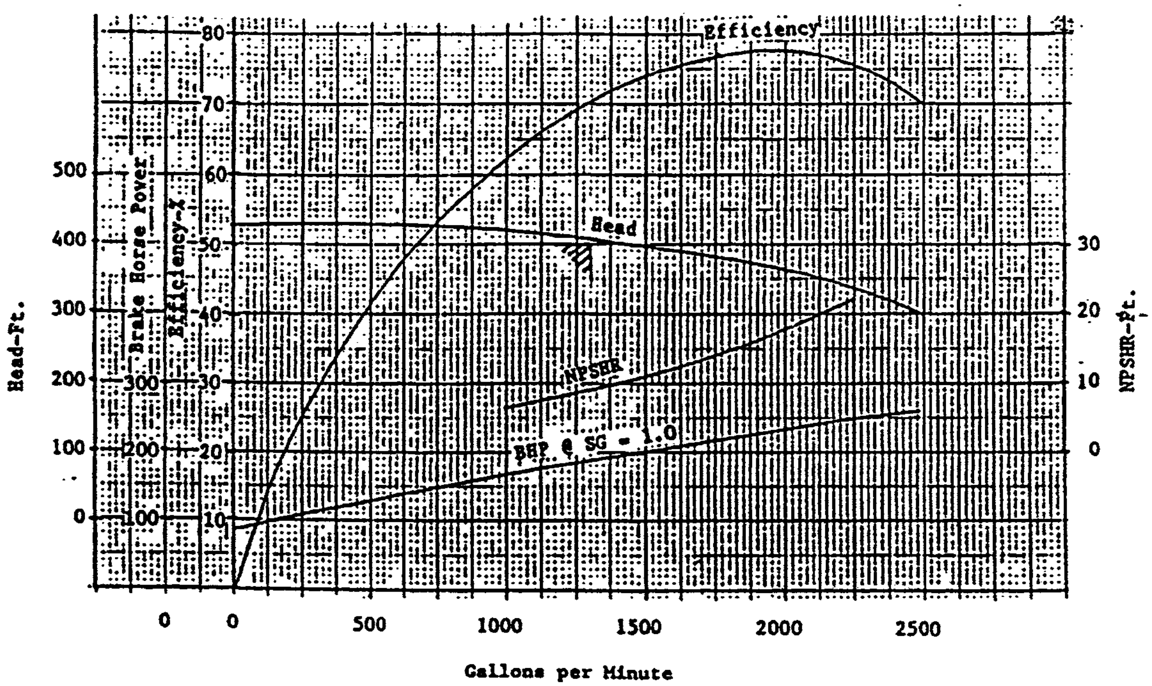
JULY 1982

Removed in Accordance with RIS 2015-17

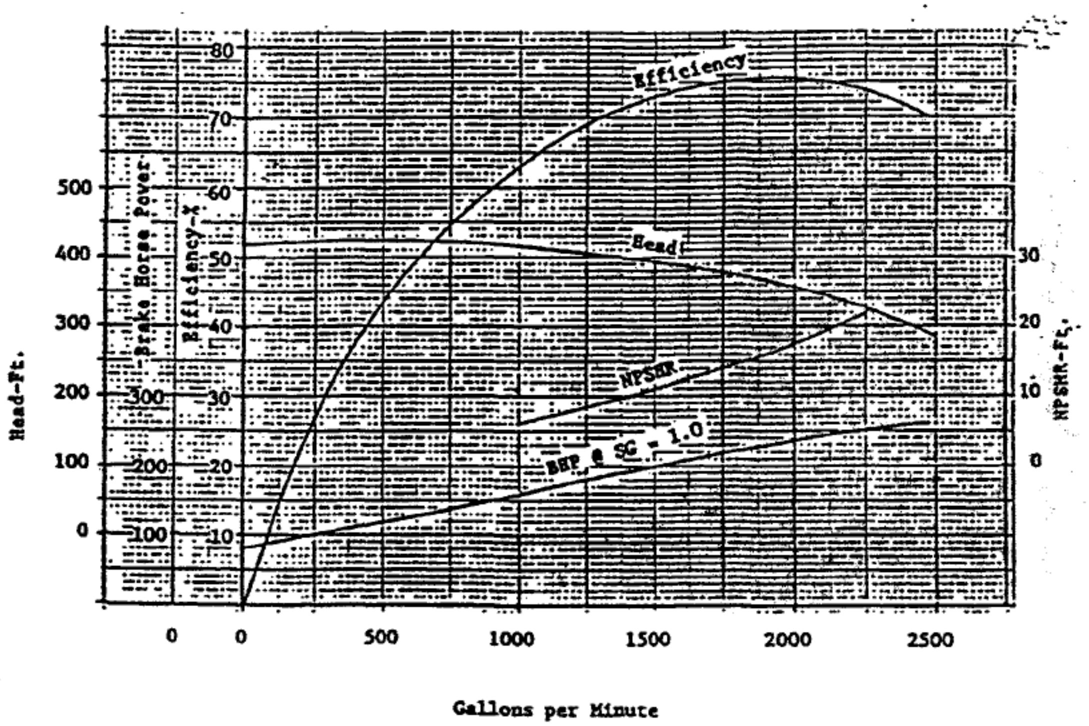
DAVIS-BESSE NUCLEAR POWER STATION

CONTAINMENT VESSEL COOLING SYSTEM
PLAN, SECTIONS AND DETAILS
M-431
FIGURE 6.2-27



DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT SPRAY PUMP NO. 1
CHARACTERISTICS
FIGURE 6.2-28

REVISION 20
DECEMBER 1996



DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT SPRAY PUMP NO. 2
CHARACTERISTICS
FIGURE 6.2-29

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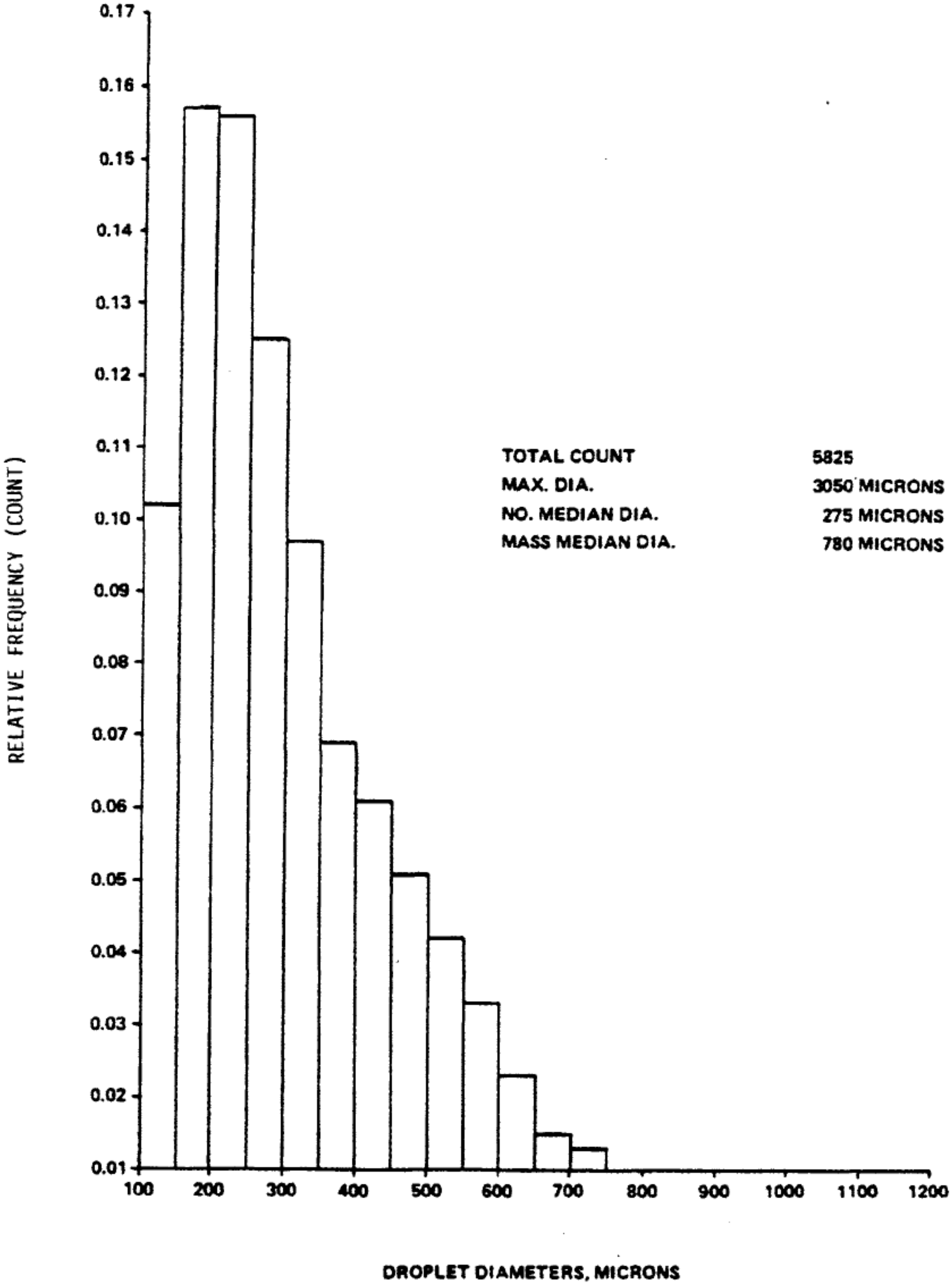
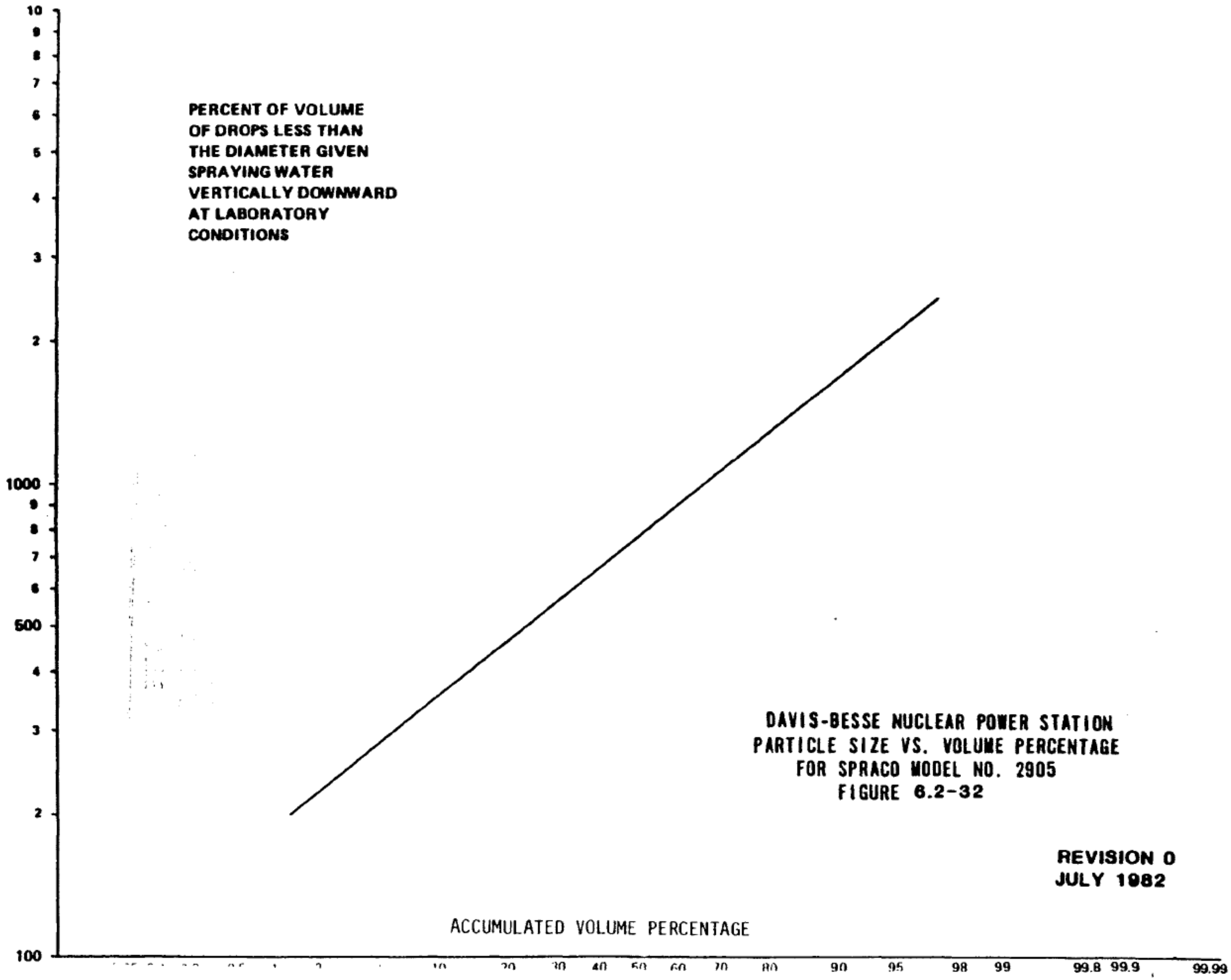


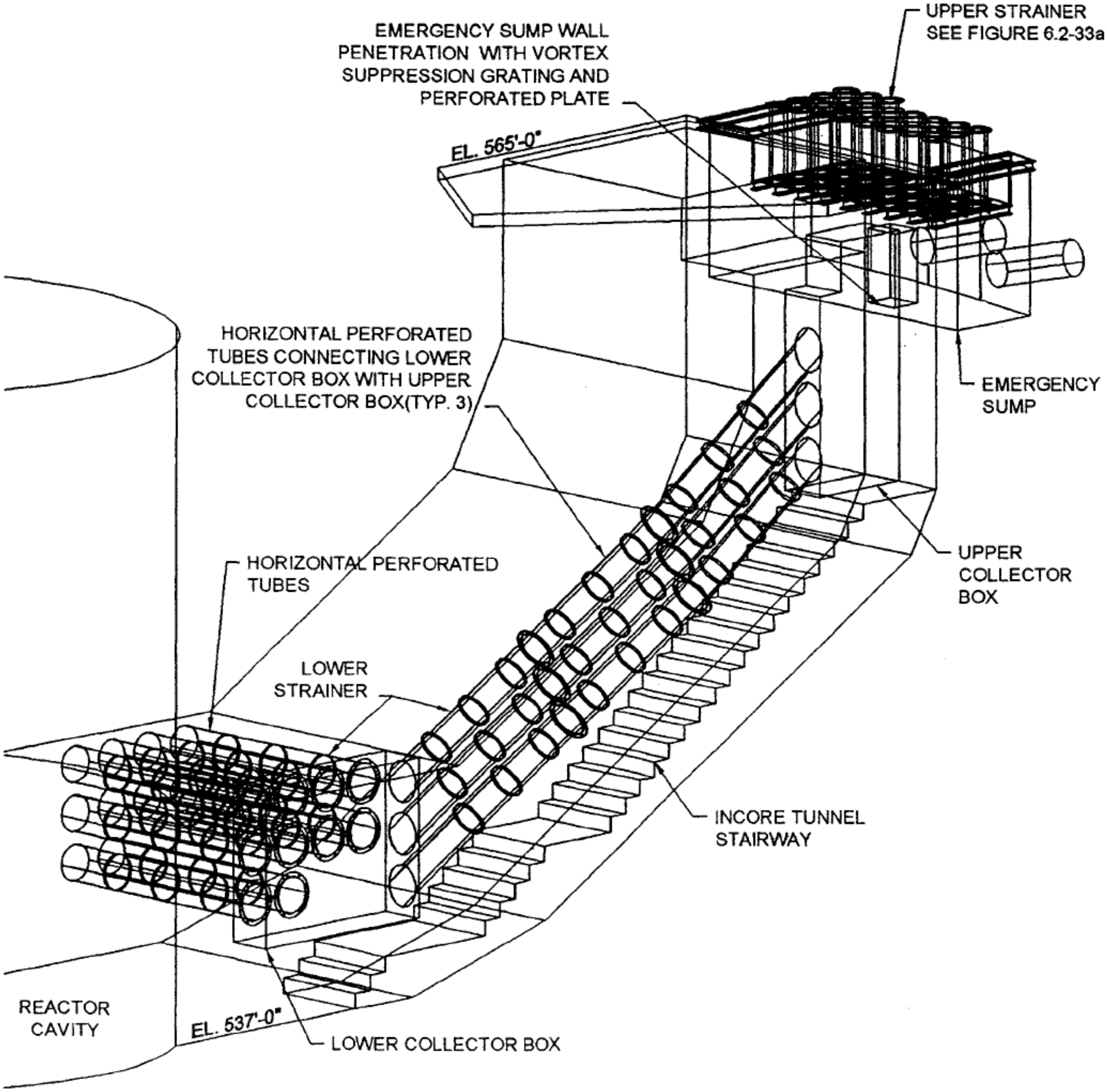
DAVIS-BESSE NUCLEAR POWER STATION
HISTOGRAM FOR SPRACO MDL. 2905
FIGURE 6.2-31

REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
PARTICLE SIZE VS. VOLUME PERCENTAGE
FOR SPRACO MODEL NO. 2905
FIGURE 6.2-32

REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION

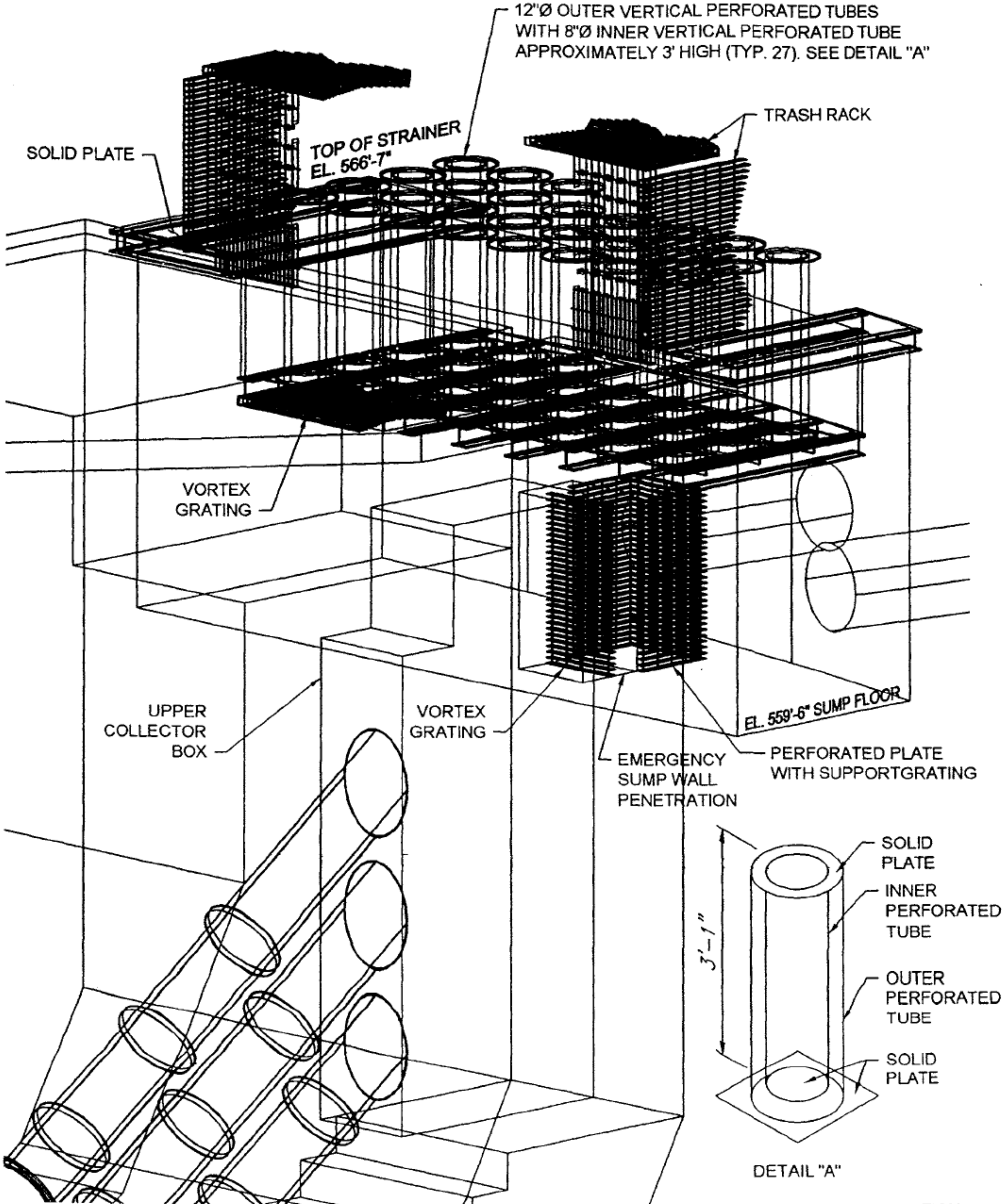
CONTAINMENT VESSEL EMERGENCY SUMP

USAR FIGURE 6.2-33

REVISION 24

JUNE 2004

DB 06-03-04 DFN G:/USAR/UFIG6233UFIG6233A.DWG



DAVIS-BESSE NUCLEAR POWER STATION
EMERGENCY SUMP UPPER VESSEL DETAIL
USAR FIGURE 6.2-33a
REVISION 24
JUNE 2004

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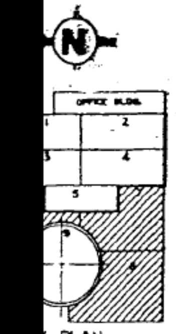


Y PLAN
L 623'-0"

W DIAGRAM THIS SYSTEM
NGS NO. M-027, M-028 & M-029.

LS. ABBREVIATIONS &
OTES SEE DWG. NO. M-400

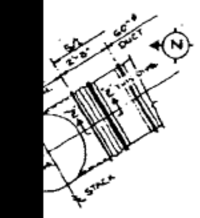
S F19-1, F19-2, F24, & F46
FURNISHED BY OWNER,
BY CONTRACTOR 7749-41.

HEATING SYSTEM P&I.D.
NO. M-021.

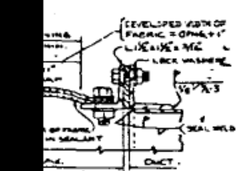
TES LIMIT OF DUCTWORK
NG SEISMIC I ANALYSIS
THAN 'Q' LISTED)

MPER SCHEDULE SEE DRAWING

GINEERING MUST BE CONSULTED
SING DESIGN AIR FLOW RATES FOR
TERS DUE TO THE POTENTIAL
E MAIN STEAM TO AUX. FEED PUMP
S ANALYSIS (REF.:C-NSA-000.02-007)).



L "B"



Z-Z"

C" (OPTIONAL)

BESSE NUCLEAR POWER STATION

LIARY BUILDING VENTILATION
UIPMENT AND DUCT LAYOUT,
PLAN AT EL. 623

M410

FIGURE 6.2-34

REVISION 18
OCTOBER 1993

Removed in Accordance with RIS 2015-17

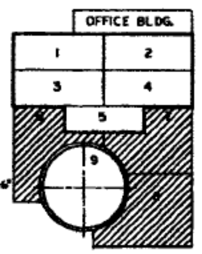
DAVIS-BESSE NUCLEAR POWER STATION

AUXILIARY BUILDING VENTILATION
EQUIPMENT AND DUCT LAYOUT

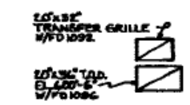
PLAN ELEV. 603'-0"
M-411
FIGURE 6.2-35

DB 11-14-17 DFN:/H/PLDSGN/M411.DGN/TIF

[redacted]



KEY PLAN
EL. 585'-0"



SECTION 'B-B'

NOTES:

1. FOR AIR FLOW DIAGRAM THIS SYSTEM SEE DRAWINGS NO. M-027, M-028 & M-029
2. FOR SYMBOLS AND ABBREVIATIONS SEE DRAWING NO. M-400.
3. FOR EXHAUST REGISTER SIZES NOT INDICATED, SEE SCHEDULE ON DRAWING NO. M-418.
4. IN GENERAL, ALL DUCTWORK SHOWN IMMEDIATLY ADJACENT TO A WALL MUST BE INSTALLED AT MINIMUM DISTANCE FROM THE WALL. DUCT LOCATIONS MAY ALSO BE ESTABLISHED BY BLOCKOUTS PROVIDED IN WALLS WHICH ARE NOTED BY THE SYMBOL. (SEE DWGS. C-218, C-219, C-220, C-240, C-241 & C-242) OR, BY LOCATION OF THE VENTILATION EQUIPMENT (SEE DWGS. M-120 THRU M-131) ALL OTHER DUCTWORK HEADERS ARE LOCATED BY DIMENSION ON THE DRAWING AS REQUIRED.
5. FOR FIRE DAMPER SCHEDULE SEE DRAWING M-467.
6. VENTILATION DUCTWORK FOR NUCLEAR FILTER COMPARTMENTS TO BE INSTALLED AFTER COMPARTMENT COVER INSTALLATION IS COMPLETE.

ENCE DWGS:

M.1 AUXILIARY BUILDING PARTIAL PLANS, SECTIONS & DETAILS

DAVIS-BESSE NUCLEAR POWER STATION

AUXILIARY BUILDING HEATING, VENTILATING AND AIR CONDITIONING EQUIPMENT PLAN ELEV. 585'-0"

M-412

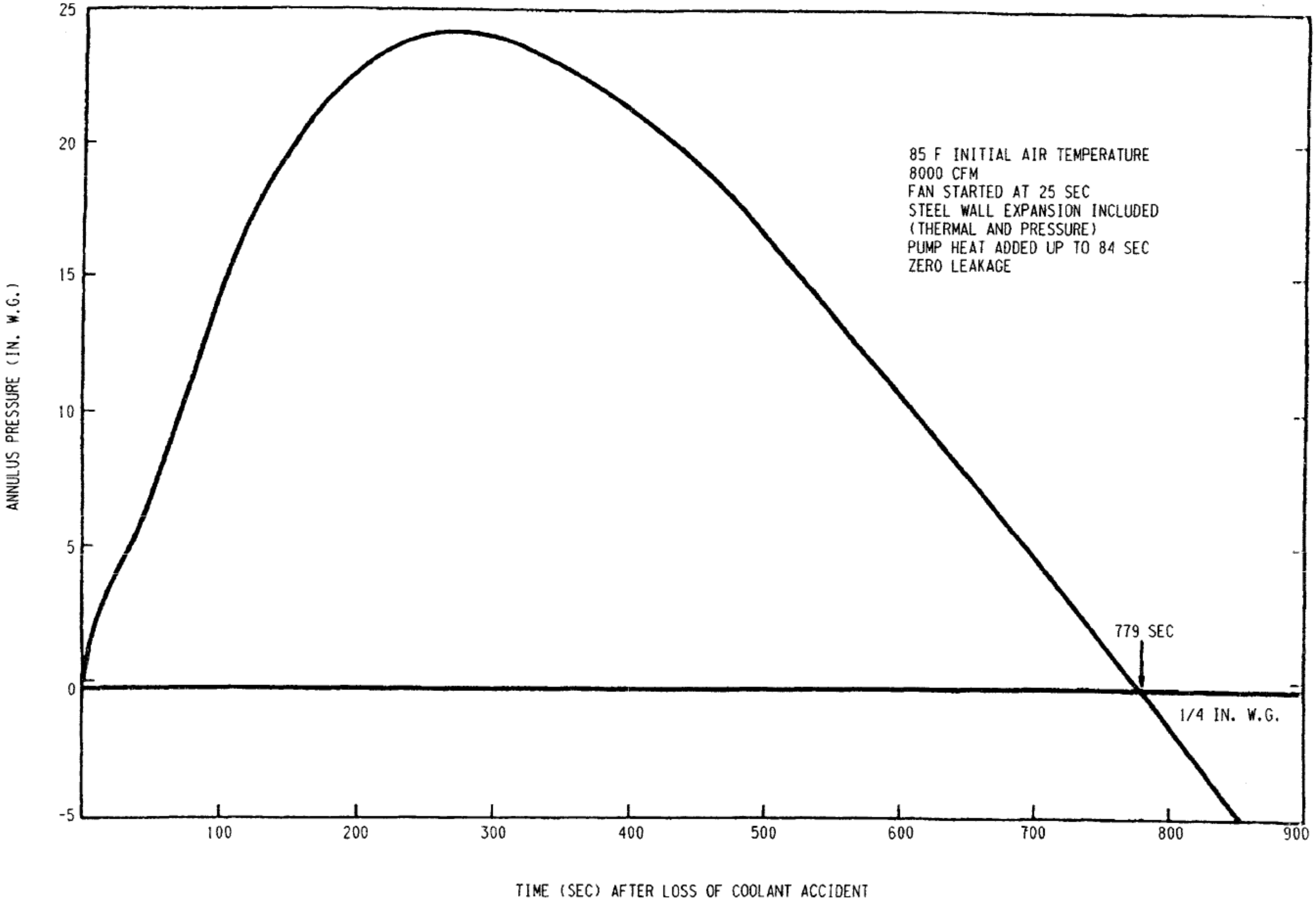
FIGURE 6.2-36

REVISION 30

OCTOBER 2014

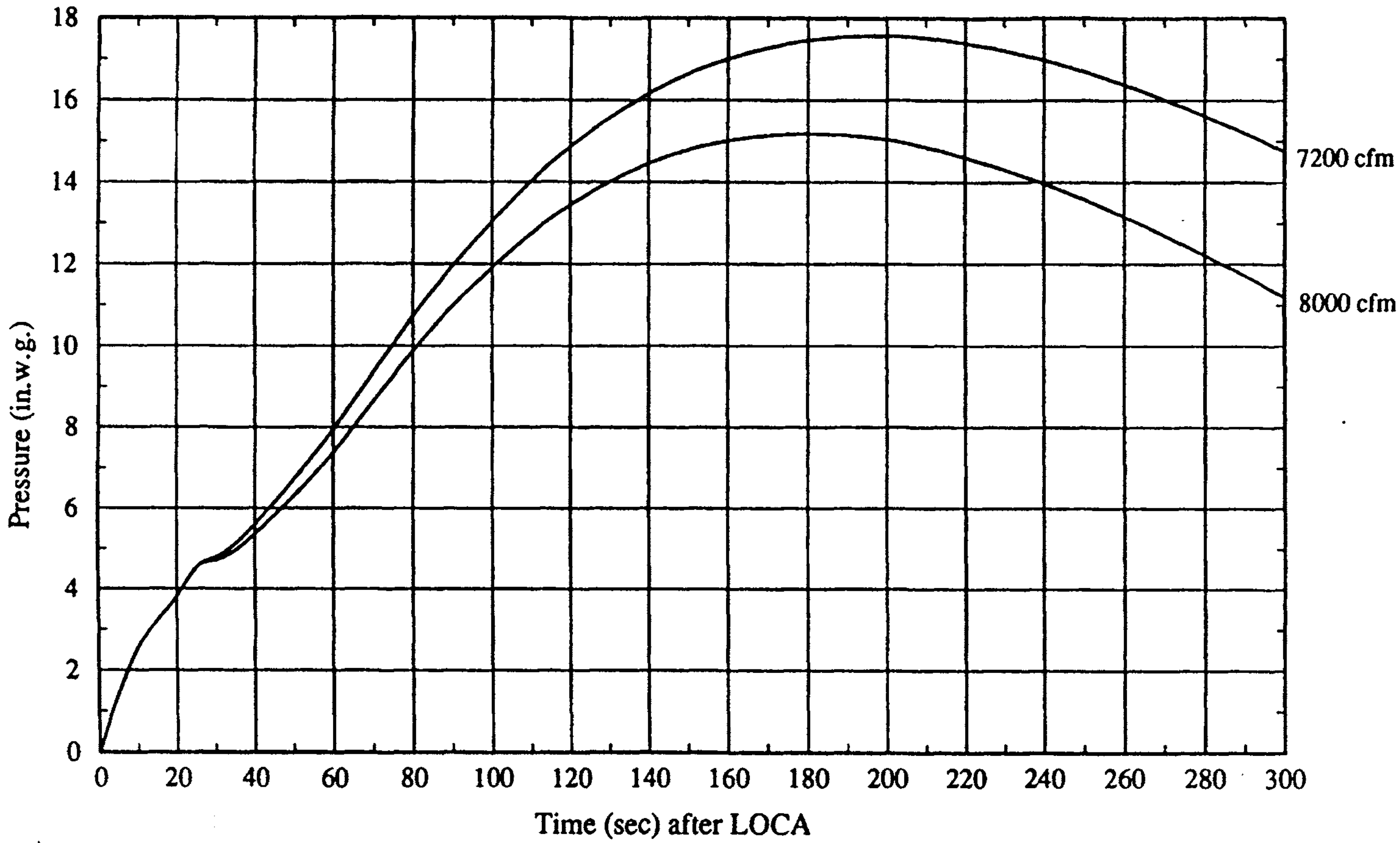
Q

DB 02-11-14 DFN=/J/RASDGN/M412.DGN/CIT



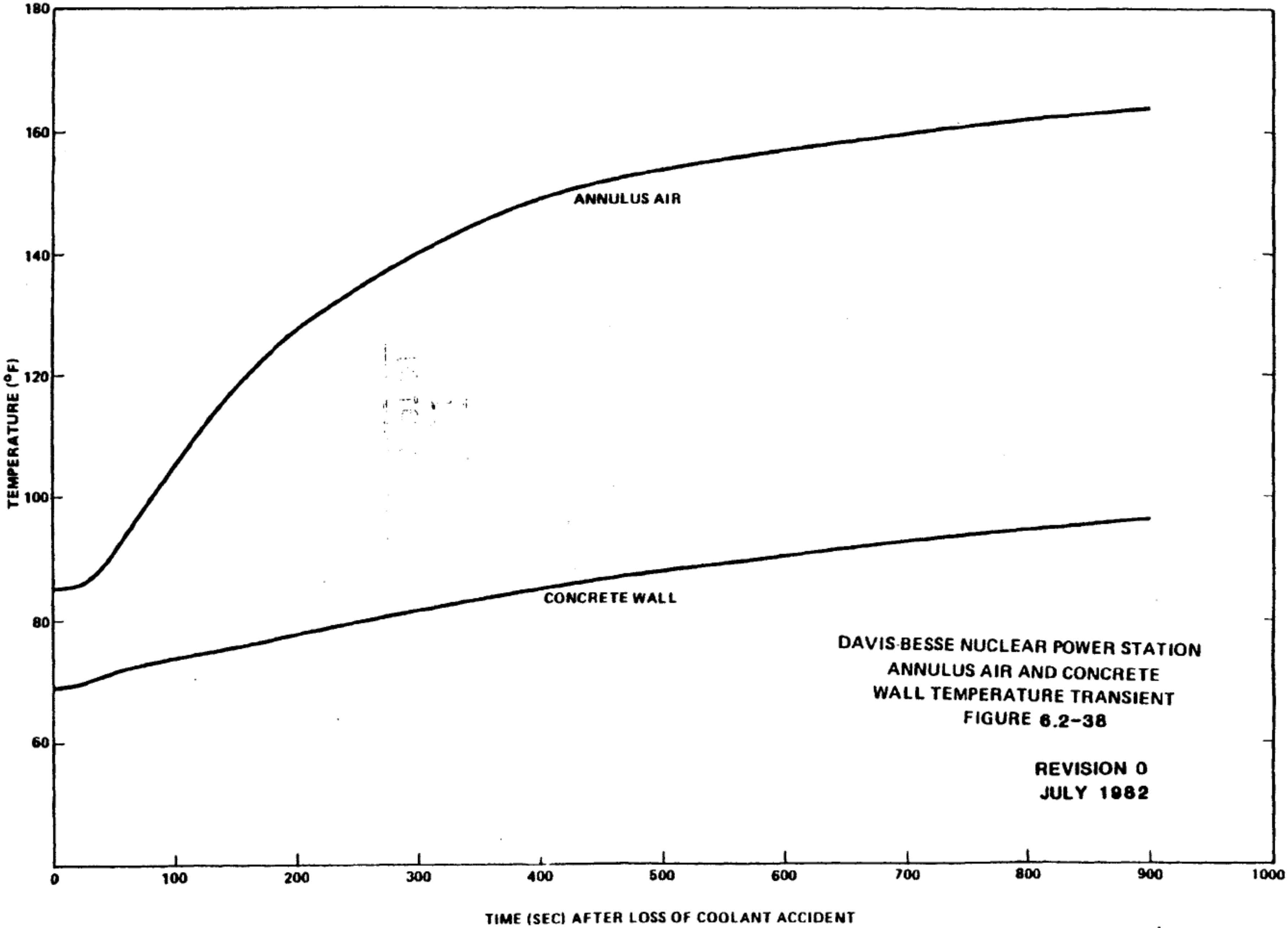
DAVIS-BESSE NUCLEAR POWER STATION
ANNULUS PRESSURE TRANSIENT
FIGURE 6.2-37

REVISION 25
JUNE 2006



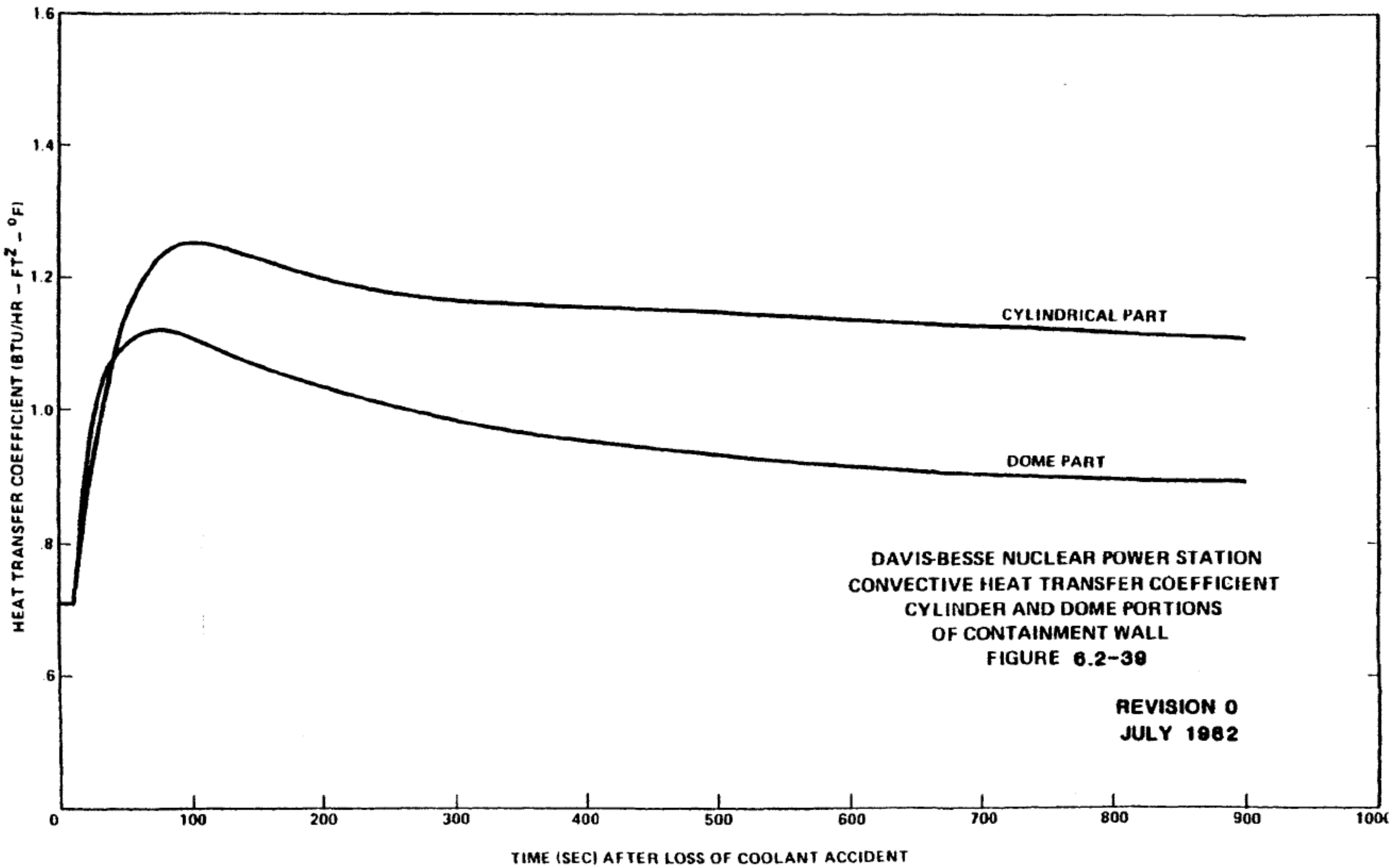
Davis-Besse Nuclear Power Station
Negative Pressure Boundary Pressure
Response Following LOCA
Figure 6.2-37A

REVISION 18
NOVEMBER 1993



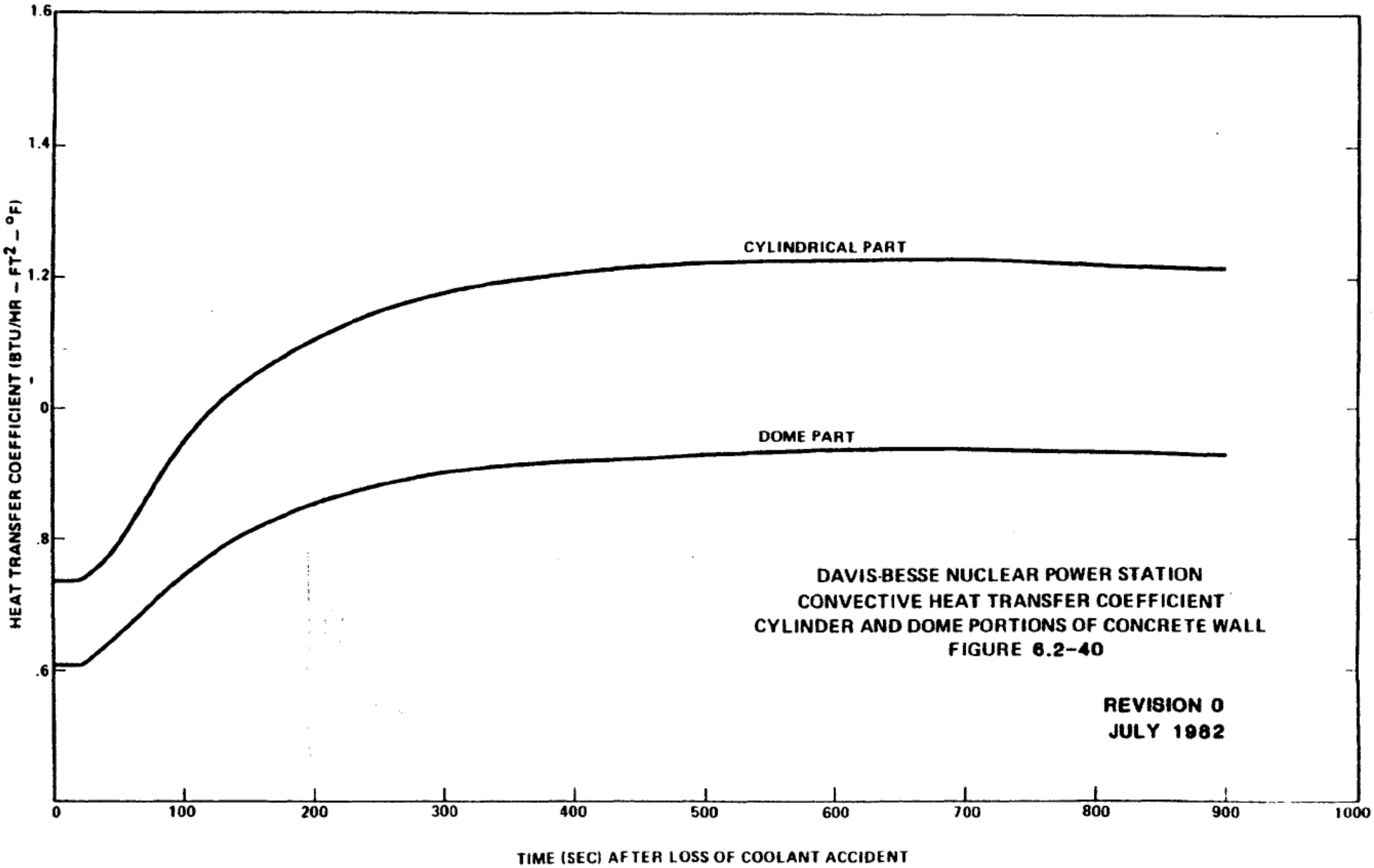
DAVIS-BESSE NUCLEAR POWER STATION
ANNULUS AIR AND CONCRETE
WALL TEMPERATURE TRANSIENT
FIGURE 6.2-38

REVISION 0
JULY 1982



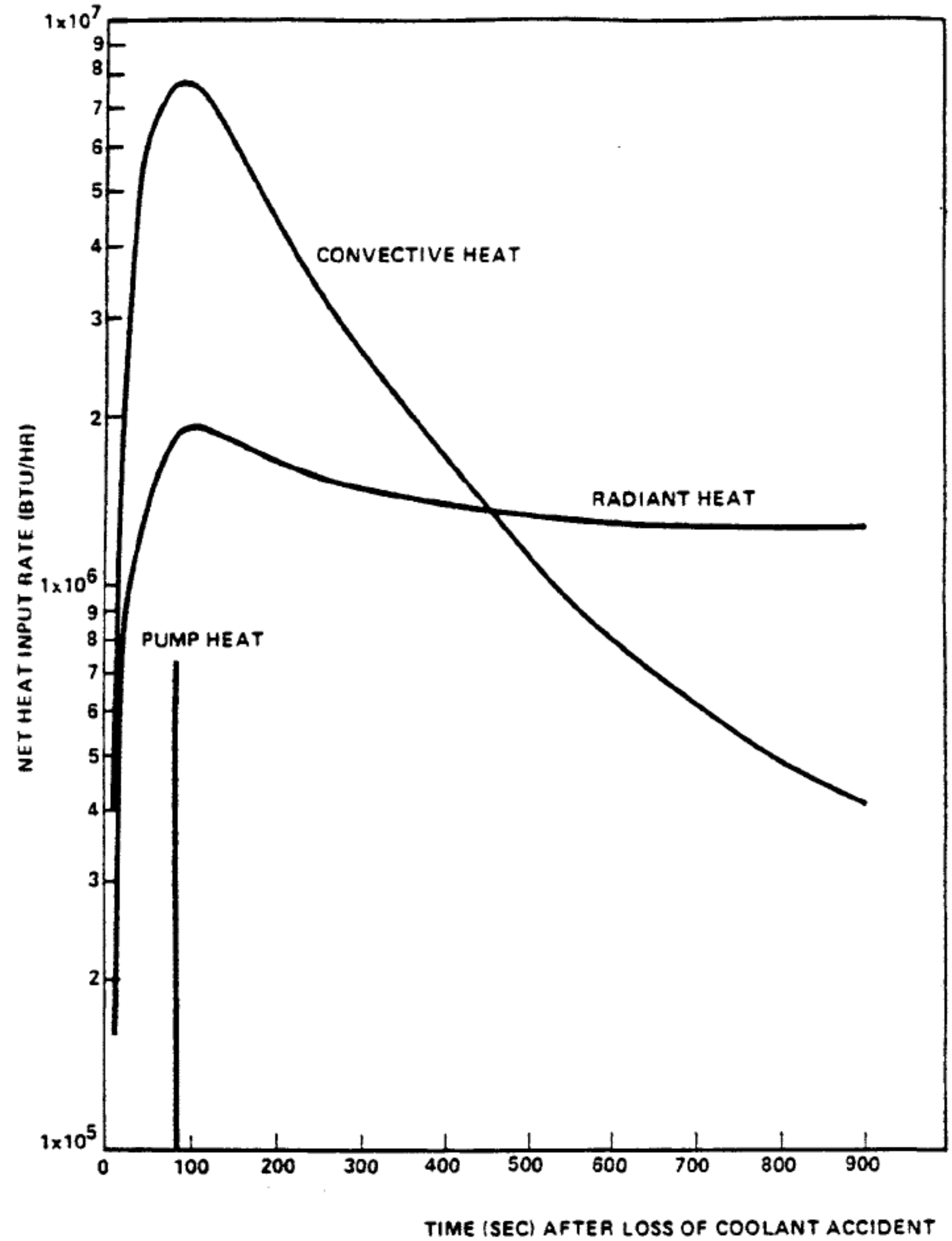
DAVIS-BESSE NUCLEAR POWER STATION
CONVECTIVE HEAT TRANSFER COEFFICIENT
CYLINDER AND DOME PORTIONS
OF CONTAINMENT WALL
FIGURE 6.2-39

REVISION 0
JULY 1982



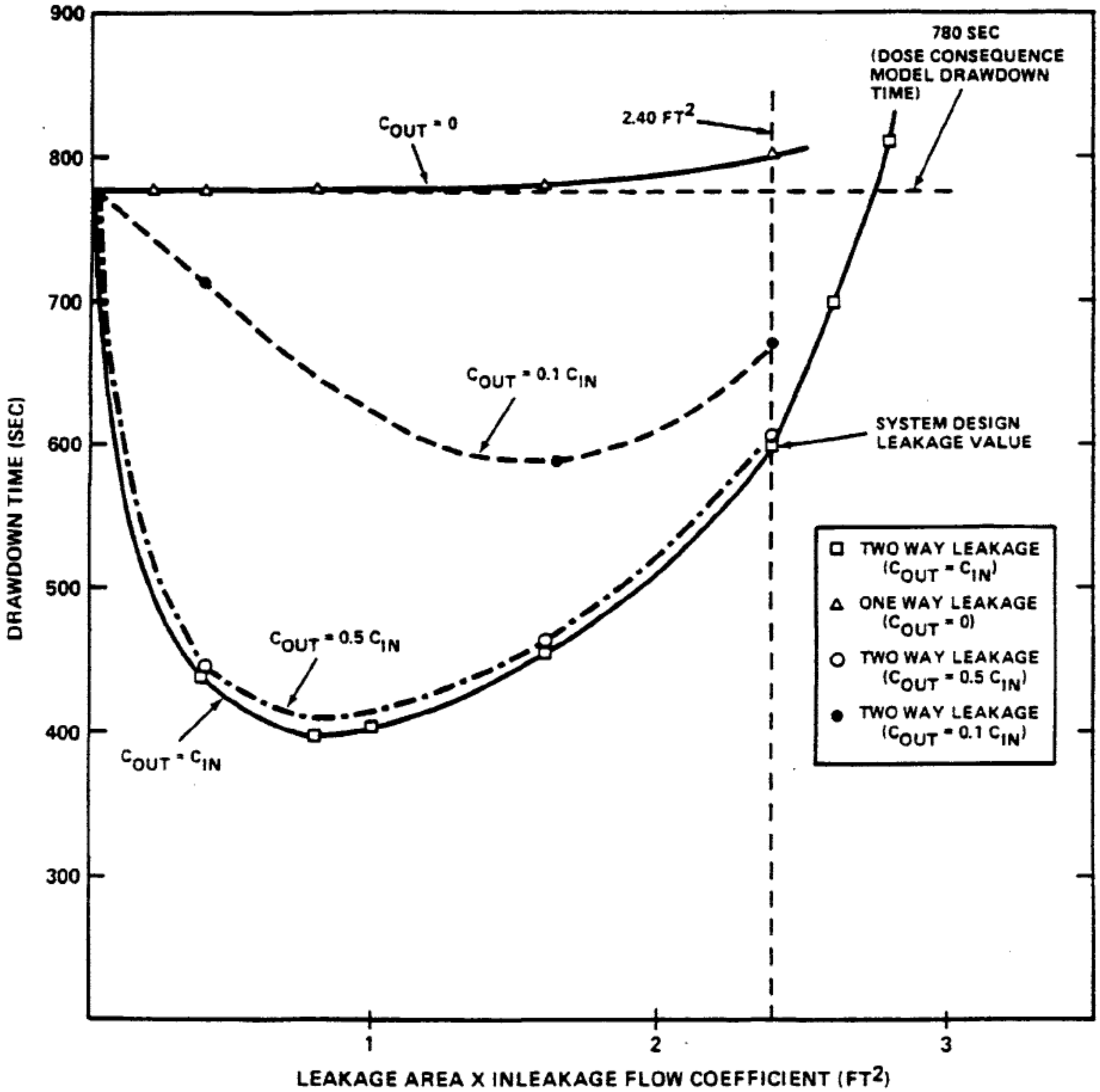
DAVIS-BESSE NUCLEAR POWER STATION
CONVECTIVE HEAT TRANSFER COEFFICIENT
CYLINDER AND DOME PORTIONS OF CONCRETE WALL
FIGURE 6.2-40

REVISION 0
JULY 1982



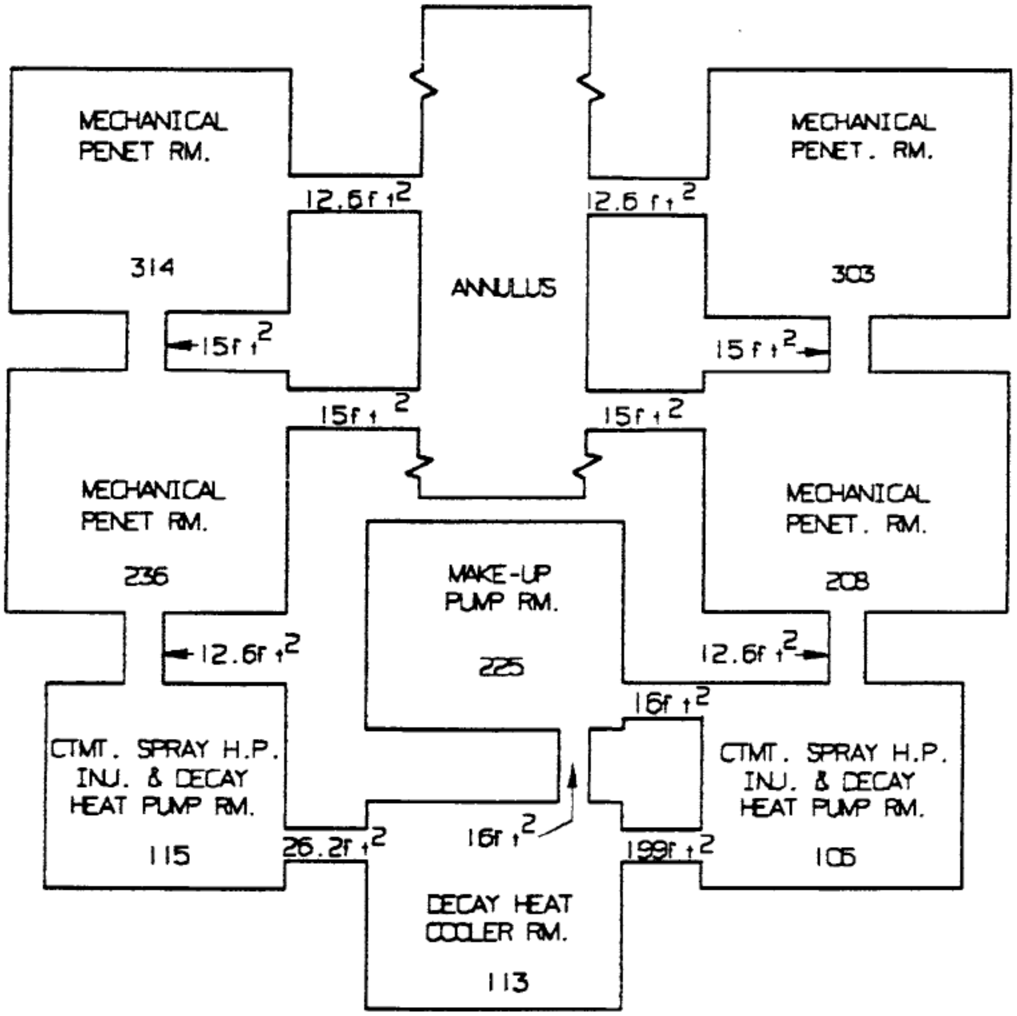
DAVIS-BESSE NUCLEAR POWER STATION
HEAT INPUT RATE TO ANNULUS AIR
FIGURE 6.2-41

REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
LEAKAGE EFFECT ON ANNULUS DRAWDOWN TIME
POST-LOCA
FIGURE 6.2-42

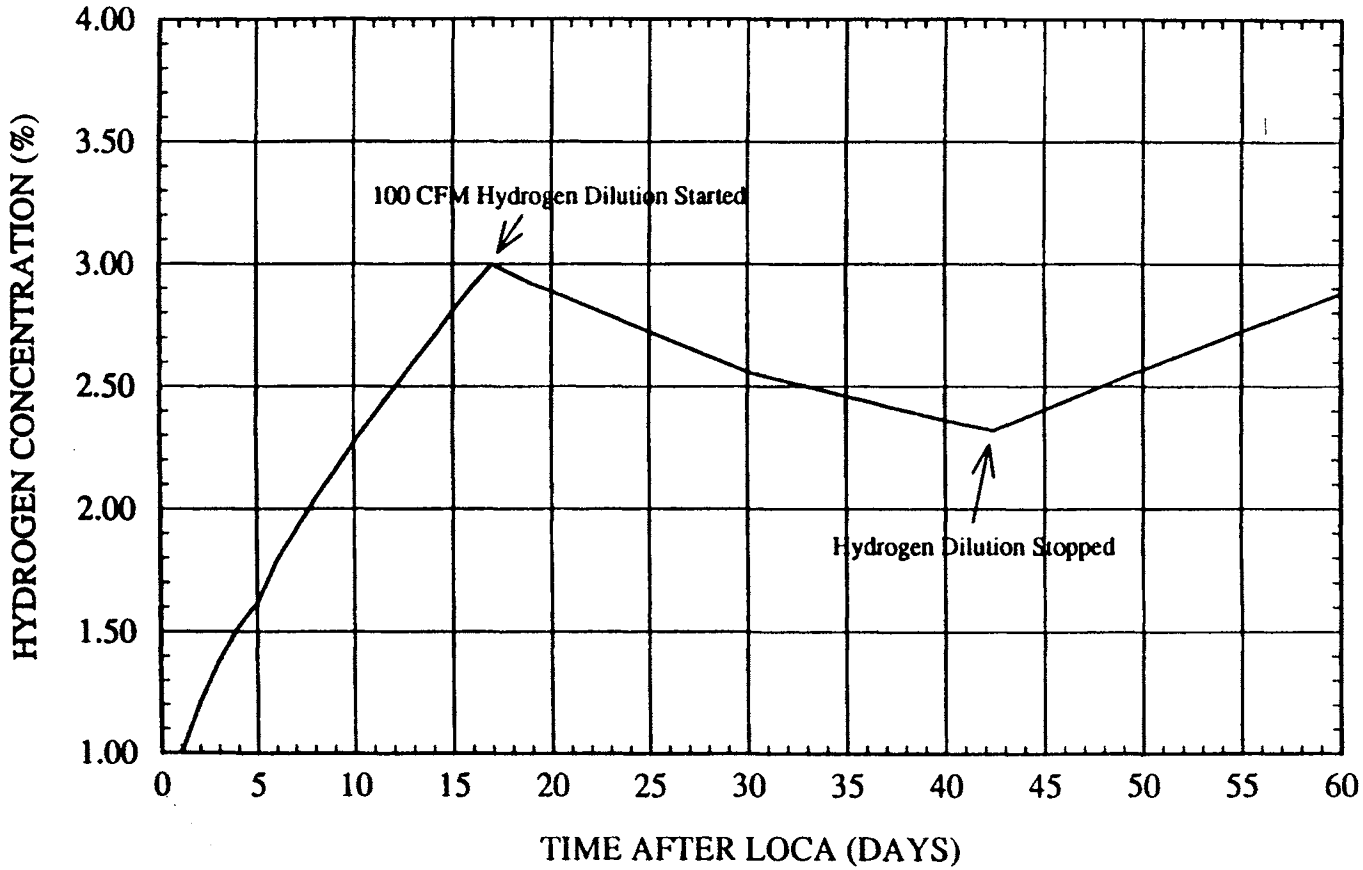
REVISION 0
JULY 1982



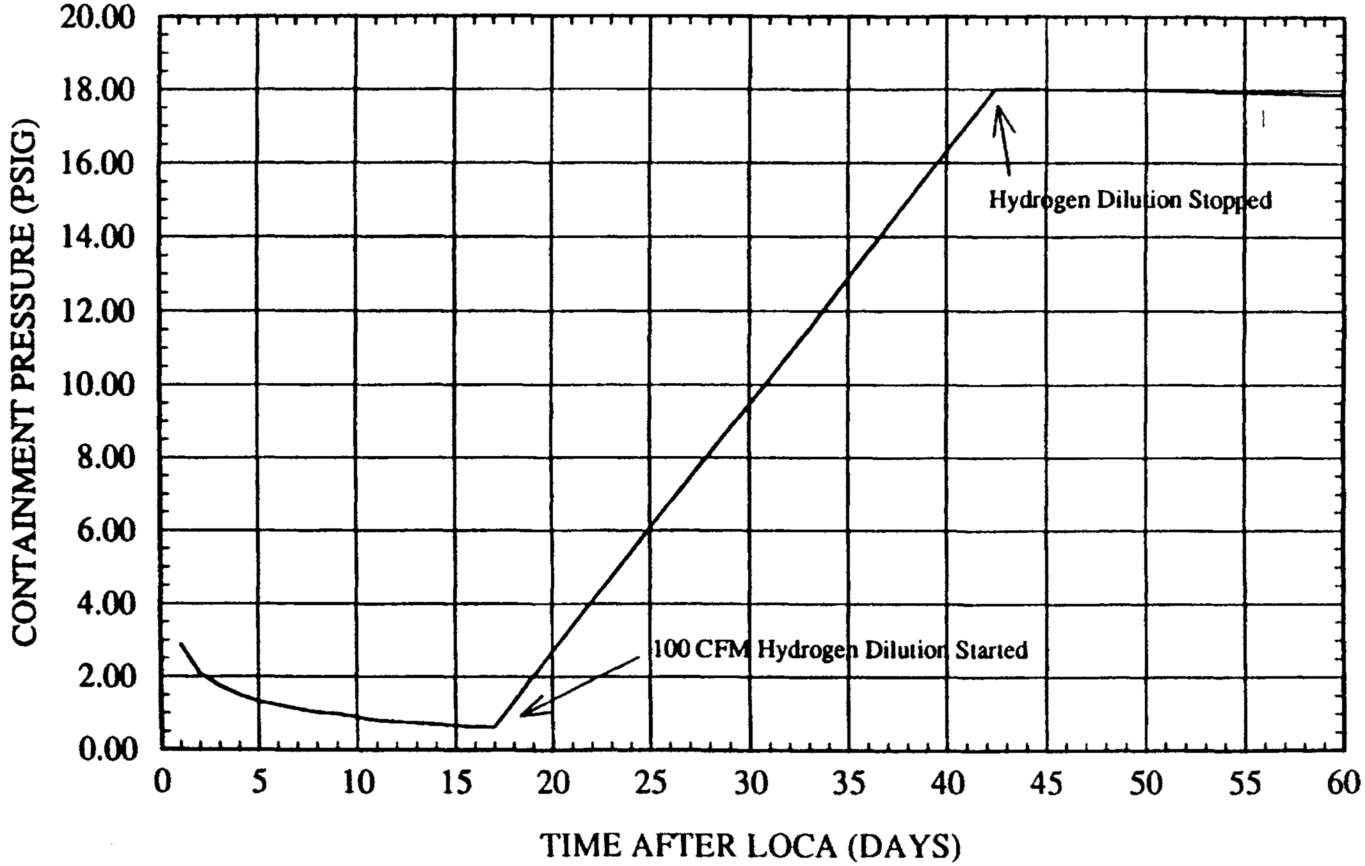
DAVIS-BESSE NUCLEAR POWER STATION
VENT AREAS BETWEEN THE ROOMS
SERVED BY EVS

FIGURE 6.2-43

REVISION 16
JULY 1992

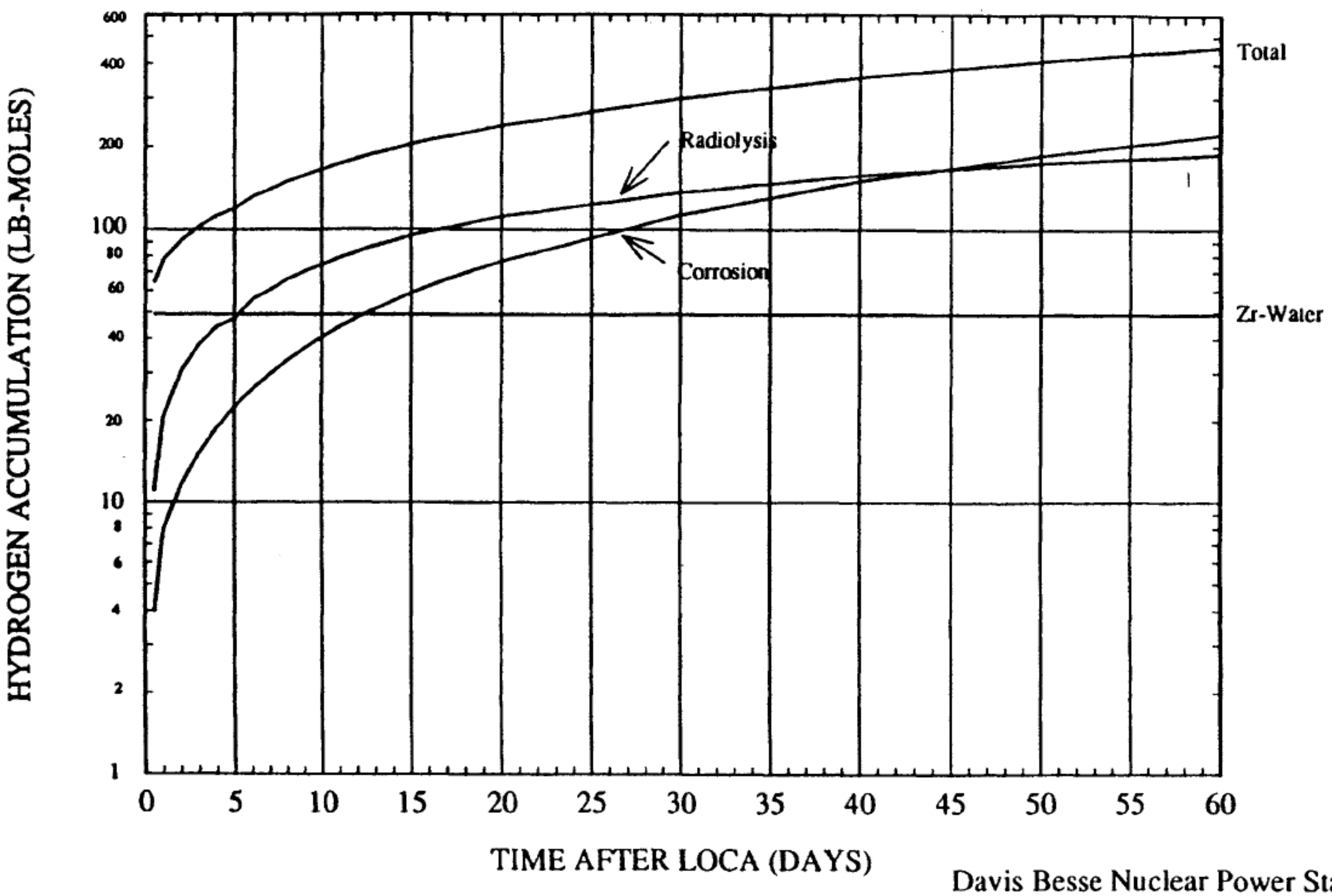


Davis Besse Nuclear Power Station
Hydrogen Concentration in Containment
Figure 6.2-45

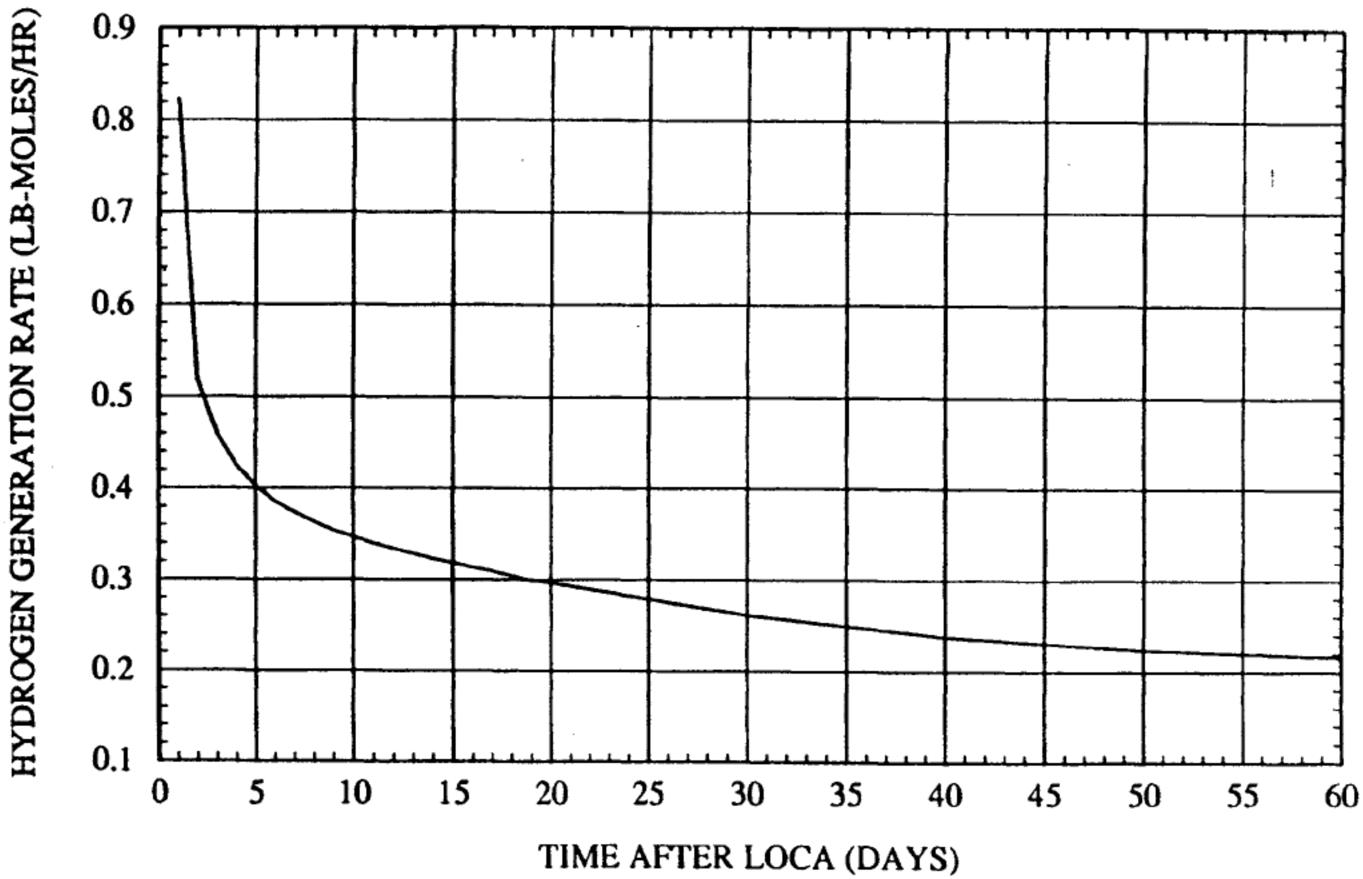


Davis Besse Nuclear Power Station
Containment Pressure with CHD
System in Operation
Figure 6.2-46

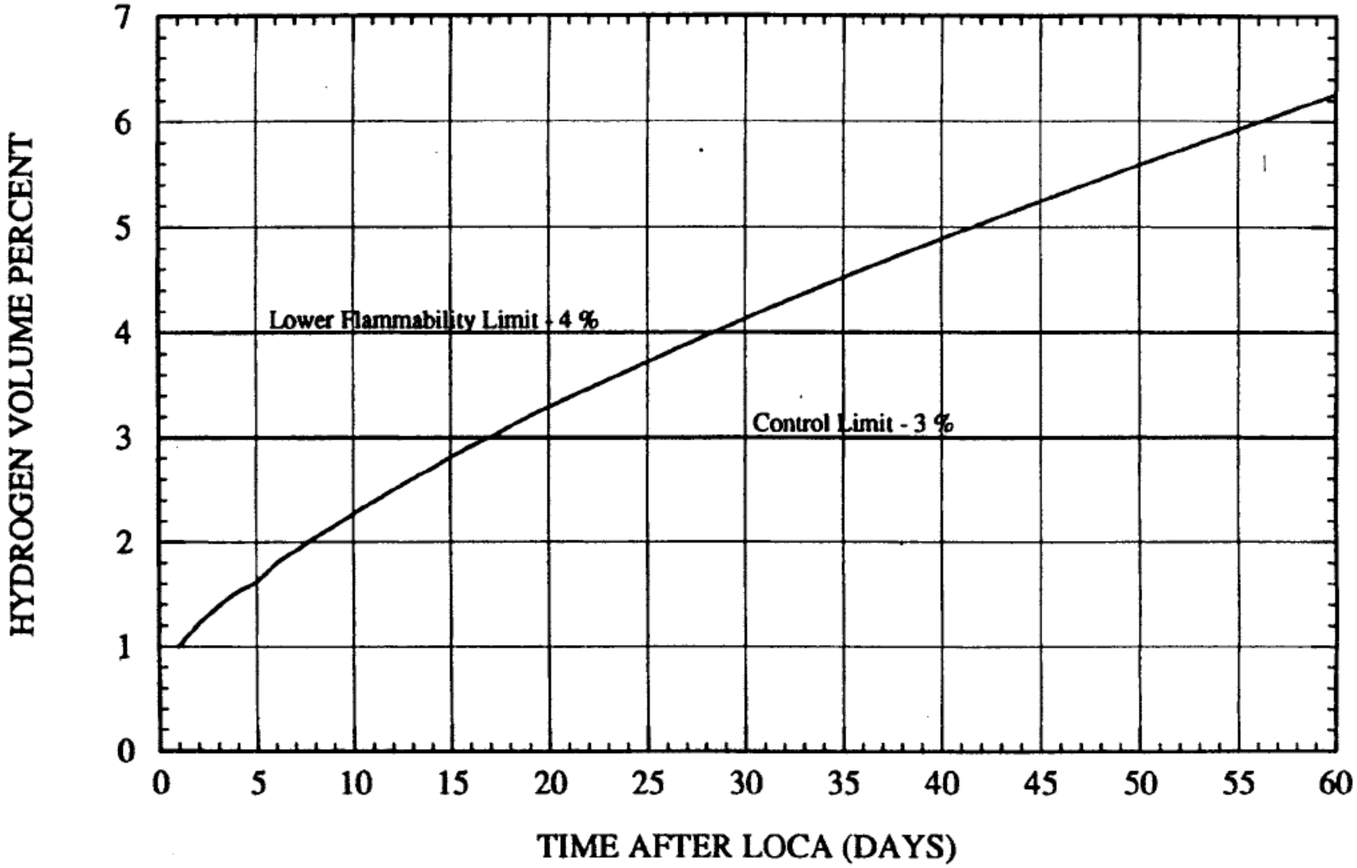
REVISION 18
NOVEMBER 1993



Davis Besse Nuclear Power Station
Post-LOCA Hydrogen Accumulation
Figure 6.2-48



Davis Besse Nuclear Power Station
Post-LOCA Hydrogen Generation Rate
Figure 6.2-49



Davis Besse Nuclear Power Station
Post-LOCA Hydrogen Percent by Volume
(Without Control Measures)
Figure 6.2-50

6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 Design Bases

6.3.1.1 Reactor Coolant System Rupture Spectrum

The Emergency Core Cooling System (ECCS) is designed to mitigate the consequences of all breaks of the Reactor Coolant System pressure boundary which result in loss of reactor coolant at a rate in excess of the capability of the Reactor Coolant Makeup System up to and including a break equivalent in area to the double-ended guillotine rupture of the cold leg or hot Reactor Coolant System pipe. The break spectrum also considers breaks in the HPI and Core Flood lines.

Loss of Coolant Accidents (LOCAs) are defined by 10 CFR 50.46 as hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, due to breaks in pipes of the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS. LBLOCAs correspond to a break size that will rapidly depressurize the RCS to containment pressure. SBLOCAs have been defined to be as small as a break that does not interrupt natural circulation, to as large as a break that depressurizes the RCS to nearly containment pressure. The threshold between SBLOCAs and LBLOCAs has historically been approximately 0.75 ft^2.

The large and small break LOCA spectrum was reanalyzed using the RELAP5/MOD2-B&W-based evaluation model (Reference 36, BAW-10192PA) as modified by Supplement 1. The analysis results and more information about the methodology summarized in Reference 35, demonstrate that the acceptance criteria of 10CFR 50.46 are met.

6.3.1.2 Fission Product Decay Heat

Core decay heat for the LOCA analyses are described in the LOCA evaluation model BAW-10192P-A as modified by Supplement 1. Calculations of the fission product decay heat are also made for emergency conditions and normal operation for sizing the decay heat removal coolers. The decay heat removal coolers are sized to handle the most restrictive load. Utilizing the curve for fission product decay published by K. Shure of Bettis, December 1961, the decay heat production can be determined as follows:

For emergency conditions the decay heat production at 30 minutes after shutdown is calculated as follows:

$$(0.019)(2817\ \text{MWt})\left(0.057\ \frac{\text{Btu/min}}{\text{watt}}\right)(60) = 183.0 \times 10^6\ \text{Btu/hr}$$

For normal operation the decay production at twenty hours after shutdown is calculated as follows:

$$(0.0063)(2817\ \text{MWt})\left(0.057\ \frac{\text{Btu/min}}{\text{watt}}\right)(60) = 60.7 \times 10^6\ \text{Btu/hr}$$

The coolers are sized to remove 60 x 10^6 Btu/hr with the reactor coolant at 140°F and the cooling water at 95° F. The coolers thus sized can remove 210 x 10^6 Btu/hr for emergency conditions as shown in Table 6.3-3. This exceeds the required capability of 183 x 10^6 Btu/hr.

Performance testing in 1988 showed that the overall heat transfer coefficient for the Decay Heat Removal Coolers under emergency conditions is below the design value assumed in the original analyses. DHR cooler characteristics for the revised overall heat transfer coefficient have been added in parentheses next to the original values in Table 6.3-3. The impact of degraded cooler performance on normal operation of the plant is discussed in Sections 9.1.3 and 9.3.5.

The tube side flow rate (reactor coolant) is set by the emergency core cooling low pressure injection requirements at 3000 gpm per cooler with the shell side flow set at twice this amount or 6000 gpm.

6.3.1.3 Reactivity Requirements

The reactor is in the cold shutdown condition when it is subcritical by at least 1 percent Δk/k and the average core fluid temperature is no more than 200°F. The boron concentration is set at the amount required to maintain the core 1 percent subcritical at 70°F while taking credit for 50% of the control rods' worth. This concentration is less than the minimum value specified in the Technical Specifications of 2600 ppm boron.

6.3.1.4 System Short- and Long-Term Capability

The Emergency Core Cooling System (ECCS) provides the capability to meet the functional requirements over both the short and long term duration of the accident. Following a LOCA, assuming a simultaneous loss of normal power source, the HPI and LPI systems are powered by the emergency diesel generators and will operate with no loss of function to maintain core cooling. The Core Flooding System (CF) will begin supplying water to the reactor when the RC system pressure falls below the CF Tank pressure. Separate and independent flow paths are provided in the ECCS, and redundancy in the active components ensures that the required functions will be performed if a single active failure occurs. Separate essential power sources are supplied to the redundant active components, and separate SFAS channels are used to actuate the system.

Although not considered part of the ECCS, AFW is also important in mitigating certain size- and location-specific small breaks. The limiting break location is one in which a portion of the HPI flow is not available for core cooling, i.e., a break in an HPI line or in the cold leg reactor coolant pump discharge piping downstream of the HPI injection nozzle. If the break size is too small to pass all of the steam produced in the core, the RCS pressure will increase. As the pressure increases, HPI flow will be reduced. AFW flow is important in filling the OTSGs and maintaining a level that is high enough to support boiler-condenser cooling (BCC). BCC, in combination with the break area, will limit the increase in RCS pressure, and will subsequently reduce the pressure to where HPI is sufficient to remove the residual energy in the system.

During the injection phase, the HPI system and LPI system will operate to provide full protection over the entire spectrum of break sizes. Operator action is required to trip the reactor coolant pumps following a loss of subcooling margin. Once HPI injection begins, manual operation is initiated to balance HPI flow between each of the injection lines if only one HPI pump is operating. The predicted reactor vessel mixture level may decrease below the top of the core for most break sizes. The HPI system, however, is sufficient to limit the fuel-clad heat up to less than 1,600°F (small-break), which is well below the acceptance criteria. As the postulated break

size is increased, the RC pressure will tend to decrease to lower levels because the break can pass all of the steam that is generated in the core. At the lower RC pressures, the CF and LPI systems along with HPI will inject borated water into the core and ensures adequate core cooling.

During the recirculation phase, the LPI system will recirculate the spilled reactor coolant and injection water from the containment emergency sump to the reactor vessel through the CF lines and/or the HPI line, if required, to maintain long-term core cooling and through the Decay Heat Removal Drop Line or Auxiliary Pressurizer Spray Line via the HPI pump for post-LOCA boron precipitation management.

For small breaks, the RC system pressure may be higher than the maximum LPI pump head at the time of Containment Vessel Emergency Sump water recirculation. Under these circumstances a crossover connection permits alignment of the HPI pumps to take suction from the outlet of the decay heat removal coolers to provide for Containment Vessel Emergency Sump water recirculation to the reactor core. The valves are motor operated with handswitches in the control room (see Figure 6.3-2A).

In the event of a LOCA, although the ECCS system is initially actuated by SFAS, some manual action is eventually required. Before the BWST is depleted, the operator is required to re-energize the BWST outlet isolation valves (DH7A and DH7B), the containment emergency sump recirculation valves (DH9A and DH9B), and the HPI Pump 1-2 Recirculation Stop Check Valve (HP-31). These valves are de-energized during plant Modes 1 through 4 to preclude an inadvertent change of position in the event of a fire. Re-energizing of these valves is required in order to switchover suction from the BWST to the emergency sump, as described in further detail below.

At a primary system pressure equivalent to the SFAS Low RCS pressure setpoint a SFAS signal starts the High Pressure Injection pumps and opens the High Pressure Injection discharge valves. The High Pressure Injection pumps are now taking suction from the BWST. If the primary system pressure continues to drop and reaches the SFAS RCS low low pressure setpoint or the containment pressure increases to a predetermined setpoint the Low Pressure Injection/Decay Heat (LPI/DH) removal pumps will start. The system will remain in this lineup until the level in the BWST drops to approximately 9 feet at which time a permissive signal is provided to allow the manual opening of the emergency sump valves after blocking the SFAS incident level 2 signal to the sump valves. Prior to transferring suction to the emergency sump, the HPI system is either shutdown or piggybacked to LPI. If the HPI system is piggybacked to LPI, the operator manually isolates the normal HPI minimum flow line to the BWST before taking suction on the emergency sump. In the case of certain very small break LOCAs where primary pressure can cycle above the discharge pressure of the HPI pump, the manual valves in the alternate minimum flow lines are opened before the normal minimum flow valves are closed. The BWST outlet valves close automatically when the sump valves open, thus switching suction to the emergency sump. After the transfer to the emergency sump is initiated, the operator verifies the transfer has started and that the transfer was completed properly in accordance with the plant emergency procedure (DB-OP-02000, Reference 19).

The rationale involved with the relative order of operation is that the switchover of suction from the BWST to the emergency sump occurs without loss of ECCS function, namely loss of pump suction. This objective is accomplished by opening the sump suction valves before the BWST valves are closed.

In addition to preserving ECC injection flow to the reactor vessel for long term cooling, operator action is required to limit the core boron concentration to prevent boron precipitation as soon as possible after the sump recirculation phase begins. Post-LOCA boron concentration management for Davis Besse is described in Section 6.3.3.1.2.1.

Reference 35 describes all operator actions that were credited in the large and small break LOCA analyses.

6.3.2 System Design

6.3.2.1 Functional Drawings and Logic Diagrams

Functional drawings of the Emergency Core Cooling System, showing components, piping, storage facilities and system interconnections are given in Figures 6.3-1A, 6.3-2 and 6.3-2A. Safety features Actuation signal logic diagrams are given in Chapter 7.

6.3.2.2 Equipment and Component Description

The components of the ECCS, with the significant design parameters, are listed in Table 6.3-1.

6.3.2.3 Codes and Classifications

The safety classifications are shown in Chapter 3, Table 3.2-2 (see Subsection 3.9.2). The ASME Code classifications for the major components are also listed in Table 6.3-1.

6.3.2.4 ECCS Materials

Materials of construction for ECCS components are noted in Table 6.3-1.

6.3.2.5 Bases for Design Temperatures and Pressures

The design pressures and temperatures for all components in the ECCS are shown on the system diagrams in this section and are listed in the attached Table 6.3-2. The bases for selection of the design temperature and pressure is to ensure that the ECCS operates satisfactorily within the design conditions. The design temperatures and pressures are so selected that they are not exceeded taking into account the maximum pressure and highest corresponding temperature actually experienced under any reactor operating conditions. The range of pressures and temperatures for the expected operating modes of the system are shown in Subsection 6.3.2.9.

The LPI lines and components are designed for normal reactor cooldown operating conditions since they are part of the Decay Heat Removal System.

These system pressure and temperature requirements are greater than those encountered during ECCS operation. The HPI system and CF system lines and components are designed for ECCS operation only, since these systems have no normal reactor operating function.

6.3.2.6 ECCS Coolant Storage

The Borated Water Storage Tank (BWST) is described in Subsection 9.3.5. The tank normally contains a minimum volume of 500,100 gallons of borated water at a minimum concentration of 2600 ppm boron. This ensures that a minimum of 360,000 gallons of borated water will be

available to provide ECCS injection to the core. During the winter, the tank contents are recirculated through steam heaters to maintain the temperature above 35°F at all times.

Each Core Flooding Tank contains a nominal volume of 7779 gallons (1040 ft^3) of borated water. In order to minimize the likelihood that the injected BWST fluid will be diluted post-LOCA, the minimum BWST and CFTs boron concentrations should be the same. Therefore, the CFT concentration is at a minimum concentration of 2600 ppm boron.

6.3.2.7 Pump Characteristics

The total dynamic head, NPSH, brake horsepower, and efficiency are shown in Figure 6.3-3 for the High Pressure Injection pumps. The total dynamic head, NPSH and brake horsepower are shown in Figures 6.3-4 and 6.3-5 for the low pressure injection/decay heat pumps.

Table 6.3-1 lists the original minimum head-capacity purchase specification for the HPI pumps while Figure 6.3-3 is the manufacturer's head-capacity curve for the pumps. For analysis purposes, the actual head capacity curve is used. Analysis dependent adjustments may be applied to the actual curve. Actual head capacity curves for the High Pressure Injection pumps are provided in plant procedures.

6.3.2.8 Heat Exchanger Characteristics

The decay heat removal coolers are designed to remove decay heat generated during a normal shutdown. In addition, each cooler is capable of cooling the injection water during the recirculation mode following a loss of coolant accident. The heat transfer characteristics of the decay heat removal cooler when cooling the core by recirculating the water from the Containment Vessel Emergency Sump are given in Table 6.3-3.

6.3.2.9 Process Information

The flow diagram for the ECCS is shown in Figure 6.3-6 with node points, located to describe emergency and test conditions. The temperature, pressure, and flow rates at these nodes are given in Table 6.3-4.

6.3.2.10 Relief Valve Capacity and Settings

The capacities, settings, and functions of the relief valves in the Emergency Core Cooling System are listed in Table 6.3-5.

6.3.2.11 Reliability Considerations

System reliability is ensured by the system functional design including the use of normally operating equipment for safety functions, test ability provisions, and equipment redundance; by proper component selection; by physical protection and arrangement of the system; and by compliance with the intent of the NRC General Design Criteria. There is sufficient redundancy in the Emergency Core Cooling System to ensure that no credible single failure can lead to significant physical disarrangement of the core. This is demonstrated by the single-failure analysis presented in Table 6.3-6. This analysis was based on the assumption that a major loss-of-coolant accident has occurred and coincidentally an additional malfunction or failure occurred in the engineered safety features system. For example, the analysis included malfunctions or failures such as an electrical circuit or motor failures, valve operator failures, etc. Although it is considered unlikely that valves would change to the opposite position

by accident if they were in the required position when the accident occurred, such inadvertent actuation has been included. Table 6.3-6 also presents an analysis of possible malfunctions of the Core Flooding Tanks that could reduce their post-accident availability. It is shown that these malfunctions result in indications that would be obvious to the operators, and so appropriate action can be taken. In general, failures of the type assumed in this analysis are considered highly improbable since a program of periodic testing is incorporated in the station operating procedures. The adequacy of equipment sizes in the ECCS is demonstrated by the post-accident performance analysis described in Subsection 6.3.3. This analysis shows that only one High Pressure Injection pump, one low pressure injection/decay heat pump, and one decay heat removal cooler in combination with both Core Flooding Tanks is required to protect against the full spectrum of break sizes.

An analysis of a double-ended rupture of a Core Flooding Tank line, postulated to occur between the reactor vessel and the first valve in that line, has been performed and is reported in Subsection 6.3.3.1.3. Minimum ECCS, one Core Flooding Tank and one High Pressure Injection pump, was assumed, and no adverse consequences were calculated to occur.

Crossover lines have been added to the Low Pressure Injection system to connect the two trains. Remote motor operated valves (switches in the control room) in the crossover lines can be actuated to provide additional protection in the unlikely event of this break.

Special design features have been incorporated to prevent spurious operation of the core flood line isolation valves. Redundant position indicators are supplied on each valve; the valves are automatically opened before reactor coolant pressure exceeds 800 psig and valves cannot be closed with reactor coolant pressure above 800 psig.

During Emergency Core Cooling System (ECCS) operation, the suction flow for the High Pressure Injection (HPI), Low Pressure Injection (LPI), and Containment spray pumps is initially provided by the Borated Water Storage Tank (BWST). Prior to the exhaustion of the BWST, LPI and Containment spray suction flow is transferred to the Containment Vessel Emergency Sump. Before this transition, the operator will know if he needs to cross-connect the LPI and HPI systems by the LPI flow indicator and by the low flow alarm. If LPI flow is not above a predetermined rate, the operator will connect the LPI/DH pumps to operate as booster pumps for the HPI pumps. This connection is necessary to ensure adequate NPSH for the HPI pumps to operate from the Containment Vessel Emergency Sump and results in continued ECCS flow from the Containment Vessel Emergency Sump even though RC system pressure may be above the discharge pressure of LPI system. Since LPI injection is not flowing in this situation, it will take at least 100 minutes to empty the BWST with both HPI and Containment Spray pumps running at design capacity, which gives the operator an ample amount of time to determine the need for, and, perform the required actions. Motor operators on the valves and hand switches in the control room are installed to allow alignment from the control room.

In the event of a very small break LOCA (0.00206 to 0.0045 ft^2), RCS cyclic repressurization above the shut-off head of the HPI pump could exist beyond the time that the pump suction is aligned to the BWST. As discussed above, the ECCS pump suction will be swapped from the BWST to the containment sump after the contents of the BWST are exhausted. The normal HPI minimum flow line which returns to the BWST is isolated upon swapover to the containment sump. An operator action is required to open the valves in the alternate minimum flow lines which discharge to the outlet of the Decay Heat pumps, and to disable the close function of the piggyback valves. This will ensure that the HPI pumps have a minimum flow line available tin the event of RCS repressurization events.

The possible flow paths and flow rates are identified by the following nodes and can be traced using the Davis-Besse Nuclear Power Station Emergency Core Cooling System Flow Diagram, Figure 6.3-6.

Low Pressure Injection with Suction from Containment Vessel Emergency Sump

Node No.	Press. (psig)	Temp (°F)	Flow* (gpm)
3	36-ATM	250-120	4300
5	36-ATM	250-120	3000
6	150	180-120	3000
8	150	180-120	3000

High Pressure Injection with Suction from Containment Vessel Emergency Sump (via LPI Pumps)

3	36-ATM	250-120	1800
5	36-ATM	250-120	500
6	186	180-120	500
10	186	180-120	500
11	1358	180-120	500
12	1358	180-120	250

* These are nominal values, flow will vary with various RCS pressure.

The concept of low pressure flow equalization by use of operator action has been introduced into the Davis-Besse NSSS Design to supply abundant core cooling in the event of a break in a core flood line coincident with a loss of offsite power. The analysis of the core flood line break, presented in Subsection 6.3.3.1.3, was performed assuming minimum ECCS available (1 HPI and 1 CFT). No credit was taken for operator action to open the LPI cross-over lines. With these assumptions, the core remained covered and the peak fuel clad temperature remained near the corresponding saturated fluid temperature.

Analyses of the core flood line break using minimum ECCS have been shown to produce acceptable results with no significant increase in cladding temperatures. Due to the low-head, high flow HPI Pumps, the LPI crossover lines, for which no time requirements for operator action are stipulated, represent an additional safety margin. A faster refill of the vessel with water can be achieved with the manually operated LPI cross-over lines which would fulfill the requirements of abundant core cooling.

In order to provide bearing cooling to the High Pressure Injection (HPI) pumps an external circulating lube oil system is provided for the HPI pumps thrust bearings. The forced lube oil system is of a closed-loop, water-cooled design. Two lube oil pumps are included: one 460VAC/60Hz 3 phase and one 125VDC. Normal operation utilizes the AC pump and motor. The DC pump and motor is actuated by a flow switch when the lubricating system oil pressure decreases below a set point value. The same flow switch also shuts off the DC pump and motor during system startup when both the AC and DC pumps and motors are operating and the lubricating oil system flow is greater than a given set point. The lubricating oil system is complete with pressure gauges, thermometers, reservoirs, heat exchangers, and valves necessary to adequately cool the high-pressure injection pump bearings.

Separate essential power sources provide power to the separate and redundant strings of the HPI and LPI systems as discussed in Chapter 8.0. Separate instrument channels are used to actuate each string within the HPI and LPI systems as discussed in Section 7.3. The core flooding system is self-contained, self-actuating, and passive in nature and is divided into two separate strings with a Core Flooding Tank in each string.

Normal operating procedures require that both Core Flooding Tanks be available to supply borated water to the reactor vessel in the event of a loss-of-coolant accident. A loss of a Core Flooding Tank nitrogen pressure during normal operation would require plant shutdown using normal shutdown procedures.

Figure 6.3-8 provides a block diagram summary of the sequence of events following a postulated pipe break within containment. This figure demonstrates that, with the redundant High Pressure Injection subsystems, the redundant Low Pressure Injection subsystems, and a passive core flood system, adequate initial and extended core cooling is provided, even in the event of a single active failure. The redundancy of the High Pressure Injection subsystems and low pressure injection subsystems is shown in Figures 6.3-2, 6.3-2A, and 6.3-6. Required support systems for the High Pressure and Low Pressure Injection Systems (ventilation, cooling water, and electric power) are redundant.

Figure 6.3-8 also shows the following details:

a. Reactor trip. Pressure relief is not a required safety action during LOCA mitigation.

b. The Core Flooding Tanks are assumed in the small break analysis to discharge to the RCS at the appropriate pressure and hence are included in the diagram.

c. It is impossible to define the exact sequence of events for every possible break size. The break sizes shown are those which have been analyzed as part of the spectrum of breaks considered in the ECCS analysis of the plant.

d. The analysis for the core flooding line break, as described in Subsection 6.3.3.1.3 (reference Figure 6.3-11), does indicate depressurization below the low Pressure Injection system shutoff head in the short term. Because of this, the Low Pressure Injection system has been included on the diagram at this point.

e. The SFAS channels.

f. The Auxiliary Feedwater System.

Auxiliaries required for each engineered safety feature are listed below:

Engineered Safety Feature	Auxiliaries Required
HPI	
	1. Essential Power
	a. Offsite

b. Onsite

1. emergency diesel
2. air starting system (tanks)
3. fuel oil supply (day tanks)

2. Borated Water Storage Tank water in proper amount and at proper boron concentration

3. ECCS room coolers

4. Service Water System

5. Component Cooling Water System

6. HPI-LPI crossconnect piping

7. Instrumentation to monitor system performance

LPI

1. Essential Power

a. Offsite

b. Onsite

1. emergency diesel
2. air starting system (tanks)
3. fuel oil supply (day tanks)

2. Borated Water Storage Tank water in proper amount and at proper boron concentration

3. ECCS room coolers

4. Service Water System

5. Component Cooling Water System

6. SFAS to provide an alarm and permissive to alert control room operator that BWST level has dropped to approximately 9 ft. so that operator can manually transfer LPI suction from the BWST to the Containment Emergency Sump.

7. Deleted

8. Instrumentation to monitor system

CFT

No auxiliaries required

6.3.2.12 Missile and Flooding Protection

Missile Protection:

Protection against missile damage is provided by either direct shielding or by physical separation of duplicate equipment. The major active components of the ECCS are external to the Containment Vessel and therefore are not exposed to post-accident CV environment. Since most of the ECCS piping within the containment is located outside the primary and secondary shield, it is protected from missiles originating from these areas. ECCS piping that is not fully protected against LOCA missile damage, utilizes dual lines to preclude loss of the protective function. Missiles that may be generated in one cubicle cannot rupture ECCS lines in the other loop.

The HPI lines enter the Containment Vessel through penetrations in different sections of the vessel. The discharge from each HPI line splits into two lines outside the Containment Vessel to provide four injection paths to the RC system. The four connections to the RC system are located between the reactor coolant pump discharge and the reactor vessel inlet nozzles. Four injection lines penetrate the secondary shield wall so that the effect on injection flow is minimized in the unlikely event of missile damage to an injection line inside the secondary shield wall.

Protection from missiles is given to the low pressure injection lines within the Containment Vessel. The portion of the Low Pressure Injection system located in the Containment Vessel consists of two redundant injection lines which are connected to injection nozzles located on opposite sides of the reactor vessel. Both are located outside of the secondary shielding and is additionally protected by a grating.

The Containment Vessel Emergency Sump, constructed of concrete, is located inside the Containment Vessel on El. 565 feet 0 inches. As shown in Figure 3.6-4, the sump is protected from missiles by the secondary shield wall, the refueling canal wall, the floor at El. 578 feet 0 inches, and the Containment Vessel wall. The strainers over the sump are constructed of tubular steel framing and grating to which 3/16 inch diameter stainless steel perforated plate is attached. The strainers are described in Subsection 6.2.2.6.2 and are shown in Figures 6.2-33 and 6.2-33A. The Incore tunnel portion of the Emergency Sump Strainer is also supported by tubular steel. It is protected from missile hazards by surrounding concrete structures. In the event of a LOCA in the vessel cavity area the sump strainer is designed to support recirculation without the Incore tunnel portion of the strainer intact.

Both the Containment Vessel Emergency Sump, its strainers and the anti-vortex grating are Seismic Class I and Q-listed.

The entire Core Flooding System is located within the Containment Vessel. The Core Flooding Tanks and two of the three valves in each core flooding lines are located outside of the secondary shield wall.

Flooding Protection:

A total of three (3) sumps are available in the two ECCS rooms and the DH cooler room. Each sump has one (1) duplex sump pump, and two (2) level switches. One level switch is for level control and the other switch is for high-high level alarm.

The first sump pump will start automatically when sump level reaches 2'-0" above the sump bottom. The second pump will start when the sump level reaches 2'-4". Qualified pump running lights are provided on the main control board in the control room for each of the ECCS room sump pumps. If the level reaches the high level set point of 2'-8", an alarm will be actuated in the control room. The electrical source of the sump pumps and control are from the essential power supply.

Each ECCS room essential component, i.e., pumps, heat exchangers, control valves, can be individually isolated by closing the isolation valves (manual or power operated). The power operated valves have position indication in the control room.

6.3.2.13 Performance Testing

Provisions are made to facilitate performance testing of ECCS components during operation of the station. In the LPI system, test flow paths are available for recirculation of water from and to the BWST to demonstrate the operability of the LPI/DH removal pumps. The LPI/DH pumps are tested singularly for operability by opening the BWST outlet valves and the test line valves to the BWST. The pumps take suction from the BWST and discharge through the test flow path to the BWST.

In the HPI system, a recirculation line from each of the HPI pumps is available to test the delivery capability. The pumps take suction from the BWST and discharge through the test line back to the BWST.

6.3.2.14 Net Positive Suction Head Requirement

The available NPSH for the High Pressure Injection and low pressure injection/decay heat pumps have been calculated to include a safety margin above the requirements of these pumps. The calculation has assumed conservatively that minimum level exists in the Borated Water Storage Tank and in the Containment Vessel Emergency Sump. Following are the bases used for the NPSH calculation during the post LOCA recirculation phase in accordance with Safety Guide 1:

a. As-Built piping drawings;

b. Pipe and fittings losses calculated using the information in Crane Technical Paper No. 410 except for the pipe suction fitting, which uses information from Flow Resistance: A Design Guide for Engineers, by Fried and Idelchik.

c. The friction loss for the sump strainer is based on information in Crane Technical Paper No. 410 and Flow Resistance: A Design Guide for Engineers, by Fried and Idelchik.

d. Total flow in a single train at the maximum flow for each pump;

e. Pressure above water surface of the sump is equal to vapor pressure of pumping fluid at the pump inlet;

f. $NPSH_a$ is calculated by the following general expression.

$$NPSH_a = P_a + (E_a - E\rho) - H_f - \frac{Pv}{\rho}$$

Where, $NPSH_a$ = Available net positive suction head, ft.

P_a = Pressure above water surface, ft.

E_a - E_p = Elevational difference between water surface and pump center line, ft.

H_f = Friction losses in suction piping, ft.

$\frac{Pv}{\rho}$ = Vapor pressure of the pump inlet, ft.

The table below shows the required and available NPSH for the various safety features pumps:

Pumps	BWST Required NPSH ft		BWST Minimum Available NPSH, ft		Emergency Sump** Required NPSH ft		Emergency Sump** Minimum Available NPSH, ft	
	Train 1	Train 2	Train 1	Train 2	Train 1	Train 2	Train 1	Train 2
LPI/DH	14.3	11.0	51.3	46.1	11.9	9.8	14.4	13.2
High Pressure Injection	30.3	33.9	48.4	40.8	33	33	156*	173*
Containment Spray	16.8	14.5	50.3	45.1	11.1	10.9	16.9	15.5

* If required in the recirculation mode, the High Pressure Injection Pump takes suction from the discharge of the low pressure injection/decay heat pump.

** After 30 days, NPSH available will be reduced by 0.21 ft due to assumed ECCS leakage outside containment.

6.3.2.15 Core Flooding Tank Isolation Valve Control Circuits

The discharge pipe from each Core Flooding Tank (CFT) is attached directly to a reactor vessel core flooding nozzle. Each core flooding line at the outlet of the CFT contains an electrically operated isolation valve adjacent to the tank and two in-line check valves in series. The isolation valves at the Core Flooding Tank outlet are open when RCS pressure is above 800 psig. During power operation when Reactor Coolant System pressure is higher than the core flooding system pressure, the two check valves in the line to the CFT prevent high pressure reactor coolant from entering the Core Flooding Tanks.

The control circuits for the motor operated isolation valves include the following features:

a. Position switches on each valve actuate open and close valve position indication for each valve. The indicators are located in the control room.

b. Two separate alarms, one for each valve, are actuated if a valve is open and reactor coolant pressure is reduced to a valve that could cause emptying of the Core Flooding Tanks; these alarms alert the operator to an impending situation where he could inadvertently discharge the Core Flooding Tanks during station shutdown.

c. Two independent computer alarms, one for each valve, are provided from contacts on the motor operator to indicate when the valve is not fully open and from two-out-of-four wide-range reactor coolant pressure sensors when they sense a pressure in excess of 700 psig. Two redundant computer and station annunciator alarms are also provided, one for each valve, using contacts mounted on the yoke of the valve and redundant, independent reactor coolant pressure signals to indicate when the valve is not fully open and pressure is greater than 735 psig.

d. If the isolation valve has not been opened by the operator previously, the SFAS will automatically give an opening signal to the valve before exceeding 800 psig. If the valve does not open, an alarm will sound.

e. The isolation valve is interlocked to prevent inadvertent closing when reactor coolant pressure is above 800 psig. The open circuit is not inhibited by this interlock.

f. After the Core Flooding Tank isolation valves are fully open, the breaker of the combination starter of each isolation valve is manually tripped open and padlocked.

   The tripped position of the breakers is monitored on the main control board by one blue indicating light for each breaker.

With the source of power to the motor operator padlocked, there is no possibility of the valves closing with the reactor at power.

Additional limit switches, independent of those provided in the motor housing, are furnished for the Core Flooding Tank isolation valves. The switches were added to assure accurate indication of the valve position. These switches are yoke mounted and actuate on actual stem travel, thereby providing a redundant means of indicating and alarming in the control room if the valve is not wide open and the reactor coolant pressure is above 735 psig.

Assuming that the unit is undergoing cooldown from the hot standby condition, the following events will take place:

As RC pressure decreases below 675 psig, the alarm is actuated in the control room if the operator has not closed the valves prior to this pressure. The operator would then close the valves to deactivate the alarms. After closing the valves power is removed.

With power removed, the possibility of the valves opening and causing either the pressure-temperature limits of the RC system or the design pressure limits of the DHR system to be exceeded, is precluded. Power may be restored and the valves opened during shutdown when the CFT has been depressurized to the point where the pressure limits will not be exceeded.

Assuming that the unit is undergoing startup from a depressurized condition, the following events will take place:

As RC pressure increases, the alarm is actuated in the control room if the operator has not opened the valves before 700 psig. At this time, the operator would open the valves to deactivate the alarms. If the valves are not opened manually, they will automatically open before exceeding 800 psig RC pressure. Once the valves are opened, power is removed.

When power is removed from the valves, in the cases described above, the breaker of the combination starter of each isolation valve will be manually tripped open and padlocked. The tripped position of the breakers are monitored on the main control board by one blue indicating light for each breaker.

6.3.2.16 Decay Heat Removal System Valve Control Circuit

The design of the Decay Heat Removal System includes interlocks on each of the two high-pressure motor-operated valves in the suction line from the Reactor Coolant System (Valves DH-11 and DH-12). These independent controls are designed to automatically close the valves or to prevent the opening of the valves when the RC pressure is above the design pressure of the DH suction piping. This prevents overpressurizing the DH system in the event the valves are inadvertently left open during heatup or if an operator prematurely tries to open the valves during cooldown. These interlocks are explained further in Chapter 7. Control power is removed from DH-11 and DH-12 when the decay heat removal system is in operation to prevent inadvertent closure during cooldown. In addition, administrative controls require that power be removed from either DH-11 or DH-12 when the plant is in Modes 1, 2, and 3 when these valves are closed and RCS pressure is higher than the Decay Heat Removal System design pressure. These administrative controls preclude inadvertent or spurious opening of at least one of these valves during certain plant conditions, such as a fire.

In addition to the suction valve interlocks, there are relief valves provided in the suction line to protect the DHR system from overpressure. See Subsection 9.3.5.5 for further details.

6.3.3 Performance Evaluation

All components of the Reactor Coolant System (RCS) have been designed and fabricated to ensure high integrity and thereby minimize the possibility of the failure. The system, the safety factors used in its design, and the special provisions taken in its fabrication to ensure quality are described in Chapter 5.

However, in the unlikely event of a RCS piping failure, emergency core cooling is provided to ensure that the core continues to be cooled and does not lose its geometric configuration. This engineered safety features system is provided by the core flooding system and two independent, full-capacity HPI and LPI systems. While not considered as part of the ECCS, SG cooling provided by auxiliary feedwater is also required for successful SBLOCA mitigation.

For the spectrum of break sizes considered, the results of the analysis show that the requirements of 10CFR50.46 are met.

6.3.3.1 Results of Analysis

6.3.3.1.1 Evaluation Model (EM)

Original FSAR:

The computer model CRAFT (Model for Equilibrium LOCA Analysis) was originally used to describe the Loss of Coolant Accident (LOCA), described in BAW-10104PA. All other methods and assumptions that were used are described in BAW-10034, Rev. 3 (May, 1972). Subsequent LOCA analyses in accordance with AEC ECCS "final acceptance criteria" were performed with methods and assumptions as described in BAW-10104, Rev. 5 (November 1988) and BAW-10105, Rev. 1 (July, 1975).

Post-TMI LOCA:

After the TMI incident SBLOCA analyses were placed under additional scrutiny and more detailed analyses were added. For the Davis-Besse raised-loop design (SGs elevated with respect to the RV), which includes low-head high-flow HPI pumps, successful SBLOCA mitigation was accomplished with balanced flow from one HPI pump through two injection lines. Some clad heatup was experienced, but the peak clad temperatures were well below those calculated for LBLOCA.

Current Analysis:

The current Small Break (SB) LOCA and Large Break (LB) LOCA analyses use the NRC-approved methods contained in Volume 1 of BAW-10192P-A as modified by BAW-10192P-A, Supplement 1P-A.

Since the approval of BAW-10192P-A, the codes and methods have evolved through approved code revisions, identification of specific codes not identified in the EM, and the addition of new methods and error corrections made under 10 CFR 50.46. The following NRC-approved topical reports have been added as part of the EM methodology for SBLOCA and LBLOCA analysis. For information about the code topicals see Table 15.1-1.

- BAW-10192P-A Rev. 0, Supplement 1P-A Rev. 0, EM Supplement

The BAW-10192P-A, Supplement 1P-A updates the EM methodology to account for fuel burnup-dependent TCD at all TILs. Changes to the TACO3 and GDTACO methods of analyses and the approach for increasing the VAFT uncertainties are made to assure that the burnup-dependent fuel TCD is adequately accounted for at moderate to high burnups. The approaches described in this supplement alter or replace certain fuel pin calculation techniques for TIL analyses on the B&W-designed plants with BAW-10192P-A and associated codes.

The LBLOCA analyses used several EM changes and error corrections made by AREVA/Framatome under 10 CFR 50.46 to ensure that Appendix K of 10 CFR 50 requirements are met. Those 10 CFR 50.46 changes have not been approved within a revised topical report. These included use of:

1. Uncertainty adjusted core flood tank parameters (PSC 5-94) discussed in the 1994 and 1995 Draft B&W Annual ECCS Report,

2. LBLOCA Reactor Coolant Pump (RCP) two-phase degradation modeling (PSC 1-99) discussed in the 1998 and 1999 Draft B&W Annual ECCS Reports,

3. Modeling upper plenum column weldments discussed in the 177 fuel-assembly plant Request for Additional Information (RAI) responses to a 30-Day 10 CFR 50.46 report of significant PCT change.

4. Updated M5® Swelling and Rupture Model (SRM).

The SBLOCA analyses used several EM changes made by AREVA/Framatome under the NRC regulation, 10 CFR 50.46, to ensure that 10 CFR 50 Appendix K requirements of that regulation are met. Those 10 CFR 50.46 changes that have not subsequently been approved within a revised topical report include use of:

1. Uncertainty-adjusted core flood tank parameters (PSC 5-94) discussed in the 1994 and 1995 Draft B&W Annual ECCS Reports.

2. SBLOCA RCP two-phase degradation modeling (PSC 2-00) was described in the 2000 and 2001 B&W Annual ECCS Reports. The SERs relating to PSC 2-00 are not associated with a specific topical report. The original SER imposed a limitation that required that the two-phase degradation model used in the SBLOCA analyses be demonstrated to the NRC to justify application of the pump model to the B&W plants. Additional information was provided to the NRC to justify generic applicability of the model to the B&W plants. In response to this information, the NRC revised the SER to remove the limitation. Therefore, the results of PSC 2-00 and associated SER are generically applicable to the B&W plants.

3. The prescribed SBLOCA axial power shape in the EM, which is peaked at the 9.536-ft elevation, is bounding at Beginning of Cycle (BOC), but it is not limiting for EOC conditions. Since no explicit burnup studies are performed, the limiting EOC peaking at 11-ft is used for all SBLOCA analyses to bound the power shape at any time-in-cycle.

4. Detailed Column Weldment (CW) model in the upper reactor vessel head as discussed in the 2012 Annual ECCS Report and addressed in the NRC RAIs.

5. Updated M5® SRM. The SBLOCA analyses contained within this report do not include this update in the analysis inputs. However, the limiting PCT for SBLOCA is below the rupture temperature where the new SRM will change the results. Therefore, the new SRM is effectively included in the SBLOCA results.

Beginning with Cycle 13, the generic LOCA EM for both large and small breaks has changed to BAW-10192P-A (Reference 36). The results of these LOCA analyses with the boundary conditions and assumptions are described in Reference 35. The LOCA EM was modified in Cycle 21 by BAW-10192P-A, Supplement 1P-A, Reference 48, to address identified concerns including fuel thermal conductivity degradation.

The RELAP5/MOD2-B&W code (Reference 37, BAW-10164P-A) calculates system thermal-hydraulics, core power generation, and the clad temperature response during blowdown. The REFLOD3B code (Reference 38, BAW-10171P-A) determines the length of the refill period and the core flooding rate during reflood. BEACH (Reference 39, BAW-10166P-A), which is the RELAP5/MOD2-B&W core model with the reflood fine-mesh rezoning option

activated, determines the clad temperature response during the reflood period with input from REFLOD3B. The CONTEMPT code (Reference 40, BAW-10095A) is used to determine the minimum containment pressure response based on the mass and energy release from the RCS as predicted by RELAP5 and REFLOD3B.

BAW-10192P-A, Supplement 1P-A (Reference 48) updates the EM methodology to account for fuel burnup-dependent thermal conductivity degradation (TCD) at all times in life. Changes to the TACO3 and GDTACO methods of analyses and the approach for increasing the volue averaged fuel temperature uncertainties are made to ensure that the burnup-dependent fuel TCD is adequately accounted for at moderate high burnups. The approaches described in this supplement alter or replace certain fuel pin calculation techniques for time in life analyses.

The application of the EM, which provides a demonstration of the methods (based on fuel and fuel cycle designs at the time of approval), shows that all the requirements of 10CFR50.46 are met. The latest LOCA analyses that have been performed for Davis Besse are summarized in Reference 35 and are based on the Mark-B-HTP fuel design. The LOCA limits for each fuel cycle are described in the cycle-specific reload report, Appendix 4B.

6.3.3.1.2 Results of Analysis (Large Break)

The ECCS is designed to accommodate a continuous range of rupture sizes, from the smallest pipe connected to the reactor coolant system (RCS) up to a double-ended rupture of the largest pipe at any location in the reactor coolant system. A LOCA occurs as the result of postulated rupture of the primary coolant piping. LOCAs can be categorized as being either small or large depending on the cross-sectional area of the break. LOCAs have typically been defined as small when the break cross-sectional area is approximately 0.75 ft^2 or less and the rate of loss exceeds the capability of the MU system. LBLOCAs are defined as large when the break cross-sectional area is greater than approximately 0.75 ft^2 and up to a double-ended rupture of the hot leg pipe. Analyses indicate that piping ruptures at the Reactor Coolant Pump (RCP) discharge impose the most severe requirements for the ECCS because a portion of the HPI flow is lost through the break and is not available to provide core cooling. Studies have been performed that show SBLOCAs do not exhibit the same large break phenomena such as ECCS bypass and core reflood. These phenomena are related to the break flow during blowdown, which is a function of the critical break flow model being used. If the critical flow model is changed, the transition break size between large and small breaks will also change.

The evaluation of the Babcock & Wilcox-designed Nuclear Steam System, with a core of 177 fuel assemblies and a loop arrangement in which the steam generators are raised in relation to the reactor vessel, during a hypothetical loss-of-coolant accident (LOCA) is presented in Reference 35. This report describes the application of the RELAP5-based LOCA evaluation model, with appropriate sensitivity studies as required by BAW-10192P-A as modified by Supplement 1. A complete set of calculations are performed to determine the linear heat rate (in kW/ft) limits, as a function of core elevation and time in core life, that result in a peak cladding temperature (PCT) of less than 2200°F. The LOCA limit analyses conservatively covers all possible maximum local powers that can be encountered during actual operation. Calculations are also presented to demonstrate that the Davis-Besse 1 NSS complies with the five acceptance criteria of 10CFR 50.46 for ECCS. In brief, the five acceptance criteria and a general overview of the compliance are as follows:

a. The peak cladding temperature (PCT) shall not exceed 2200°F.
   Compliance: A spectrum of breaks is evaluated, and allowable linear heat rates as a function of core elevation for the most limiting break are determined. These results, form part of the data base from which administrative controls and

procedures during power operation are established. The conservative nature of the evaluation model combined with the imposed restrictions during power operation ensure the effectiveness of the ECCS to limit cladding temperatures to values less than 2200°F in the unlikely event of a LOCA. Operating limitations based on LOCA analysis are analyzed for each fuel cycle and results are summarized in the Reload Report. The most current Reload Report is included in USAR Appendix 4B.

b. The percentage of local cladding oxidation shall not exceed 17 percent.
Compliance: The analysis performed to satisfy the first criterion also provides the percentage of local cladding that oxidizes. These results show that oxidation of the hot pins is less than 17%.

c. The percentage of hydrogen generation resulting from whole-core cladding oxidation shall not exceed 1%.
Compliance: A calculation is performed, which conservatively estimates local oxidation versus local power. The results of this evaluation are integrated over the whole-core power distribution, resulting in whole-core hydrogen generation less than 1%.

d. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
Compliance: The changes in core geometry calculated during the analyses in support of the first criterion are examined to ensure that no gross core blockage or disfiguration will occur. Changes in the cladding swelling models (Reference 15, NUREG-0630) and fuel pin performance analyses have been incorporated into the determination of LOCA operating limits as described in the Reload Report (USAR, Appendix 4B).

e. The mode of long-term core cooling shall be established.
Compliance: The analysis for the first criterion is continued until the cladding temperatures at all locations in the core are decreasing, the fluid level in the core is rising, and a clear path to long-term cooling is foreseen. In addition, analyses are performed and procedures written to ensure that the chemical additive concentration in the reactor vessel will remain dilute throughout the post-accident cooling period. Section 6.3.3.1.2.1 provides the discussion of the establishment and maintenance of long-term core cooling.

Large break loss-of-coolant accidents can be treated analytically in three separate phases: blowdown, refill, and reflood. The blowdown phase is characterized by the rapid depressurization of the reactor coolant system to a condition nearly in pressure equilibrium with its containment surroundings. Core flow is variable and dependent on the nature, size, and location of the break. For the limiting double-ended guillotine Cold Leg Pump Discharge break, Departure from nucleate boiling (DNB) is calculated to occur very quickly at the high power locations, and core cooling is by a film boiling process. Since film boiling accounts for only a small fraction of the core decay heat cooling, the cladding temperature increases. CFT flow begins after the RCS depressurizes below the CFT fill pressure. Steam condensation caused by the CFT liquid accelerates the negative core flows and reduces the fuel pin temperatures during the middle blowdown period. During the last phases of blowdown, cooling is by convection to steam, and the cladding temperature begins to rise again.

Following blowdown, a period of time is required for the CFTs to refill the bottom of the reactor vessel before final core cooling can be established. During this period, core cooling is marginal, and the cladding experiences a near-adiabatic heatup. This period is designated as the refill phase. When the water level reaches the bottom of the core, the reflood phase begins. Core cooling is by steam generated below the rising core water level with entrained liquid droplets that help limit the steam superheating. The cladding temperature excursion is generally terminated before a particular elevation is covered by two-phase mixture level. A two-phase mixture eventually covers the core, and the path to long-term cooling is established through initiation of LPI near the time that the CFTs empty (Reference 35) and subsequent operator action to maintain pumped injection through continued LPI injection. The LPI injection began following a conservative pumped injection ECS delay time that allows flow prior to the time that the CFT empties. During Long Term Core Cooling (LTCC), operator actions are required to maintain the pumped injection fluid source as the suction is switched from the BWST to the RB emergency sump.

In accordance with the SER restrictions of the RELAP5-based LOCA EM, sensitivity studies were performed to determine which ECCS configuration, maximum or minimum flow, would produce the most conservative calculation in terms of calculated peak clad temperature. In addition, studies are also performed to determine if a loss of offsite power will produce the most conservative results. The loss of offsite power is assumed to occur coincident with the opening of the break and also ensures that the RCPs are modeled as having no electrical power during the accident. The studies confirmed that maximum ECCS flow coupled with a loss of offsite power yields the highest clad temperatures for the large break LOCAs. Modeling the maximum ECCS flow during the transient results in the minimum RB pressure. The RCS and the RB are in equilibrium and have nearly the same pressure. The lower pressure creates the greatest steam binding effect in the piping loops, which inhibits the core reflood and refill rates and results in the higher calculated peak clad temperatures. The analyses demonstrate that the pumped and passive ECCS injection flow is sufficient to recover the core. The fuel peak clad temperatures, clad oxidation, and whole-core hydrogen generation rates are shown to be less than 10CFR 50.46 limits.

Further analyses have been performed regarding the ability to keep the chemical additive concentration below its precipitation limit throughout the post accident cooling period to demonstrate continued compliance with item (e) above. See Subsection 6.3.3.1.2.1 for further details.

The Core Flooding Tanks (CFTs) are isolated from the RCS at pressures below approximately 700 psig so that they do not empty into the RCS during normal operation. If a LOCA were to occur while CFTs are isolated, the worst time would be during shutdown just after CFT isolation at approximately 700 psig RCS pressure. If a hypothetical LOCA is postulated to occur just after the CFTs are isolated, the LPI pumps would still be available for automatic actuation when the appropriate SFAS system high CTMT building pressure or low-low RCS pressure setpoints are exceeded. (HPI pumps would start if the SFAS high CTMT building pressure setpoint is reached, but the operator would have blocked the low RCS pressure SFAS trip. Therefore, HPI is not credited.) The blowdown transient would be longer in duration than that calculated at full power due to the lower system pressure. Since a substantial amount of time after shutdown is required to reach 700 psig, the decay heat generation would be very low compared to that of a large break LOCA occurring from power operation. Postulating that the accident occurs 1.6 hours after shutdown with isolated CFTs, and only assuming flow from a single LPI pump (with a 25 second delay to allow for loss of offsite power and emergency diesel generator startup), fuel cladding heatup would take place primarily after blowdown and during the downcomer/lower reactor vessel head refill period. In the postulated transient, the core average heat flux would

be less than 3000 BTU/hr- ft^2. Assuming a heat transfer coefficient of only 10 BTU/hr- ft^2, a fuel cladding temperature of only 300°F above saturation would be required to terminate the cladding heatup. The decay heat power being generated would be at a low enough level that the fuel could sustain an adiabatic heatup for more than 200 seconds before reaching only 1500°F. However, even in the unlikely event that the reactor vessel is fully voided at the end of the blowdown transient, one LPI pump would require only approximately 80 seconds to fill the lower head and commence reflooding the core. Since steam cooling would be available during the blowdown and reflood portions of the transient, the consequences of a LOCA during shutdown (even with isolated CFTs) would be insignificant compared to a break of similar size at full power.

6.3.3.1.2.1 Boron Precipitation Control

Two active means of ensuring the chemical additive concentration remains below its solubility limit throughout the post-accident cooling period are provided. The primary method of Boric Acid Precipitation Control (BPC) uses the discharge of Decay Heat Removal/Low Pressure Injection (DHR/LPI) Pump 1 through a line that bypasses DH-1517 and allows reverse flow into the Decay Heat Drop Line. This line permits the required flow to enter the RCS Hot Leg when the containment isolation valves DH-11 and DH-12 are opened following transfer to the emergency sump. The backup BPC method uses High Pressure Injection (HPI) Pump 2, in piggy back with DHR/LPI Pump 2 to supply water to the Auxiliary Spray Line via a tie-line, providing the required dilution flow to the Pressurizer.

Both BPC methods have been analyzed to be effective for any size LOCA initiated at any power level up to 102% of 2772 MWt.

During normal power operation the systems are lined up to support both the primary and backup BPC mode. During normal shutdown cooling operation, the primary BPC bypass line is manually isolated by closing DH10 and the backup BPC tie-line and DHR/LPI system valves are manually positioned to allow normal cool down of the pressurizer utilizing DHR/LPI. These valves (HP209, HP210, DH200 and DH201) are administratively controlled to prevent all four from being open simultaneously, which could cause overpressurization of the Decay Heat/Low Pressure Injection System.

Control Room BPC flow indication is provided for monitoring the success of both the primary and backup BPC flow path.

Both the primary and backup BPC flow rates have been analyzed to be sufficient to exceed the core boil-off rate prior to the boric acid concentration reaching the solubility limit. The primary BPC method initiates a reverse core flow through the hot leg using Decay Heat Pump 1. For the back up BPC method, the Auxiliary Pressurizer Spray flow initiates a reverse core flow. In both cases the reverse core flow reduces the core boric acid concentration and precludes potential precipitation by transporting the fluid with high boric acid concentrations backward through the downcomer and out the break.

BPC is not required when the RCS is above 322°F, as insufficient boric acid is available in the entire RCS and Emergency Core Cooling System (ECCS) to reach the solubility limit. BPC is also not required when the RCS is below 322°F and the core exit temperature is adequately subcooled, as no boiling occurs in the core region which could concentrate the boric acid. Confirmation of RCS conditions is accomplished using Control Room instruments or the plant computer.

Either method is initiated after the DHR/LPI system suction source is switched over from the borated water storage tank (BWST) to the ECCS emergency sump, once the RCS is within the design pressure and temperature range for the DHR drop line piping and components. Analysis has shown that RCS cold leg pump discharge break sizes of 0.09 ft^2 and larger will cooldown (without operator assistance) below the DHR drop line design range by the time of sump switchover. Smaller breaks may not evolve to these conditions at the time of sump switchover, however, they will allow the flow from both DHR/LPI Pumps to refill the reactor vessel such that passive core boric acid dilution is obtained through Reactor Vessel Vent Valve (RVVV) liquid spillover. The reactor vessel refill will occur prior to the RCS reaching the DHR drop line initiation range or the core solubility limit. Once RVVV liquid dilution has been established, it halts any core boron concentration increase, and begins to dilute the core concentration. The RVVV liquid dilution flow prevents precipitation in the core and reduces the core concentration until the DHR drop line design conditions are reached and the primary method can be established.

Thus, Davis-Besse has two active methods of demonstrating adequate post LOCA boron dilution, effective from any initial power level. Both the primary and backup BPC methods are in full compliance with 10CFR50, Appendix K analytical requirements. In addition since both BPC methods are not single failure proof the NRC has granted an Exemption from 10CFR50, Appendix K, Section I.D.I, "Single Failure Criterion" requirements. (Log No. 6255 & 5659). This continues to fulfill the requirement of Section 6.3.3.1.2.e.

6.3.3.1.3 Results of the Analysis (Small Break)

Event Progression:

For the SBLOCA events, the highest clad temperatures are calculated assuming that minimum ECCS flow is available. An SBLOCA generally progresses through five phases: (1) subcooled depressurization (2) reactor coolant pump and loop flow coastdown and natural circulation, (3) loop draining, (4) boiling pot, and (5) refill and long-term cooling.

The subcooled depressurization phase begins at the leak initiation. This phase is characterized by the period of time before the RCS begins to saturate and voids begin to collect in the RV upper head and hot leg U-bends. During this period, the pressurizer will begin to empty, the RCS will depressurize to the low RCS pressure reactor trip setpoint, and the turbine will trip. With the assumption of a loss of offsite power coincident with reactor trip, the MFW pumps and RC pumps will trip and AFW will be initiated following a time delay.

Following the RCP coastdown, the RCS flow tends to evolve to a natural circulation flow condition. The energy generated by the core is transferred by convection to the steam generators during the flow phase. The continued loss of the RCS liquid inventory allows steam voids to form in the upper reactor vessel head and the upper hot leg U-bends. Natural circulation ends when the U-bend steam void displaces the hot leg mixture levels below the U-bend spillover elevation. Flow is usually interrupted first in the hot leg containing the pressurizer surge line connection, because of the additional flashing of the saturated pressurizer liquid that enters during the subcooled depressurization. Near the end of the flow phase, alternating periods of RCS repressurization can cause intermittent spillovers of hot-leg liquid into the steam generator primary region for the smaller break sizes.

With the interruption of the RCS loop flow, the loop-draining phase begins. As the entire RCS approaches saturated conditions, the onset of subcooled and saturated nucleate boiling occurs in the core because of the high decay heat levels and the RCS depressurization. The flashing

within the hot legs increases the size of the voids in the U-bends and eventually interrupts RCS flow and decreases the primary-to-secondary heat transfer. For larger SBLOCAs, the RCS will continue to depressurize as the loops drain. For smaller breaks, however, the reduced heat transfer can interrupt the RCS depressurization. Also for these smaller breaks, the volumetric expansion of the RCS, due to continued steam formation, can exceed the volumetric discharge from the break, causing the RCS pressure to temporarily stabilize or even increase.

In the reactor vessel, the steam void in the upper head displaces enough liquid to uncover the reactor vessel internals vent valves (RVVVs), creating a manometric imbalance between the core and the downcomer. The imbalance forces the RVVVs to open and pass steam into the reactor vessel downcomer. The downcomer steam volume grows until the cold leg nozzle is exposed to steam. As soon as the downcomer liquid level decreases below the cold leg nozzle spill under elevation, a steam venting path develops from the core through the RVVVs to the cold leg break, enhancing the RCS depressurization.

During the loop draining phase, the steam voids that develop in the U-bends can become large enough that the primary liquid level is displaced into the steam generator tube region below the AFW nozzles. If AFW is injecting, an improved primary-to-secondary heat transfer can then be restored, through condensation on the tubes wetted by the AFW. This heat transfer process within a once-through steam generator (OTSG) is referred to as Boiler Condenser Cooling (BCC). When BCC cooling takes place near the location of the AFW nozzles, it is referred to as high-elevation BCC cooling. If high-elevation BCC cooling occurs, the RCS depressurization rate will be increased. Later in the loop draining phase, a different form of BCC cooling can occur if the RCS tube liquid level decreases below the secondary liquid level. This cooling process is referred to as pool BCC, and will continue if (1) RCS condensation and ECCS injection do not cause the RCS liquid level to increase above the secondary level and (2) the secondary fluid temperature is maintained below the temperature of the steam on the primary side of the OTSG tubes. For the Davis-Besse raised-loop design, any condensate formed during this process can augment the ECCS flow to the core. For the smaller breaks, the combination of leak flow (with upper-RV venting through the RVVVs), BCC cooling, and HPI cooling will cause the RCS pressure to again decrease.

Also during the loop draining phase, the reactor vessel outlet annulus mixture level will decrease to the hot leg nozzle spill-under elevation. If the top of the hot leg nozzles void, steam will flow up the hot leg riser section, and liquid from the hot leg risers will drain back into the vessel. This hot leg draining allows the mixture level in the outlet annulus to remain near the top of the hot leg nozzle until the hot leg level drops into the RV exit nozzle horizontal piping.

After the hot legs empty, another path for the direct venting of steam to the break can be opened if the loop seals in the RCP suction piping are cleared. For the larger break sizes, the RCS depressurization can be rapid enough to cause significant flashing in the suction piping, causing the liquid level to decrease below the suction piping spill under elevation. The loop seals will then be clear, creating another steam relief path, in addition reopening to the path through the RVVVs.

When loop draining ends, the break site void fraction will be based on core steam plus broken loop HPI flow. At that point, the only RCS liquid available for core cooling is the liquid remaining in the reactor vessel and the ECCS flow plus any SG condensate from the intact loop if the loop seal has not cleared. This portion of the transient is defined as the "boiling pot" phase. The increased void fraction at the break will further increase the volumetric discharge rate which typically increases the RCS depressurization rate. The reactor vessel levels will continue to

decrease, however, if the ECCS injection plus SG condensate cannot match the reactor vessel liquid loss from flashing, decay heat, and passive metal heat removal.

The break flow allows the RCS depressurization to progress until either the CFT pressure is reached or the HPI flow rate exceeds the core boil off rate, allowing the RCS to refill to the break elevation. Before either of these conditions occur, the mixture levels may descend into the core heated region resulting in a heat up of the fuel cladding in the uncovered portion of the core.

The clad temperature increase calculated for the upper core elevations are conservative because a power shape skewed to the core exit is used. The radial peaking factor is selected to match or bound the peaking used for the positive imbalance limits at the limits of normal operation. During the period of partial core uncovering, the clad may swell and possibly rupture if the clad temperatures exceed approximately 1300°F. This 1300°F does not constitute a limit, but rather a general approximation of possible rupture. A comparison of the rupture temperature and the calculated cladding temperature is performed at each burnup step used in the analysis. The potential for clad rupture is evaluated in the SBLOCA analytical model by modeling multiple EM pin channels with initial internal pin pressures that span BOL to EOL.

An SBLOCA transient analysis is normally terminated at some point after the entire core is refilled, the cladding temperatures returned to within a few degrees of RCS saturation temperature and ECCS heat removed from the core matches decay heat. For the level to increase, core inflow (ECCS plus SG condensate) must exceed the liquid loss rate. Continued RCS depressurization permits higher ECCS injection rates that hastens core refill. The additional ECCS flow assures that the core can be kept covered. Once the core has been completely quenched, the analytical results are checked to ensure a path to long-term cooling is established. For long-term cooling to be assured, the HPI flow and/or LPI flow must match core boiling due to decay heat and wall metal heat plus flashing. When long-term cooling is assured, the LOCA analysis is terminated.

The limiting breaks are generally those that result in the largest bypass of emergency core cooling system (ECCS) flow directly out of the break. The break size range includes any break that can exceed the makeup system flow up to and including that of a full, double-ended guillotine rupture of the cold leg or hot leg pipe. The mitigation of the break consequences is accomplished by a cooperative effort of makeup flow from high pressure injection (HPI), core flood tanks (CFT), and low pressure injection (LPI), plus ultimate core decay heat removal via auxiliary (emergency) feedwater (AFW), and long-term cooling via decay heat coolers with ECCS recirculation from the containment sump. These systems are activated and managed by both automatic trips and controls or manual operator actions identified in the plant emergency operating procedures (EOPs).

Although not considered part of the ECCS, AFW is important in mitigating certain size- and location-specific small breaks. The limiting break location is one in which a portion of the HPI flow is not available for core cooling, i.e., a break in an HPI line or in the cold leg reactor coolant pump discharge piping downstream of the HPI injection nozzle. If the break size is too small to pass all of the steam produced in the core, the RCS pressure will increase. As the pressure increases, HPI flow will be reduced. AFW flow is important in filling the OTSGs and maintaining a level that is high enough to support BCC. BCC, in combination with the break area, will limit the increase in RCS pressure, and will subsequently reduce the pressure to where HPI is sufficient to remove the residual energy in the system.

The limiting break location is typically in the bottom of the cold leg pump discharge pipe. The RCS response to the cold leg pump discharge breaks is dependent on the break size being analyzed. For the small size break areas, the combinations of break size, core power, and steam generator heat transfer result in a relatively slow depressurization. The reactor will trip on low pressure within the first minute of the event. During the subcooled blowdown portion of the transient, however, the pressure may not decrease below the low RCS pressure SFAS actuation setpoint. If SFAS is not actuated by this time, it may be several minutes before boiler-condenser cooling process will decrease the RCS pressure below the SFAS actuation setpoint. During this time, enough primary coolant may be lost to cause the core to uncover. This establishes that not only is the magnitude of HPI flow important, but the integrated amount of HPI flow delivered to the RCS is also important. The RCS behavior for these smaller cold leg pump discharge breaks are unique in that the RCS pressure will initially decrease due to the release of subcooled liquid. The higher temperature regions of the RCS will saturate. The break area by itself is not large enough to pass all of the steam generated by boiling in the core and the RCS pressure will increase until a quasi-equilibrium is reached. Steam generator cooling is necessary to supplement the energy released through the break. As core decay heat decreases and boiler-condenser cooling process is established, the RCS pressure (and temperature) will gradually decrease to near that of the active steam generator(s).

Results Discussion:

The results of the complete spectrum of small breaks are described in Reference 35. The core cooling provided by the unaffected injection lines, and in combination with SG cooling, is adequate for meeting 10CFR 50.46 requirements.

A special type of SBLOCA, which minimizes available ECCS to mitigate the LOCA, is the core flood line break. For this SBLOCA, a break is postulated at the core flood tank (CFT) line to reactor vessel nozzle interface. The loss of RC fluid is limited in flow area to 0.44 ft^2 by the CF nozzle flow restrictor, located inside the reactor vessel core flood nozzles. The ECCS analysis takes credit for one CFT and one HPI pump for the core flood line break. The Reference 35 analysis indicates that the criteria in 10CFR 50.46 are met.

A core flood line break and larger cold-leg pump discharge breaks could be worse if offsite power is available. When offsite power is available, the operators must trip the RCPs within two minutes after loss of subcooling margin (LSCM) to prevent the loss of RCS liquid inventory that is needed later in the transient to augment the pumped ECCS injection. Therefore, in addition to analyzing these breaks with loss of offsite power, additional analyses are performed with RCP trip at two minutes after LSCM.

A second special type of SBLOCA is the HPI line break. For this accident, a break is postulated in one of the injection lines with the active HPI pump, between the last check valve in the line and the RCS cold leg pipe. The HPI flow rate assumptions for this SBLOCA are different than for the classical RCS cold leg SBLOCA and CFT line break since the back pressure in the broken and intact HPI lines vary dramatically. At Davis Besse, one HPI pump feeds two cold legs and due to the difference in back pressure, no HPI flow is initially available for core cooling. Operator action is credited at 10 minutes after loss of subcooling margin to balance the flow between the legs.

The analyzed small-break LOCA spectrum consisted of: (a) four HPI line break sizes (0.015 ft^2, 0.020 ft^2, 0.022 ft^2, and 0.02463 ft^2), (b) twelve cold leg pump discharge line breaks with LOOP ranging from 0.010 ft^2 to 0.5 ft^2, (c) two cold leg pump discharge line breaks with offsite power available (0.30 ft^2 and 0.50 ft^2) and, (d) two 0.44 ft^2 CFT line breaks (one with LOOP and one

with offsite power available). The highest peak clad temperature for the small-break LOCA spectrum was predicted for a 0.02463 ft^2 HPI line break. For this break, the peak clad temperature remained below 1,600 °F.

6.3.3.1.4 Discussion of Noncondensible Gases

Sources of Noncondensible Gases in the Primary System:

Table 6.3-8 lists the potential sources and historical amounts of noncondensible gases for a 177 fuel assembly plant. However, most of these gases would not be released for small break transients. Appendix K evaluations performed for the 177FA plants demonstrate that cladding temperatures remain low and no cladding rupture nor metal water reaction occur. Thus, these sources can be neglected. The Core Flooding Tanks discharge into the RCS only for breaks large enough to depressurize the RCS (Reference 12). Also, the steam generator (SG) is a heat sink only if primary system pressure is above that which corresponds to the secondary system safety valve setpoint (approximately 1050 psig). Therefore, gases present in the Core Flooding Tank can be neglected in addressing the effect of noncondensibles. The only sources of noncondensibles which might separate in the RCS are the gases dissolved in the coolant, the gases in the pressurizer, gases in the makeup and Borated Water Storage Tank and gases released from an allowed 1% failed fuel in the core.

Effects of Noncondensible Gases on the Primary System Response Following a Small Break LOCA:

There are two possible ways in which the release of noncondensible gases in the primary system could interfere with the condensation heat transfer processes which occur in the steam generator during small loss-of-coolant accidents. If noncondensible gases filled the U bend at the top of the hot leg, then water vapor would have to diffuse through the noncondensible gases before they could be condensed in the steam generator. This would be a very slow process and would effectively inhibit natural circulation. Lesser amounts of noncondensibles would reduce the heat transfer by condensation because the vapor would have to diffuse through the noncondensibles to get to the condensate on the tubes.

The only sources of noncondensibles which might separate in the RCS are the gases dissolved in the coolant, the gases in the pressurizer, gases in the makeup and Borated Water Storage Tank and gases released from an allowed 1% failed fuel in the core. Thus, the maximum amount of noncondensible gases in the system, assuming all gas comes out of solution, no noncondensibles are lost through the break flow, that there was one percent failed fuel, and the injection of 6.4 x 104 lbm from the makeup tank and BWST (typical of approximately 1500 sec of HPI), would be:

Dissolved in coolant	563 scf
In pressurizer	166
Fission gas	2
Fuel rod fill gas	11
MU tank	24
BWST	14
Total	780 scf

This gas would occupy a volume of 22.4ft^3 at a pressure of 1050 psia, the lowest pressure condition in the primary system for which condensation heat removal will occur. It should be noted that the assumed integrated injection flow does not have a significant effect on the total

volume of noncondensibles which might be present in the primary system. Since the volume required to completely fill the U-bend in the hot leg is 125ft^3, the noncondensible gases will not impede the flow of vapor to the steam generator.

The heat transfer during condensation is made up of the sensible heat transferred through the diffusion layer and the latent heat released due to condensation of the vapor reaching the interface (see Figure 6.3-22). The model of Colburn and Hougen (Reference 9) gives the following equation for the heat transfer in the vapor phase:

$$\phi = h_g (T_{go} - T_{gi}) + K_g M_g h_{fg} (P_{go} - P_{gi})$$

Where

ϕ = condensation heat flux, Btu/hr- ft^2

h_g = heat transfer coefficient for vapor layer, Btu/hr- ft^2-°F

T_{go} = bulk temperature, °F

T_{gi} = temperature of interface, °F

K_g = mass transfer coefficient, $\frac{\text{lb - mole}}{\text{lb - hr}}$

M_g = molecular weight, lbm/lb-mole

h_{fg} = latent heat of vaporization, Btu/lbm

P_{go} = partial pressure of vapor at bulk conditions, $\frac{\text{lbf}}{\text{ft}^2}$

P_{gi} = partial pressure of vapor at the interface, $\frac{\text{lbf}}{\text{ft}^2}$

$$K_g = \frac{1.02\,D}{z\,RT}\left[\frac{gz^3\rho\left[\frac{\rho_o}{\rho_i}-1\right]}{\mu D}\right]^{.373} \qquad \rho/\text{pam} \qquad (1)$$

D = diffusion coefficient, ft^2/hr

z = height, ft

R = gas constant, 1545 $\frac{\text{lbf - ft}}{\text{lb} - \text{mole} - {}^\circ\text{R}}$

T = absolute temperature at bulk conditions, °R

g = acceleration of gravity, ft/hr^2

ρ = density, lbm/ft^3

ρ_o = density at bulk conditions, lbm/ ft^3

ρ_i = density at interface conditions, lbm/ft^3

μ = viscosity, lbm/hr-ft

$$pam = \frac{\underline{pai - pao}}{\lambda n\left[\frac{pai}{pao}\right]}$$

pai = partial pressure of gas at interface, $\frac{lb}{ft^2}$

pao = partial pressure of gas at bulk conditions, $\frac{lb}{ft^2}$

For the application to OTSG condensing heat transfer during small break transients, the term $h_g(T_{go} - T_{gi})$ can conservatively be neglected since the vapor velocities would be very low. Thus,

$$\phi = K_g M_g h_{fg} (P_{go} - P_{gi}). \qquad (2)$$

The heat transfer with noncondensible gases present is obtained by iteration. An interface temperature T_{gi} is assumed, which fixed P_{gi}, and the heat transfer across the liquid condensate film is computed from

$$\phi = h_f (T_{gi} - T_w) \qquad (3)$$

Where

$$h_f = .943 \left[\frac{\rho_f (\rho_f - \rho_g) g h_{fg} k_f^{\ 3}}{\mu_f z(T_{gi} - T_w)}\right]^{1/4}$$

ρ_f = density of fluid, lbm/ft^3
ρ_g = density of vapor, lbm/ft^3
k_f = thermal conductivity of fluid, Btu/hr-ft-°F
T_w = wall temperature, °F

The partial pressure of the gas at the bulk conditions can be calculated from the mole fraction of noncondensible gases. When the heat flux computed from equation 2 matches that computed by equation 3, the proper interface temperature has been found.

The impact of noncondensibles on the condensation heat transfer process during a small break was examined for the 0.04 ft^2 and 0.01 ft^2 cold leg breaks analyzed for the 177-FA plants. The breaks utilize the SG for heat removal for a significant portion of the transient. Hand calculations were performed, using the theory presented above, to ascertain the effect of non-condensibles on the transient.

The amount of noncondensible gases, assuming that all gases come out of solution, would be 2.61 moles. The effect of these gases is to raise the pressure and primary temperature to obtain the same heat transfer. Assuming that the noncondensibles accumulated only within the steam generator upper plenums and the steam generator tubes, the system pressure increase, due to noncondensibles, would only be 25 psi for a 0.04 ft^2 break and 40 psi for a 0.01 ft^2 break. It should be noted that this effect is predominantly due to the inclusion of the partial pressure of the noncondensibles, which is 24 psi for the 0.04 ft^2 break and 34 psi for the 0.01 ft^2 break in the total system pressure. These calculations represent the maximum impact as they were computed at the time of maximum condensation heat flux for the respective cases.

As shown, the influence of noncondensibles does not significantly effect the condensation heat transfer process. The estimates made are conservative in that they assumed all the gas is located in the steam generators (none is in the top of the reactor vessel or pressurizer) and no gases escape through the break. Thus, it is shown that the presence of noncondensible gases in the system considering the effect on condensation heat transfer, system pressure and natural circulation should not significantly affect the small break transient.

Actions to Preclude Introduction of Noncondensible Gases into the Primary System:

Introduction of significant quantities of noncondensible gases into the primary system following a small break LOCA is prevented if the core is not uncovered during a small break. The ATOG based emergency procedures (Reference 19), are designed to prevent core uncovery by assuring continued ECCS injection. Thus, the amount of noncondensibles which might separate in the RCS is small and would not significantly effect the small break transient.

Operator Actions During Accumulation of Noncondensible Gases in the Primary System:

A significant accumulation of noncondensible gases within the primary system during a small break is not expected. This position is confirmed by small break transient predictions, using conservative assumptions, which shows that fuel clad temperature excursions are limited to less than 1800°F. Fuel clad failure is not expected and significant H_2 gas formation due to metal water reaction will not occur.

Small amounts of noncondensible gases can be released into the primary system during a small break. For the break size range where noncondensible gases could have a detrimental effect (i.e., breaks where natural circulation is required for energy removal) the quantities of gases that are predicted to exist within the primary system are not significant. For larger quantities of noncondensible gases to exist, a core transient that is not predicted must occur. The probability for such an occurrence is believed to be small because of the detailed emergency procedures for post-LOCA conditions that have been developed and the extensive operator training that has been conducted in their use.

Emergency procedures have been developed to accommodate noncondensible gases, to maintain plant control, and to achieve a stable long term cooling condition. The plant control measures contained in the ATOG-based emergency procedures (Reference 19) will counteract the effects of noncondensible gases. It also contains guidance for operator action developed for an inadequate core cooling condition.

To upgrade the RCS venting and/or degasing capabilities, remotely operated hot leg vents have been designed and installed. The ATOG-based emergency procedures describe the use of the hot leg high point vents to aid in the event that makeup/high pressure injection cooling may be needed.

Current Procedural Actions:

During a small break, the principle effect of noncondensible gases is to minimize the performance of the steam generators during natural circulation (either single phase water flow or reflux boiling). A restart of the RC pumps (one per loop) is the optimum action. A return to forced circulation will aid in condensation of existing steam and removal of noncondensible gas (if present) within the hot leg piping. Noncondensible gases, originally within the looping piping, would then tend to be suspended within the coolant stream and collect within the upper regions of the reactor vessel (RV). A Reactor Vessel Head Vent System, described in Section 5.5.16, is installed to continuously vent steam and non-condensible gases from the Reactor Vessel to Steam Generator No. 2 where the steam could be condensed and non-condensible gases could be vented by the high point vent. A substantial quantity (approximately 1000 ft^3) of gas can be accommodated within the upper region of the RV; therefore, there is good assurance that natural circulation can be maintained if RC pump operation must be terminated. If the RC pumps cannot be started and/or no secondary side heat sink is available, the operator will utilize the PORV, HPI and Makeup Pumps for core cooling and RC pressure control until the RC pumps can be restarted and/or normal secondary cooling is re-established.

The procedural actions required to accommodate noncondensible gases are described in the ATOG-based emergency procedures (Reference 19).

Small Break - Inadequate Core Cooling Conditions:

An inadequate core cooling condition is not expected for B&W 177-FA plants. However, procedures which identify the symptoms and operator actions for several circumstances, including a small break, are contained in the ATOG-based emergency operating procedures (Reference 19).

6.3.3.1.5 Evaluation of Fuel Rod Performance

Fuel rod performance during a LOCA is calculated under the guidelines of 10CFR50, Appendix K, as interpreted by B&W's evaluation model documented in BAW-10192P-A as modified by Supplement 1. The application of B&W's evaluation model to Davis-Besse 1 is reported in Reference 35.

6.3.3.1.5.1 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 47) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that, for both the large break and small break LOCAs, the net effect of Steam Generator differences is beneficial, and is due to the lower limit on steam generator tube plugging in the replacement Steam Generators (5%) versus the original Steam Generators (20%). The small break LOCA response is also improved by the reduced Auxiliary Feedwater (AFW) bypass flow in the replacement Steam Generators.

For large break LOCAs, the above AREVA evaluation concluded that a net reduction in predicted Peak Clad Temperature (PCT) would result from the replacement Steam Generators on the analyses of record. The local and whole-core oxidation predictions would also be reduced, because these are directly related to cladding temperature and its time at temperature. Since PCT is reduced and the time at temperature is similar or better due to the reflooding rate improvement, the local oxidation and whole-core hydrogen generation rates from the existing analysis (20% steam generator tube plugging) remain bounding for the replacement Steam

Generators (5% steam generator tube plugging). Therefore, the replacement Steam Generators do not impact the large break LOCA analyses.

The evaluation of the small break LOCA (SBLOCA) with replacement Steam Generators concluded that the lower tube plugging limit and improved AFW bypass performance assures that the existing small break LOCA analysis with the original Steam Generators remains bounding for operation with the replacement Steam Generators.

6.3.3.2 Additional Considerations for ECCS Performance

Eutectic Formation

The temperature transient in the core can produce significantly higher than normal temperatures in components other than fuel rods. Therefore, a possibility of eutectic formation between dissimilar core materials exists. Considering the general area of eutectic formation in the entire core and reactor vessel internals, the following dissimilar metals are present, with major elements being in the approximate proportions shown:

Type 304 stainless steel:

19 percent chromium
10 percent nickel
Remainder iron

Control rod poison material:

80 percent silver
15 percent indium
5 percent cadmium

Zircaloy-4:

98 percent zirconium
1-3/4 percent tin

Inconel:

53 percent nickel
19 percent chromium
3 percent molybdenum
5 percent columbium-tantalum
1 percent titanium
0.5 percent aluminum
Remainder iron

All these alloys have relatively high melting points (greater than 2700°F) except for the silver-indium-cadmium alloy, whose melting point is about 1470°F.

The binary phase diagram indicates that zirconium in the proportion of 75 to 80 percent has a eutectic point with either iron, nickel, or chromium at temperatures of approximately 1710, 1760, and 2370°F, respectively. If these dissimilar metals are in contact and if those eutectic points

are reached, then the materials could theoretically melt even though the temperature is below the melting point of either material taken singly.

Historically, one point of such dissimilar metal contact was between zircaloy-clad fuel rods and Inconel-718 spacer grid. In the analysis of the loss-of-coolant accident, some of the cladding exceeded the zirconium-iron and the zirconium-nickel eutectic points. Since the spacers are located at 21-inch intervals along the assembly and each grid has a very small contact area, only a fraction of the hottest fuel rods would be in contact with Inconel-718 spacer grids.

B&W conducted experimental tests in which specimens of zircaloy-4 tubing in contact with sections of spacer grid material were subjected to a thermal transient closely approximating that of the cladding hot spot following a LOCA. These tests verified that the eutectic reaction is limited to the small region of contact between the cladding and the spacer grid tips (dimples), and that it terminates as these materials melt at the point of contact. Both the cladding and the grid material maintained their structural integrity because the amount of material involved was small and melting was localized. Since the intermediate spacer grids are now made of zircaloy, this eutectic formation is no longer pertinent.

Chloride Stress Corrosion

The solution used in the ECCS is water containing 2600 ppm to 2800 ppm boron with Na_3PO_4 (TSP) added to produce a pH of 7 or higher. The pH is adjusted with TSP in order to reduce the possibility of chloride stress corrosion of austenitic stainless steels in the event that the chloride levels increase as a result of contamination on the surfaces in the reactor containment building. Tests performed by Westinghouse Electric Corporation have shown that boric acid solutions with no pH adjustment and chloride concentrations as low as 10 ppm can cause chloride stress corrosion of stainless steel (Reference 16). For boric acid solutions adjusted with sodium hydroxide to a pH of 8.0 or greater, no cracking was observed at chloride concentrations of up to 500 ppm.

Thermal Shock Considerations

The ability of the cladding to maintain its strength and structural integrity during core reflooding has been confirmed by experimental work at B&W involving the rapid quenching of unirradiated zircaloy tubing specimens from temperatures as high as 2300°F. Test results show that for temperatures up to 2300°F, the cladding material retains its strength and does not experience brittle fracture upon quenching. Irradiation would increase the tensile strength of the material even though it would also promote the tendency toward a brittle fracture mode of deformation at higher pressure and temperatures toward the end of core life.

The margin of safety against brittle fracture of the reactor pressure vessel is controlled as specified in Appendix G and H of 10CFR50 and Regulatory Guide 1.2, particularly with regard to specific guidelines for the treatment of heat-up and cool-down conditions and for analysis of the thermal shock transient.

Thermal sleeves are installed, where required, to limit the thermal stresses developed because of rapid changes in fluid temperatures. They are provided in the four HP injection nozzles on the reactor inlet pipes.

The ECCS takes suction from either heated storage tanks or the reactor building sump so that there are no thermal shock effects within the system itself.

6.3.3.2.1 Control Rod Effects

Davis-Besse employs an extended life control rod assembly design. The absorber material is Silver-Indium-Cadmium and the clad (i.e., sheath) material is Inconel-625. Each control rod is inserted into a guide tube within the fuel assembly. The guide tube material for the current fuel assembly design, i.e., Mark-B-HTP, is M5.

The melting temperature of the Silver-Indium-Cadmium control rod absorber is approximately 1470°F. The M5 guide tube material is a Zirconium-based alloy and the control rod cladding material contains Iron and Nickel. The limiting eutectic melting temperature for Zirconium/Iron and Zirconium/Nickel is 1715°F, which is lower than the individual constituent melting temperatures. Because the control rod absorber melting and cladding/guide tube eutectic temperatures are less than the 2200°F peak fuel rod clad temperature criterion, calculations of the control rod and guide tube temperatures were necessary to ensure a coolable geometry and long term cooling capability is maintained. Previously, the fuel assembly spacer grids were constructed of Inconel-718. The Mark-B-HTP design uses M5 spacer grids, so a eutectic interaction is not a concern.

A generic control rod temperature response analysis (Reference 45) that is applicable to the Davis-Besse Station with a 15 x 15 fuel assembly design has been completed for limiting Small and Large Break LOCAs. The methods, inputs and results of the analyses are discussed in the following sections. Burnable Poison Rod Assemblies (BPRAs) were scoped out of this analysis based on non-limiting melting points.

6.3.3.2.1.1 Analytical Methods for Evaluating Control Rod Effects

The generic control rod temperature response analysis was performed using a model of the control rod absorber, clad and guide tube, the fuel rod cladding and the associated annuluses and gaps between each of these components. These components were added to the existing Davis-Besse LBLOCA and SBLOCA limiting case models, which were then run in order to provide both the LOCA boundary conditions and the control rod temperatures in one integrated calculation. Based on the code capabilities and its NRC approved use in performing LOCA analyses for Davis-Besse, the RELAP5/MOD2-B&W code was utilized.

The modes of heat transfer used in the model are shown in Figure 6.3-21. Temperature-dependent conduction heat transfer was modeled within the absorber, Helium gap, control rod clad and guide tube. The water between the control rod clad and guide tube as well as between the fuel clad and the guide tube was modeled using RELAP5/MOD2-B&W flow channels. These two-fluid flow models include radiant heat to steam for the post-DNB fluid conditions. Convective heat transfer to the flow channels was considered at the control rod and fuel rod cladding outer surfaces, as well as at the guide tube inner and outer surfaces. Heat generation due to neutron and gamma attenuation was modeled internally to the absorber, cladding and guide tube. Heat generation due to beta attenuation was modeled internally to the absorber.

All uncertainties associated with the analysis are accounted for by conservative approximations.

1) Thermal radiation heat transfer was neglected between the control rod guide tube and both the control rod and fuel cladding. This was done because other conservatisms were put in place. Notably, the decay heat standard model was increased by 20 percent, minimum ECCS flow was used, no control rod worth was modeled during a LBLOCA and minimal control rod worth was modeled during a SBLOCA.

6.3.3.2.1.2 Control Rod Effects During a Large Break LOCA

The LBLOCA analyses considered a power level of 3026 MWt with a linear heat rate of 17.8 kW/ft and an axial power shape peaked at the 9.536 ft elevation.

The results of the Large Break LOCA indicated no control rod melt is predicted to occur. The maximum control rod absorber temperature was predicted to be 1435°F and occurred at 152 seconds. This temperature is 35°F less than the absorber melt temperature of 1470°F. The maximum guide tube temperature was predicted to be 1397°F and occurred at 100 seconds. This value is less than the control rod sheath eutectic melt temperature of 1715°F. Thus, even if contact occurs between the control rod and guide tube, no melting will occur.

6.3.3.2.1.3 Control Rod Effects During Small Break LOCA

The SBLOCA analyses considered power levels of 3026 MWt and 2827 MWt, with a linear heat rate of 17.3 kW/ft and an axial power shape peaked at the 11 ft elevation. In general, the Small Break LOCA predictions show that the upper regions of the core uncover, increase in temperature and remain at elevated temperatures for a longer period of time compared to the Large Break LOCAs. Consistent with the current licensed power level for Davis-Besse, only the results of the analysis at 2827 MWt will be discussed.

For the generic SBLOCA analysis at 2827 MWt it was determined that the control rod absorber temperature would reach, but not exceed the melting temperature of 1470°F. A control rod guide tube temperature of 1490°F was predicted, which is less than the eutectic melt temperature of 1715°F for contact between the control rod sheathing and guide tube. The control rod AIC melt temperature was locally reached in the top heated fuel segment region. The control rod power in this axial blanket region location would realistically have considerable leakage and much lower energy contributions than that which was modeled in the RELAP5 source terms for this segment. Although no localized melting is expected, if there was some, it would not change the rod negative reactivity and no structural breach of the sheath is expected since the control rod sheath and guide tube temperatures are well below the limiting eutectic temperature (1715°F). Control rod and surrounding guide tube integrity and reactivity contributions were preserved. Therefore, these components will remain intact and will perform their design functions during a Small Break LOCA.

6.3.3.2.2 Interrelationship of ECCS Subsystems

The Emergency Core Cooling System is made up of the High Pressure Injection System, the Low Pressure Injection System, and the Core Flooding System. These subsystems are described in Subsection 6.3.1. The High Pressure Injection System provides pumped injection of water into the cold leg piping. The system operates independently of other ECCS subsystems and is used to provide cooling water to the core when the low pressure system is unavailable, which could occur as a result of small leaks.

The Core Flooding System, made up of two pressurized storage tanks, provides water at medium to low pressure directly to the reactor vessel downcomer through a set of nozzles. The system is short term and ceases to operate when the tanks empty which may take forty to several hundred seconds depending on the nature of the accident. The Low Pressure Injection System provides pumped injection of low pressure water directly into the vessel downcomer. The system is used for long term cooling of the reactor core during a loss-of-coolant accident. The system injects into the vessel through the same nozzle as the Core Flooding System.

6.3.3.2.3 Acceptable Lag Times

The Core Flooding Tanks are self actuating and discharge directly into the reactor vessel when the Reactor Coolant System pressure falls below a nominal value of 600 psig.

The current LOCA analysis (Reference 35) uses a range of 582 psia to 648 psia for the CFT pressure and High and Low Pressure Injection delays of greater than or equal to 40 seconds.

Subsection 6.3.4 specifies that the valves would be in their commanded (open) position in 30 seconds and the pumps (HPI and DH) would be operating and delivering flow within 30 seconds. These values are consistent with the numbers used in the analysis.

6.3.3.2.4 Minimum Conditions of ECCS

To be consistent with single active failure criteria, the safety analysis provided earlier in this section was conducted assuming the operation of two Core Flooding Tanks, one High Pressure Injection train, and one Low Pressure Injection train. The Technical Specifications require that two Core Flooding Tanks, two High Pressure Injection trains and two Low Pressure Injection trains be operable except for short periods when one High Pressure Injection train and/or one Low Pressure Injection train may be unavailable due to maintenance. The B&W Owners Group topical report, "Justification of Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems," BAW-2295, Revision 1 and 2, justified the extension of the allowed outage time (AOT) to 7 days for one Low Pressure Injection (LPI) train. Since Davis-Besse Nuclear Power Station has an AOT for an inoperable ECCS train, the LPI AOT was separated from the High Pressure Injection (HPI) AOT, using the 7 day AOT for LPI and retaining the 72 hour AOT for the HPI. The 7 day AOT applies when the LPI train is the only reason for the inoperability of the HPI train. The plant configuration can accommodate the inoperability of multiple components in either train of High Pressure Injection or Low Pressure Injection, as long as 100% of the flow equivalent to one High Pressure Injection train and one Low Pressure Injection train is available.

In the latest analysis (Reference 35) each of the two core flooding tank's fluid volumes were analyzed over a range of 1000 ft^3 to 1080 ft^3 at 120°F. Sensitivity studies were performed to identify the most limiting volume and pressure combination for the CFTs. Those values were assumed in the LOCA analysis. The required volume to preserve the analysis and required boron concentrations are provided in the Technical Specifications.

Normally, the BWST contains a minimum volume of 500,100 gals. of water with a minimum boron concentration of 2600 ppm.

The BWST is not required to take the Davis-Besse Unit 1 to cold shutdown following a postulated tornado. In going to cold shutdown, the reactor coolant contracts 3166 cubic feet. This "void" can be filled with water which is available from the boric acid addition tank, the makeup tank, the Core Flooding Tanks, the clean waste receiver tanks and the pressurizer, all located within Seismic Class I, tornado protected structures.

6.3.3.2.5 Provisions to Protect the ECCS

A three dimensional seismic, thermal, and dead load analysis was performed on the ECCS to determine piping and component nozzle stresses. Nozzle attachments to components are treated as anchors, but the specified deadweight and thermal displacements are applied at the attachment points. Thermal expansion, dead load, and seismic analyses were performed using the same mathematical model. The seismic analysis utilizes the normal mode, response

spectra approach. The inertia forces determined for each mode were supplied mathematically to the model. Results of the analyses were compared to allowable stresses per ASME Section III as follows.

The total longitudinal stress due to weight and pressure was compared with the allowable stress value for the applicable material at design temperature as specified by the code. Thermal expansion stresses were also calculated, based on maximum temperature, and compared to code allowables. The total longitudinal stress due to weight, pressure, and seismic conditions were then combined and compared to 1.2 times the allowable stress value at design temperature as directed by the code.

For those systems penetrating containment, the effect of LOCA on containment vessel expansion is also taken into account. Loadings caused by the vessel movement were added to those used for hanger and restraint design to ensure the adequacy of the hangers and restraints under all conditions.

Loadings imposed on nozzles of components were also tabulated and verified as not exceeding allowables.

During normal Station operation, the ECCS lines will be maintained full by the static head created by the relative elevations of the BWST (bottom at elevation 585') and the emergency sump valves (elevation 560'-8"). During the postulated accident, the minimum water level in the BWST before transfer to the emergency sump is above the tank discharge line. Since the tank discharge line is never drained during the postulated accidents, water hammer due to line filling will not occur. The highest point in any discharge piping (593’-0 ¾" for the Auxiliary Spray Line) is well below the operating water level in the BWST, thus providing a significant positive head on the system. A small amount of gas or vapor could be trapped at the closed HPI discharge valves (at the containment vessel) or closed LPI check valves (at the reactor coolant piping). This small volume of voiding could not cause a water hammer. Manual venting capability is provided at the ECCS pump casings and discharge high points. Even in the event that the piping downstream of the motor-operated HPI discharge throttle valves is completely void of liquid water, an analysis has been performed to verify that no unacceptable forces on the lines will occur in the event of an HPI system actuation.

During normal station power operation, the lines in the ECCS system are in a no flow condition (except for the RCS makeup flow path) and are full of water. Since the check valves in these lines remain closed except for testing or when called on to function, they are not subject to the abuse of flowing operation and thus do not experience significant wear of moving parts. When the check valves are called on to prevent reverse flow, the pressure differential is not sufficient to cause high impact loads or resultant failure of the valve disc. Due to these reasons and the fact that the check valves are inherently tested with the ECCS system, the check valves are expected to function.

6.3.4 Test and Inspections

Portions of the Emergency Core Cooling System are not normally operating. In order to affirm that the normally idle emergency equipment is in a state of readiness to operate in the event of an accident, periodic tests are conducted which verify the operability and function of that equipment.

The Core Flooding System, the High Pressure Injection System, and the Low Pressure Injection System are tested at the frequencies specified in the Technical Specifications.

Each system is tested by itself and it is evaluated so that the system's emergency core cooling functional requirements are confirmed to be fulfilled.

Two separate tests are performed on the Core Flooding System. The first test verifies proper flow from each Core Flooding Tank to the RCS. Partially pressurized Core Flooding Tanks are individually aligned to the RCS with the refueling canal partially filled in Mode 6 and the reactor head removed. The flowrate is determined from the rate of change of Core Flooding Tank level. The second test verifies that the Core Flooding System check valves CF-28, CF-29, CF-30, and CF-31 satisfactorily isolate the relatively high pressure RCS from the relatively low pressure Core Flooding System. This is accomplished by individually leak testing each check valve in Mode 3.

The test of the High Pressure Injection System is typically performed when the reactor is shut down for normal refueling. One train of the equipment which would be called upon to operate in the event of an SA actuation accident is tested. An SA signal is applied separately to the HPI pump motor breaker and the HPI valves which are required to move at the initiation of the accident. Each of these devices is considered to have operated satisfactorily when it obeys the SA signal as noted. The test is considered to be acceptable when the devices requiring active motion obey their respective SA signals within the specified time interval. The valves which are required to move are to be in their safety position within 30 seconds. The HPI pumps are tested in accordance with the Inservice Testing Program to assure the capability of the pumps to perform their SFAS function as verified by pumps reaching and maintaining a specified point on their head-capacity curves. HPI Pump testing may be performed injecting to the RCS from the BWST, recirculating the BWST or recirculating the RCS in the piggyback mode with the associated LPI Pump. Valves in the HPI system which are required to move are stroked in accordance with the Inservice Testing Program to verify their capability to function.

Once per 31 days, each ECCS manual, power operated and automatic valve in the flowpath that is not locked, sealed, or otherwise secured in position, is verified to be in its correct position.

The positions of the valves are monitored by the valve position lights in the control room. The status of the pumps is monitored by the status indicating lights and the station computer. The HPI flow is monitored by the flow indicators and alarms by the station computer and annunciator.

The system test of the Low Pressure Injection System is typically performed when the reactor is shut down for normal refueling. One train of the equipment which would be called upon to operate in the event of an SA actuating accident is tested. An SA signal is applied to the LPI/DH pump motor breaker and the LPI system valves which are required to move at the initiation of the accident. Each of these devices is considered to have operated satisfactorily when it obeys the signal as noted. The test is considered to be acceptable when the devices requiring active motion obey their respective SA signal within the specified time interval. The valves which are required to move are to be in their safety position within 30 seconds. All LPI valves are normally in their SFAS position. The decay heat pumps are tested in accordance with the Inservice Testing Program in a recirculation mode to the BWST or to the RCS to assure the capability of the pumps to perform their SFAS function as verified by pumps reaching and maintaining a specified point on their head-capacity curves. Valves in the LPI system are stroked as required by the Inservice Testing Program to verify their capability to function.

Once per 31 days, each ECCS manual, power operated and automatic valve in the flowpath that is not locked, sealed, or otherwise secured in position, is verified to be in its correct position.

The testing frequency of the systems related to emergency core cooling is specified in the Technical Specifications. The test frequencies are considered to be satisfactory. The testing is considered to give a demonstration of emergency equipment readiness. The method of conducting the test is by manually actuating the component from the control room. The device is considered to have operated acceptably when it goes to its SFAS status.

The leak tight enclosure protecting decay heat suction valves DH 11 and DH 12 from post LOCA flooding is tested in accordance with the Technical Requirements Manual to verify its leakage-prevention capability subsequent to being breached. The test creates a vacuum differential pressure equal to post LOCA hydrostatic pressure on the enclosure. The acceptance criterion is that the leakage must be less than that leakage which would result in the motor operators becoming flooded based on an accumulation of leakage over a 7 day period. High and high-high level switches (nominal setpoints 3"$^{+0}_{-1}$ and 6"$^{+0}_{-1}$ respectively from the decay heat pit bottom) notify operators that operational leakage in the valve pit has reached these levels. A 4" diameter inspection port allows inspection of the decay heat pit.

Valves CF-28, CF-29, CF-30, CF-31, DH-76 and DH-77 are leak tested to demonstrate operability. Valves CF-28, CF-29, DH-76 and DH-77 are tested at a pressure above the minimum Core Flooding Tank operating pressure which is 580 psig. Valves CF-30 and CF-31 are tested with the Reactor Coolant System pressure above 1200 psig.

The acceptance criteria for valves CF-28, CF-29, CF-30, CF-31, DH-76 and DH-77 allows a maximum leak rate of 5 gpm.

In order to protect the Decay Heat Removal/Low Pressure Injection system from overpressurization, the leak rate of check valves DH76 and DH77 must be less than the capacity of relief valves PSV-1508 and PSV-1509. Each relief valve has a design capacity of 49.0 gpm.

In order to protect the Core Flooding Tanks from overpressurization the capacity of the Core Flood Tank relief valves (PSV-CF7A and PSV-CF7B) must accommodate the maximum flow rate into the tanks. The Core Flooding Tanks may receive water from (1) filling of the tanks or (2) leakage from the Reactor Coolant System (RCS). Of these two events core flood tank filling (Table 6.3-5) is the most restrictive. Allowable check valve leakage is less and is controlled through surveillance requirements to Davis-Besse Technical Specifications.

It should be emphasized that the above leak rate values, related to ensuring overpressure protection is provided, are for the limiting conditions. For normal plant operating conditions, the acceptable leakage is in accordance with the above leak testing acceptance criteria.

6.3.4.1 Management of Gas Accumulation

On January 11, 2008, the NRC issued Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.” Generic Letter 2008-01 requested licensees to evaluate the licensing basis, design, testing, and corrective action programs for the emergency core cooling, decay heat removal, and containment spray systems.

As a result, the company performed evaluations that included the review of gas susceptible piping locations, the development of activities to monitor various piping locations as appropriate based on industry experience and plant specific experience, and acceptance of some generic locations that normally accumulate voids that do not adversely affect the design function(s) of the system, such as relief valve dead legs.

The company established a gas accumulation prevention and management program to ensure that gas accumulation is reasonably prevented or maintained less than the amount that challenges the functionality of the applicable systems and that appropriate action is taken when conditions adverse to quality are identified.

6.3.5 Instrument Application

The instrumentation provisions, design details, and logic for various methods of actuation are discussed in Chapter 7.

The following process information is available in the control room to assist the operator in assessing post-LOCA conditions:

1. Containment Vessel pressure.
2. Containment Vessel radiation.
3. Containment Vessel temperature.
4. Annulus negative pressure.
5. Hydrogen concentration and particulate activity from samples taken from containment.

In the event of an ECCS injection signal, the operator has sufficient information available in the control room to determine if a real need for cooling water does exist. The following list delineates the variables monitored by SFAS to initiate ECCS injection in the event an unsafe condition is detected. Under each item are the minimum number of normally available indications the operator may consult to determine if a real need does exist. The redundancy and diversity would make misinterpretation of all the available information highly unlikely.

1. Deleted
2. Reactor coolant pressure.

   Four reactor coolant pressure indicators (two on each leg)

   Four containment pressure indicators

   Two containment vessel post accident radiation monitors

   Two reactor coolant flow indicators (one on each leg)

3. Containment pressure.

   Four containment pressure indicators

Four reactor coolant pressure indicators (two on each leg)

Two main steam header pressure indicators (one on each header)

Two reactor coolant flow indicators (one on each leg)

Two pressurizer level indicators

One makeup tank level recorder

Analytical Basis for Pressure Setpoints Associated With Core Flooding Tanks:

The normal operating pressure is 600 psig ±20 psi. The design pressure is 700 psig.

The alarm setpoints are high: either valve closed and RC pressure ≥700 psig on the computer and ≥735 psig on the station annunciator; low: valves open and RC pressure < 675 psig.

The valves will automatically open before RC pressure exceeds 800 psig, and they are interlocked so they cannot be closed.

6.3.6 Operator Actions

Manual actions required by the operator for proper ECCS operation are described in the ATOG-based emergency operating procedures Reference 19 (DB-OP-02000). The following are some of the major manual operations required by the operator for the ECCS.

No actions are immediately required by the operator in large pipe LOCA (ones in which the primary system depressurizes over a period on the order of minutes or less). The operators must, however, promptly trip the reactor coolant pumps. The SFAS automatically starts the Emergency Core Cooling Systems, Containment Vessel Cooling Systems and their support systems. Longer term required operator actions are discussed in Sections 6.3.1.4 and are carried out as specified in the emergency operating procedures Reference 19 (DB-OP-02000). The operator actions assumed in the LOCA analyses are described in Reference 35.

For the Core Flooding Tank nozzle break, no operator action is required, except to meet longer term requirements as discussed above. One HPI pump and one Core Flooding Tank will be sufficient to refill the core and still maintain peak fuel cladding temperature within acceptable limits, i.e. less than 10CFR50.46 limits. The operator may, however, achieve a faster refill rate and abundant cooling by cross connection and flow balancing of LPI.

For some small breaks, the RC pressure might “hang up.” This may occur for very small break areas of approximately 0.1 ft^2 or less. During this time the BWST may be emptied and manual cross connection between the LPI and HPI is necessary to ensure that pump minimum NPSH requirements are met. For some very small break areas (0.00206 to 0.0045 ft^2), RCS cyclic repressurization above the shut-off head on the HPI pump could exist beyond the time that the pump suction is aligned to the BWST. To prevent damage to the HPI pumps as a result of operating at shutoff head, the operator would open the manual valves in the alternate minimum flow lines which discharge to the outlet of the LPI pumps. While not credited in the LOCA analyses, the operator would initiate cooldown using the auxiliary feedwater system thereby cooling down the primary system. This may eliminate or minimize the necessity for this cross connect if the HPI termination criterion in the plant EOPs is met. For HPI line

breaks, once actuated, operator action is required to balance the HPI flow between the injection lines.

TABLE 6.3-1

ECCS Components

Core Flooding Tanks	
Number	2
Design pressure, psig	700
Design temperature, °F	300
Operating pressure, psig	600 $\pm$20
Operating temperature, °F	110
Total volume, ft^3	1410
Normal water volume, ft^3	1040
Minimum boron concentration in water, ppm	2600
Materials of construction	CS Clad SS
Shell	SA 516, grade 70
Cladding	SS-304
Code	ASME Section III-C
HPI PUMPS	
Number	2
Type	Multi-stage centrifugal
Required NPSH (design), ft.	15
Pump material	SS
Barrel Casing	A182, F304
Casing head	A182, F304
Casing end cover	A182, F304
Pump suction nozzle	A376, SS-304
Pump disch nozzle	A376, SS-304
Disch volute section	A351, CF8
Suction volute section	
Central volute section	
Shaft	ASTM, A564, GXM 25-H 1150
Impeller first stage	A351, CF8

TABLE 6.3-1 (Continued)

ECCS Components

Impeller suction end	A351, CF8
Impeller disch end	A351, CF8
Inboard bearing hsg	A216, grade WCB
Design temp/pres., °F/psig	300/2000
Capacity for HPI requirements, gpm/ft	500/2700
Code	ASME Pump & Valve Code Class II
Bolted Bonnet Globe Valve HP-HV2A, HP-HV2C HP-HV2B, HP-HV2D	
Body	SS, grade F316, ASTM A182
Bonnet	SS, grade F316, ASTM A182
Yoke	Cast CS, grade WCB, ASTM A216
Disc	Stellite No. 6
Stem	Type 630, ASTM A461
LPI/Decay Heat Pump	
Number	2
Type	Single-stage centrifugal
Required NPSH (design), ft	8.5
Pump material	SS
Casing	A351, CF8
Impeller	A351, CF8
Shaft	A479, SS-304
Casing ring	ASTM A461, grade 630, cond H-1150
Impeller ring	A182, F304
Backcover	A351, CF8
Shaft sleeve	A351, CF8
Gland plate	A182, F316
Impeller key	A276, 304A
Design temp/pres., °F/psig	350/450
Code	ASME Pump & Valve Code Class II
Capacity for LPI requirements gpm/ft	3000/350

TABLE 6.3-1 (Continued)

ECCS Components

Decay Heat Removal Cooler

Number	2
Type	Shell & Tube
Capacity (@140°F), 10^6 Btu/hr	30 (26.9)*
Reactor coolant flow, gpm	3000
Component cooling water flow, gpm	6000
Cooling water inlet temp., °F	95
Material, shell/tube	CS/SS

Decay Heat Coolers

Rolled plate	CS, SA-285-C
Baffles	CS, SA-36
Tie rods	CS, SA-36
Spacers	CS, SA-214
Nozzle flanges	FS, SA-181-1
Nozzle necks	CS, SA-53-B
Formed heads	CS, SA-515-60
Unit flanges	FS, SA-105-II
Formed heads	SS-304, SA-240
Plate	SS-304, SA-240
Unit flanges	FS, SS-304, SA-105-11
Nozzle flanges	SS-304, SA-182
Tubesheets	SS-304, SA-240
Tubes	SS-304, SA-249
Tube-side reinf pads	SS-304, SA-240
Shell-side reinf pads	CS, SA-285-C
Design pressure, psig, shell/tube	150/450
Design temperature, °F, shell/tube	250/350

* The values in parentheses are from B&W Document 51-1172856-00, dated August 3, 1988, based on input from Atlas Industrial Manufacturing Company for the design normal case heat load assuming degraded Decay Heat Removal Cooler performance as discussed in Section 6.3.1.2.

TABLE 6.3-1 (Continued)

ECCS Components

Code ASME Code Section, tube/shell	ASHE Sec. III Class C/ASME Section VIII TEMA Class R
Borated Water Storage Tank	
Capacity, gal	550,000 Net
Material	SS
Design Press.	Atmospheric
Design Temp., °F	125
Gate Valve DH-HV7A, DH-HV9A DH-HV7B, DH-HV9B	
Body	SS, grade CF8, A351
Bonnet	SS, grade CF8, A351
Disc	SS w/P-100 Stellite facing, grade CF8, A351
Stem	SS-304, A276
B'less Globe Valve CF-HV2A, CF-HV5A CF-HV2B, CF-HV5B	
Body	SS, grade F316, ASTM A182
Stem	Type 630, ASTM A461
Disc	Stellite No. 6
BB Gate Valve CF-HV1A. CF-HV1B	
Body	SS, grade F316, ASTM A182
Bonnet	SS, grade F316, ASTM A182
Stem	Type 630. ASTM A461
Yoke	Cast CS, grade WCB, ASTM A216
Rotary Disc Valve DH-HV13A, DH-HV14A DH-HV13B, DH-HV14B	
Body	SS-316
Shaft	17-4PH
Trim	SS-316

TABLE 6.3-1 (Continued)

ECCS Components

BB Gate Valve
DH-HV1A, DH-HV1B

Body	SS, grade F316, ASTM A182
Bonnet	SS, grade F316, ASTM A182
Yoke	Cast CS, grade WCB, ASTM A216
Stem	Type 630, ASTM A461

TABLE 6.3-2

ECCS Component Design Conditions

A. Core Flooding System

		Pressure (psig)	Temp ° (F)
1.	Core Flooding Tanks	700	300
2.	Piping from CF tanks up to the EMO valves	700	300
3.	From EMO isolation valves HV-1A&B up to second check valve	2500	300
4.	From second check valves to reactor vessel	2500	650

B. Low Pressure Injection

		Pressure (psig)	Temp ° (F)
1.	Piping from the BWST outlet up to EMO valves HV-7A&B.	75	150
2.	From EMO valves HV7A & B and piping up to check valves DH-81 & 82.	75	150
3.	From check valves DH-81 & 82 and piping to EMO valves HV9A & B and HV2733 & HV2734.	75	300
4.	From EMO valves HV2733 & HV2734 and piping to LPI/DH pumps.	320	350
5.	From LPI/DH pumps and piping including DH coolers up to EMO valves HV1A & B.	450	350
6.	From EMO valves HV1A & B and piping up to check valves DH-76 & 77.	2500	350

TABLE 6.3-2 (Continued)

ECCS Component Design Conditions

C. High Pressure Injection

		Pressure (psig)	Temp ° (F)
1.	From BWST suction to check valves HP-10 & 11.	75	150
2.	From check valves HP-10 & 11 to HPI pumps.	450	350
3.	HPI pumps.	2000	300
4.	From HPI pump discharge flange to check valves HP-22 and HP-23.	2000	300
5.	From check valves HP-22 and HP-23 to EMO valves HP-2A, B, C and D.	2600	200
6.	From EMO valves HP-2A, B, C & D to check valves HP-48, 49, 56 & 57.	2500	650
7.	From check valves HP-50, 51, 58 & 59 to R. C. system piping.	2500	650

TABLE 6.3-3

Decay Heat Removal Cooler Characteristics

(These data are for SFAS Level 4 conditions)**

	Shell Side	Tube Side
Fluid	Component Cooling Water	Reactor Coolant
Flow, lb/hr	3.0×10^6	1.5×10^6
Temperature in, °F	119.3	250
Temperature out, °F	154.3 (146.0)*	180 (196.7)*
Number of passes	1	2
Pressure drop, psi	13.3	3.2
Heat transfer Surface area, ft^2	3092	
Overall heat transfer coefficient, Btu/hr-°F-ft^2	478 (service)	(300)*
	562 (clean)	
Heat transfer rate, Btu/hr	105×10^6	(81×10^6)*
LMTD (corrected), °F	71.2	(87.3)*

* NOTE: The values in parentheses are from Toledo Edison Calculation C-NSA-049.02-13 for a revised service overall heat transfer coefficient of 300 BTU/hr-°F-ft^2.

** SFAS level 4 conditions are the bounding case. Lower Component Cooling Water flow rates are acceptable for SFAS level 3 conditions as analyzed in approved engineering calculations.

TABLE 6.3-4

Process Information

	EMERGENCY				TEST			
Mode No.	Press (psig)	Temp (°F)	Flow (gpm)	Comments	Press (psig)	Temp (°F)	Flow (gpm)	Comments
1	30	80 (40 min.)	6050-5000	[2]	30	80 (40 min.)	3000/1500	[1]DH/CS Pump Oper.
2	30	80 (40 min.)	5550-5000	[2]	30	80 (40 min.)	3000/1500	[1]DH/CS Pump Oper.
3	36-atm	250-120	4500(500)[4]	[3]	NA	NA	NA	
4	30	80 (40 min.)	1800-1500	[2]	30	80 (40 min.)	1500	[1]CS Pump Oper.
	36-atm	250-120	1560	[3]	NA	NA	NA	
5	30	80 (40 min.)	3750-3000	[2]	30	80 (40 min.)	3000	[1]DH Pump Oper.
	36-atm	250-120	3000(700) [4]	[3]	NA	NA	NA	
6	180	80 (40 min.)	3750-3000	[2]	180	80 (40 min.)	3000	[1]DH Pump Oper.
	166-150	180-120	3000(700)[4]	[3]	NA	NA	NA	
7	NA	NA	NA		30	80 (40 min.)	3000	[1]DH Pump Oper.
8	100	80 (40 min.)	3750-3000	[2]	NA	NA	NA	
	100-atm	180-120	3000	[3]	NA	NA	NA	
9	30	80 (40 min.)	500	[2]	NA	NA	NA	
10	186-150	240-120	500[4]	[3]	NA	NA	NA	
11	1000	80 (40 min.)	500	[2]	NA	NA	NA	Tested during normal operation for partial flow
	1000	240-120	500[4]	[3]	NA	NA	NA	
12	600	240-120	250[4]	[3]	NA	NA	NA	
	600	80 (40 min.)	250	[2]	NA	NA	NA	
13	600-0	CV Ambient	----	Tank empty in 30 sec.		NA	NA	Tested for opening of check valves during cooldown

TABLE 6.3-4 (Continued)

Process Information

NOTES:

(1) All pressures, flows, & temperatures shown are estimated values.
(2) Suction from the BWST.
(3) Suction from the containment vessel emergency sump.
(4) Used only for small. breaks where recirculation is through HPI pumps.

TABLE 6.3-5

Relief Valves in ECCS

Mark No. Location	Set Press. psig	Capacity 10% Accumulation	Function
PSV-1508, 1509 CV emergency sump- BWST header	75	~49 gpm	Protect against over-pressurization due to ambient temperature change and leakage from normal DH flow path through closed valves during normal DH system operation.
PSV-1529, 1550 In DH injection line	450	~38 gpm	Protect against leakage from RC system thru normally closed DH injection valves during normal RC system operation and protect against overpressurization due to ambient temperature change.
PSV-2762 Vacuum Breaker BWST	1 oz/in^2	1200 scfm	Protect tank from vacuum at maximum draining rate and protect against external pressure during tornado.
PSV-CF7A, CF7B Core Flooding Tank	700	226 scfm (minimum capacity)	Protect CF tank from maximum fill rate from High Pressure Injection pumps.

TABLE 6.3-6

Single Failure Analysis - Emergency Core Cooling System

Component Identification: No.	Component Identification: Description	Component Position: Normal Condition	Single Failure	Evaluation
A. High Pressure Injection System (Fig. 6.3-2)				
DH7A or DH7B	Suction valve for HPI pump from BWST	Open (also SA signal to open)	Closes	Breakers BF1148 and BE1157 will be maintained in the open position for valves DH7A and DH7B respectively. This has been evaluated and is acceptable in resolving the Appendix R concerns of a fire induced fault causing valves DH7A and DH7B to actuate. However, in the event of inadvertent closure, the other HPI train or subsystem provides 100% of design flow.
			Fails to Close	During CTMT sump recirculation reverse flow to BWST is prevented by HP10 and HP11. These valves are included in the Periodic Check Valve Testing Program to ensure function.
HP2A, HP2B, HP2C, or HP2D	High pressure injection line valve	SA Signal to Open	Closes	Other line of affected train provides more than 50% of design flow and unaffected train provides 100% of design flow.
HP Pump 1 or HP Pump 2	High pressure injection pump	Not operating	Fails to start	Other HPI train or subsystem provides 100% of design flow.
Failure of one of the two redundant diesels or electrical fault on 4KV or 480 V bus	---	---	---	Other HPI train or subsystem provides 100% of design flow.
HP-31 (See Note 2) or HP-32	High pressure injection pump minimum Recirculation line isolation valve	1) Open (Injection Phase)	Closed	No effect unless the pump is pumping at shutoff. If the pump is at shutoff and is damaged, the pump will be stopped. The redundant train is adequate to mitigate the consequences of the accident.

TABLE 6.3-6 (Continued)

Single Failure Analysis - Emergency Core Cooling System

Component Identification: No.	Description	Component Position: Normal Condition	Single Failure	Evaluation
	2)	Closed (Piggy Back Mode)	Open	If the valve will not close the HPI pump will be stopped and it's LPI to HPI cross-tie will be closed. One train is adequate to mtigate the consequences of the accident (See Note 1)
B.	Low Pressure Injection System (Fig. 6.3-2A) for break other than in core flood line			
DH7A or DH7B	Suction valve for HPI/DH pump from BWST	Open (also SA Signal to open)	Closes	Breakers BF1148 and BE1157 will be maintained in the open position for valves DH7A and DH7B respectively. This has been evaluated and is acceptable in resolving the Appendix R concerns of a fire induced fault causing valves DH7A and DH7B to actuate. However, in the event of inadvertent closure, the other HPI train or subsystem provides 100% of design flow.
			Fails to Close	During CTMT sump recirculation, reverse flow to the BWST Is prevented by DH82 and DH81. These valves are included in the Periodic Check Valve Testing Program to ensure function.
DH1A or DH1B	Low pressure injection line valve	Open (control power discon-nected)	No single active failure credible	---

Note 1: The dose rate at the site boundary due to "shine" from the Borated Water Storage Tank (BWST) has been evaluated for this case and found to be 300 mr/hr. This was based on a site boundary minimum distance of 737 meters. The flow through the line to the BWST was assumed to be 500 gallons (expected flow rate is 35 gpm). The activity entering the BWST was the activity in the Containment Vessel Emergency Sump water, containing 50 percent of the core saturation inventory consistent with TID-14844 specifications. This activity had decayed for 90 minutes, the time at which the recirculation mode is initiated, based on the worst RCS break (0.1 ft^2) for which the piggyback mode may be required. The dose rate was determined by considering the BWST as a point source conservatively neglecting self-attenuation of the water in the tank and not taking any credit for dilution with water already in the lower portion of the tank.

Note 2: Breaker BF1194 will be maintained in the open position for valve HP-31. This has been evaluated and is acceptable in resolving the Appendix R concerns of a fire induced fault causing valve HP-31 to actuate.

TABLE 6.3-6 (Continued)

Single Failure Analysis - Emergency Core Cooling System

Component Identification		Component Position		
No.	Description	Normal Condition	Single Failure	Evaluation
DH14A or DH14B	LPT line flow control valve	Open (Also SA signal to open and control power disconnected hand wheels are locally locked)	No single active failure credible	---
DH13A or DH13B	Decay heat removal cooler bypass line flow control valve	Closed (Also SA signal to close)	Opens	Would reduce heat removal rate of that cooler but other LPI train or subsystem is not affected and provides100% of design heat removal.
DH12 or DH11	Isolation valves in DH suction line (from hot leg)	Closed	Opens	No effect on LPI capability.
DH1517 or DH1518	Isolation valves in DH suction line (from hot leg)	Closed	Opens	No effect on LPI capability.
DH9A or DH9B	Isolation valve in containment emergency sump outlet line	1. Closed (Also SA signal to close) (injection phase from BWST)	1. No single active failure credible.	1. Breakers BF 1142 and BE 1112 will be maintained in the open position for valves DH9A and DH9B respectively. This has been evaluated and is acceptable in resolving the Appendix R concerns of a fire induced fault, causing valves DH09A and DH09B to actuate. For a break necessitating alignment of the HPI pump for operation in the piggyback mode, if the single failure opening of DH9A(9B) occurred after the LPI-HPI cross-tie was acccomplished by opening DH63 or 64), the HPI pump on the connected train

TABLE 6.3-6 (Continued)

Single Failure Analysis - Emergency Core Cooling System

Component Identification		Component Position		
No.	Description	Normal Condition	Single Failure	Evaluation
				could be affected due to possible air binding, depending on the amount of water in the containment emergency sump. It is most probable, however, that there will be adequate water in the sump to preclude such a problem since the HPI pump will be aligned after sufficient water has accumulated. At any rate, there will be no effect on the other LPI or HPI train. At Lo-Lo level in the BWST, the suction for LPI and containment spray pumps will be manually transferred from the BWST to the containment emergency sump.
		2. Open (during recirculation phase) (Also SA signal to permit opening on Lo-Lo BWST level)	2. Closes	2. Stops flow in one LPI train or subsystem but other LPI train is not affected and provides 100% of design flow. If HPI cross connect is being used, to only one HPI pump can be affected.
DH2736 or DH2735	Isolation valve in auxiliary Pressurizer spray line	Closed	Opens	No effect on LPI capability.
DH2733 or DH2734	Isolation valve in suction line of LPI/DH pump	Open (Also SA signal to open)	Closes	Other LPI train or subsystem provides 100% of desgn flow.

TABLE 6.3-6 (Continued)

Single Failure Analysis - Emergency Core Cooling System

Component Identification		Component Position		
No.	Description	Normal Condition	Single Failure	Evaluation
DH830 or DH831	Isolation valves in cross-connect between the two LPI trains.	Closed	Opens	No effect on LPI capability.
LPI/DH Pump 1 or LPI/DH Pump 2	Low pressure injection/decay heat pump	Not operating	Fails to start	Other LPI train or subsystem provides 100% of design flow.
Failure of one of the two redundant diesels or . electrical fault on 4KV or 480 V bus	---	---	---	Other LPI train or subsystem provides 100% of design flow.

C. Low Pressure Injection System (Fig. 6.3-2A) for core flood line break

No.	Description	Normal Condition	Single Failure	Evaluation
DH1A or DH1B	Low pressure injection line valve	Open (Control power disconnected)	No single active failure credible	---
DH14A or DH14B	LPI line flow control valve	Open (Also SA signal to Open and control power disconnected hand wheels are locally locked.	No single active failure credible	---

TABLE 6.3-6 (Continued)

Single Failure Analysis - Emergency Core Cooling System

Component Identification		Component Position		
No.	Description	Normal Condition	Single Failure	Evaluation
DH13A or DH13B	Decay heat removal cooler bypass line flow control valve	Closed (Also SA signal to Close)	Opens	During injection phase, if failure is in unbroken train there is no effect on core cooling since flow is from BWST and no cooling through HX is required. During recirculation phase, flow is re-established in both LPI trains by opening manual cross-connect using valve DH830 or DH831 (operable from control room).
DH7A or DH7B	Suction valve for LPI/DH pump from BWST	Open (Also SA signal to Open)	Closes	Breakers BF1148 and BE1157 wiil be maintained in the open position for valves DH7A and DH7B respectively. This has been evaluated and is acceptable in resolving the Appendix R concerns of a fire induced fault causing valves DH7A and DH7B to actuate. However, in the event of inadvertent closure, the flow is re-established in both LPI trains by opening manual cross-connect using valve DH830 or DH831 (operable from control room).
			Fails to Close	During CTMT sump recirculation, reverse flow to the BWST is prevented by DH82 and DH81.
DH12 or DH11	Isolation valves in DH suction line (from hot leg)	Closed	Opens	No effect on LPI capability.

TABLE 6.3-6 (Continued)

Single Failure Analysis - Emergency Core Cooling System

Component Identification		Component Position		
No.	Description	Normal Condition	Single Failure	Evaluation
DH1517 or DH1518	Isolation valves in DH suction line (from hot leg)	Closed	Opens	No effect on LPI capability.
DH9A or DH9B	Isolation valve in containing emergency sump outlet line	1. Closed (Also SA signal to close) (injection phase from BWST)	1. Opens	1. Same as in B above, except that flow is re-established in both LPI trains by opening manual cross-connect using valve DH830 or DH831 (operable from control room).
		2. Open (Also SA signal to open along with manual initiation) (recirculation phase)	2. Closed	2. Stops flow in affected loop. Flow is re-established in both LPI trains by opening manual cross-connect using valve DH830 or DH831 (operable from control room).
DH2736 or DH2735	Isolation valve in auxiliary pressurizer spray line	Closed	Opens	No effect on LPI capability.
DH2733 or DH2734	Isolation valves in suction line of LPI/DH pump	Open (Also SA signal to open)	Closes	Flow-is re-established in both LPI trains by operating manual cross-connect using valve DH830 or DH831 (operable from control room).

TABLE 6.3-6 (Continued)

Single Failure Analysis - Emergency Core Cooling System

Component Identification		Component Position		
No.	Description	Normal Condition	Single Failure	Evaluation
Failure of one of the two redundant diesels or electrical fault on 4KV or 480 V bus	---	---	---	Valves on affected train are inoperable, but remain in normal position (see Normal Valve Position column), which is the safe position. If failure is in unbroken train, that pump is inoperable. Flow is re-established in both LPI trains by opening manual cross-connect using valve DH830 or DH831 (operable from control room).
LPI/DH Pump 1 or LPI/DH Pump 2	Low pressure injection/decay heat pump	Not operating	Fails to start	Flow is re-established in both LPI trains by opening manual cross-connect using valve DH830 or DH831 (operable from control room).
D. Core Flooding System (Fig. 6.3-1A)				
CF1A or CF1B	CF tank discharge line isolation valve	Open (Also control to open and interlock to prevent closing and power removed from motor control center	No single failure credible	---
CF2A or CF2B or CF2C	CF tank drain line isolation valve	Closed	Opens	No effect as there is a normally closed "SA" valve in series with the motor operated valve. The "SA" valve is the containment outside isolation valve.

TABLE 6.3-6 (Continued)

Single Failure Analysis - Emergency Core Cooling System

Component Identification		Component Position		
No.	Description	Normal Condition	Single Failure	Evaluation
CF5A or CF5B	CF tank vent line isolation valve	Closed	Opens	No effect as there is a normally closed "SA" valve in series with the motor operated valve. The "SA" valve is the containment outside isolation valve.
Failure of one of the two redundant diesels or electrical fault on 4KV or 480 V bus	---	---	---	No effect on injection capability. System does not depend on powered component movement.
CF7A or CF7B	CF tank relief Valve	Closed	Opens during normal operation	Loss of nitrogen pressure. Shutdown as per Technical Specification.
---	Check valves in discharge line	Closed	Excessive leak detected during normal Operation	For details, refer to response to position question 6.3.2 (4/18/75).

TABLE 6.3-8

Sources of Noncondensibles – 177 FA Plant

		Total Available				% Failed Fuel				1% Metal - Water Reaction			
Source	Gas	Total Volume scf	Individual Gas Volumes scf	Total Mass lb.	Individual Gas Masses lb.	Total Vol. scf	Ind. Gas Vol. scf	Total Mass lb.	Indivi--dual Gas Masses lb.	Total Vol. scf	Ind. Vol. Gas scf	Total Mass lb.	Individual Gas Masses lb.
Dissolved in reactor coolant	H_2 & N_2	563	H_2 – 305 N_2 – 158	14	H_2 – 1.7 N_2 – 12.3								
Pressurizer steam space	H_2 & N_2	136	H_2 – 65 N_2 – 71	5.9	H_2 – 0.4 N_2 – 5.5								
Pressurizer water space	H_2 & N_2	30	H_2 – 20 N_2 - 10	0.91	H_2 – 0.11 N_2 – 0.8								
Fission gases in core	Kr & Xe	186	Kr – 20 Xe - 166	65.5	Kr – 4.8 Xe – 60.7	1.9	Kr – 0.2 Xe – 1.7	0.66	Kr – 0.05 Xe – 0.61	1.9			
Fuel rod fill gas	He & some H_2 & O_2	1133	He – 1092 N_2 – 32 O_2 - 9	14.8	He – 11.5 N_2 – 2.8 O_2 – 0.8	11.3	He – 10.9 N_2 – 0.3 O_2 – 0.1	0.16	He – 0.12 N_2 – 0.03 O_2 – 0.01	11.3			
Metal water* reaction (100%)	H_2	416,500	-	2320	-					4165	-	23.2	-
MU tank gas space	H_2 & N_2	726	H_2 – 421 N_2 – 305	26.1	H_2 – 2.3 N_2 – 23.8								
MU tank water space	H_2 & N_2	24	H_2 – 16 N_2 - 8	0.71	H_2 – 0.09 N_2 – 0.62								
BWST	Air (N_2 & O_2)	1383	N_2 – 902 O_2 – 481	121.2 -50.9	N_2 – 70.3 O_2								
CF tank gas space (two tanks)	N_2	26,248	-	2047	-								
CF tank water space (two tanks)	N_2	984	-	75	-								

TABLE 6.3-8 (Continued)

Sources of Noncondensibles - 177 FA Plant

Assumptions

1. RCS contains 40 std. cc N_2/Kg water & 20 std. cc N_2/Kg water, with water volume = 10,690ft^3 at 583°F and 2200 psia.

2. Pressurizer water contains 40 std. cc H_2/Kg water & 20 std. cc N_2/Kg water with Henry's Law relation between water space and steam space at 650°F. Water volume = 825ft^3 and steam volume = 716ft^3.

3. Fission gases based on inventory in core at 292 EFPD.

4. Fuel rod gas based on each rod containing 0.0375 gmol He, 0.0011 gmol N_2 and 0.00029 gmol/O_2.

5. Metal-water reaction based on 52,000 lb. Zr cladding.

6. MU tank values based on tank containing 200ft^3 gas space and 400ft^3 water space at 120°F with the water containing 40 std. cc H_2/Kg and 20 std. cc N_2/Kg with Henry's Law relationship between gases in water and in gas space.

7. BWST contains 450,000** gallons of water saturated with air, i.e., 15 std. cc N_2/Kg and 8 std. cc O_2/Kg.

8. Each CF tank contains 1040ft^3 water and 370ft^3 gas space with 625 psig N_2 at 120°F with Henry's Law relation between water and gas.

9. Values for 1% failed fuel based on Xe and Kr fission product inventory and fuel rod fill gas (He) in 1% of fuel rods being released to coolant.

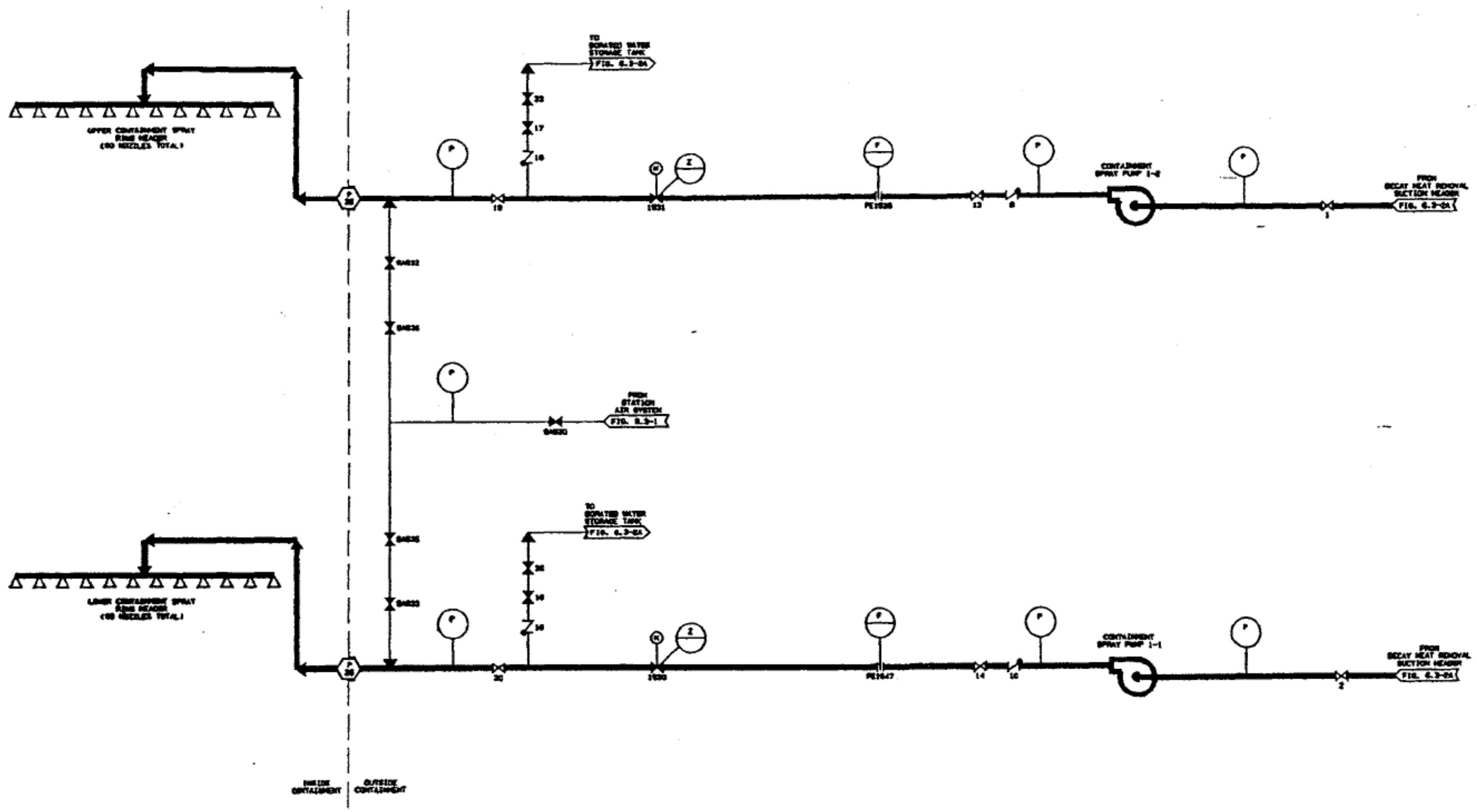
10. Values for 1% metal-water reaction based on gases in Item 9 above and H_2 released from 1% of Zr cladding (520 lb.) reacting with coolant.

* The hydrogen generation due to the fuel design change which incorporated zircaloy spacer grids was shown to be acceptable as analyzed in B&W document 32-1175316-00. The fuel design change for the use of M5 cladding instead of zircaloy is also considered bounded by the above and considered acceptable as analyzed in FTI document 86-5006232-00.

** The 450,000 gallons is an original representative value. Variations to this value are insignificant with respect to total non-condensable gas volume.

NOTES

1. ALL VALVES ARE PREFIXED WITH "CS" UNLESS OTHERWISE NOTED.



REVISION 14
JULY 1991

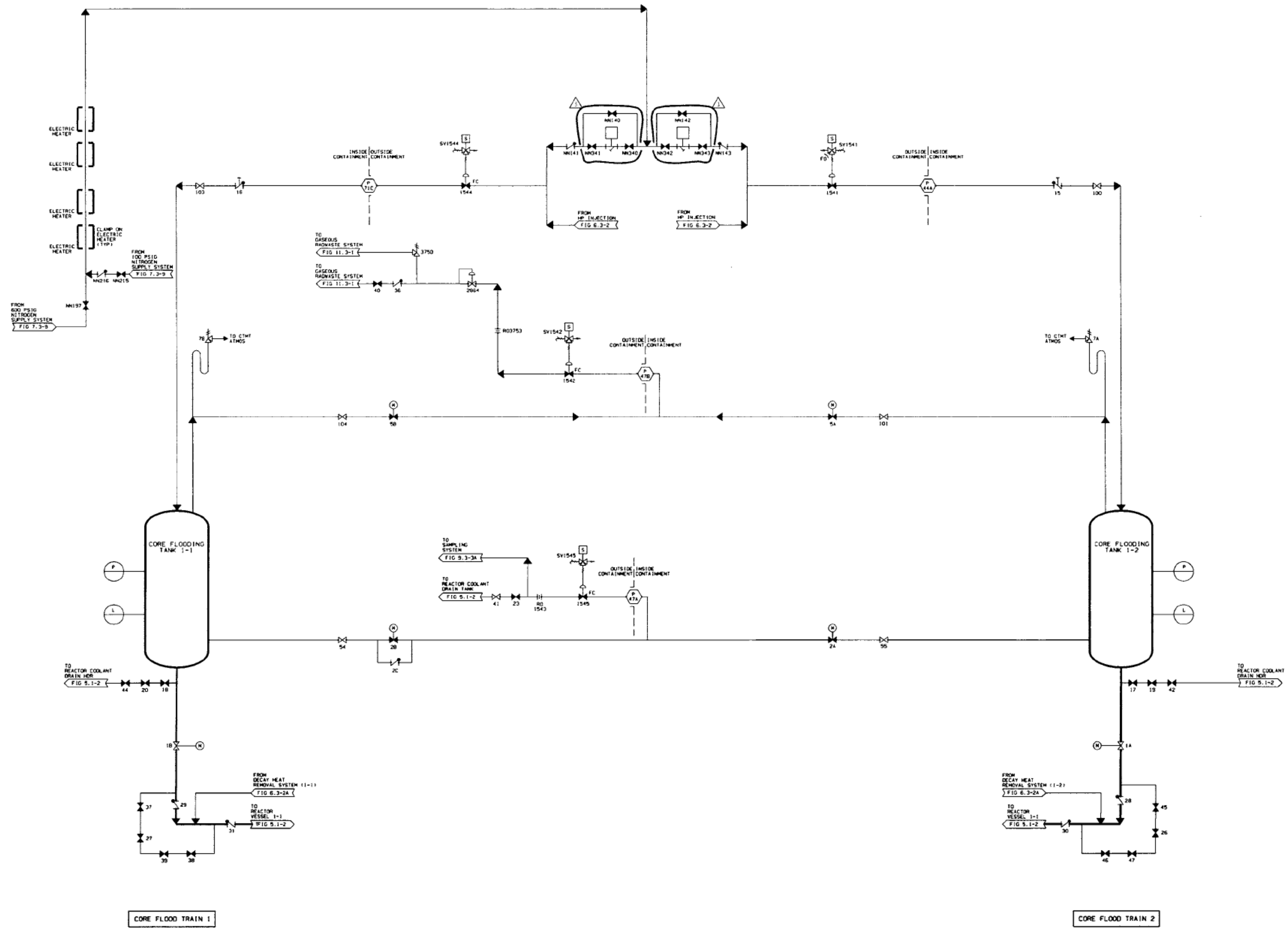
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
CONTAINMENT SPRAY SYSTEM

FIGURE NO: FIGURE 6.3-1 | REV: 0

NOTES

1. ALL VALVES ARE PREFIXED WITH "CF" UNLESS OTHERWISE NOTED.



REVISION 30
OCTOBER 2014

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	MED MGR
3		INC. CN 11-231, 11-232, 14-193	SBW				
2		INC UCN 97-111J	"INITIALS ON FILE"				
1		INC UCN 94-047	"INITIALS ON FILE"				
0	5-21-90	INFORMATION ONLY	"INITIALS ON FILE"				

SCALE NONE | DESIGNED NJD | DRAWN CPH | DATE 04-10-90

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
CORE FLOODING SYSTEM

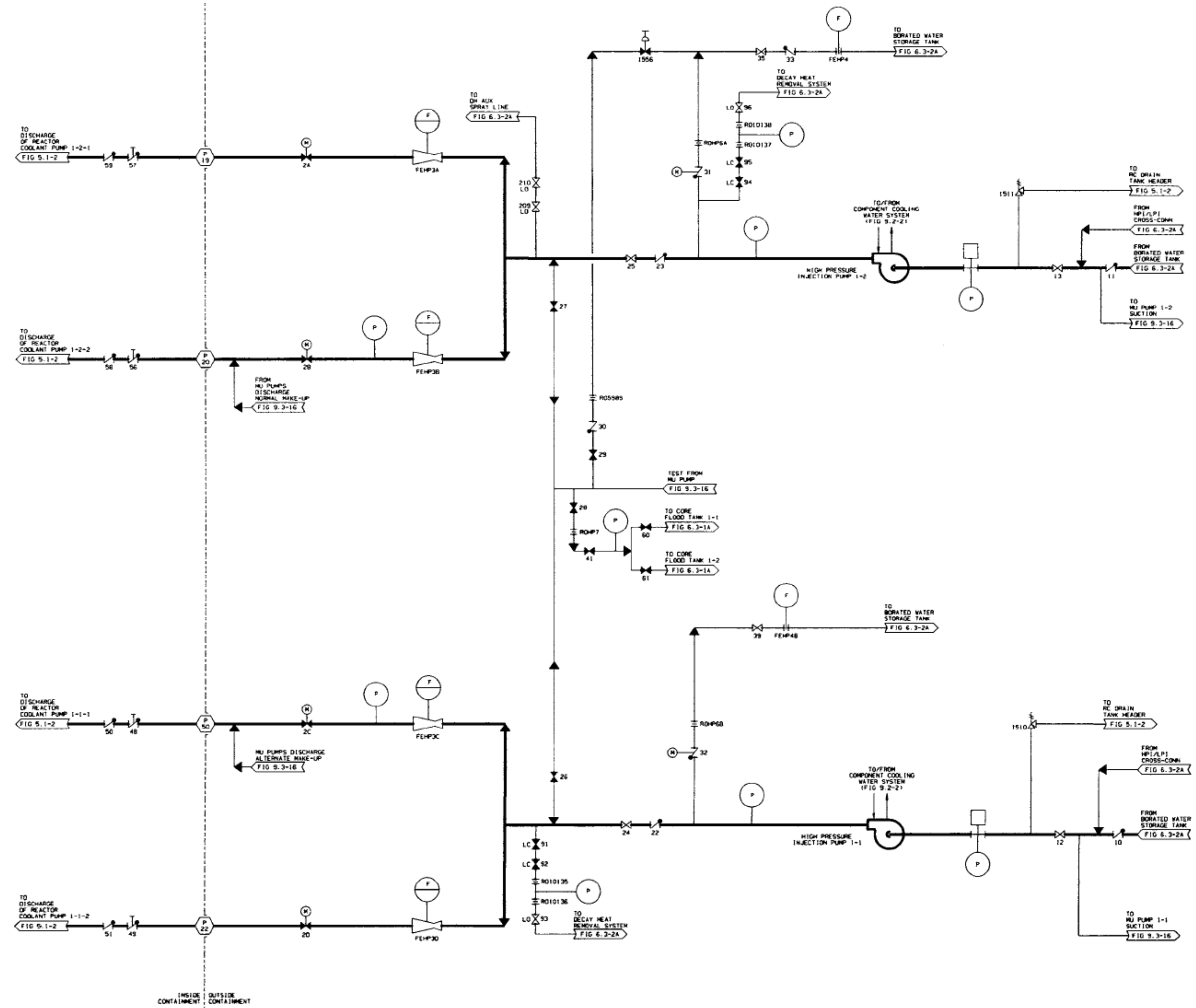
FIGURE NO: FIGURET 6.3-1A | REV: 3

052-175-27310S006.DGN

DFN=/USAR/UFIG631A.DGN
DB 08-22-14

NOTES

1. ALL VALVES ARE PREFIXED WITH "HP" UNLESS OTHERWISE NOTED.



THE CN CHANGE FOR THIS FIGURE WAS TO REPRINT IT TO SHOW CORRECT LINE WEIGHT AS REQUIRED BY DBBP-DM-1002, ATTACHMENT 11. THERE WAS NO PHYSICAL OR TECHNICAL CHANGES MADE. WHOLE DRAWING NOT CLOUDED FOR CLARITY. REMOVE THIS NOTE AT NEXT UPDATE.

REVISION 30
OCTOBER 2014

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	NES MGR
4		INC. UCN 10-004	"INITIALS ON FILE"				
3	/ /	INC. UCN 07-053	"INITIALS ON FILE"				
6	/ /	INC. CN 14-245	SBW				
5	/ /	INC. CN 07-197	"INITIALS ON FILE"				
0	5/29/91	INFORMATION ONLY	"INITIALS ON FILE"				

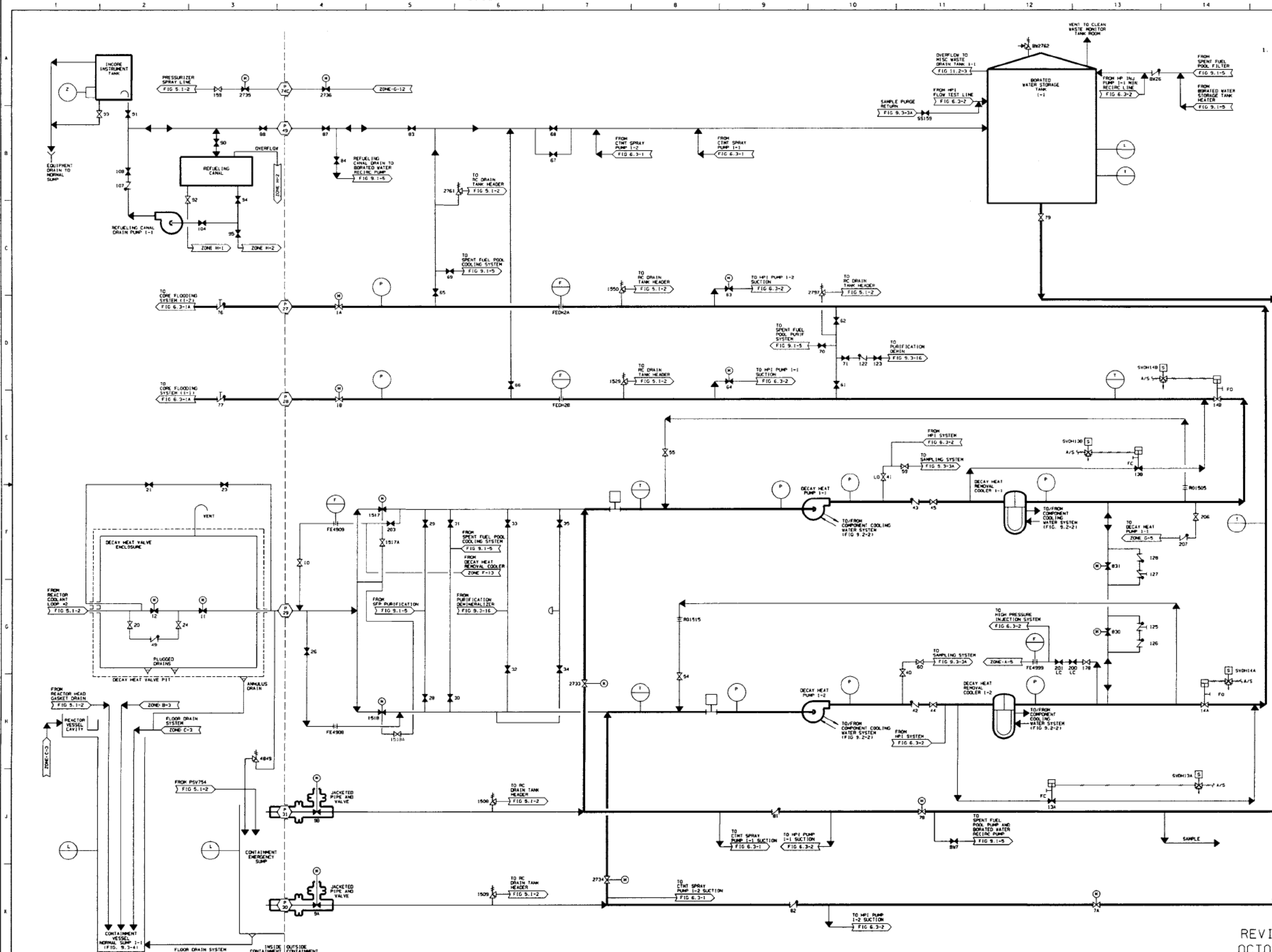
SCALE NONE | DESIGNED AJD | DRAWN AGM | DATE 04-09-90

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
HIGH PRESSURE INJECTION SYSTEM

FIGURE NO: FIGURE 6.3-2 | REV: 6

DFN=/G/USAR/UF1G632.DGN
DB 09-30-14



NOTES

1. ALL VALVES ARE PREFIXED WITH "DH" UNLESS OTHERWISE NOTED.

THE CN CHANGE FOR THIS FIGURE WAS TO REPRINT IT TO SHOW CORRECT LINE WEIGHT AS REQUIRED BY DBBP-CM-1002, ATTACHMENT 11. THERE WAS NO PHYSICAL OR TECHNICAL CHANGES MADE. WHOLE DRAWING NOT CLOUDED FOR CLARITY.
REMOVE THIS NOTE AT NEXT UPDATE.

REVISION 30
OCTOBER 2014

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	MED MGR
4		INC. UCN'S 03-032U, 03-076U, 03-084U AND 03-0119U	"INITIALS ON FILE"				
7		INC. CN 14-245	SBW				
6		INC. CN 11-144	"INITIALS ON FILE"				
5		INC. UCN 04-051U	"INITIALS ON FILE"				
3		INFORMATION ONLY	"INITIALS ON FILE"				

SCALE NONE | DESIGNED RSM | DRAWN JDR | DATE 04-09-90

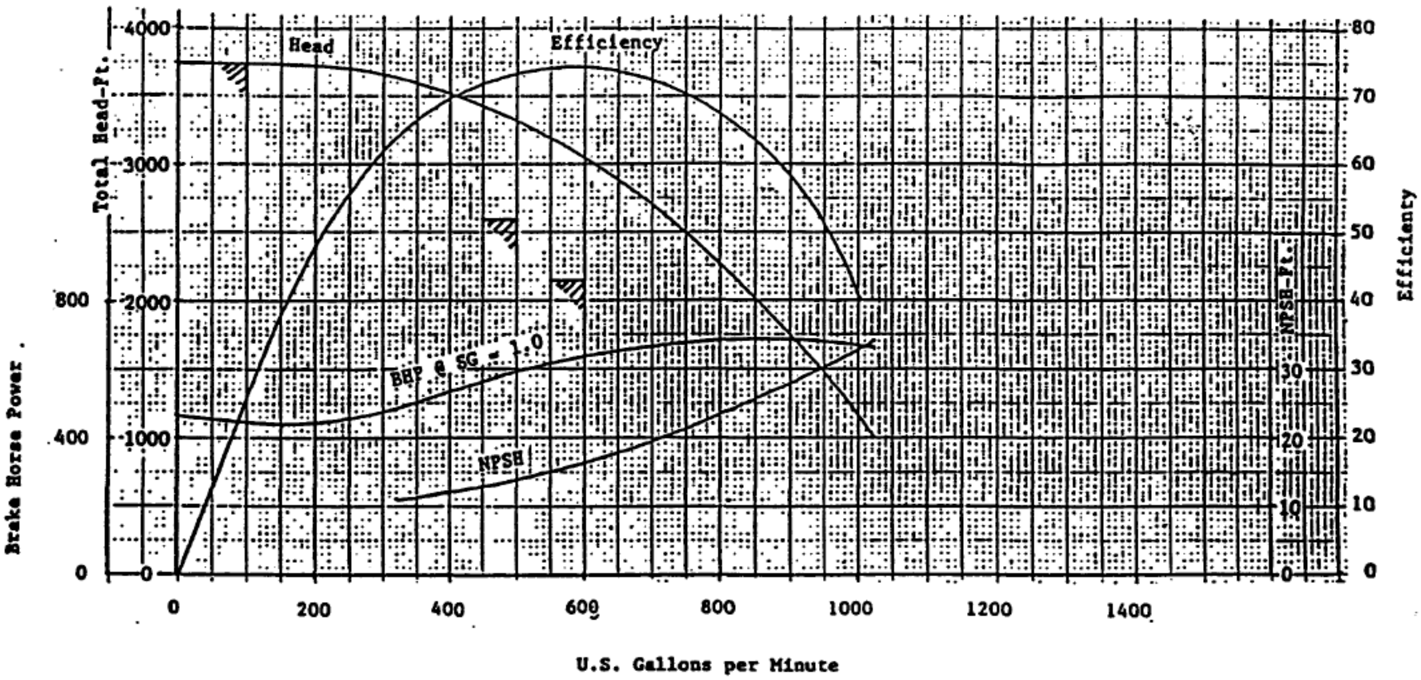
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
DECAY HEAT REMOVAL/
LOW PRESSURE INJECTION SYSTEM

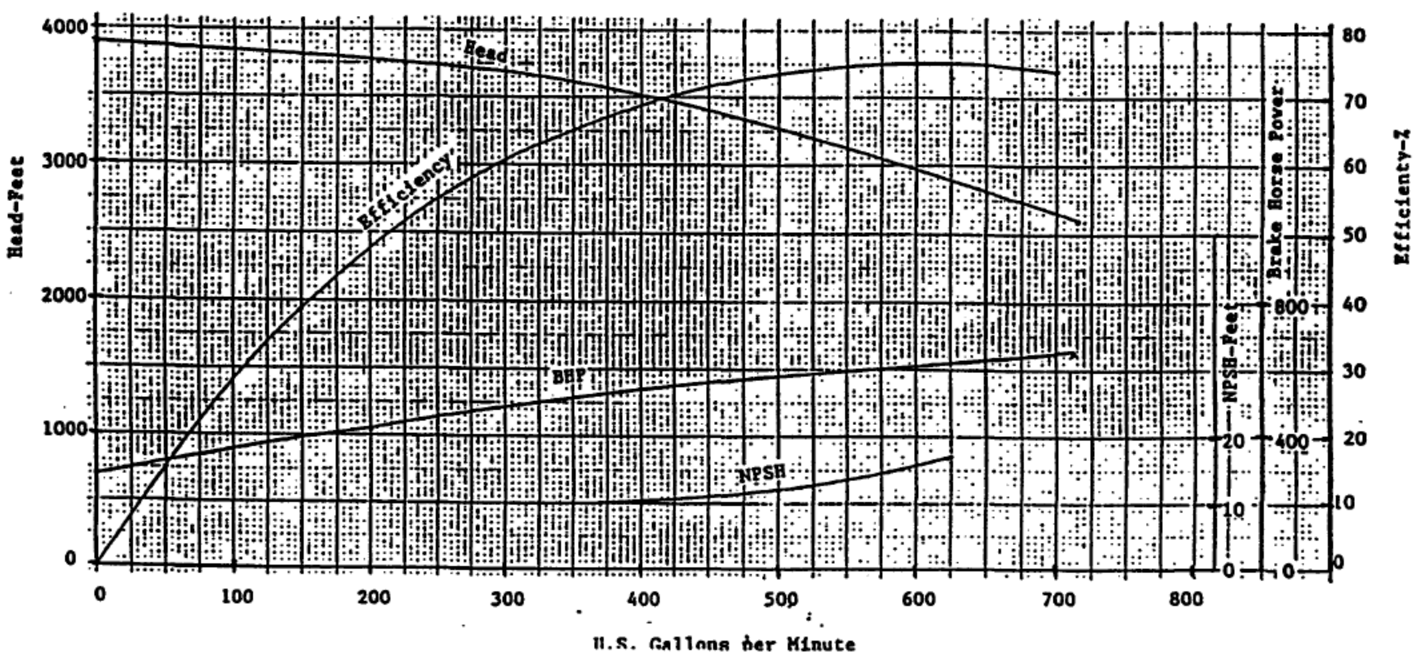
FIGURE NO: FIGURE 6.3-2A | REV: 7

DFN=/G/USAR/JFIG632A.DGN
DB 10-01-14

Equipment Name: HPI Pump 1-1

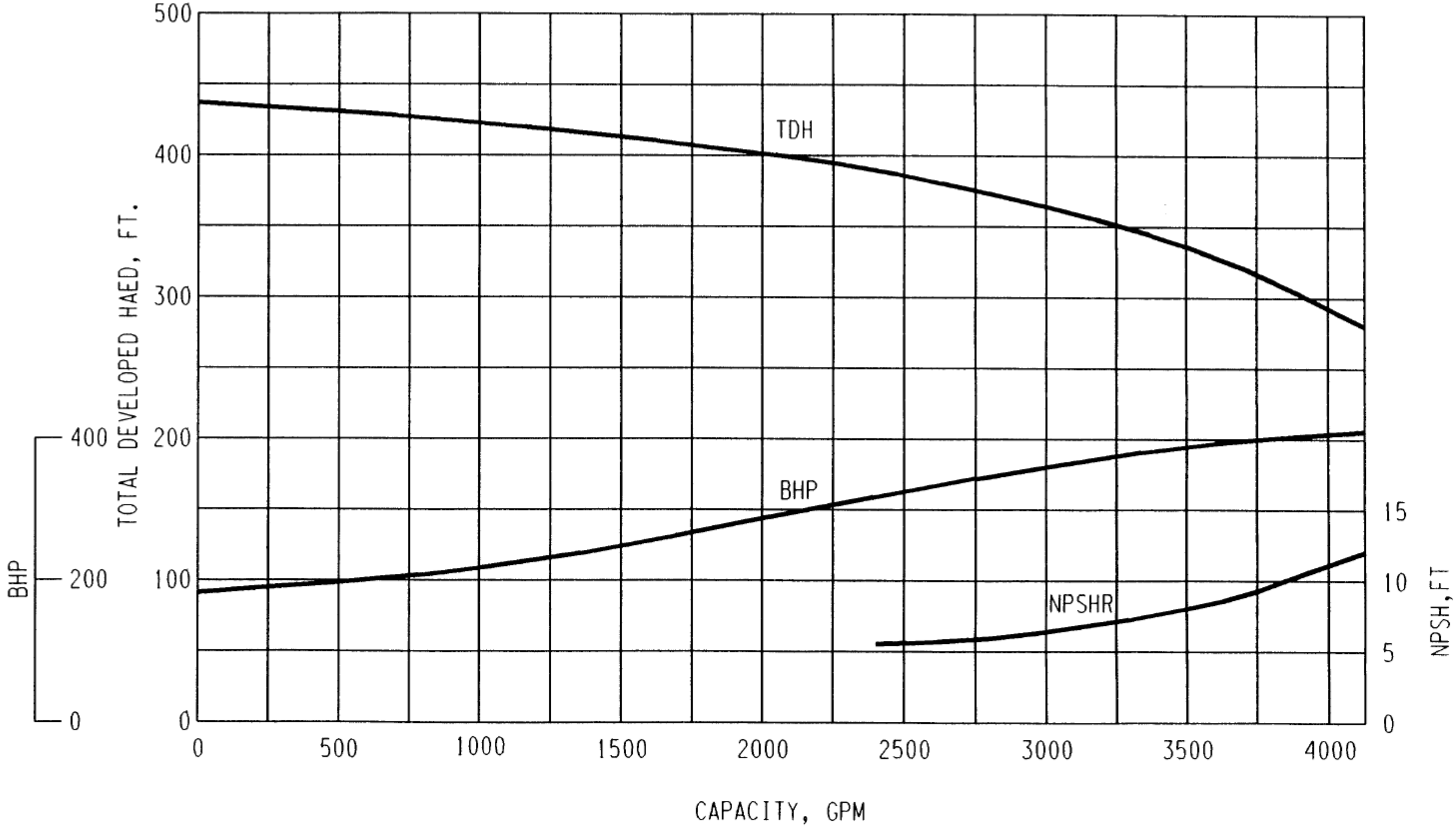


Equipment Name: High Pressure Injection Pump 1-2



DAVIS-BESSE NUCLEAR POWER STATION
HIGH PRESSURE INJECTION PUMP DATA
FIGURE 6.3-3

REVISION 20
DECEMBER 1996



DAVIS-BESSE NUCLEAR POWER STATION
DECAY HEAT PUMP NO. 1 DATA
FIGURE 6.3-4

REVISION 22
NOVEMBER 2000

DB 06-28-00

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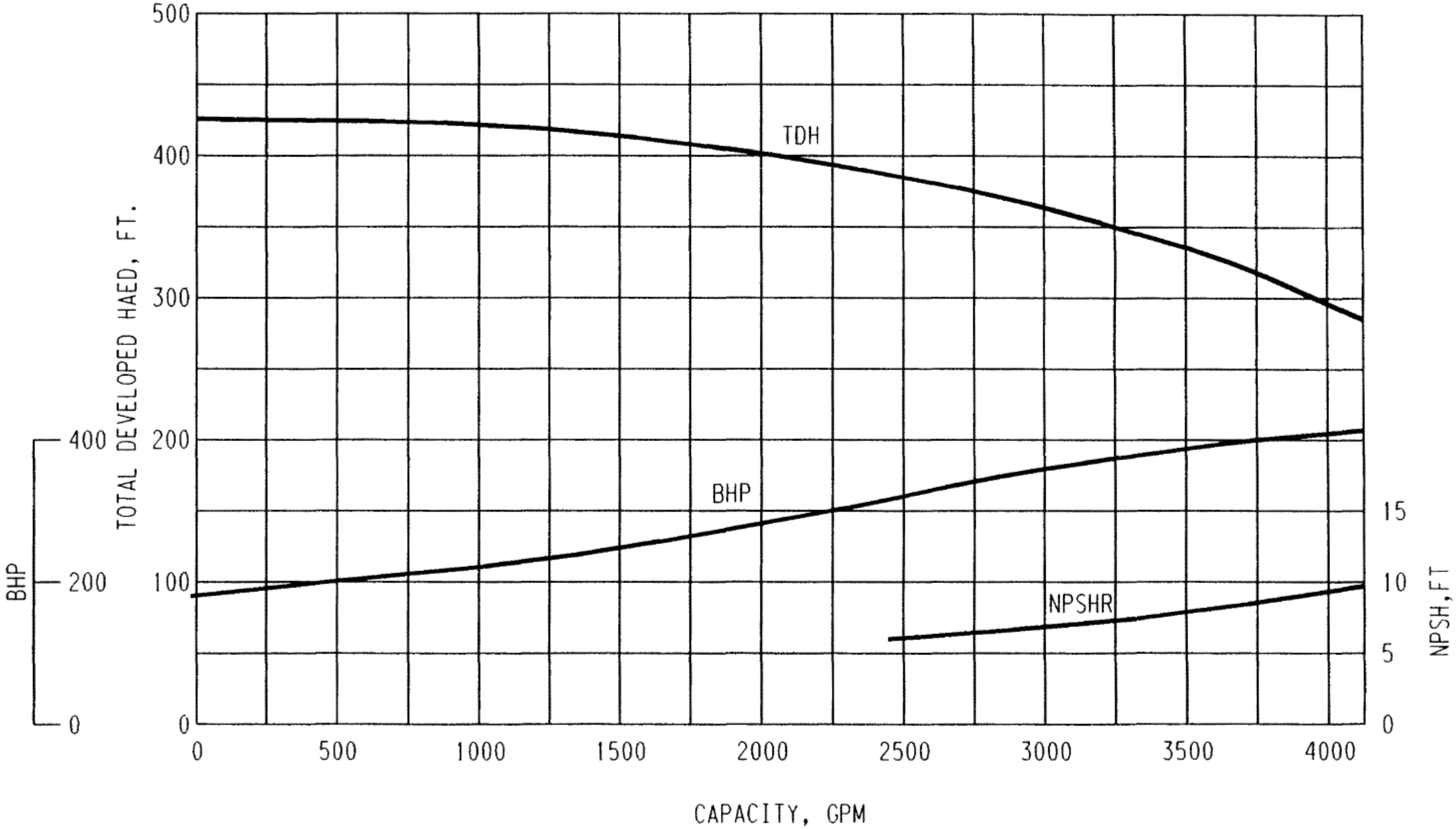

DAVIS-BESSE NUCLEAR POWER STATION
DECAY HEAT PUMP NO. 2 DATA
FIGURE 6.3-5

REVISION 22
NOVEMBER 2000

DB 06-28-00

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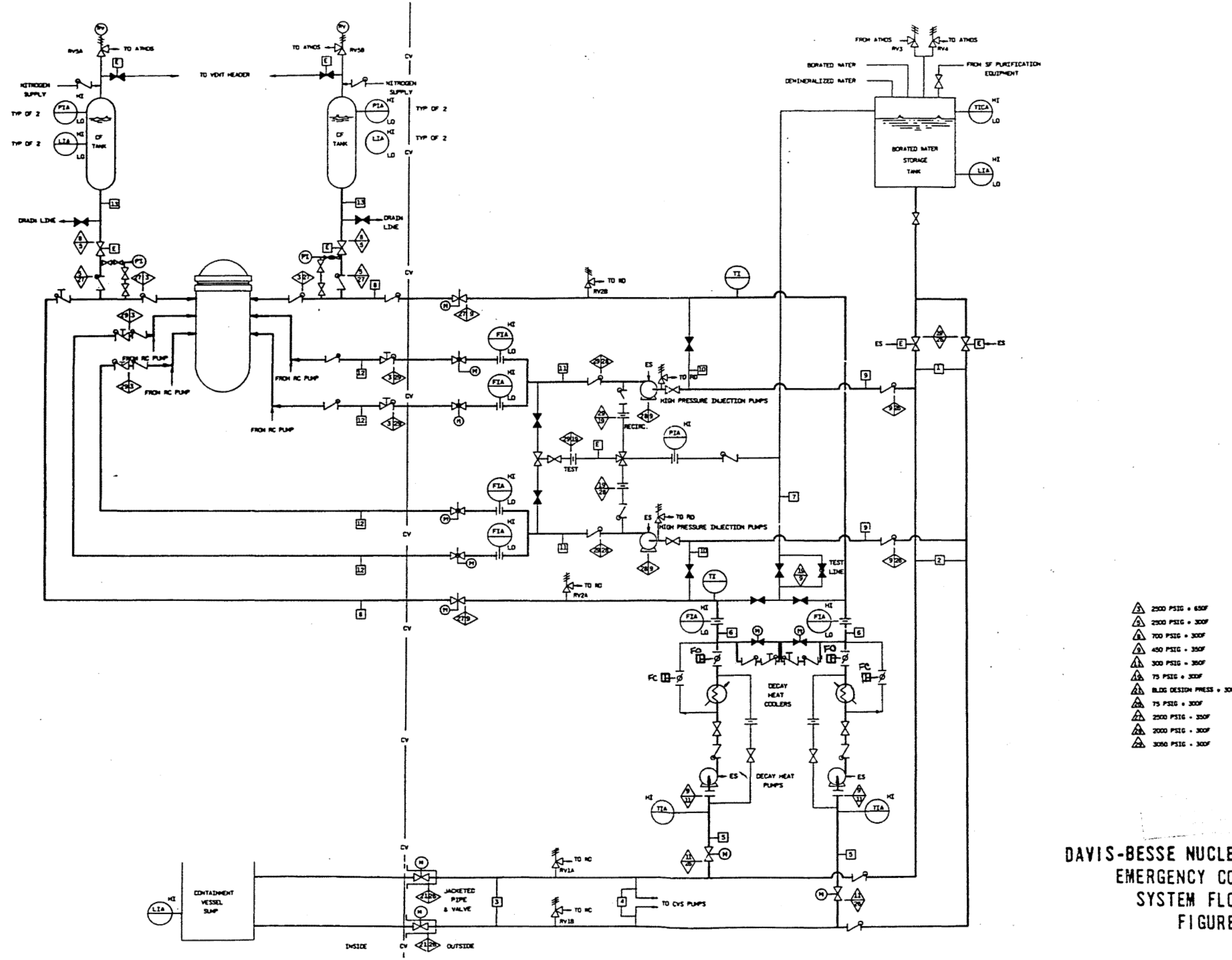
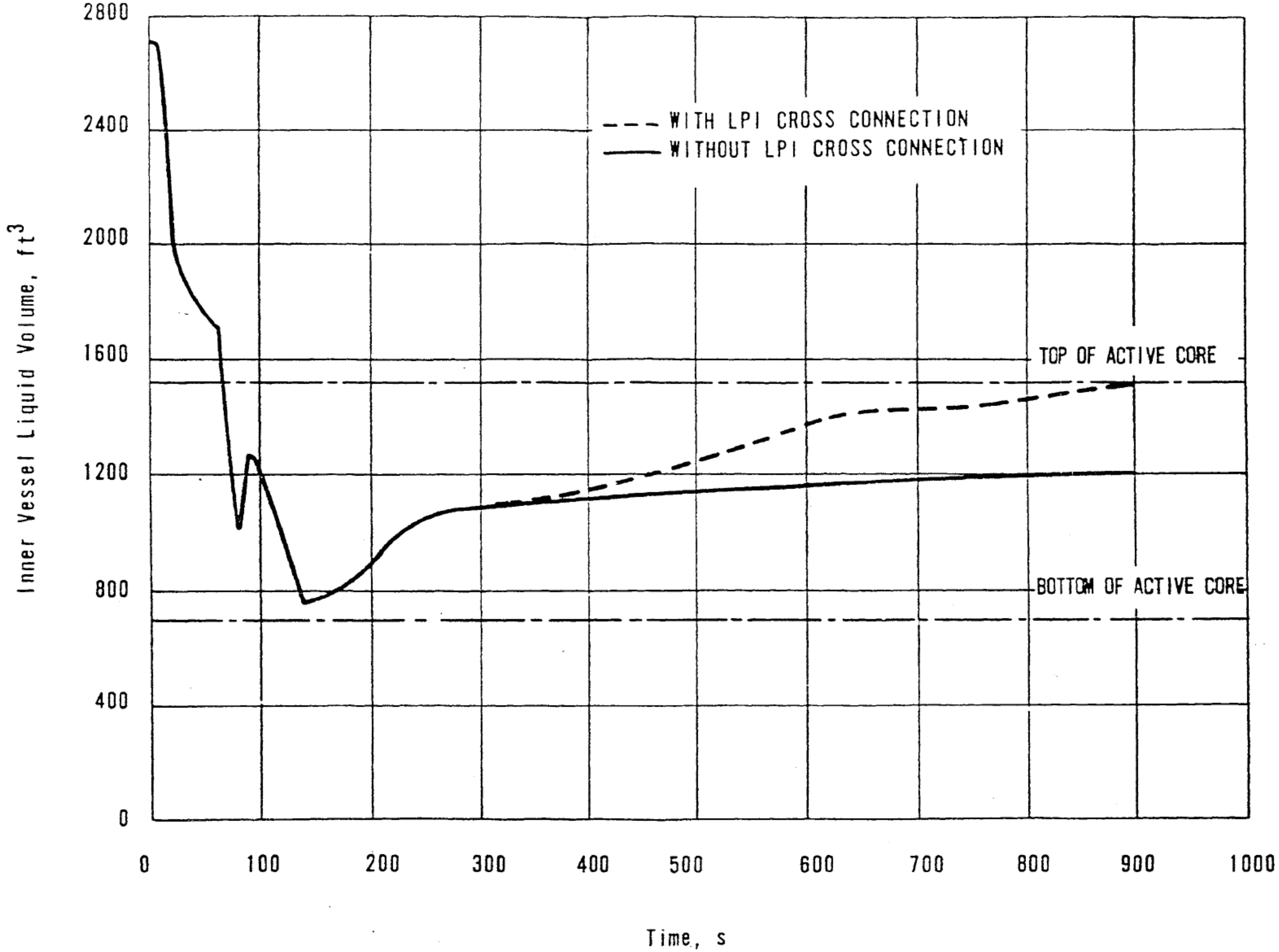


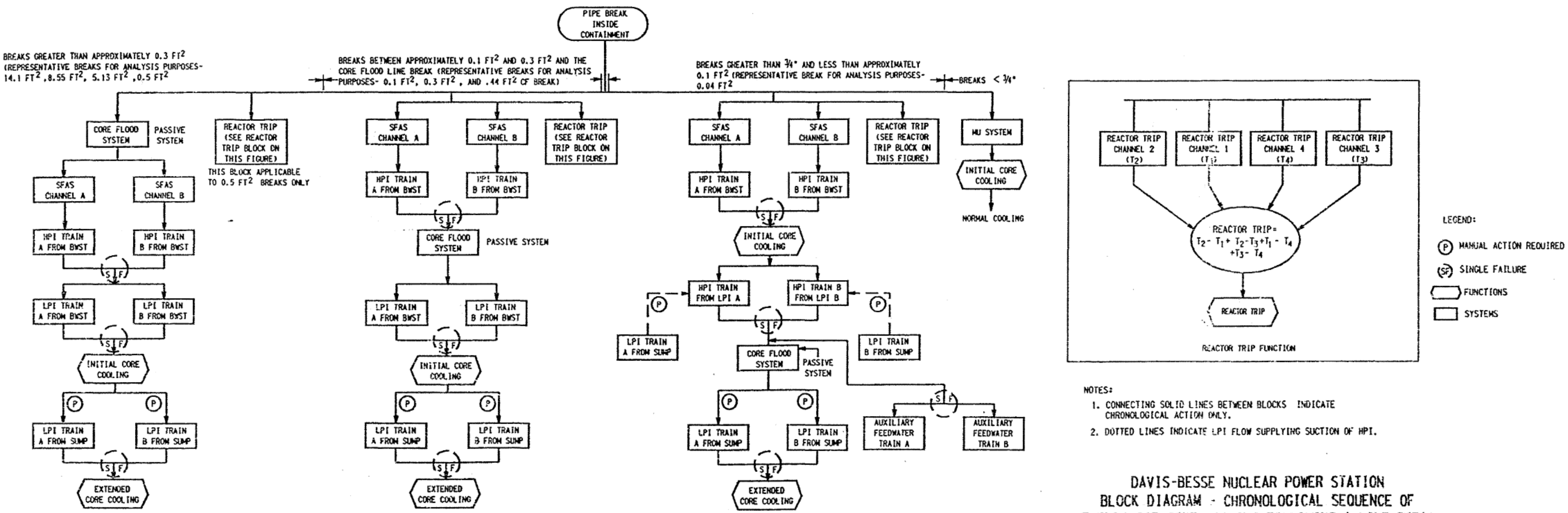
DAVIS-BESSE NUCLEAR POWER STATION
EMERGENCY CORE COOLING
SYSTEM FLOW DIAGRAM
FIGURE 6.3-6

REVISION 9
JULY 1989



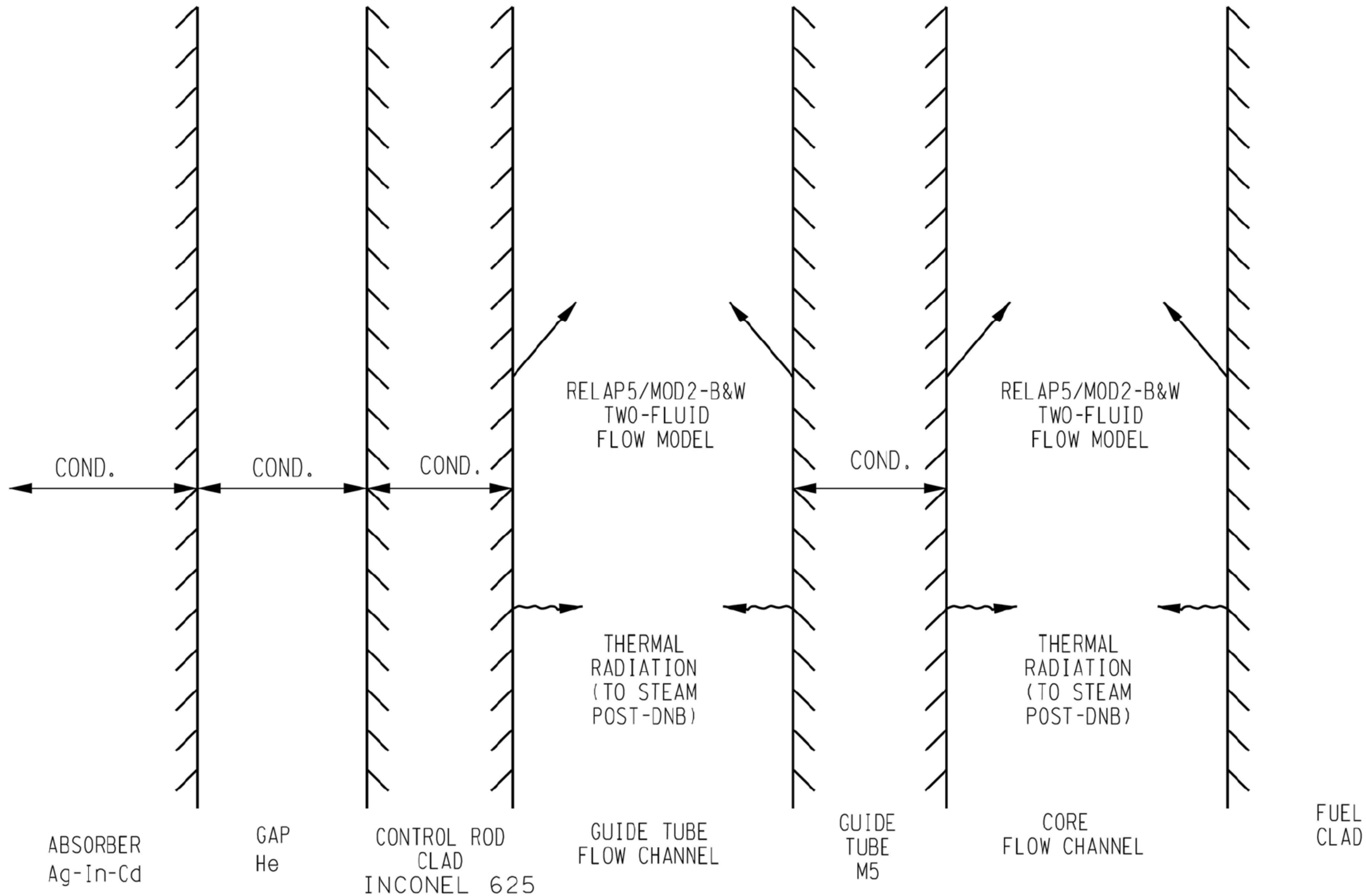
DAVIS-BESSE NUCLEAR POWER STATION
INNER VESSEL LIQUID VOLUME FOR 0.44 $FT.^2$
CFT LINE BREAK
FIGURE 6.3-7

REVISION 0
JULY 1982



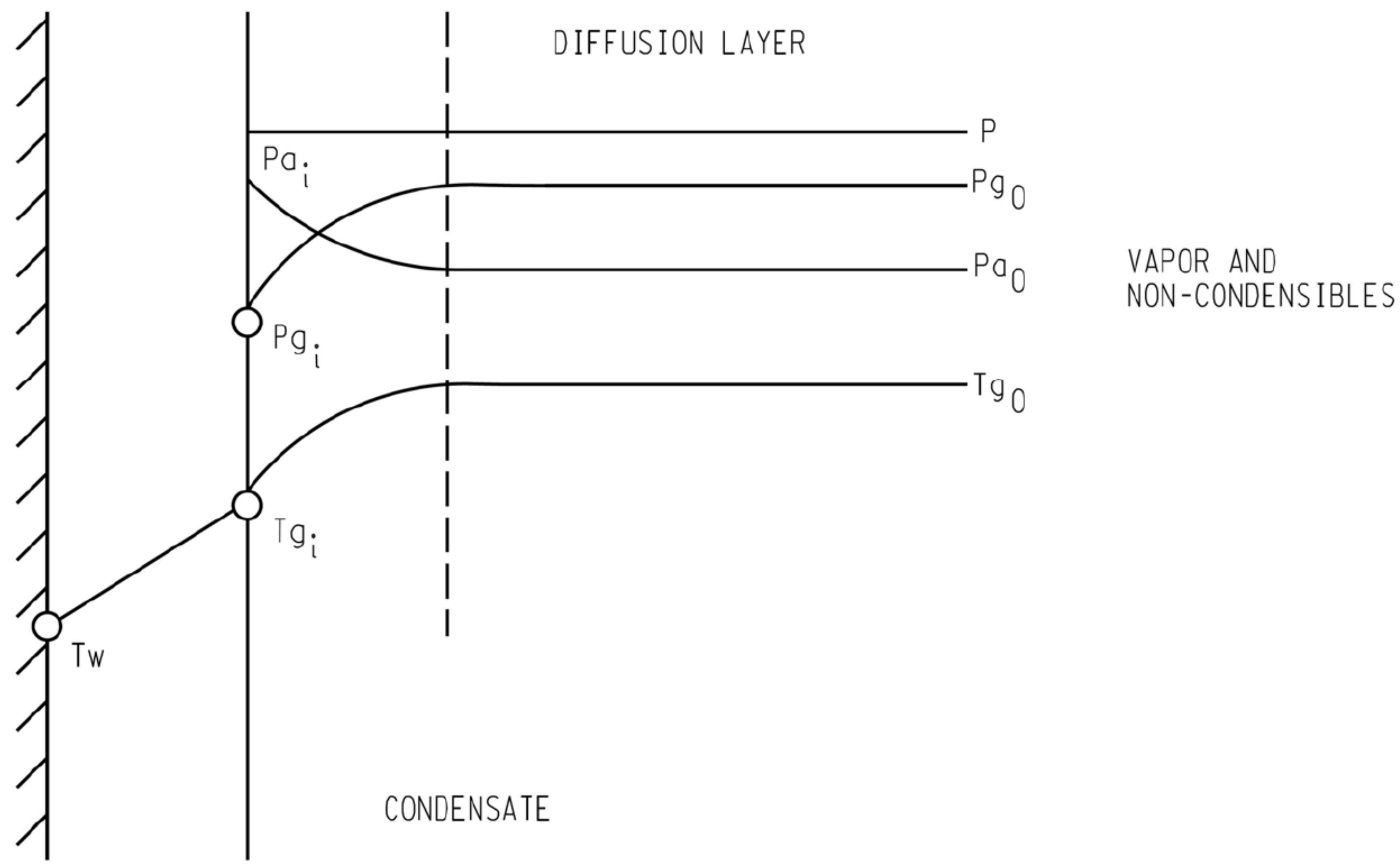
DAVIS-BESSE NUCLEAR POWER STATION
BLOCK DIAGRAM - CHRONOLOGICAL SEQUENCE OF
EVENTS FOR CORE COOLING FOLLOWING A LINE BREAK
FIGURE 6.3-8

REVISION 18
NOVEMBER 1993



DAVIS-BESSE NUCLEAR POWER STATION
CONTROL ROD ASSEMBLY DURING LOCA
MODES OF HEAT TRANSFER
FIGURE 6.3-21

REVISION 33
SEPTEMBER 2020



P = TOTAL PRESSURE

Pa_i = PARTIAL PRESSURE OF GAS AT INTERFACE, lb_f/ft^2

Pa_0 = PARTIAL PRESSURE OF GAS AT BULK CONDITIONS, lb_f/ft^2

Pg_i = PARTIAL PRESSURE OF VAPOR AT INTERFACE, lb_f/ft^2

Pg_0 = PARTIAL PRESSURE OF VAPOR AT BULK CONDITIONS, lb_f/ft^2

Tw = WALL TEMPERATURE, °F

Tg_i = TEMPERATURE AT INTERFACE, °F

Tg_0 = BULK TEMPERATURE, °F

DAVIS-BESSE NUCLEAR POWER STATION
CONDENSATION HEAT TRANSFER
FIGURE 6.3-22

REVISION 33
SEPTEMBER 2020

6.4 HABITABILITY SYSTEMS

Control room systems are designed so that habitability of the control room can be maintained under normal and accident conditions in accordance with the general guidance in General Design Criteria 19 of 10CFR 50, Appendix A. The control room ventilation systems are described in Section 9.4.1.

6.4.1 Radiation Monitoring

The radiation shielding and control room layout are described in Section 12.1. Control room airborne radioactivity monitoring is described in Section 12.2. The evaluation of radiological exposure to control room personnel for postulated accident conditions are presented in Section 15.4.

6.4.2 Toxic Gas Protection Provisions

Davis-Besse commits to the regulatory position of Regulatory Guide 1.78 (December 2001).

The habitability of the control room was evaluated using procedures described in Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release." As indicated in Section 2.2, analysis of off-site storage or transport of chemicals and hazardous materials stored onsite demonstrate that no toxic or explosive materials are stored in volumes or locations which pose a control room habitability hazard exceeding emergency system capabilities. Administrative procedures are in place to control the allowable amount of transient hazardous materials in the vicinity of the control room. A sodium hypochlorite biocide system is used, thus eliminating an onsite chlorine hazard, therefore, special protection against toxic gases will not be required. Self-contained breathing apparatus is provided for the emergency crew to provide assurance of control room habitability in the event of occurrences such as smoke hazards.

6.5 REFERENCES

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24. Deleted

25. Deleted

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