CHAPTER 6 – ENGINEERED SAFEGUARDS

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APPENDICES

6.0 ENGINEERED SAFEGUARDS

Engineered safeguards are those systems and components designed to function under accident conditions to prevent or minimize the severity of an accident or to mitigate the consequences of an accident. Thus, if reactor coolant is lost, the engineered safeguards act to provide emergency cooling inventory to ensure structural integrity of the core, to maintain the integrity of the Reactor Building, and to reduce the fission products expelled to the Reactor Building. Because of their importance in ensuring the health and safety of the general public in the vicinity of the site, special precautions are taken to assure high quality in the components and in system design to assure reliable operation. Separate and independent engineered safeguards are provided for TMI-1.

The engineered safeguards include provisions for:

a. High pressure injection by the Makeup and Purification System.

b. Low pressure injection by the Decay Heat Removal System.

c. Core flooding by the Core Flooding System.

d. Reactor Building cooling by the Reactor Building Emergency Cooling System.

e. Reactor Building cooling and pressure reduction by the Reactor Building Spray System.

f. The removal of fission products in the Reactor Building atmosphere by the Reactor Building Spray System.

g. Reactor Building isolation by the Containment Isolation System.

Figure 6.0-1 schematically depicts the portion of the engineered safeguards system related to core and building protection (see Items a. through e. above). A general description of the engineered safeguards provisions is presented below, and a more detailed description is presented in the latter portion of this Chapter.

The purpose of this Section is to describe the physical arrangement, design, and operation of the engineered safeguards systems as related to their safety functions.

The Makeup and Purification System, the Decay Heat Removal System, and the Core Flooding System are designated as the Emergency Core Cooling System (ECCS), and collectively prevent melting or physical disarrangement of the core over the entire spectrum of Reactor Coolant System (RCS) break sizes. Figure 6.0-2 shows the ECCS. The Makeup and Purification System is arranged so that two of the three pumps are available for emergency use. The Decay Heat Removal System includes two redundant pumps and two coolers. The Core Flooding System is composed of two separate pressurized tanks containing borated water. These tanks automatically discharge their contents directly into the reactor vessel at a preset RCS pressure without reliance on any actuating signal or any externally actuated component.

Reactor Building integrity is ensured by two full capacity, independent, Reactor Building pressure reducing systems operating on different principles: The Reactor Building Spray

System and the Reactor Building Emergency Cooling System (refer to Drawings C-302-712 and C-302-831). These systems each have the redundancy required to meet the single failure criterion. One cooler and spray path are operated concurrently to prevent Reactor Building pressure from exceeding design limits over the entire spectrum of RCS break sizes and to rapidly reduce the driving force for leakage of radioactive materials from the Reactor Building. The Reactor Building Spray System also utilizes chemical additives to reduce postaccident fission product concentration in the building atmosphere. (Refer to Section 6.3.1 for details)

Air sampling for hydrogen concentration is accomplished by the Hydrogen Monitoring System.

Operability of engineered safeguards equipment is ensured in several ways. Much of the equipment in these systems serves a function during normal reactor operation. Where equipment is used for emergency functions only, such as the Reactor Building Spray System, the system has been designed to permit meaningful periodic tests. Operational reliability is achieved by using proven component designs wherever possible and by conducting tests where either the component or its application was considered unique. Quality control requirements were invoked during the manufacture and installation of the engineered safeguards components and systems to assure that a high quality level is maintained. These requirements include use of accepted codes and standards as well as supplementary test and inspection requirements to ensure that all components will perform their intended function under environmental conditions following an accident.

A high degree of isolation has been provided between redundant components in the engineered safeguards systems and between the different engineered safeguards systems. The three makeup pumps (high pressure injection) are located in individual cubicles on elevation 281 ft of the Auxiliary Building, and the two decay heat pump and cooler combinations (low pressure injection) and the two Reactor Building spray pumps are located in individual cubicles at elevation 261 ft of the extreme north end of the Auxiliary Building.

The three makeup pumps are separated from one another and all other items of equipment by reinforced concrete walls. Their suction and discharge lines are separated within and routed through a large valve room abutting the three cubicles housing the pumps. The pump cubicles are individually drained to the Auxiliary Building sump located below them at elevation 261 ft with provisions for detecting major leaks within each cubicle. Valves are provided to isolate the makeup pump suction and discharge lines in the event of a major leak.

The two high pressure injection lines supplying the primary loop in the east steam generator lobe of the secondary shield enter containment in quadrant III (aircraft missile protected area between the Auxiliary and Reactor Buildings) below the 308 ft elevation floor in the Reactor Building. These lines remain below the 308 ft elevation floor until after they pass through the secondary shield wall. The high pressure injection lines for the loop in the west lobe enter containment in quadrant III above the 308 ft elevation floor. Therefore, these two sets of high pressure injection lines are separated outside the secondary shield by the 308 ft elevation floor and outside containment by the 305 ft elevation floor in the Auxiliary Building, except in the valve room, where separation is accomplished by distance.

The two decay heat removal (low pressure injection) pumps, heat exchangers, and associated valves are located in separate pump vaults at elevation 261 ft. These vaults each drain to the Auxiliary Building sump and have leakage detectors that alarm in the Control Room if a high rate of leakage occurs. Within each pump vault is a separate cubicle, with its own drain and

leak detector, which houses the Reactor Building spray pump and valves. Piping between a spray pump cubicle and the associated decay heat removal pump vault runs through an opening a minimum of 7 ft above floor level. This provides an added degree of protection from flooding in that the volume of the sump, and the volume of the cubicle must fill up before overflow occurs into the adjacent cubicle. Openings between the two decay heat removal pump vaults are about 10 ft above floor level. The penetrations between the A and B decay heat removal pump vaults are sealed in accordance with the fire protection program requirements.

Recirculation piping from the Reactor Building sump is embedded in concrete from the sump to the decay heat removal pump vaults. Each recirculation line enters directly into its associated pump vault.

These lines are equipped with guard pipes which cover the exposed piping up to and including the recirculation isolation valve. This guard pipe limits leakage to an amount which prevents an unisolatable leak from flooding the pump vaults and draining the sump.

Low pressure injection discharge piping exits the pump vault through the top and is enclosed in a guard pipe up to the remotely operated isolation valve. This guard pipe prevents flooding the adjacent vault in the event of a pipe rupture outside the vault upstream of the remotely operated valve. The two low pressure injection lines enter the Reactor Building through penetrations greater than 20 ft apart and are run around opposite sides of the outside of the west lobe of the secondary shield until they connect to the core flooding lines.

Spray pump discharge piping exits the pump cubicle in guard pipes similar to those used for the decay heat removal lines. The two spray lines enter the Reactor Building through penetrations approximately 25 ft apart and run up the inside of the containment liner with the same or larger separation to the spray rings.

Two analyzing and sample conditioning systems installed in the Intermediate Building are self-sufficient and independent of each other, one H_2 analyzer is on elevation 295 ft and the other is on elevation 322 ft.

Reactor Building isolation is described in Chapter 5. Other Chapters of this FSAR contain information which is pertinent to the engineered safeguards systems. Chapter 7 describes the actuation instrumentation for these systems. Chapter 14 describes the analysis of the engineered safeguards systems' capability to provide adequate protection during accident conditions. Sections 6.1 through 6.5 and Chapter 9 discuss functions performed by these systems during normal operation and give further design details and descriptive information concerning them.

6.1 EMERGENCY CORE COOLING SYSTEM

6.1.1 DESIGN BASES

The principal design basis for the ECCS, as described in proposed AEC General Design Criteria (GDC) 44 and FSAR Section 1.4.44 has been met. Protection for the entire spectrum of break sizes is provided. Separate and independent flow paths containing redundant active components are provided in the ECCS. Redundancy in active components ensures that the required functions will be performed when a single failure occurs in any of the active components. Separate power sources are supplied to the redundant active components. Separate instrument channels are used to actuate the systems. The capability of the installed ECCS to prevent fuel and clad damage is discussed in Chapter 14.

The ECCS has been designed to operate in the following possible modes:

a. High pressure injection of borated water from the borated water storage tank by the Makeup and Purification System.

b. Rapid injection of borated water by the Core Flooding System.

c. Low pressure injection of borated water from the borated water storage tank by the Decay Heat Removal System.

d. Long term core cooling by recirculation of injection water from the Reactor Building sump to the core through the Decay Heat Removal System (in combination with Makeup and Purification System in the Piggyback Mode).

Refer to the following drawings:

Drawings C-302-660 and C-302-661, Makeup and Purification System
Drawing C-302-640, Decay Heat Removal System
Drawing C-302-711, Core Flooding System

Although the Makeup and Purification System and the Decay Heat Removal System operate to provide full protection across the entire spectrum of break sizes, each system may operate individually and each is initiated independently. The Makeup and Purification System prevents uncovering of the core for small coolant piping leaks where high system pressure is maintained and delays uncovering of the core for intermediate sized leaks. The Core Flooding and Decay Heat Removal Systems are designed to re-cover the core at intermediate to low pressures, and to ensure adequate core cooling for break sizes ranging from intermediate breaks to the double-ended rupture of the largest pipe. The Decay Heat Removal System is also designed to permit long term core cooling in the recirculation mode after a LOCA.

Pumped injection is subdivided so that there are two separate and independent strings, each including both high pressure and low pressure injection, and each capable of providing 100 percent of the necessary core injection with the core flooding tanks. The redundant protection afforded by the ECCS components, subsystems, and systems for the spectrum of reactor coolant pipe break sizes is discussed in Chapter 14.

6.1.2 SYSTEM DESIGN

6.1.2.1 ECCS Operation

The schematic diagram for the ECCS is shown on Figure 6.0-2. The supporting systems are described in Sections 9.5 and 9.6. The ECCS is automatically actuated upon detection of a low reactor coolant pressure of either less than 1600 psig or less than 500 psig or a Reactor Building pressure greater than 4 psig. The 500 psig set point is used as a backup signal to the 1600 psig set point to provide protection over the whole range of Reactor Coolant System (RCS) pressure.

a. Makeup and Purification System (Including High Pressure Injection)

During normal reactor operation, the Makeup and Purification System recirculates reactor coolant for purification and for supply of seal water to the reactor coolant pumps. This normal operating mode and component data are described in Section 9.1.

High pressure injection of borated water is initiated at: (1) a low RCS pressure less than 1600 psig or less than 500 psig (backup signal) or (2) a Reactor Building pressure greater than 4 psig. Automatic actuation of the valves and pumps by the actuation signals switches the system from its normal operating mode to deliver water from the borated water storage tank into the reactor vessel through the reactor coolant lines. The following automatic actions accomplish this change:

1) The makeup pumps start.

2) The valve in each high pressure injection line opens.

3) The valves in the lines connecting to the borated water storage tank open.

4) The valves in the pump minimum recirculation line close.

5) The normal makeup line isolation valve closes.

In addition to the automatic action described above, the pumps and valves can be remotely operated from the Control Room.

Operation of the Makeup and Purification System in the emergency injection mode will continue until the system action is switched by operator action into the recirculation mode or manually terminated.

b. Decay Heat Removal System (Including Low Pressure Injection)

The Decay Heat Removal System is designed to maintain core cooling for larger break sizes. This system provides low pressure injection independent of and in addition to the high pressure injection provided by the Makeup and Purification System. A description of the normal operating mode and component data for the system is given in Section 9.5.

Automatic actuation of the low pressure injection system is initiated at: (1) a low RCS pressure of less than 1600 psig or less than 500 psig (backup signal) or (2) a Reactor Building pressure greater than 4 psig. Low pressure injection is initiated at 1600 psig, which is the same pressure at which the high pressure injection is initiated. Although low pressure injection is not required at this pressure, the pump has the ability to operate in the bypass mode and will thus be available when RCS pressure decreases below the pump discharge pressure. Having the same set points for high pressure injection and low pressure injection allows simplification of the actuation circuitry. During power operation, the Decay Heat Removal System is available for emergency use. Automatic or manual initiation of the emergency operation provides the following actions:

1) The decay heat removal pumps start and operate in the bypass mode until the RCS pressure decreases below the pump discharge pressure. The design pump head is nominally 350 ft or 151 psi at 3000 gpm (See Figure 6.1-3). One train of the low pressure injection system is capable of delivering the flow assumed in the LOCA analysis as described on Table 14.2-27. (Bypass is defined as the operation of a low pressure injection pump in the recirculation mode. There is a locked-open valve in each line.)

2) The valve in each low pressure injection line opens.

3) The valves in the lines connecting the borated water storage tank to the decay heat removal pump suction headers open.

4) Decay heat river water and decay heat closed cycle pumps start.

5) Decay heat river water pump discharge valves open. (Decay heat closed cycle cooling water system valves to the decay heat removal coolers are normally open.)

Low pressure injection will be accomplished through two separate flow paths, each including one pump and one heat exchanger and terminating directly in the reactor vessel through core flooding nozzles located on opposite sides of the vessel.

After initiation during a LBLOCA or during sump recirculation operation, LPI flow is throttled to ensure adequate available NPSH. The flow rate can be controlled from the Control Room by throttling valves DH-V4A and DH-V4B. There is “jog” valve control and flow indication in the Control Room. If access is permitted, long term throttling of the decay heat flow is performed using manually operated globe valves DH-V19A and DH-V19B (normally wide open). Extension handles on these two valves permit their operation from floor elevation 281 ft in the Auxiliary Building.

A cross connection between the two decay heat removal lines, as shown on Figure 6.0-1, provides the capability of injecting an adequate supply of borated water for core cooling even in the event of a core flood line rupture.

The initial injection of water by the Decay Heat Removal System involves pumping water from the borated water storage tank into the reactor vessel. With all engineered safeguards pumps operating, and assuming the maximum break size, this mode of

operation lasts for a minimum of about 25 minutes. When the borated water storage tank reaches a low level, the operator will take action to open the suction valves from the Reactor Building emergency sump, permitting recirculation of the spilled reactor coolant and injection water from the Reactor Building sump.

c. Core Flooding System

The Core Flooding System provides core protection continuity for intermediate and large RCS pipe failures. It automatically floods the core when the RCS pressure drops below 600 psig. The Core Flooding System is self-contained, self-actuating, and passive in nature. The combined coolant volume in the two tanks is sufficient to re-cover the core hot spot, assuming no liquid remains in the reactor vessel following the LOCA. The discharge pipe from each core flooding tank is attached directly to a reactor vessel core flooding nozzle. Each core flooding line at the outlet of the core flooding tanks contains an electric motor operated stop valve adjacent to the tank and two inline check valves in series. The stop valves at the core flooding tank outlet are fully open during reactor power operation. Valve position indication is shown in the Control Room. During power operation when the RCS pressure is higher than the Core Flooding System pressure, the two series check valves between the flooding nozzles and the core flooding tanks prevent high pressure reactor coolant from entering the core flooding tanks.

The following procedures will be followed to isolate the core flooding tanks during routine depressurization, and to ensure that the isolation valves cannot inadvertently close during normal operation or following a LOCA. As the RCS is depressurized to less than 700 psig but greater than 650 psig, both core flood isolation valves will be closed by the operator. The breakers for each of these valves will then be manually opened and tagged open at their respective motor control center, thereby deenergizing the control circuit, in order to prevent inadvertent opening of these valves during periods that the reactor is shutdown and depressurized. Should the operator neglect to close these valves at 700 psig, an alarm (CF-V1A and/or 1B Position Abnormal) will be energized before RCS pressure decreases to 600 psig.

During normal operation, these valves will remain open. The breakers for each of these valves will be tagged open at their respective motor control center to prevent inadvertent closing of these valves during periods when the reactor is pressurized. In addition, a protector mounted on the console prevents the pushbutton from being pushed accidentally and thus precludes accidental closing of these valves. If for any reason, during normal operation, the valve is closed with the RCS pressure above 700 psig, an alarm (CF-V1A and/or 1B Position Abnormal) will be energized. The valves are designed to fail in the "as is" condition.

The driving force to inject the stored borated water into the reactor vessel is supplied by pressurized nitrogen which occupies approximately one third of the core flooding tank volume. Connections are provided for adding both borated water and nitrogen during power operation so that the proper level and pressure may be maintained.

Each core flooding tank is protected from overpressurization by a relief valve installed directly on the tank. The size of these relief valves is based upon maximum water makeup rate to the tank. Redundant level and pressure indicators for each tank are provided on the main control console. High and low pressure and level are alarmed in the main control room.

The inservice inspection of the check valves and isolation valves in the core flood tank discharge lines shall be performed with the reactor shutdown in accordance with ASME B&PV Code Section XI and applicable addenda as required by 10CFR50 Section 50.55a(g) and Technical Specification 4.5.2 to verify the Core Flooding System operability.

During the initial plant startup pre-operational test program, a core flooding system flow path test was performed. The purpose of this test was to demonstrate CF System discharge capability (i.e., unobstructed flow). The verification of the calculation methods used to predict the core flood system flow rate under accident conditions was successfully completed at Oconee Unit 1 and was not repeated at Three Mile Island Unit 1.

The Core Flooding System flow path is periodically checked by discharging the contents of one core flooding tank at a time into the reactor vessel. At the start of the test, the core flooding tanks are at low pressure and contain nominal volumes of water. The test verifies the core flood system flow capability assumed in the LOCA analysis.

Design data for major system components are shown in Table 6.1-1.

6.1.2.2 Codes And Standards

The Makeup and Purification, the Decay Heat Removal, and the Core Flooding Systems are designed and manufactured to the codes and standards in Tables 6.1-2, 9.1-2, and 9.5-2.

6.1.2.3 Material Compatibility

All components with surfaces in contact with water containing boric acid are protected from corrosion and deterioration. The Makeup and Purification System, which operates continuously with borated reactor coolant, is constructed entirely of stainless steel. The borated water storage tank and the major components in the Decay Heat Removal System are constructed of stainless steel. The core flooding piping and valves are stainless steel, and the tanks are constructed of stainless clad carbon steel.

6.1.2.4 Component Design

a. Piping

The makeup and purification lines and components are designed for the normal reactor operating and emergency conditions. The Decay Heat Removal System piping and valves are subjected to more severe conditions during normal operation than during emergency operation and, therefore, operate well within the design conditions. The

design pressures and temperatures of these systems are shown in Table 6.1-3. To assure system integrity, major piping has welded connections, except where flanges are dictated for maintenance reasons. The piping shown on Figure 6.0-2 is designed in accordance with USAS Power Piping Code B31.1.0 and fabricated, erected, and inspected in accordance with USAS Nuclear Power Piping Code B31.7 - February 1968 - Draft, including June 1968 Errata.

b. Pumps

The pumps used in the emergency injection systems are of proven design and have been used in many other applications. The makeup pump has been used in boiler feed pump service. The decay heat removal pump is used extensively in refinery service. The decay heat removal pump seals have been tested satisfactorily under the temperature and pressure conditions that would be encountered during the LOCA, per Reference 1, Section 6.7. The makeup pump and decay heat removal pump casings are liquid penetrant tested by methods described in the ASME Boiler & Pressure Vessel (B&PV) Code, Section VIII, and to acceptance standards of Appendix VIII of that code. Both pumps have been hydrotested and qualified to be able to withstand pressures as great as or greater than 1.5 times the design pressure. The pumps are designed so that periodic testing may be performed to ensure operability and ready availability. The characteristics of each engineered safeguards pump are verified by shop testing before installation.

c. Heat Exchangers

The decay heat removal heat exchangers are designed and manufactured to the requirements of the ASME Sections VIII and III-C and the TEMA-R standards. In addition to these requirements, uniformity of the tubes is assured by eddy current testing, and the tubes are seal welded to the tube sheet to decrease the possibility of leakage. All tube welded ends are liquid penetrant tested to ensure the absence of welding flaws.

d. Valves

All valves in the emergency injection systems, as shown on Figure 6.0-2, are inspected in accordance with the intent of the USAS Nuclear Power Piping Code B31.7 - February 1968 - Draft, including June 1968 Errata. Liquid penetrant, radiography, ultrasonic, and hydro testing are performed as the code classification requires.

The seats and discs of these valves are manufactured from materials which are free from galling and seizing.

e. Seismic Design

Components and piping in the emergency injection systems are designated as Class I equipment and are designed to maintain their functional integrity during a safe shutdown earthquake (SSE). Chapter 5 defines the acceptable stress limits for the Class I equipment.

f. Instrumentation

The engineered safeguards actuation instrumentation for the emergency injection system employs redundant channels and signals as described in Chapter 7. The Control Room layout is arranged so that all indicators and alarms are grouped in one sector at a convenient location for viewing. The status of all ECCS equipment can be monitored from this area both during the accident and in the postaccident cooling mode.

g. Quality Control

Quality standards for the emergency injection systems components are given in Table 6.1-2.

6.1.2.5 Coolant Storage

The makeup tank has a total volume of 600 ft^3 and normally contains a volume of water sufficient to provide NPSH to the makeup pumps. During normal operation this tank provides water to the makeup pump in operation (MU-P-1A, MU-P-1B, or MU-P-1C if manually aligned). The makeup tank is designed and inspected in accordance with the requirements of ASME Section III-C.

The borated water storage tank (BWST) is located outside the Reactor Building and the Auxiliary Building. Inventory and boron concentration requirements are described on Table 6.2-1. The BWST is used for emergency core injection and filling the fuel transfer canal during refueling. The borated water storage tank also supplies borated water for emergency cooling to the Reactor Building Spray System, Decay Heat Removal System, and the Makeup and Purification System. Redundant level indication is provided on the main control console. High, low, and low-low level alarms are provided on the main annunciator panel. Provisions are made for sampling the water and for adding concentrated boric acid solution or demineralized water. The borated water storage tank is heated with redundant immersion heaters to prevent freezing. Although not required to keep the BWST from freezing and not part of the original licensing basis, heat tracing exists on the BWST. The 24 inch pipe from the tank is heat traced to the point where it enters the Auxiliary Building.

Each core flooding tank nominally contains approximately 940 cu ft of borated water at a minimum concentration of 2270 ppm of boron.

6.1.2.6 Pump Characteristics

Curves of total dynamic head, net positive suction head (NPSH), and brake horsepower versus flow are shown on Figure 6.1-2 for the makeup pumps and on Figure 6.1-3 for the decay heat removal pumps.

6.1.2.7 Heat Exchanger Characteristics

The decay heat coolers are designed to remove sensible heat and the decay heat generated during a normal shutdown. In addition, each cooler is capable of cooling the injection water during the recirculation mode following a LOCA to remove decay heat and thus provide adequate core cooling. The heat transfer capability of the decay heat cooler as a function of recirculated water temperature is illustrated on Figure 6.1-4. This figure also illustrates the capability margin between the design requirements and the actual predicted performance.

6.1.2.8 Relief Valve Settings

Relief valves are provided to protect the low pressure piping and components from overpressure. The relief valves are set to protect system components consistent with their design pressures. Some component design pressures are shown in Chapter 9.

6.1.2.9 Reliability Considerations

System reliability is assured by the system functional design, including the use of normally operating equipment for safety functions, testability provisions, and equipment redundancy; by proper component selection; by physical protection and arrangement of the system; and by compliance with the intent of the AEC General Design Criteria. There is sufficient redundancy in the ECCS to assure 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 6.1-4.

Table 6.1-4 also presents an analysis of possible malfunctions of the Core Flooding System that could reduce its postaccident availability. It is shown that these malfunctions result in indications that will be obvious to the operators so appropriate action can be taken. In general, failures of the type assumed in this analysis are considered highly improbable because a program of periodic testing is included in the station procedures. The adequacy of equipment sizes in the ECCS is demonstrated by the postaccident performance analysis described in Chapter 14. This analysis shows that only one makeup pump, one decay heat removal pump, and one decay heat removal cooler (two of each are normally available) are required in combination with the Core Flooding System to protect against the full spectrum of break sizes. For more details see Reference 3.

6.1.2.10 Missile Protection

Protection against missile damage is provided either by shielding or by physical separation of duplicate equipment as described in Section 6.0

6.1.2.11 Actuation

Injection from the Makeup and Purification System is automatically actuated by a low RCS pressure less than 1600 psig or less than 500 psig (backup signal), or by a high Reactor Building pressure of 4 psig. The required pumps and automatic valves can also be remotely operated from the Control Room.

Injection from the Decay Heat Removal System is automatically actuated by a low RCS pressure less than 1600 psig or less than 500 psig (backup signal), or by a high Reactor Building pressure of 4 psig. The required pumps and automatic valves can also be remotely operated from the Control Room.

The Core Flooding System is actuated at an RCS pressure of 600 psig. At this point, the differential pressure across the in-line check valves allows them to open, releasing the contents of the tanks into the reactor vessel. A description of the Engineered Safeguards Actuation System is given in Section 7.1.3. Table 7.1-3 gives actuation set points for all of the systems discussed.

6.1.2.12 Environmental Considerations

The major operating components of the ECCS are external to the Reactor Building and will not be exposed to the postaccident building environment.

The ECCS & RB cooling electrical and mechanical components within the Reactor Building which are required to be operable during and subsequent to a LOCA and/or a steam line break inside the Reactor Building are:

a. RCS pressure transmitters.

b. Reactor Building isolation valves and associated position indicators.

c. Reactor Building ventilation unit fans and cooling coils.

d. Instrument cables for Items a and b above.

e. Power cables for the Reactor Building ventilation fan motors and isolation valves.

Non-nuclear instrumentation (Item a., above) in the Reactor Protection System and the Engineered Safeguards Actuation System is located inside the Reactor Building and has been environmentally qualified.

In addition, compliance with 10CFR50.49 provides additional assurance that the ESAS components are environmentally qualified to perform as designed for safe plant shutdown for accident/postaccident conditions.

6.1.2.13 ECCS Sump Strainer Design Considerations

In response to Generic Letter 2004·02, the Reactor Building sump, trash rack, and strainer was reconfigured to provide additional assurance that the ECCS and BS pumps would not experience a debris-induced loss of NPSH margin during sump recirculation. The sump was divided into a normal "wet" sump and an ECCS "dry" sump. The normal sump design has two (2) boxes that are approximately 3 ft x 2.5 ft in plan and 6 ft high, located in the corners at the west end of the existing sump pit on either side of the ECCS suction pipes. The ECCS sump is the remaining volume of the pit. Redundant level instrumentation is available for both the normal "wet" sump and the ECCS "dry" sump. (Reference 23)

The strainer assembly is an array of "top hat" modules. The top hat module is comprised of two perforated plate tubes (8" and 12" diameter) forming an annular flow area between. The perforated plate has 3/32" holes and the annular region contains a debris bypass eliminator (wire mesh filter element). Downstream Effects Evaluations have concluded that the downstream components are capable of providing the necessary long term cooling. (Reference 23)

The effects (including chemical) of debris loading the RB sump strainers, the minimum RB sump water level, and head loss for the maximum allowed flow rate plus instrument error were considered when evaluating the NPSH available at the DH pumps. Operators must throttle DH flow to ensure adequate available NPSH. (Reference 11)

The strainer assembly qualification evaluation includes Dead Weight, Thermal, Seismic (SSE) including hydrodynamic mass, and Differential Pressure (7 psi). To ensure structural adequacy of the strainer assembly, the analysis assumes 1 train of Building Spray is secured 1 hour after swap over to recirculation mode; the last train of Building Spray is secured 24 hours after swap over to recirculation mode, and throttling of the DH pump if strainer DP is high. (Reference 11).

6.1.2.14 GL 2008-01 Gas Entrainment Ptrogram

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 (Reference 26). Generic Letter 2008-01 requested "...that each addressee evaluate its ECCS, DHR system, and containment spray system licensing basis, design, testing, and corrective actions to ensure that gas accumulation is maintained less than the amount that challenges operability of these systems, and that appropriate action is taken when conditions adverse to quality are identified." Technical Evaluation 728092-76, "TMI System Evaluations for NRC GL 2008-01 Managing Gas Accumulation in
Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems" (Reference 27) identifies gas susceptible locations and establishes a system monitoring program. Acceptance criteria for the monitoring program are established in Technical Evaluation 09-00241 "Technical Evaluation Addressing Air Void 14 Gates" (Reference 28).

6.1.3 DESIGN EVALUATION

6.1.3.1 Makeup And Purification System (Includes High-Pressure Injection)

Two pumps are actuated upon receipt of an emergency safeguards initiation signal. One pump is capable of delivering the HPI flow assumed in the LOCA analysis as described on Table 14.2-28. The safety analysis in Chapter 14 has shown that one makeup pump is sufficient to prevent core damage for those smaller leak sizes which do not allow the RCS pressure to decrease rapidly to the point at which low pressure injection is initiated. The LOCA analysis assumes HPI operation 35 seconds after actuation. This allows 10 seconds to energize the on-site emergency power source, 1 second for instrumentation response and 24 seconds for makeup system component response time. One of the makeup pumps is normally in operation, and a positive static head of water assures that all pipe lines are filled with coolant. The makeup and purification injection lines contain thermal sleeves at their connections into the reactor coolant piping to prevent overstressing the pipe juncture.

The high pressure injection lines are cross-connected to assure required flow in the event of a break in a high pressure injection line. These cross-connects will permit an acceptable flow distribution during high pressure injection in the event of a high pressure injection line break. Cavitating venturis on the high pressure injection lines will limit flow through a ruptured high pressure injection line. Cavitating venturis provide for the proper flow split.

The core flood line break establishes the maximum size acceptable for a cavitating venturi. Under these accident conditions, low RCS pressure occurs and high pressure injection is required. The cavitating venturis will limit flow to less than runout flow of a single pump, if the pump is only providing HPI. Pump runout for a pump also providing seal injection is prevented by operator action to throttle HPI based upon HPI Flow indication.

The TMI-1 configuration has cavitating venturis installed in each HPI leg to provide for balancing the flow to the unbroken legs without Control Room operator action. The HPI system design, including the venturi design, ensures that the minimum required HPI flow is injected into the RCS. Table 14.2-28 shows HPI flow as a function of RCS pressure at the HPI nozzle for breaks in RCS piping and for the worst case HPI line break condition. To ensure that the required injection flow is achieved, if only one train of HPI is available, flow through the makeup line must be isolated since it enters the HPI leg “B” downstream of the cavitating venturi. The isolation of makeup flow is accomplished by closing valve MU-V-18 when HPI is actuated.

MU-V-18 is a pneumatic, DC operated valve which fails closed on loss of air and open on loss of power (i.e., must be energized to close). It is powered from 125 VDC distribution panel 1M. The DC system is powered from onsite sources and therefore, is unaffected by LOOP. The 1M DC distribution panel is powered through an automatic transfer switch from either the A or B DC system. This transfer switch will not transfer following an ES actuation. In accordance with the design requirements for LOCA mitigation, a single failure is assumed to occur in addition to the initiating event. Loss of DC power is the identified limiting single failure associated with the MU-V-18 close safety-related function.

In this event, HPI system performance is assured by maintaining the electrical power lineup to 1M DC panel in coordination with the mechanical lineup of the normal makeup supply line. When the normal MU line is connected to the HPI train A, the power supply to 1M DC panel is from 1A ES power. If the normal MU line is connected to the HPI train B, then the power supply to 1M DC panel is from 1B ES power.

Operation of this system does not depend on any portion of another engineered safeguard. However, the system can be operated in series with the Decay Heat Removal System if needed, in the recirculation mode.

Analysis (Reference 17) has been performed which demonstrated that the required MU pump NPSH margin will be maintained in the event of a SBLOCA at the B HPI nozzle, and that the MU tank gas volume will not jeopardize the operation of the HPI pumps when minimum BWST level is reached during a LBLOCA. These conclusions are based upon operation with the MU tank pressure and level within limits and MU-V-222 throttled to limit flow as defined in the analysis.

6.1.3.2 Decay Heat Removal System (Includes Low Pressure Injection)

Two pumps will deliver greater than 5500 gpm to the reactor vessel through two separate injection lines when the vessel pressure is at approximately 100 psig. One pump can deliver greater than 2750 gpm to the reactor vessel. Refer to Table 14.2-27 for the various LPI flows as a function of Core Flood/LPI Nozzle Pressure. Assuming the reactor had been operating at full power prior to the accident, the decay heat being generated in the core at 30 minutes after the accident is approximately 1.9 percent of full power.

Following a LOCA, assuming a simultaneous loss of normal power sources, the emergency power source and the Decay Heat Removal System will be in full operation within 35 seconds after initiation. The components of this time delay are as follows:

a. Total instrumentation lag 1 sec

b.	Emergency power source start	10 sec
c.	Pump motor startup (from the time the pump motor line circuit breaker closes until the pump attains full speed)	10 sec
d.	Injection valve opening time (in parallel with c.)	24 sec
e.	Borated water storage tank (in parallel with c.) outlet valves	20 sec

The Decay Heat Removal System is connected with other safeguards systems in four respects: (1) the Makeup and Purification System, the Decay Heat Removal System, and the Reactor Building Spray System take their suction from the borated water storage tank; (2) the decay heat removal pumps and the Reactor Building spray pumps share common suction lines from the Reactor Building sump during the coolant recirculation mode; (3) the decay heat removal system and the Core Flooding System utilize common injection nozzles on the reactor vessel; and (4) the Makeup and Purification System may be interconnected with the Decay Heat Removal System if recirculation through the makeup pumps is required.

Following a LOCA, if one decay heat removal (low pressure injection) pump fails to start, the cross-connection line valves shown on Figure 6.0-2 will be opened by the plant operators in compliance with the Emergency Operating Procedures. This action will ensure that at least 1000 gpm of borated water is injected into the reactor vessel for long-term core cooling in the event of a core flood line break with the operable LPI pump on the side with the break. Short-term cooling is provided by HPI and core flood through the intact line. It is conservative to take this action because this specific event may not be easily identified by interpreting Control Room indications. If the LOCA is not a core flood line break, injection of borated water into the reactor vessel for core cooling will not be affected by opening the cross-connection line valves.

When the BWST level reaches the Low Level Limit, the operator opens the RB sump isolation valves to transfer the DH pump suction to the RB sump. To preclude the potential for air entrainment in the pumps and to ensure that the minimum BWST inventory is injected into the Reactor Building, the operator closes the BWST isolation valves after the BWST level reaches Lo-Lo Alarm setpoint. The evaluation of this transition included consideration for instrument error, operator response time, minimum RB pressure and valve stroke time. The Lo-Lo Level Alarm setpoint (76”) establishes the minimum quantity of water which is available for RB sump recirculation.

When operating in sump recirculation mode with two trains of Decay Heat, each pump is throttled to 3000 gpm (indicated flow) or less to ensure adequate pump NPSH. If one train of Decay Heat is operating with the cross connect valves open, the flow is throttled to 2800 gpm (total indicated flow) or less to ensure adequate pump NPSH. The effects (including chemical) of debris clogging the RB sump strainers, the minimum RB sump water level, and head loss for the maximum allowed flow rate plus instrument error were considered when evaluating the NPSH available at the DH pumps. The NPSH evaluation conservatively assumes a RB pressure equal to vapor pressure of the sump liquid when the sump temperature exceeds 208°F and an RB pressure of -1 psig at sump temperatures lower than 208°F. The reactor

building conditions and the system and pump parameters for the NPSH evaluation are listed on Table 6.1-6. (Reference 11).

The decay heat pump flow rate can be adjusted from the Control Room by throttling valves DH-V4A and DH-V4B. There is "jog" close valve control in the Control Room for these valves. If finer control or long term flow throttling of the decay heat flow rate is desired, manually operated globe valves DH-V19A and DH-V19B (normally wide open) may be utilized to achieve this. Extension handles on these two valves permit their operation from floor elevation 281 ft in the Auxiliary Building.

The required NPSHs indicated in Table 6.1-6 reflect measured test values. The pump performance has been verified prior to fuel loading (see Appendix 13A, page 5 for Decay Heat Removal System Functional Test).

6.1.3.3 Core Flooding System

Injection response of the Core Flooding System is dependent upon the rate of reduction of RCS pressure. The capability of the Core Flooding System to reflood the core is described in Chapter 14.

The core flooding nozzles and lines are designed to ensure that they will perform adequately under the differential temperature imposed in the injection mode and the expansion in the recirculation mode.

6.1.4 TESTS AND INSPECTIONS

All active components, listed in Table 6.1-5, of the emergency injection system will be tested periodically to demonstrate system readiness. Performance of active systems can be tested by establishing flow and observing pressures and flows during scheduled shutdowns. The Makeup and Purification System will be inspected periodically during normal operation for leaks from pump seals, valve packing, and flanged joints. In addition to the ASME Section XI Code requirements, the HPI nozzle assemblies are inspected under an augmented Inservice Inspection Program which incorporates enhanced IGSCC techniques and a more frequent inspection schedule than required by the Code. During operational testing of the decay heat removal pumps, the portion of the system subjected to pump pressure will be inspected for leaks.

Items to be inspected for leaks to atmosphere will be pump seals, valve packing, flange gaskets, heat exchangers, and safety valves. The check valves of the low pressure injection system (DH-V-22A/B) will be leak tested per Technical Specification 4.2.7.

TABLE 6.1-1
(Sheet 1 of 2)

CORE FLOODING SYSTEM COMPONENT DATA

Core Flooding Tanks (CF-T-1A/B)	
Number	2
Design pressure, psig	700
Operating pressure, psig	600
Maximum pressure, psig	625
Minimum pressure, psig	575
Design temperature,ºF	300
Operating temperature, ºF	90
Total volume, ft^3	1410
Normal water volume, ft^3	940
Minimum water volume, ft^3	910
Maximum water volume, ft^3	970
Material of construction	Carbon steel lined with SS
Check Valves (CF-V-4 & 5A/B)	
Number per flood line	2
Size, inches	14
Material	304 SS
Design pressure, psig	2500
Design temperature:	
Valve nearest reactor, ºF	650
Valve in CFT line, ºF	300

TABLE 6.1-1
(Sheet 2 of 2)

CORE FLOODING SYSTEM COMPONENT DATA

Isolation Valves (CF-V-1A/B)

Number per flood line	1
Size, inches	14
Material	316 SS
Design pressure, psig	2500
Design temperature, ºF	300

Piping	Reactor to First Check Valve	First Check Valve to Isolation Valve	Isolation Valve to Tank
Size, inches	14	14	14
Material	316 SS	304 SS	304 SS
Design pressure, psig	2500	2500	700
Design temperature, ºF	650	300	300

TABLE 6.1-2
(Sheet 1 of 4)

QUALITY CONTROL STANDARDS FOR ENGINEERED SAFEGUARDS SYSTEMS

Summary of Requirements for
Core Flooding Tanks

Classification: ASME SECTION III, Class C, Paragraph N-2113 and the requirements of ASME SECTION VIII, Paragraph UW-2(a) (lethal substances)

Inspection Requirements	Acceptance Standard
1. Inspection of raw materials and review of material certificates	ASME SECTION III
2. Hydro test	ASME SECTION III
3. Radiograph	ASME SECTION VIII

Summary of Requirements for Decay Heat
Removal Heat Exchanger

Classification: Shell ASME SECTION VIII, Tube ASME SECTION III, Class C (lethal)

Inspection Requirements	Acceptance Standard
1. Inspection of raw materials and review of material certificates	ASME SECTION II, III, VIII
2. Seal weld on tube-to-tube sheet	TEMA-R-7 and additional requirements
3. Liquid penetrant inspection on tube-to-tube sheet weld	ASME SECTION III, N-627 and additional requirements
4. Hydro test	ASME SECTION III, VIII, TEMA-R
5. Leak test and seal weld (air)	--

TABLE 6.1-2
(Sheet 2 of 4)

QUALITY CONTROL STANDARDS FOR ENGINEERED SAFEGUARDS SYSTEMS

Summary of Requirements for Valves

Inspection Requirements	Acceptance Standard
Class I and II Valves by USAS B31.7	
1. Radiographic inspection of the body casting	USAS B31.7*
2. Inspection of material and review of material certificates	USAS B31.7*
3. Liquid penetrant inspection of the valve body casting	USAS B31.7*
4. Hydro test of valve assembly requirements	USAS B16.5 and additional
5. Seat leakage test	MSS-SP-61 and additional requirements
Class III Valves by USAS B31.7	
1. Inspection of material and review of material certificates	USAS B31.7*
2. Hydro test of valve assembly	USAS B16.5
3. Seat leakage test MSS-SP-61 and additional requirements	

In addition to the inspections listed above, all valve materials must meet the ASTM material specification.

* USAS B31.7 - February 1968 - Draft - Including June 1968 Errata

TABLE 6.1-2
(Sheet 3 of 4)

QUALITY CONTROL STANDARDS FOR ENGINEERED SAFEGUARDS SYSTEMS

Summary of Requirements for
Engineered Safeguards Systems Pumps

Inspection Requirements	Acceptance Standard
1. Inspection of materials and review of material certificates	ASTM (applicable specification)
2. Liquid penetrant inspection of castings	ASME SECTION VIII
3. Performance test Standard	Hydraulic Institute

Additional Requirements:

Decay Heat Removal Pumps

Hydrotest casing to 900 psig. Test pressure is held for 30 minutes per inch of thickness with a minimum holding time of 30 minutes. This exceeds the hydrotest requirements of ASME SECTION VIII, Paragraph UG-99 (greater than 1.5 times design pressure).

Reactor Building Spray Pumps

Hydrotest casing to 900 psig. Test pressure is held for 30 minutes per inch of thickness with a minimum holding time of 30 minutes. This exceeds the hydrotest requirements of ASME SECTION VIII, Paragraph UG-99 (greater than 1.5 times design pressure).

Makeup Pumps

Hydrotest pump casing to 4500 psig. Test pressure is held for 30 minutes per inch of thickness with a minimum holding time of 30 minutes. This meets the hydrotest requirements of ASME SECTION VIII

Paragraph UG-99 (1.5 times design pressure).

TABLE 6.1-2
(Sheet 4 of 4)

QUALITY CONTROL STANDARDS FOR ENGINEERED SAFEGUARDS SYSTEMS

Summary of Requirements for
Borated Water Storage Tank

Inspection Requirements	Acceptance Standard
Welds	AWWA-D-100*

Additional Requirements:

All girth and seam welds are full-penetration butt welds.

Summary of Requirements for
Sodium Hydroxide Tank

Inspection Requirements	Acceptance Standard
Welds	ASME SECTION VIII

* Except ASME Section VIII, welding requirements were used for field erection welding.

TABLE 6.1-3
(Sheet 1 of 2)

ENGINEERED SAFEGUARDS PIPING DESIGN CONDITIONS

	Temperature (ºF)	Pressure (psig)
1. Makeup and Purification System		
a. From the pump discharge to downstream of the isolation valves outside the Reactor Building	200	3050
b. Makeup pump	200	3000
c. From downstream of the isolation valves to upstream of the check valves outside the secondary shielding	300	2500
d. From upstream of the check valves outside the secondary shielding to upstream of the check valves inside the secondary shielding.	554	2500
e. From upstream of the check valves inside the secondary shielding to the reactor inlet line	650	2500
2. Decay Heat Removal System		
a. From the borated water storage tank outlet to upstream of the electric-motor-operated valves in the borated water feed lines	150	75
b. From upstream of the electric-motor-operated valves to upstream of the check valves in the borated water feed lines	250	225
c. From upstream of the check valves to upstream of the valves at the decay heat removal pumps' inlets	300	370

TABLE 6.1-3
(Sheet 2 of 2)

ENGINEERED SAFEGUARDS PIPING DESIGN CONDITIONS

	Temperature (ºF)	Pressure (psig)
d. From upstream of the valves at the pump inlets to upstream of the Reactor Building isolation valves	300/250	470/505
e. From upstream of the Reactor Building isolation valves to upstream of the check valves in the core flooding lines	300	2500
f. From upstream of the check valves in the core flooding lines to the reactor vessel	650	2500
g. From the Reactor Building emergency sump to upstream of the valves in the recirculation lines	300	55 (building design pressure)
3. Reactor Building Spray System		
a. From the intertie with the Decay Heat Removal System at the isolation valve to downstream of the Reactor Building valves	300	350
b. From downstream of the inlet valves through the nozzles	300	200
c. Deleted		

TABLE 6.1-4
(Sheet 1 of 3)

SINGLE FAILURE ANALYSIS - EMERGENCY CORE COOLING SYSTEM

Component	Malfunction	Comments
1. Makeup and Purification System		
a. Suction valve for makeup pump from borated water storage tank	Fails to open	The parallel valve will supply the required flow to one pump string.
b. Emergency injection valve	Fails to open	The alternate line will provide the flow required for protection.
c. Makeup pump (operating)	Fails (stops)	Adequate injection is provided by the redundant pump.
d. Makeup pump	Fails to start	Adequate injection is provided by the redundant pump.
e. Seal return isolation valve	Fails to close on ES signal	The other isolation valve will close, eliminating this fluid path.
f. Letdown line isolation valve	Fails to close on ES signal	The other isolation valve will close the flow path.
g. HPI pump recirculation line valve	Fails to close on ES signal	The other valve will close the flow path.

TABLE 6.1-4
(Sheet 2 of 3)

SINGLE FAILURE ANALYSIS - EMERGENCY CORE COOLING SYSTEM

Component	Malfunction	Comments
2. Decay Heat Removal System		
(Injection From Borated Water Storage Tank)		
a. Decay heat removal pump	Fails to start	Adequate injection is provided by the redundant pump.
b. Emergency injection valve	Fails to open	Other line admits necessary flow.
c. Valve in suction line from BWST	Fails to open	Other line admits necessary flow.
(Recirculation From Reactor Building Emergency Sump)		
d. Valve in suction line from emergency sump	Fails to open	Other line admits necessary flow.
e. Valve in suction line from BWST	Fails to close after initiating	Check valve prevents flow into BWST.
f. Decay heat removal pump	Loss of pump	Reactor core protection will be maintained by alternate pump and low pressure injection string.

TABLE 6.1-4
(Sheet 3 of 3)

SINGLE FAILURE ANALYSIS - EMERGENCY CORE COOLING SYSTEM

Component	Malfunction	Comments
3. Core Flooding System		
a. Isolation valve in discharge line	Closes during normal operation	If the valve cannot be opened, the reactor must be shut down as specified in Technical Specs.
b. Tank relief valve	Opens during normal operation	Loss of nitrogen pressure and consequent loss of ability of tank to perform. Reactor must be shut down as specified by Technical Spec.
c. Check valves in discharge line	Excessive leak detected during normal reactor operation	It is extremely unlikely that both check valves would permit excessive leakage. Leakage would be indicated by core flooding tank pressure and level changes. If leakage becomes progressively worse or is unacceptably high, reactor must be shut down while the check valves are repaired.

TABLE 6.1-5
(Sheet 1 of 2)

EMERGENCY INJECTION SYSTEMS COMPONENT TESTING

Makeup pumps	One pump operates continuously. The other pumps will be tested periodically.
Makeup and purification emergency injection valves	The remotely operated stop valves in each line are opened one at a time.
Makeup pump suction valves	These valves are opened and closed individually and console lights monitored to indicate valve position.
Decay heat removal pumps	The pumps are used in normal service for shutdown cooling. Pump recirculation valves are normally open. Pumps will be run periodically and their discharge pressure observed.
Borated water storage tank outlet valves	These valves are opened and closed individually and console lights monitored to indicate valve position.
Decay heat removal emergency injection valves	These valves can be opened and reclosed and console lights monitored to indicate valve position.
Sump recirculation suction valves	With the decay heat removal pumps shut down, operation of these valves can be checked.

TABLE 6.1-5
(Sheet 2 of 2)

EMERGENCY INJECTION SYSTEMS COMPONENT TESTING

Check valves in core flooding injection lines	With the reactor shut down, the check valves in each core flooding line are checked for operability by closing the isolation valves, reducing the Reactor Coolant System pressure below CF Tank pressure, and opening the isolation valves. Check valve operability is shown by tank pressure and level changes.
Gate valves in cross connect of decay heat lines	With decay heat pumps shut down, operation of these manual valves can be checked.
Decay heat line flow indication	Flow will be monitored upon decay heat pump testing.

TABLE 6.1-6
(Sheet 1 of 1)

DESIGN BASIS ACCIDENT DH PUMP NPSH EVALUATION SUMMARY

RB sump water level	283.9 ft (elev.)
RB Pressure Used in NPSH Analysis	Vapor pressure when RB sump water temperature > 208^{o}F -1 psig when RB sump temperature < 208^{o}F
System Configuration	"A" LPI Pump operating with Cross-Connection Open 2 BS Pumps operating at 1180 gpm
Pump Flow, gpm [(a)]	3351 (LPI indicated flow of 2800 gpm)
NPSH Margin (ft H_2O)	0.1 ft
NPSH Required (3351 gpm)	12.93 ft

[(a)] Pump flow includes instrument error and minimum flow recirculation.

6.2 REACTOR BUILDING SPRAY SYSTEM

6.2.1 DESIGN BASES

The Reactor Building Spray System is designed to furnish building atmosphere cooling to limit postaccident building pressure to less than the design value and to reduce the building pressure to nearly atmospheric.

The Reactor Building Spray System removes energy from the environment by transferring heat from the higher temperature atmosphere to the lower temperature droplets injected from the BWST. Heat transfer continues until the spray droplets reach the saturation temperature associated with the pressure in the Reactor Building. In this manner, energy is redistributed from the atmosphere to the sump. The building spray system is switched to the sump-recirculation mode following the depletion of the BWST.

Trisodium phosphate is primarily used to control the RB Sump pH for long term corrosion control and to maintain elemental iodine in the sump water, thus reducing the inventory of airborne fission products (primarily iodine) available for leakage to the environment.

When the Reactor Building Spray system is taking suction from the RB Sump the spray will also remove elemental iodine from the RB atmosphere.

6.2.2 SYSTEM DESIGN

6.2.2.1 Piping And Instrumentation Diagram

A schematic diagram of the system is shown on Drawing C-302-712. Corresponding interfaces are shown on the Decay Heat Removal System schematic, Drawing C-302-640.

The Reactor Building Spray System serves only as an engineered safeguard and performs no normal operating function. In the event of a major LOCA, the system sprays borated water into the Reactor Building atmosphere to reduce the postaccident energy and to remove airborne fission product iodine. The system consists of two pumps, two Reactor Building spray headers, and the necessary piping, valves, instrumentation, and controls.

Once the RB Sump is placed in the recirculation mode the Reactor Building Spray system sprays a chemical solution that removes elemental iodine from the RB atmosphere. Trisodium phosphate stored in baskets on elevations 281'- 0" of the containment is provided for iodine removal and establishment of a pH > 7.0 during the event.

The pumps and remotely operated valves can be operated from the Control Room. Both paths operate independently, and the Reactor Building spray system also operates separately from the Reactor Building emergency cooling units which independently possess full postaccident cooling capability.

A crossover is provided between the spray system supply lines and contains double manual valves with a test line for recirculation of borated water from the building spray pumps.

The Reactor Building Spray System is maintained on standby during power operation. The operation of this system is initiated by high Reactor Building pressure. At a Reactor Building pressure of 4 psig, all electric operated valves automatically open.

A high Reactor Building pressure signal of 30 psig from the Engineered Safeguards Actuation System initiates operation.

The initiating system design minimizes the possibility of inadvertent initiation and is in accordance with the requirements of the IEEE Standard 279 (see 7.5 Reference 2).

During the injection phase, the two Building Spray pumps start and take suction from the borated water storage tank through the interface with the Decay Heat Removal System. During this phase, the BS pumps pump water from the BWST up to the spray rings and DH pumps pump water out of the RCS break. The water from these sources accumulates in the Reactor Building sump on elevation 281'-0". As the water level rises in the RB, it surrounds the baskets of TSP on elevation 281'-0" with borated water having a minimum pH of 4.6, and the TSP begins to dissolve into solution. The dissolving of the TSP is a completely passive process.

Once the BWST reaches the Lo-Lo Level, the recirculation phase is begun by opening the outlet valves on the RB sump and the injection phase is ended by closing of the BWST outlet valves. The Decay Heat and Building Spray pumps recirculate the water in the RB sump. Because the TSP has dissolved into solution, the pH of the water is between 7.3 and 8.0.

Trisodium phosphate reacts with the iodine in the RB Sump and converts it to nonvolatile iodine which is retained in the solution.

The Trisodium phosphate raises the pH of the borated sump water into the alkaline range, between 7.3. and 8.0 within four hours after the initiation of Building Spray.

The outlet valves from the RB sump are opened and the outlet valves from the borated water storage tank are closed remotely by the Control Room operator after the BWST reaches its minimum level.

Reactor Building sump level is monitored by two independent instrument channels which are qualified for operation with post accident environmental conditions.

6.2.2.2 Codes And Standards

The equipment is designed to the applicable codes and standards given in Tables 6.1-2 and 6.2-1.

6.2.2.3 Material Compatibility

All materials are compatible with the reactor coolant, sodium hydroxide, Trisodium Phosphate (TSP), and other borated water solutions. The major components of the system are constructed of stainless steel. Minor parts, such as pump seals, utilize other corrosion resistant materials.

6.2.2.4 Component Design

a. Pumps

The Reactor Building spray pumps are similar to those used in refinery service. These pumps are liquid penetrant tested by methods described in the ASME Boiler & Pressure Vessel Code, Section VIII, and are hydrotested and qualified to be able to withstand pressures greater than 1.5 times the design pressure. The system is designed so that periodic testing of the pumps may be performed to ensure operability. The certified performance test data for reactor building spray pump BS-P-1A is shown on Figure 6.4-1. In addition, the pump performance has been verified prior to fuel loading (see Appendix 13A, page 7 Reactor Building Spray System Functional Test).

b. Valves

The valves of the Reactor Building Spray System are inspected to the same requirements as the valves in the Emergency Core Cooling Systems. (Refer to Section 6.1.2.4.)

An electric motor operated globe valve is provided outside the building and a swing check valve is located inside the building for reactor building isolation.

c. Spray Headers and Nozzles

The spray nozzles are arranged on each of the two sets of concentric Reactor Building spray headers. The spray nozzles in each header are spaced to give uniform spray coverage of the Reactor Building volume above the operating floor.

d. Piping

Except for the section of lines requiring flanged connections for maintenance, the entire system is of welded construction. Table 6.1-3 lists the design conditions for this system. The piping shown on Drawing C-302-712 is designed in accordance with USAS Power Piping Code B31.1.0 and fabricated, erected, and inspected in accordance with USAS Nuclear Power Piping Code B31.7 - February 1968 - Draft, including June 1968 Errata.

e. Seismic Design

Components and piping in the Reactor Building spray system are designated as Class I equipment and are designed to maintain their functional integrity during a safe shutdown earthquake (SSE). Chapter 5 defines the acceptable stress limits for Class I equipment.

f. Instrumentation

The engineered safeguards actuation instrumentation for the Reactor Building spray system employs redundant channels as described in Chapter 7.

g. Quality Control

Quality standards for the Reactor Building Spray System components are given in Table 6.1-2.

6.2.2.5 Coolant Storage

This system shares borated water storage tank capacity with the Makeup and Purification System and the Decay Heat Removal System. The Sodium Thiosulfate and Sodium hydroxide tanks have been isolated from the Building Spray and Decay Heat systems.

6.2.2.6 Hydraulic Characteristics

Each pump is capable of delivering the required flow at peak Reactor Building pressure and BWST level at the minimum allowed during operation. Flow orifices have been sized to balance ECCS performance with iodine removal for dose consequences. These restrict actual spray flow to 800-1180 GPM per train.

Curves of total dynamic head and NPSH versus flow are shown on Figure 6.4-1.

6.2.2.7 Reliability Considerations

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is shown in Table 6.2-2.

6.2.2.8 Missile Protection

Protection against missile damage is provided by direct shielding or by physical separation of duplicate equipment. The spray headers are located outside and above the primary concrete shield.

6.2.2.9 Actuation

The Reactor Building spray system will be activated at a Reactor Building pressure of 30 psig. The system components may also be actuated by operator action from the Control Room for performance testing.

6.2.2.10 Environmental Considerations

None of the active components of the Reactor Building spray system are located within the Reactor Building, so none are required to operate in the steam-air environment produced by the accident.

6.2.3 DESIGN EVALUATION

The Reactor Building Spray System, acting independently of the Reactor Building Emergency Cooling System, is capable of limiting the containment pressure after a LOCA to a level which is below the design pressure and of eventually reducing containment pressure to near atmospheric level. The Reactor Building spray system is at least equivalent in heat removal capacity to the Emergency Cooling System. In combination with emergency cooling units, it affords redundant alternative methods to maintain containment pressure at a level below design pressure. Any of the following combinations of equipment will provide sufficient heat removal capability to accomplish this:

a. The Reactor Building Spray System alone.*

b. Three Reactor Building Emergency Cooling units alone.

c. One Reactor Building Emergency Cooling unit and the Reactor Building Spray System operating at one half capacity.

The Reactor Building Spray System will deliver 1600-2360 gpm, based on two pump operation and diesel loading delays, through the spray nozzles within 160 seconds after RB pressure exceeds 30 psig. The building spray system design flow rate of at least 800 GPM per train is verified by periodic test of the pumping capability.

This required time includes emergency diesel start, ES block loading, BS pump start, BS valve stroke times and the time to fill the piping and the spray header. In the event that spray pumps are fed by the auxiliary transformers in lieu of the diesels, all safeguards loads will be started essentially 10 seconds sooner (see Item b. of Section 8.2.3.1.).

The Reactor Building spray flow rate can be reduced from the Control Room by throttling valves BS-V1A and BS-V1B. There is "jog" close valve control in the Control Room for these valves. With the flow orifices sized to supply 800-1180 gpm per train, throttling of BS-V-1A/B is not required in any analyzed event.

Evaluations were performed of the RB spray and RB sump pH for the range of post-LOCA conditions. The BWST chemical composition (Table 6.2-1), limiting BWST levels, and a range of trisodium phosphate quantities were considered. The evaluations determined that the sump and spray pH range will be 7.3 to 8.0 during the post-LOCA recirculation phase.

The maximum hypothetical accident dose consequence analysis in Chapter 14 considers the post accident iodine retention in the RB Sump and removal effectiveness of reactor building spray.

6.2.4 TESTS AND INSPECTION

6.2.4.1 Initial Testing

During the preoperational testing program, the Reactor Building Spray system was tested by recirculation to the borated water storage tank.

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* This is beyond single-failure assumptions for system design. Other accident mitigation systems are providing core cooling and RB heat removal via RB sump recirculation.

6.2.4.2 Periodic Testing

a. Reactor Building Spray Pumps

The delivery capability of one pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

b. Borated Water Storage Tank Outlet Valves

These valves were tested in performing the pump test above.

c. Reactor Building Spray Injection Valves

With the pumps shut down and the borated water storage tank outlet valves closed, these valves can each be opened and closed by operator action.

d. Reactor Building Spray Nozzles

With the Reactor Building spray inlet valves closed, low pressure air or smoke can be blown through the test connections. Provisions are made so that visual observation will indicate if the nozzle flow paths are open.

TABLE 6.2-1
(Sheet 1 of 1)
REACTOR BUILDING SPRAY SYSTEM DATA

Tanks	Borated Water Storage (DH-T-1)
Design temperature, °F	150
Design pressure, ft H_2O	66
Code	AWWA-D-100[a]
Contents	
Boron Conc.	2500-2800 ppm B
Min Level	56.0 ft [b]
Capacity, gal	367,600[c]
Material	SS

Reactor Building Spray Pumps (BS-P-1 A/B)

Number	2
Nominal flow, gpm	1100[d]

Spray Headers

Number	2

Spray Nozzles

Number (total)	192
Type	Sprayco 1713A
Orifice diameter, inches	3/8

(a) Except ASME Section VIII, welding requirements were used for field erection welding.

(b) Technical Specifications require a minimum BWST volume of 350,000 gallons be maintained. (Reference 13)

(c) The tank volume at the 59 foot level, which is the bottom of the overflow pipe at the highest point of the inverted "J" loop. The tank capacity, based on B&W calculations (Reference B&W Letter No. REM-I-128, dated July 24, I975) is approximately 368,600.

(d) Flow restricting orifices establish the required spray flow. The design minimum is 800 gpm. The pumps may be tested up to 1500 gpm during recirculation to the BWST.

TABLE 6.2-2
(Sheet 1 of 1)

SINGLE FAILURE ANALYSIS - REACTOR BUILDING SPRAY SYSTEM

Component	Malfunction	Comments
1. Reactor Building spray pump	Fails to start	Since each of the two strings of the Reactor Building spray system is equally sized, the remaining string will provide heat removal capability at a reduced rate. In combination with the Reactor Building emergency cooling system, heat removal capability in excess of the requirements will be provided. Iodine removal is adequate with one string operating (Chapter 14 sensitivity analysis)
2. Building isolation valve	Fails to open	Same as above.
3. Check valve in discharge line	Fails to open	Same as above.
4. Suction isolation valve	Fails to open	Same as above.

6.3 REACTOR BUILDING EMERGENCY COOLING SYSTEM

6.3.1 DESIGN BASES

Reactor Building Emergency Cooling System is provided to limit postaccident ambient pressures to design values and maintain Reactor Building temperature within limits established to ensure the availability of safety related equipment in the Reactor Building.

Reactor Building air recirculation and cooling units, backed up by the Reactor Building spray system as described in Section 6.2, are used during emergency cooling periods. The systems are designed so that the heat removal capability required during the postaccident period can be attained by operating spray systems and cooling units in the emergency mode with the following combinations:

Spray Systems	Emergency Cooling Units
Full Capacity	None
Half Capacity	One
None	Three

The pressure-temperature results of the design basis accident (DBA) have been considered in the design of the Reactor Building coolers, fans, and motors. These are designed to operate in the postaccident environment and to withstand the maximum hypothetical earthquake, and are physically located outside the secondary shield wall, thus being isolated from the primary loop. The return air ductwork has been designed and analyzed to withstand simultaneously, both the design basis earthquake (DBE) and DBA pressure pulse forces. During a LOCA, the units will be subject to an external pressure. The units are designed to withstand an implosive load of 2 psi. Relief valves, specifically designed, built, and tested for this application, protect the units from the maximum predicted external pressures by permitting rapid equalization of the interior with the exterior pressure.

A single failure analysis of the RBEC system is shown in Table 6.3-1. The limiting single failure is loss of electrical power. As described in Appendix 6B the configuration with one RBEC cooling unit and one train of the spray system operating results in higher RB pressures and temperature than other single failures.

TMI-1 Environmental Qualification Temperature and Pressure profile analysis, described in FSAR Appendix 6B, is based on operation of only one ECCS train, one Reactor Building Spray train, and one fan cooler.

6.3.2 DESCRIPTION

The schematic flow diagram of the Reactor Building Emergency Cooling System with associated instrumentation is given on Drawings C-302-611 and C-302-831.

Emergency and normal cooling is performed with the same basic units, which are components of the Reactor Building Ventilation System described in Section 5.6. Each unit contains an emergency cooling coil, a normal cooling coil, and a two speed fan. For emergency cooling, the

units will operate at a reduced speed under postaccident conditions with the heat being rejected to river water. The back pressure regulating valve on the emergency cooling coil discharge line maintains emergency system pressure above maximum design basis accident containment pressure and prevents leakage out of the containment through a damaged system. Figure 6.3-1 shows the original design heat exchanger performance versus reactor building ambient conditions for these units. The original design data for the cooling units are shown in table 6.3-2. The analysis of post accident (LOCA) and MSLB) containment conditions (peak pressure and post accident pressure and temperature EQ profiles) is described in Appendix 6B. The Reactor Building Emergency Cooler heat removal rate is calculated in the GOTHIC model. The containment conditions are determined assuming volumetric RB air flow of 25000 cfm for each unit, a cooling water temperature of 95ºF, and a cooling water flow of 1450 gpm. The GOTHIC fan cooler model calculates the fan cooler heat removal rate based on the calculated containment conditions (i.e. pressure, temperature, and humidity). This fan cooler model is used to determine the Reactor Building post-accident temperature profile. The inputs and assumptions for this analysis are described in Table 6B-6 for LOCA and Table 6B-7 for MSLB. These analysis results are the EQ pressure and temperature profiles shown in Figure 6B-19 and 6B-20. Normal operation of the reactor building fan units is described in Section 5.6.

Seismic, pressure, and temperature factors as well as radiation and chemical degradation factors pertinent to the application have been accounted for in design. In addition, factory performance tests of components under simulated emergency conditions have been run. Tested components include the emergency cooling coil, housing, relief dampers, motor insulation, motor shaft seal, and fan/motor combination. A synopsis of the tests (items a. b. and c. below) is included as Appendix 6A. A summary of tests conducted is as follows:

a. Cooling Capacity of Coil at Postaccident Conditions

   Theoretical and experimental analyses were made of a representative coil sample at 281ºF, 68.3 psia; 245ºF, 14.7 psia; and 187ºF, 24.7 psia.

b. Relief Valve Test Program

   Theoretical and experimental analyses were made of relief capability of relief valve under dynamic pressure transient conditions.

c. Recirculation Fan and Motor Test

   Operation of fan and motor under simulated emergency conditions of temperature, pressure, humidity, and chemical spray was tested.

d. Motor Seal Test Operation of motor shaft seal was tested under simulated LOCA conditions of pressure by GPUNC.

e. Testing of Electrical Insulation

   Electrical insulation was irradiation tested by Knolls Atomic Power Laboratory.

6.3.3 ACTUATION

Receipt of the Reactor Building isolation signal (4 psig Reactor Building ambient pressure or a low reactor pressure of either less than 1600 psig or less than 500 psig backup signal) automatically switches the Reactor Building emergency cooling system to the emergency mode. This includes:

a. Energizing the three recirculating air handling units.

b. Operating the three units at the lower speed.

c. Starting the Reactor Building emergency cooling pumps.

d. Opening the emergency cooling coil isolation valve on the outlet side of the coil. Inlet valves are normally open for leak monitoring purposes.

e. Closing the normal cooling coil isolation valve.

TABLE 6.3-1
(Sheet 1 of 1)

SINGLE FAILURE ANALYSIS - REACTOR BUILDING EMERGENCY COOLING SYSTEM

	Component	Malfunction	Comments
1.	One of the emergency air handling units	Fails to start or fails to start at low speed	When any of the air emergency air handling units are not operating, they are automatically isolated from the system by gravity dampers. The remaining units will operate and in combination with the spray system provide RB heat removal.
2.	Cooling water supply and return lines or emergency cooling coil	Rupture	If a rupture or major leak is detected through indication of outlet flow, cooler outlet pressure, changes in RB sump level, then the coil and lines can be isolated from the Control Room.
3.	Supply or return valve	Fails to operate	The remaining units will operate and in combination with the spray system provide heat removal.
4.	One Reactor Building Emergency Cooling Water Pump	Fails to start	The remaining Reactor Building Emergency Cooling Water Pump provides the required flow.
5.	Back pressure valve in return water line from emergency cooling coils	Fails closed	A parallel valve has been provided which can be operated from the Control Room. The RB Spray system provides cooling until the bypass valve is opened. The bypass valve will be opened before sump recirculation is initiated.
6.	Emergency Diesel Generator	Diesel fails to start	At least one fan, and one pump will remain operating with RR flow through two or three coolers. Reactor Building cooling is provided by a single RR pump, cooling water flow through three coolers, and a single fan in combination with one train of Building Spray.

TABLE 6.3-2
(Sheet 1 of 2)

REACTOR BUILDING COOLING UNIT DESIGN PERFORMANCE AND EQUIPMENT DATA
(capacities are on a per-unit basis as originally designed and presented by manufacturer)**

Design Performance Data	Duty	
	Emergency	Normal
No. installed	3	3
No. required	3*	2
Type coil	Finned tube	Finned tube
Peak heat load, Btu/hr	80×10^6	2.15×10^6
Unit capacity, Btu/hr	92.9×10^6	2.33×10^6
Fan capacity, ft^3/min	54,000	108,000
Reactor Building Atmosphere Inlet Conditions		
Temperature, ºF	281	110
Steam partial pressure, psia	49.99	-
Air partial pressure, psia	18.31	-
Total pressure, psia	68.30	Atmospheric
Cooling water flow, gpm	1,780	430
Cooling water inlet temperature, ºF	85	85
Cooling water outlet temperature, ºF	183	95.6

* One is required with one spray pump operating.

** Table 6.3-2 reflects the original design heat removal capability of the RB Emergency Coolers. The current design performance requirements are in GOTHIC analysis described in Appendix 6B.

TABLE 6.3-2
(Sheet 2 of 2)

REACTOR BUILDING COOLING UNIT DESIGN PERFORMANCE AND EQUIPMENT DATA
(capacities are on a per unit basis as originally designed and presented by manufacturer)**

Emergency Cooling Unit Construction Data

Item	Description
Coil tubes	5/8 inch o.d. seamless copper with 0.049 in. wall 7.5 sq. ft. surface area per coil 30 coils per unit
Coil headers	2 5/8 in o.d.Copper with Schedule 80 steel connectors
Coil fins (emergency coil)	0.007 inch thick copper, spaced six per inch
Plenum casing	7 gauge steel floor, 11 gauge steel walls
Fan casing	3/8 inch steel plate
Motor	Totally enclosed, water cooled

** Table 6.3-2 reflects the original design heat removal capability of the RB Emergency Coolers. The current design performance requirements are in GOTHIC analysis described in Appendix 6B.

Tables 6.3-3 thru 6.3-9

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6.4 ENGINEERED SAFEGUARDS LEAKAGE AND RADIATION CONSIDERATIONS

6.4.1 INTRODUCTION

The use of normally operating equipment for engineered safeguards functions, and the location of some of this equipment outside the Reactor Building, requires that consideration be given to direct radiation levels from the fluids circulating in these systems and the leakage from these systems after fission products have accumulated.

The shielding for components of the engineered safeguards is designed to meet the following objectives in the event of a maximum hypothetical accident:

a. To provide protection for personnel to perform all operations necessary for mitigation of the consequences of the accident.

b. To provide sufficient accessibility in all areas around the station to permit safe continued operation of the required equipment.

In order to ensure reliable operation after an accident when decay heat removal vaults are not accessible, each DH pump is provided with remote capabilities for routine pump services. External lube oil reservoirs are connected to remote fill stations which allow pump and motor oil levels to be maintained without access to the DH vaults. Remotely operated vent valves provide the capability to vent air and non-condensables from the pump casing without access to the DH vaults.

6.4.2 SUMMARY OF POSTACCIDENT RECIRCULATION

Following a postulated RCS rupture, flow is initiated in the Makeup and Purification, Decay Heat Removal, and Core Flooding Systems from the borated water storage tank and core flood tanks to the reactor vessel. Flow is also initiated by the Reactor Building Spray System to the building spray headers. When the borated water storage tank inventory is exhausted, suction valves from the Reactor Building sump are opened by the operator, thus starting recirculation for decay heat removal. The post accident recirculation flow paths include all piping and equipment external to the Reactor Building, as shown on Figure 6.0-2, up to the valves leading to the borated water storage tank and the building spray system.

6.4.3 BASES OF LEAKAGE ESTIMATE

While the reactor auxiliary systems involved in the recirculation complex are closed to the Auxiliary Building atmosphere, leakage is possible through component flanges, seals, instrumentation, and valves.

The leakage sources considered are:

a. Valves

    1) Disc leakage when valve is on recirculation system boundary

    2) Stem leakage

3) Bonnet flange leakage

b. Flanges

c. Pump shaft seals

While leakage rates have been assumed for these sources, maintenance and periodic testing of these systems will preclude all but a small percentage of the assumed amounts. With the exception of the boundary valve discs, all of the potential leakage paths may be examined during periodic tests or normal operation. These periodic tests are performed IAW TS 4.5.4. The boundary valve disc leakage is retained in the other closed systems and, therefore, will not be released to the Auxiliary Building, or will be confirmed to be less than 3 GPM by periodic testing. (Ref. 32)

6.4.4 DESIGN BASIS LEAKAGE

Chapter 14 presents an analysis of the effects associated with the release of the radioactive fluid. It was concluded from this analysis that the leakage from the engineered safeguards systems outside the Reactor Building does not result in doses in excess of those allowed by 10CFR50.67. This analysis considered ES systems leakage to the Auxiliary Building and leakage through boundary valves to tanks vented to the atmosphere outside of the Auxiliary Building.

6.4.5 SAFEGUARDS CUBICLE LEAK DETECTION

Each of the seven cubicles, which individually contain a makeup, decay heat, or Reactor Building spray pump, is provided with two individual drain lines to the Auxiliary Building sump. One of the drain lines is located about 2 inches lower than the other in each cubicle and contains a leak detection level probe of the conductivity type located in an 8 inch diameter pipe recessed in the floor of each cubicle. This pipe is drained to the Auxiliary Building sump by a 3 inch diameter line through a backwater check valve. (The other drain line from the cubicle is also provided with a backwater check valve.)

The floor drain loop seal assembly is sized so that about 4.53 gpm of water flowing through it will back water up to an electrode level probe mounted within the 8 inch pipe. Each of the electrode level probes will actuate its individual "excessive leak" alarm on a panel in the Control Room because any leak in excess of 4.53 gpm is considered to be an indication of a significant failure which may require operator action to isolate the leaking component. A Decay Heat vault leak can be isolated with valves operable from the control room when the Decay Heat vault area may be inaccessible.

In the event that the operator is not able to act immediately to isolate the leak in the affected cubicle, the water in that cubicle will ultimately back up and start to flow out the second, unobstructed, drain line from the cubicle to the Auxiliary Building sump. The bottom of the Auxiliary Building sump is about 7 ft below the lowest floor level in any of the pump cubicles, and the sump has a capacity of about 9700 gal when the level in it is 6.5 ft above the sump floor. Two Auxiliary Building sump pumps are provided, each with a capacity of 150 gpm. These pumps can be operated in the automatic or manual mode. When selected to the automatic mode, the pump starts when the sump level reaches 5.5 feet. If the Auxiliary Building

sump recirculation isolation valve is open or the pumps are in manual control, operator action is required to pump the Auxiliary Building sump to the Miscellaneous Waste Storage Tank. The Auxiliary Building sump pumps are normally in automatic control. If the Auxiliary Building sump pumps are in recirc when a LOCA occurs the pump discharge will be returned to the Miscellaneous Waste Storage Tank prior to initiating Reactor Building Sump recirculation. When the sump level reaches a liquid level of 70 inches in the sump, an alarm will sound in the Control Room. At a 72 inch level in the sump, the second Auxiliary Building sump pump will start automatically in the automatic mode to pump out the sump (also at 150 gpm). A second alarm will sound at a liquid level of 80 inches in the Auxiliary Building sump. The equipment in the other cubicles connected to the Auxiliary Building sump is protected from becoming flooded by backwater check valves in the two drain lines from each of them.

Analysis has been performed (Reference 20) that demonstrates that without any sump pump operation the system provided for detecting a leaking safeguards component will provide adequate time for the isolation of that component without threatening the function or integrity of the redundant train.

Although it is anticipated that the maximum leak that will occur in any safeguards pump cubicle is much less than 300 gpm, the system is analyzed for a 600 gpm leak to demonstrate that even with twice the design leakage, the operator still has adequate time to isolate the leaking component.

The analysis was performed assuming that the leak was in the cubicle containing Building Spray Pump B which has the smallest cross-sectional area for equipment required to operate after RB sump recirculation is initiated. The Auxiliary Building sump is maintained below 3 ft, and the bottom of the Auxiliary Building sump pump motors are at approximately the 263 ft. elevation. From the time of the initial alarm from the leak detector in the affected pump cubicle until the water level reaches the bottom of the Auxiliary Building sump pump motors, more than 28 minutes would be available for isolation of the affected equipment. From the time of the initial Auxiliary Building sump level alarm, more than 19 minutes (which is adequate) would be available to isolate the affected equipment.

In the event that the maximum design leak (300 gpm) occurs in a safeguards pump cubicle, more than 42 minutes will elapse between receipt of the pump cubicle leak detector alarm and the time the water may threaten the integrity of the operable Auxiliary Building sump pumps. In the event that the cubicle leak detector does not alarm more than 32 minutes will elapse between actuation of the first Auxiliary Building sump level alarm and jeopardizing the integrity of the operable Auxiliary Building sump pump.

All level probe amplifiers and Control Room annunciators for the cubicle leak detection system are powered by vital ac. The Auxiliary Building sump pumps are also powered by vital ac.

6.4.6 PERIODIC LEAK REDUCTION PROGRAM

NUREG-0737 included a requirement for plants to implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to ‘as-low-as-practical’ levels. The ongoing aspect of this program involves periodic leak tests of the affected systems at least once during each refueling cycle.

The review of TMI-1 systems determined that the following systems or portions thereof are included in the scope of the program.

- Make-up and Purification
- Decay Heat Removal
- Reactor Building Spray
- Reactor Coolant Sampling
- Waste Disposal – Liquid
- Waste Disposal – Gas
-

The station has developed procedures to leak test the portions of these systems outside containment that are subject to having highly radioactive fluid in any accident scenarios. For liquid systems, the leak tests include measurement of the leakage, along with identification of the leak source. For gaseous systems, the leak tests monitor for acceptable pressure decay rate. Any leaking components or excessive pressure decay is identified in the Corrective Action Program for further investigation and repair. The station maintains the leak tests as surveillance procedures. Each of these surveillance procedures is scheduled to be performed at the refueling interval, and each references the NUREG-0737 requirement to perform the test and identify the leakage. The station maintains the results of the tests as ‘life-of-plant’ records.

TABLE 6.4-1
(Sheet 1 of 1)

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(The information contained in this Table has been relocated to Table 6.1-6.)

TABLE 6.4-2
(Sheet 1 of 1)

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TABLE 6.4-3
(Sheet 1 of 1)

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6.5 REACTOR BUILDING COMBUSTIBLE GAS CONTROL

6.5.1 DELETED

6.5.2 HYDROGEN MONITORING

6.5.2.1 Design Basis

The hydrogen monitoring equipment is capable of monitoring the hydrogen concentration inside containment during normal, accident, and postaccident conditions.

The initial design requirements of this system and its components meet the applicable requirements of Regulatory Guide 1.89 and IEEE-323-74 as defined in NUREG-0588, Category I. The analyzers have been analyzed and determined to be in a mild environment, therefore environmental qualification per Regulatory Guide 1.89 is not required nor maintained. Environmental qualification of sample isolation valves inside containment is maintained.

License Amendment 240 approved changing the RB Hydrogen monitor RG 1.97 Category 1 classification to Category 3. This eliminates the need for Safety Grade design requirements.

6.5.2.2 System Description And Design

Two complete analyzing and sample conditioning systems are installed. Each system is self-sufficient and independent of the other.

The hydrogen analyzer cabinets are located in the Intermediate Building at elevation 295 ft 0 inches and 322 ft 0 inches. Two separate hydrogen sampling lines are run to each analyzing and monitoring unit. Penetration #101S and #420S are used for redundant systems.

Sample points, one per analyzer, are located near the top of the containment dome.

The piping system within containment is Class N-2, and the entire system is Seismic Class I.

Hydrogen concentration is indicated locally and indicated and recorded in the Control Room. High hydrogen concentration (2 percent of H_2) is alarmed at the local panel and in the Control Room. The hydrogen monitor measurement capability is over a range of 0 to 10 volume percent within an accuracy of ±5 percent of full scale and drift of ±2 percent of full scale at the recorder per week under both positive and negative containment pressure. Hydrogen monitors although available at all times are not in service during normal operations.

Several trouble alarms are provided at the local analyzer panel, which annunciate a common alarm in the Control Room.

6.5.2.3 Design Evaluation

Independence of redundant circuits is maintained in accordance with Regulatory Guide 1.75 and IEEE-384-1974 to the extent permitted by the design of the existing raceway and electrical systems. (License Amendment 240 approved changing the RB Hydrogen monitor RG 1.97

Category 1 classification to Category 3. This eliminates the need for Safety Grade design requirements.)

Redundant mechanical systems are separated to prevent their loss of function as a result of high energy line breaks and missiles. A minimum separation distance of 3 feet was provided.

6.5.2.4 Tests And Inspections

The equipment shall be maintained in accordance with a regular schedule and as recommended by the manufacturer. Calibration of instruments shall be performed periodically in accordance with plant procedures.

Testing shall be performed in accordance with the applicable portions of Regulatory Guide 1.118 pertaining to testing of instrument channels.

6.5.3 HYDROGEN GENERATION

6.5.3.1 Design Basis

Reference 10 conservatively assumes that hydrogen is produced instantaneously by the cladding metal-water reaction. All zinc and aluminum exposed to corrosive conditions are accounted for by the calculation. The radiolytic decomposition rate of water is modeled as a function of time.

6.5.3.2 Phenomena Description

Hydrogen is produced by several chemical reactions, resulting from radiolysis following an accident, as follows:

a. Hydrogen Produced by Metal-Water Reaction

   Hydrogen is produced by the reaction of the Zircaloy-4 fuel cladding with steam according to:

   $$Zr + 2H_2O \longrightarrow ZrO_2 + 2H_2$$

   One pound of zirconium reacts to produce 0.0219 pound-moles of free hydrogen, yielding 8.48 scf of hydrogen per pound of reacted zirconium.

   The total amount of hydrogen produced is based on the amount of zirconium fuel cladding reacting with steam, as determined by the assumptions given in Branch Technical Position CSB 6-2, and is approximately 20,000 scf. Reference 31 demonstrated that this value is bounding for the Mark-B and Mark-B12 fuel designs. The Mark-B-HTP fuel design contains the same fuel rod design with the same material and cladding dimensions as the Mark-B12; therefore the conclusions of Reference 31 remain valid for the Mark-B-HTP design.

b. Hydrogen Produced by Corrosion of Aluminum and Zinc by Solutions (TSP) Used for Emergency Cooling or Containment Spray.

According to Standard Review Plan NUREG-75/087, Section 6.2.5, the production of hydrogen by aluminum corrosion is based on surface area within the containment and the surface corrosion rate according to the reaction:

$2Al + 3H_2O \leftrightarrow 3H_2 + Al_2O_3$

Zinc Corrosion is treated in a similar fashion and follows the reaction:

$Zn + 2OH^- + 2H_2O \longrightarrow Zn(OH)_4^{-2} + H_2$

One pound of aluminum reacts to produce 19.93 scf H_2. One pound of zinc produces 5.5 scf of hydrogen gas. The amount of hydrogen produced is based on the total amounts of aluminum and zinc inside of containment. The rate of hydrogen production is calculated in reference 10 and is dependent on the surface area of aluminum and zinc, and the aluminum / zinc corrosion rates given in NUREG-74/087.

c. Hydrogen Produced by Radiolytic Decomposition of the Postaccident Emergency Cooling Solutions.

GPUN Calculation C1101-901-5360-007 (Reference 10) calculates the total amount of hydrogen produced through radiolytic decomposition following a postulated design basis LOCA.

The hydrogen production rate due to radiolysis is the final element of hydrogen production in the Reactor Building. The zirc-water reaction is calculated to take place immediately. The hydrogen produced by radiolytic decomposition will occur after the LOCA.

6.5.3.3 Design Evaluation

TMI Technical Specifications Amendment #240 deleted the specifications for the hydrogen recombiners. The safety evaluation for TS Amendment #240 concluded that the risk from hydrogen combustion, resolution of GI-121, "Hydrogen Control for PWR Dry Containments," and the TMI-1 Individual Plant Examination (IPE), successfully demonstrated that TMI-1 could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function without credit for the hydrogen recombiners or the hydrogen purge system for not only the design-basis case, but also the more limiting severe accident sequences. Therefore, the requirements for the hydrogen recombiners and the backup hydrogen purge capability as part of the TMI-1 design basis are unnecessary, and their removal from the design basis is justified. See Technical Specifications Amendment #240 for additional safety evaluation details.

The safety evaluation for TS Amendment #240 also states that the risk associated with hydrogen combustion is not from design-basis accidents but from severe accidents. Multiple containment burns due to the operation of ignition systems could pose a serious threat to safety-related equipment located in the source compartment. The multiple burn environment was found to potentially be a threat because the source compartment temperature remains elevated form the previous burn. However, for TMI-1, this is not a concern because there is ample time between burns to reduce elevated containment temperatures via containment heat

removal systems. Reference 10 places limits on the amount of hydrogen and zinc materials that can be stored inside of containment to ensure that ample time exists in between multiple hydrogen burns to reduce elevated temperatures.

6.6 REACTOR BUILDING PRESSURE-TIME RESPONSE

6.6.1 Peak Containment Pressure

Mass and energy release rates are determined by the use of a multiregion CRAFT code. The nodal arrangement of the CRAFT model is shown on Figure 6.6-1. The secondary side of the steam generators and the associated feedwater piping are represented by control volumes 6, 17, 20, and 21. Main feedwater was added directly to the secondary side of the steam generators over a period of 17 seconds as the feedwater control valve closed. At 15 seconds, the paths connecting the feedwater piping to the steam generators (which were assumed to be closed in the CRAFT model until this time) were opened so that the mass and energy trapped in the piping could enter the steam generator. Auxiliary feedwater was started at approximately 35 seconds. Sensible heat stored in the steam generator tubes was modeled by slabs of metal in the secondary control volumes. Stored heat in the primary metal was simulated by slabs of metal in the control volumes which discharge their stored energy as the RCS temperature decreases.

To ensure a conservative calculation, the CRAFT code was run at 103 percent of the initial cycle* core power level (2619 MWt) and the mode of heat transfer in the core was assumed to be nucleate boiling until the quality of the coolant was approximately 1.0. Decay heat was calculated by using the ANS Standard time 1.2. This analysis bounds the power upgrade to 2568 MWt.

The entire transient (blowdown and reflood periods) were simulated using the CRAFT code. In order to present a conservative analysis for the Reactor Building pressure response, it was assumed that the worst case single failure was a power failure which resulted in minimum Reactor Building cooling (one cooler and one spray). This assumed failure allows operation of only one HPI and one LPI pump.

Breaks in the cold leg piping at the reactor coolant pump suction and the reactor coolant pump discharge have been analyzed. The results of the cold leg break analyses presented are for a spectrum of pump suction split-type breaks. The pressure-time responses for this spectrum of breaks are shown on Figures 6.6-2 through 6.6-7. The 7.0 ft^2 cold leg break results in the highest peak Reactor Building pressure of 49.6 psig and is approximately the same pressure as the worst hot leg break pressure of 49.4 psig. Hot leg breaks are shown on Figures 6.6-9 through 6.6-12.

Although the primary path for flow from the core is through the vent valves, the secondary side energy is removed by the steam generators. This effect is obvious in the pressure-time response curves.

The reactor building pressure-temperature analysis was performed using the digital computer code "CONTEMPT" developed by Phillips Petroleum Company in conjunction with the LOFT project. This program and its capabilities are described in Reference 2.

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* Design basis rated power of the initial core was originally 2535 MWt.

In this model, the reactor building is divided into two regions: the atmosphere (water vapor and air mixture) and the sump region (liquid water). Each region is considered to be well mixed and in thermal equilibrium, but the temperature of each region may be different. The reactor building and its internal structures are subdivided into five segments and treated as slabs with a one-dimensional heat transfer. Each segment, divided into several heat conducting subregions, may act as a heat source or sink. The program includes the capability of cooling the reactor building atmosphere by air coolers (reactor building spray system), and cooling the liquid region by sump cooler (decay heat removal coolers).

During blowdown, mass and energy are added directly to the atmosphere where the liquid water present is assumed to fall to the liquid region: these input quantities to CONTEMPT are obtained from CRAFT. After blowdown is over and emergency injection has been initiated, mass and energy are also added directly to the vapor region as steam. When the water level in the reactor vessel reaches the nozzle height, all mass and energy are added directly to the liquid region since no boiling of the injection water occurs. When the supply of injection water is depleted, recirculation and cooling of sump water is simulated; all mass and energy input to CONTEMPT after blowdown is obtained from PRIT; a B&W Company code. The PRIT progam has the same core thermal model as described for QUENCH and in addition to determining fuel and clad temperatures, maintains an inventory of coolant mass and energy in the reactor vessel and that released to the reactor building.

The reactor building calculations are begun by computing steady-state results using initial atmospheric conditions as the input. Following the rates for each time step. Heat losses or gains due to the heat- conducting slabs are calculated. Then the pressure and temperature of the liquid and vapor regions are calculated from the mass, volume, and energy balance questions.

The model has been developed so that the effectiveness of the natural heat sinks and the engineered safety features can be clearly demonstrated.

Table 6.6-1 shows the peak Reactor Building pressure and the time at which the peak occurs for each of the breaks in the hot and cold legs.

The core inlet velocity during the reflood stage is oscillatory in nature but, by using the integral of the core inlet flow path, the core average velocity was determined and is shown on Figure 6.6-8 for the 7.0 ft^2 suction break. The CRAFT code conservatively calculates an average carryout rate fraction of approximately 0.9 of the core inlet flow. Table 6.6-2 shows the leak flow rate and enthalpy for the 14.1 ft^2 hot leg break at the pump suction. As can be seen from Table 6.6-3, the leak flow was zero from 36 to 40 seconds and again from 44 to 48 seconds (caused by the building pressure, as calculated by CRAFT, coming into equilibrium with RCS pressure).

In order to provide a better understanding of where the energy came from, Table 6.6-4 shows an energy balance at t = 0 second, and at the time of peak pressure at t = 120 seconds for the 7 ft^2 cold leg break at the pump suction. Energy added by the core, steam generators, ECCS, and building cooling systems is also shown. The reference temperature for these calculations was 32°F with the exception of the Reactor Building structures, where the initial building temperature of 110°F was used. As can be seen from this table and Figure 6.6-3, all of the

available energy sources have contributed substantially to the Reactor Building pressure response, yet considerable margin remains between the peak building pressure and building design pressure.

The original OTSGs were replaced at the end of Cycle 17. An evaluation of the peak containment pressure response with the replacement OTSGs for a LOCA is contained in Reference 24. It was concluded that the current post-LOCA response remains bounding.

6.6.2 MINIMUM CONTAINMENT BACKPRESSURE

The minimum containment backpressure used in the analysis of the TMI-1 emergency cooling systems was calculated using the methods outlined in Section 5.2.2 of Reference 14, Section 4.3.6.1 of Reference 15, and Section 4.1.2.3 of Reference 19. The analysis was performed for an 8.55 ft^2 double ended break at the pump discharge - the break location and size which yielded the highest cladding peak temperature. The generic values described in Tables 6.6-5 through 6.6-9 were used in this analysis.

In a previous analysis, TMI-1 was modeled identically to the generic analysis except that some TMI-1 plant specific data were used as described in Tables 6.6-5 through 6.6-9. In all other cases, generic model parameters were used and these parameters are conservative for the TMI-1 design. A comparison of the results shown in Table 6.6-6, generated from both the TMI-1 and Generic models, demonstrates that using the Generic model values is conservative.

The minimum backpressure analysis was analyzed assuming that one HPI and one LPI pump was operable. To weight the effect of this assumption on the Reactor Building pressure, runs were made using two LPI pumps. A negligible difference in Reactor Building pressure was seen. Figure 6.6-13 (Reference 16) shows a comparison of RELAP5/CONTEMPT and CRAFT2/CONTEMPT minimum containment backpressure predictions. A peak minimum backpressure of approximately 35 psig was obtained. Both cases were performed with a rated core power level of 2772 MWt. The increased containment pressure response is a result of the change in evaluation models used in the analyses. The minimum backpressure for the most recent LOCA analyses (Reference 19) generated results that are indistinguishable from the previous analysis.

The original OTSGs were replaced at the end of Cycle 17. Evaluation of LOCA containment response with the replacement OTSGs used minimum containment back pressure. Assumptions used in determination of minimum backpressure are included in Reference 25. The analysis concluded that the current post-LOCA response remains bounding.

TABLE 6.6-1*
(Sheet 1 of 1)

PEAK REACTOR BUILDING PRESSURE VERSUS
BREAK SIZE AND LOCATION

Break Size (ft^2)	Location	Peak Pressure (psig)	Time of Peak (sec)
14.1	Hot leg	49.4	99.8
11.0	Hot leg	49.3	115
8.55	Hot leg	49.1	112
5	Hot leg	49.1	146
8.55	Cold leg (pump suction)	48.8	141
7.0	Cold leg (pump suction)	49.6	120
5.13	Cold leg (pump suction)	49.1	143
3	Cold leg (pump suction)	47.0	177
2	Cold leg (pump suction)	45.9	200
0.5	Cold leg (pump suction)	39.2	260

* This Table is retained for information only. The actual design basis accident is a 7.0 ft^2 rupture at RCP suction as described in Section 6B, paragraph 1.2.

TABLE 6.6-2
(Sheet 1 of 1)

MASS RATE AND ENTHALPY TO THE REACTOR BUILDING
FOR A 14.1 ft^2 HOT LEG BREAK

Time Interval (sec)	Average Mass Flow Rate (lb/sec)	Average Enthalpy (Btu/lb)
0-2	80,249.5	599.2
2-4	60,009	595.1
4-6	46,675.5	601.9
6-8	30,404.5	636.1
8-10	14,993	736.3
10-12	4,807	1,031.8
12-14	2,376	1,181.3
14-16	1,636	1,065.7
16-18	1,326.5	611.3
18-21	259.33	780.2
21-20	0	0
26-30	321.5	269.8
30-34	953	274.1
34-38	54	263.8
38-42	546.75	304.5
42-44	168	857.1
44-46	1,540.5	296.9
46-48	2,936.5	325.3
48-50	2,084	510.5
50-52	2,337.5	459.6
52-54	2,419.5	504.2
54-56	2,963	403.4
56-58	3,391.5	369.4
58-60	3,268.5	395.5
60-70	2,938.1	404.8
70-80	2,185.1	472.7
80-90	1,778.3	510.4
90-100	1,387	420.0
100-120	493.85	455.6
120-140	403.65	376.3
140-160	366.2	366.0
160-180	363.75	350.1
180-200	368	337.0
200-220	370.85	326.2
220-240	365.8	315.3
240-260	397.25	303.0
260-280	482.7	293.3
280-300	453.8	290.6

TABLE 6.6-3
(Sheet 1 of 1)
MASS RATE AND ENTHALPY TO THE REACTOR BUILDING
FOR A 7 ft^2 SPLIT AT THE PUMP SUCTION

Time Interval (sec)	Average Mass Flow Rate (lb/sec)	Average Enthalpy (Btu/lb)
0-2	57,300	558.333
2-4	53,350	566.382
4-6	47,035	583.019
6-8	34,195	617.780
8-10	22,187	689.503
10-12	12,904	765.916
12-14	4,768	1,086.838
14-16	4,630	735.501
16-18	5,605	520.250
18-20	5,416	457.903
20-24	2,893	419.165
24-28	899	412.658
28-32	14	298.246
32-36	38	337.748
36-40	0.0	0.0
40-44	48	333.333
44-48	0.0	0.0
48-56	79	265.263
56-62	1,427	416.472
62-68	656	1,073.895
68-74	1,477	604.153
74-80	2,688	461.519
80-90	2,479	477.207
90-100	1,959	538.497
100-110	1,131	692.002
110-120	561	888.632
120-140	157	1,133.524
140-160	54	1,180.801
160-180	50	1,182.093
180-200	48	1,148.691
200-240	323	431.353
240-280	673	474.770
280-320	556	438.506
320-360	340	344.001
360-400	413	322.518
400-440	540	316.633
440-480	391	300.288
480-520	383	280.439
520-560	359	275.384
560-600	345	273.663

TABLE 6.6-4
(Sheet 1 of 1)
ENERGY DISTRIBUTION FOR THE 7 FT^2 BREAK (SPLIT)
AT REACTOR COOLANT PUMP SUCTION

DESCRIPTION	ENERGY BEFORE ACCIDENT Btu X 10^{-6}	ENERGY ADDED BETWEEN 0 6 & 120 SEC.	ENERGY AT PEAK PRESSURE (120 SEC.) Btu X 10^{-6}
Reactor Coolant System, coolant	298.17		24.28
Reactor Coolant System, Structures			
a. Fuel & Cladding	22.95		5.89
b. Vessel, piping pressurizer, and primary Side of steam generators	157.22		147.31
Core heat generation		17.32	
ECCS coolant		13.04	
From secondary system including tubes		31.03	
Reactor Building atmosphere	1.78		237.07
Reactor Building sump	0.0		82.98
Reactor Building structures	0.0		42.66
Reactor Building coolers		- 1.94	
Reactor Building sprays		0.62	
	--------	---------	--------
	480.12	60.07	540.19

TABLE 6.6-5
(Sheet 1 of 2)

A COMPARISON OF KEY PARAMETERS OF THE GENERIC EVALUATION MODEL FOR MINIMUM CONTAINMENT PRESSURE VERSUS INDIVIDUAL PLANT PARAMETERS

Parameter	Generic Model	TMI-1
Reactor Building Free Volume, ft^3	2.205×10^6	2.126×10^6
a. The Reactor Building Walls Including the Concrete Wall, Steel Liner,and Anchors[1]		
Exposed area, ft^2	67,410	63,300
Paint thickness, ft	0.00083	0.00083
Steel thickness, ft	0.05504	0.02946
Concrete thickness, ft	4.0	3.5
The surface area is 5 percent larger than the largest values reported for Category 1 plants.		
b. The Reactor Building Dome Including Concrete, Steel Liner, and Anchors[1]		
Exposed area, ft^2	18,375	18,400
Paint thickness, ft	0.00083	0.00083
Steel thickness, ft	0.06546	0.02946
Concrete thickness, ft	3.0	3.0
The surface area is 5 percent larger than the largest value reported for Category 1 plants.		
c. Painted Internal Steel[2]		
Exposed area, ft^2	249,000	311,599
Paint thickness, ft	0.00083	0.00083
Steel thickness, ft	0.03125	0.0227
d. Unpainted Internal Steel[2]		
Exposed area, ft^2	36,000	94
Steel thickness, ft	0.03125	0.0054

TABLE 6.6-5
(Sheet 2 of 2)

A COMPARISON OF KEY PARAMETERS OF THE GENERIC EVALUATION MODEL FOR MINIMUM CONTAINMENT PRESSURE PLANT PARAMETERS

Parameter		Generic Model	TMI-1
e.	Unpainted Stainless Steel[2,3]		
	Exposed area, ft^2	10,000	42,151
	Steel thickness, ft	0.03125	0.0108
f.	Internal Concrete[2,3]		
	Exposed area, ft^2	160,000	118,000
	Paint thickness, ft	0.00083	0.00083
	Concrete thickness, ft	1.0	1.54

Notes:
1. The values displayed for the RB walls and dome heat structures represent the inside surface and the total thickness of the steel and concrete.
2. The values displayed for the remaining heat structures represent the total surface area of both sides of the component with the half-thickness of the steel and concrete.
3. The values are reported in this manner to conserve the heat structure volume, obtained by multiplying the exposed area by the reported thickness.

MATERIAL PROPERTIES USED IN THE MODEL FOR THE MINIMUM CONTAINMENT PRESSURE

Material	Thermal Conductivity BTU/sec-ft-F	Heat Capacity BTU/ft^3-F
Concrete	0.2556e-3	22.62
Steel	0.75e-2	58.8
Stainless Steel	0.2551e-2	54.263
Paint	0.1726e-3	40.42

TABLE 6.6-6
(Sheet 1 of 1)

MINIMUM CONTAINMENT – BACK PRESSURE COMPARISON:
TMI-1 AND GENERIC

Time (sec)	Pressure TMI-1	(psig) Generic
2.0	13.32	12.29
4.7	18.11	17.52
6.7	22.19	21.46
8.9	25.67	24.75
9.9	27.00	26.00
12	29.24	28.11
13	30.10	28.59
15	31.34	29.87
17	32.05	30.41
19	32.46	30.68
21	32.49	30.59
23	31.87	29.54
25	30.67	28.71
27	29.43	27.69
29	28.38	26.61
31	27.53	25.77
33	26.74	25.11
35	26.09	24.53
37	25.53	24.08
39	25.04	23.69
42	24.42	23.22
46	23.70	22.77
48	23.32	22.50
52	22.81	22.10
56	22.38	21.84
72	21.55	21.41
76	21.42	21.31
78	21.38	21.28
82	21.27	21.21
88	21.18	21.14
92	21.10	21.07
94	21.07	21.06
96	21.03	21.03
98	21.00	20.99
105	20.91	20.30
115	20.82	20.79
125	20.69	20.69
135	20.60	20.52
145	20.47	20.33
155	20.36	20.23
165	20.25	20.09
175	20.14	19.24
185	20.01	19.79
195	19.89	19.63

TABLE 6.6-7
(Sheet 1 of 1)
REACTOR BUILDING DATA FOR REACTOR BUILDING
PEAK PRESSURE ANALYSIS

STRUCTURAL PROPERTIES

Segment	Material Type	Surface Area, $Ft.^2$	Thickness	Description
1.	Paint	81,700	I5 mils	Reactor Building
	Steel	81,700	3/8 in.	Wall & Dome
	Air Gap	81,700	1/32 in.	
	Conc.	81,700	3.4 Ft.	
2.	Steel	6,000	1/4 in.	Refueling Canal-
	Air Gap	6,000	1/32 in.	Stainless Steel
	Conc.	12,000	3.73 ft.	Liner on inside
	Paint	6,000	15 mils	
3.	Paint	11,000	15 mils	Reactor Building
	Conc.	11,000	2.0 ft.	Floor
4.	Paint	106,110	15 mils	Misc. Internal
	Steel	106,110	0.268 in.	Steel
	5.Paint	117,800	15 mils	Misc. Internal
	Conc.	117,800	1.435 ft.	Concrete

THERMAL PROPERTIES

Material	Density (lb/ft^3)	Thermal Conductivity ($Btu/h\text{-}ft^2\text{-}F/ft$)	Heat Capacity (Btu/°F-lb)
Paint	103	0.20	0.35
Steel	490	26.0	0.12
Air	0.0721	0.0184	0.17
Concrete	145	0.45	0.156
S. Steel	503	26.0	0.12

TABLE 6.6-8
(Sheet 1 of 2)

General Parameters Used in Reactor Building
Peak Pressure Analysis*

a.	Reactor Power Level	2,619 MWt
b.	Low Pressure Injection Flow	3,000 gpm
c.	High Pressure Injection Flow	500 gpm
d.	Borated Water Storage Tank Volume	350,000 gal.
e.	Reactor Building Initial Pressure	0 psig
f.	Reactor Building Initial Temperature	110°F
g.	Reactor Building Initial Humidity	0%
h.	Reactor Building Free Volume	2.0×10^6 ft^3
i.	Design Pressure	55 psig
j.	Design Temperature	281°F
k.	BWST Temperature (ECCS and Building Spray Temperatures before Recirculation)	90°F**

* This analysis and the corresponding assumptions remain conservative because of the model used and are not affected by the revised peak temperature profile described in Section 6.3.2. For the analysis supporting Tech Spec 3.17 temperature limits of ≤120°F below elevation 320 ft and ≤130°F above elevation 320 ft, see Chapter 6B.

** Re-analysis using BWST temperature at 120°F shows a peak pressure increase of less than 0.5 psi with no increase in peak clad temperature (Reference 6).

TABLE 6.6-8
(Sheet 2 of 2)

General Parameters Used in Reactor Building
Peak Pressure Analysis*

1.	Number of Air Coolers	1 of 3
2.	Design Heat Removal Capacity per Cooler	80.0×10^6 Btu/h
3.	Delay Time From Actuation for Fans	25 s
4.	Reactor Building Design Temperature	281°F
5.	Number of Spray Systems	1 of 2
6.	Spray Flow Rate per System	1,500 gpm**
7.	Delay Time from Actuation for Sprays from 56 to 71 sec. after break**	Ramp up to Full Flow

* This analysis and the corresponding assumptions remain conservative because of the model used and are not affected by the revised peak temperature profile described in Section 6.3.2. For the analysis supporting Tech Spec 3.17 temperature limits of ≤120°F below elevation 320 ft and ≤130°F above elevation 320 ft, see Chapter 6B.

** Re-analysis using BS flow of 800 GPM and delay of 160 seconds shows a peak pressure increase of less than 0.5 psi with no increase in peak clad temperature (Reference 22).

TABLE 6.6-9
(Sheet 1 of 1)

HEAT TRANSFER COEFFICIENTS APPLIED TO THE
HEAT STRUCTURES FOR MINIMUM BACK PRESSURE ANALYSIS

Transient Phase	Condensing Heat Transfer Coefficient (HTC)
Initial	8 BTU/hr-ft^2-F
Blowdown	Linear increase from the Initial HTC to 4.0 times Tagami Correlation
Transition	Exponential transition from the HTC at the End of Blowdown to the HTC for the Long-Term Stagnation Phase with a decay constant of 0.025
Long-Term Stagnation	1.2 times Uchida

6.7 REFERENCES

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5. Deleted

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7. Deleted

8. Deleted

9. Deleted

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29. NUREG-0737, "Clarification of TMI Action Plan Requirements."

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