6.3 Emergency Core Cooling Systems

6.3.1 Design Bases and Summary Description

Subsection 6.3.1 provides the design bases for the Emergency Core Cooling Systems (ECCS) and a summary description of the several systems as an introduction to the more detailed design descriptions provided in Subsection 6.3.2 and the performance analyses described and provided in Subsection 6.3.3.

6.3.1.1 Design Bases

6.3.1.1.1 Performance and Functional Requirements

The ECCS is designed to provide protection against postulated loss-of-coolant accidents (LOCAs) caused by ruptures in primary system piping. The functional requirements (e.g., coolant delivery rates) specified in detail in Table 6.3-1 are such that the system performance under all LOCA conditions postulated in the design satisfies the requirements of 10CFR50, Paragraph 50.46 (Acceptance Criteria for Emergency Core Cooling System for Light-Water-Cooled Nuclear Power Reactors). These requirements are summarized in Subsection 6.3.3.2. In addition, the ECCS is designed to meet the following requirements:

(1) Protection is provided for any primary system line break, including the double-ended break of the largest line.

(2) Three high-pressure cooling systems are provided, each of which is capable of maintaining water level above the top of the core and preventing ADS actuation for breaks of lines less than 25mm in diameter.

(3) No operator action is required until 30 minutes after an accident to allow for operator assessment and decision.

(4) The ECCS is designed to satisfy all criteria specified in Subsection 6.3.3.2 for a LOCA following any normal mode of reactor operation.

(5) A sufficient water source and the necessary piping, pumps and other hardware are provided so that the containment and reactor core can be flooded for possible core heat removal following a LOCA.

6.3.1.1.2 Reliability Requirements

The following reliability requirements apply:

(1) The ECCS must conform to all licensing requirements and good design practices of isolation, separation and common mode failure considerations.

(2) In order to meet the above requirements, the ECCS network has built-in redundancy so that adequate cooling can be provided, even in the event of specified failures. As a minimum, the following equipment make up the ECCS:

(a) Reactor Core Isolation Cooling System (RCIC)

(b) High Pressure Core Flooder System (HPCF)

(c) Low Pressure Flooder (LPFL) mode of Residual Heat Removal System (RHR)

(d) Automatic Depressurization System (ADS)

(3) The system shall be designed so that a single active or passive component failure, including power buses, electrical and mechanical parts, cabinets and wiring will not disable the ADS.

(4) In the event of a break in a pipe that is not used by the ECCS, no single active component failure in the ECCS shall prevent automatic initiation and successful operation of less than the following combinations of ECCS equipment:

(a) One HPCF + RCIC + two LPFL + all ADS valves

(b) Two HPCF + three LPFL + all ADS valves

(c) Two HPCF + RCIC + three LPFL + all ADS valves minus one

(5) In the event of a break in a pipe that is used by the ECCS (this includes some portion of the feedwater piping), no single active component failure in the ECCS prevents automatic initiation and successful operation of less than the combination of ECCS equipment as identified in Subsection 6.3.1.1.2 (4), minus the ECCS in which the break is assumed.

These are the minimum ECCS combinations which result after assuming any failure (from item (4) above) and assuming that the ECCS line break disables the affected system.

(6) Long-term cooling requirements call for the removal of core decay heat via the reactor building cooling water system. These requirements state that in addition to the break which initiated the loss-of-coolant event, the ECCS design must be able to sustain one failure, either active or passive and still have at least two low pressure ECCS loops with heat exchangers receiving 100% reactor building cooling water flow and one ECCS pump that provides flow to the vessel. One of the two low pressure ECCS loops can provide flow to the vessel as well as removal of decay heat, since each of the three low pressure ECCS loops include heat exchangers and the pump flow passes through the heat exchanger where it is cooled before it is injected into the reactor vessel.

(7) Offsite power is the preferred source of power for the ECCS network, and every reasonable precaution must be made to assure its high availability. However, onsite emergency power shall be provided with sufficient diversity and capacity so that all the above requirements can be met even if offsite power is not available.

(8) The onsite diesel fuel reserve is in accordance with Regulatory Guide 1.137.

(9) The diesel-load configuration provides one diesel generator for each of the three ECCS divisions. See Chapter 7 (Section 7.3) for details on loading sequence. Also, the Lungmen NPS design includes an extra diesel, named “Swing Diesel.” For details on swing diesel, see Chapter 8 (Sections 8.1 and 8.3).

(10) Systems which interface with, but are not part of, the ECCS are designed and operated such that failure(s) in the interfacing systems shall not propagate to and/or affect the performance of the ECCS.

(11) Each system of the ECCS, including flow rate and sensing networks, is capable of being tested during plant operation, including logic required to automatically initiate component action.

(12) Provisions for testing the ECCS network components (electronic, mechanical, hydraulic and pneumatic, as applicable) are installed in such a manner that they are an integral part of the design.

6.3.1.1.3 ECCS Requirements for Protection from Physical Damage

6.3.1.1.3.1 Protection of ECCS Piping

The ECCS piping and components are protected against damage from:

(1) Movement

(2) Thermal stresses

(3) Effects of the LOCA

(4) Effects of the safe shutdown earthquake and operating basis earthquake

The ECCS piping and components inside the containment are protected against the effects of pipe whip, which might result from piping failures up to and including the design basis event LOCA. This protection is provided by separation, pipe whip restraints, or energy-absorbing materials if required. One of these three methods will be applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level.

The ECCS piping and components located outside the primary containment are protected from internally and externally generated missiles by the reinforced concrete walls of the Reactor Building. In addition, the watertight doors are provided to ECCS pump rooms to protect against flooding of redundant ECCS pump rooms.

6.3.1.1.3.2 Mechanical Separation of ECCS

Mechanical separation outside the drywell is achieved as follows:

(1) The ECCS shall be separated into three functional groups:

  (a) RCIC + 1 RHR + ADS

  (b) 1 HPCF + 1 RHR + ADS

  (c) 1 HPCF + 1 RHR + ADS

(2) The equipment in each group shall be separated from that in the other two groups.

(3) Separation barriers shall be constructed between the functional groups, as required, to assure that environmental disturbances such as fire, pipe rupture, falling objects, etc., affecting one functional group will not affect the remaining groups.

6.3.1.1.4 ECCS Environmental Design Basis

Each ECCS has a safety-related injection/isolation testable check valve located in piping within the drywell, except RCIC and RHR in Division A, which connect to feedwater lines outside the drywell. In addition, the RCIC System has an isolation valve in the drywell portion of its steam supply piping. The portions of ECCS piping and equipment located outside the drywell and within the secondary containment are qualified for the environmental conditions defined in Section 3.11.

6.3.1.2 Summary Descriptions of ECCS

The ECCS injection network is comprised of RCIC, HPCF, and RHR. These systems are briefly described here as an introduction to the more detailed system design descriptions provided in Subsection 6.3.2. ADS, which assists the injection network under certain conditions, is also briefly described.

6.3.1.2.1 High Pressure Core Flooder

The HPCF pumps water through a flooder sparger mounted above the reactor core. Coolant is supplied over the entire range of system operation pressures. The primary purpose of HPCF is to maintain reactor vessel inventory after small breaks which do not depressurize the reactor vessel.

6.3.1.2.2 Residual Heat Removal

Two of the RHR loops deliver water through a flooder sparger mounted above the core shroud head. The third RHR loop delivers water to the Feedwater (FW) line which delivers water through the FW sparger above the core shroud head.

RHR has three independent loops and delivers water to the core at relatively low reactor pressures. The primary purpose of RHR is to provide inventory makeup and core cooling during large breaks, and to provide containment cooling. For small breaks, RHR provides inventory makeup following ADS initiation.

6.3.1.2.3 Reactor Core Isolation Cooling

RCIC injects water into a feedwater line, using a pump driven by a steam turbine. The RCIC steam supply line branches off one of the main steamlines leaving the reactor pressure vessel and goes to the RCIC turbine. Makeup water is supplied from the condensate storage tank (CST) or the wetwell suppression pool, with the preferred source being the CST.

6.3.1.2.4 Automatic Depressurization System

ADS utilizes eight (8) of the reactor safety/relief valves (SRVs) to reduce reactor pressure. Normally, during a small pipe break RCIC is initiated at high drywell pressure or low reactor water level (Level 2) signal. If for some reason, RCIC fails to start, then HPCF is initiated by high drywell pressure or low reactor water level (Level 1.5) signal. If both RCIC and HPCF fail to inject sufficient water into the vessel, then ADS is initiated (Level 1) to reduce the vessel pressure, and the ADS initiation may require actuation of all eight or less SRVs. When the vessel pressure is reduced to within the capability of the low-pressure cooling system, the low pressure flooder mode of RHR can start in addition to any running HPCF pump and provide additional water inventory makeup so that acceptable post accident temperatures are maintained.

6.3.2 System Design

A more detailed description of the individual systems, including individual design characteristics of the systems, is provided in Subsections 6.3.2.1 through 6.3.2.4.

The following discussion provides details of the combined systems; in particular, those design features and characteristics which are common to all systems.

6.3.2.1 Schematic Piping and Instrumentation Diagrams

The P&IDs for the ECCS are identified in Subsection 6.3.2.2. The process flow diagrams which identify the various operating modes of each system are also identified in Subsection 6.3.2.2.

6.3.2.2 Equipment and Component Descriptions

The starting signal for the ECCS is derived from four independent and redundant sensors of drywell pressure and low reactor water level. The ECCS is actuated automatically and requires no operator action during the first 30 minutes following the accident. A time sequence for starting of the systems during a typical limiting LOCA is provided in Table 6.3-2.

Electric power for operation of the ECCS is from regular AC power sources. Upon loss of the regular AC power sources, operation is from onsite emergency standby AC power sources. Emergency power sources have sufficient capacity so that all ECCS requirements are satisfied. Each of the three ECCS functional groups identified in Subsection 6.3.1.1.3.2(1) has its own diesel generator emergency power source. Section 8.3 contains a more detailed description of the power supplies for the ECCS.

Regulatory Guide 1.1 prohibits design reliance on pressure and/or temperature transients expected during a LOCA for assuring adequate NPSH. The requirements of this Regulatory Guide are applicable to the HPCF, RCIC and RHR pumps.

The Lungmen NPS design conservatively assumes 0 kPaG containment pressure and maximum expected temperatures of the pumped fluids. Thus, no reliance is placed on pressure and/or temperature transients to assure adequate NPSH.

Requirements for NPSH are given in Tables 6.2-2c(HPCF), 5.4-2 (RCIC) and 6.2-2b (RHR). Vessel pressure versus system flow curves are given in Figures 6.3-4 (HPCF), 6.3-5 (RCIC) and 6.3-6 (RHR).

The design parameters for the HPCF and the RHR components are provided in Tables 6.3-8 and 6.3-9, respectively.

6.3.2.2.1 High Pressure Core Flooder System (HPCF)

HPCF is composed of two HPCF loops (B and C) flooding water to the RPV above the core. Each of the two loops belongs to a separate division; electrical and mechanical separation between the two divisions is complete. Physical separation is also assured by locating each division in a different area of the Reactor Building. The two loops are both high pressure pumping systems (i.e., they are capable of injecting water into the reactor vessel over the entire operating pressure range). Rated flow at both high and low pressure is the same for each loop. The piping and instrumentation diagram and process flow diagram are given in Figures 6.3-7 and 6.3-1, respectively.

The reference pressure for the operating performance of the system at high pressure is the lowest spring (safety) setpoint of the SRVs.

Both HPCF divisions take primary suction from the CST and secondary suction from the suppression pool. In the event CST water level falls below a predetermined setpoint or suppression pool water level rises above a predetermined setpoint, the pump suction will automatically transfer from the CST to the suppression pool. Both HPCF loops have suction lines that are separate from the RHR loops.

The HPCF pumps are located at an elevation which is below the water level in the suppression pool. This assures a flooded pump suction. The motor-operated valve in the suction line from the suppression pool on each division is normally closed, since primary suction is taken from the CST. This valve automatically opens on receipt of either of the suction transfer signals noted above. The suppression pool suction valves on each loop are capable of being closed from the control room if a leak develops in the system piping downstream of the isolation valves. Overpressure protection of the pump suction line is provided by a relief valve with discharge to the suppression pool.

Each of the two high pressure flooder loops discharges water into the core via a separate flooder sparger. Internal piping connects each sparger to the vessel nozzle.

Each HPCF discharge line to the reactor is provided with two isolation valves in series. One of these valves is a testable check valve located inside the drywell. HPCF injection flow causes this valve to open during LOCA conditions; thus, no power is required for valve actuation during LOCA. If an HPCF line should break outside the drywell, the check valve in that line inside the drywell will prevent loss of reactor water outside the drywell. The other isolation valve (which is also referred to as an HPCF injection valve) is a motor-operated gate valve located outside the drywell and as close as practicable to the HPCF discharge line drywell penetration. This valve is capable of opening with the maximum pressure differential across the valve expected for any system operating mode including HPCF pump shutoff head. This valve is normally closed as a backup to the inside testable check valve for containment integrity purposes. A vent line is provided between the two valves. A normally open manual isolation valve inside the drywell is provided for HPCF loop maintenance during a plant refueling or maintenance outage.

For each loop, a full flow line is provided with discharge to the suppression pool to allow for full flow test of the system during normal operation. The valves in these lines are closed during normal operation. A full flow test return line is consistent with established BWR practice. There is no Regulatory Guide requiring this feature, but all BWRs have a 100% capacity test return line, and the Chapter 16 Technical Specifications specify periodic full flow system functional tests. There are no specific requirements for testing at runout flow; however, the system does have this capability.

For each loop, a pump minimum flow bypass line is also provided to return water to the suppression pool to prevent pump damage due to overheating when the injection valves on the main discharge lines are closed. The bypass line connects to the main discharge line between the main pump and the discharge check valve. A motor-operated valve on the bypass line automatically closes when sufficient flow in the main discharge line has been established. A flow element in the main discharge line measures system flow rate during LOCA and test conditions and automatically controls the motor-operated valve on the minimum flow bypass line.

HPCF is designed to operate from normal offsite auxiliary power or from emergency diesel generators if offsite power is not available. If normal auxiliary power is lost, the onsite power source (diesel generator) for that division is started. The onsite power source for any division is capable of carrying all of the division emergency loads, including the HPCF pump and valve motors. Manually operated remote controls for system components (such as HPCF pumps, valves, etc.) and diesel generators are provided in the plant control room.

Full flow functional tests of HPCF can be performed during normal plant operation or during plant shutdown by manual operation of HPCF from the control room. For testing during normal plant operation, the pump suction is transferred to the suppression pool, the pump is started, and the test discharge line to the suppression pool is opened. A reverse sequence is used to terminate this test. Upon receipt of an automatic initiation signal while in the flow testing mode, the system will automatically return to injection mode and flow will be directed to the reactor vessel.

6.3.2.2.2 Automatic Depressurization System (ADS)

If RCIC and HPCF cannot maintain the reactor water level, ADS, which is independent of any other ECCS, reduces the reactor pressure so that flow from RHR operating in the low pressure flooder mode enters the reactor vessel in time to cool the core and limit fuel cladding temperature.

ADS employs nuclear system pressure relief valves to relieve high pressure steam to the suppression pool. The design number, location, description, operational characteristics and evaluation of the pressure relief valves are discussed in detail in Subsection 5.2.2. The instrumentation and controls for ADS are discussed in Subsection 7.3.1.1.1.2.

6.3.2.2.3 Reactor Core Isolation Cooling System (RCIC)

RCIC consists of a steam-driven turbine which drives a pump assembly. The system also includes piping, valves, and instrumentation necessary to implement several flow paths. The RCIC steam supply line branches off one of the main steamlines (leaving the reactor pressure vessel) and goes to the RCIC turbine with drainage provision to the main condenser. The turbine exhausts to the suppression pool with vacuum breaking protection. Makeup water is supplied from the CST or the suppression pool with the preferred source being the CST. RCIC pump discharge lines include the main discharge line to the feedwater line, a test return line to the suppression pool, a minimum flow bypass line to the pool, and a cooling water supply line to auxiliary equipment. The piping configuration and instrumentation is shown in Figure 5.4-8, and the process flow diagram is shown in Figure 5.4-9.

Following the reactor scram, steam generation in the reactor core will continue (at a reduced rate) due to the core fission product decay heat. The main steam turbine bypass system will divert the steam to the main condenser, and the feedwater system will supply the makeup water required to maintain reactor vessel inventory.

In the event that the reactor vessel is isolated, and the feedwater supply is unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat. Upon reaching a predetermined low level, RCIC is initiated automatically. The turbine-driven pump will supply water from the suppression pool or from the CST to the reactor vessel. The turbine will be driven with a portion of the decay heat steam from the reactor vessel, and the turbine exhaust steam discharges to the suppression pool.

In the event that there is a LOCA, RCIC, in conjunction with the two HPCF loops, is designed to pump water into the vessel while it is fully pressurized. This combination of systems will provide adequate core cooling until vessel pressure drops to the point at which the Low Pressure Flooder (LPFL) Subsystems of RHR can be placed in operation.

During RCIC operation, the suppression pool acts as the heat sink for steam generated by reactor decay heat. This will result in a rise in pool water temperature. Heat exchangers in RHR are used to maintain pool water temperature within acceptable limits by cooling the pool water directly.

A design flow functional test of RCIC may be performed during normal plant operation by drawing suction from the suppression pool and discharging through a full flow test return line back to the suppression pool. The discharge valve to the vessel remains closed during the test, and reactor operation remains undisturbed. Should an initiation signal occur during test mode operation, flow will be automatically directed to the vessel. All components of RCIC are capable of individual functional testing during normal plant operation.

6.3.2.2.4 Residual Heat Removal System (RHR)

RHR is a closed system consisting of three independent pump loops which inject water into the vessel and/or remove heat from the reactor core or containment. Each of the pump loops contains the necessary piping, pumps, valves, and heat exchangers. The piping and instrumentation diagram and process flow diagram are shown in Figures 5.4-10 and 5.4-11, respectively. In the core cooling mode, each loop draws water from the suppression pool and injects the water into the vessel outside the core shroud (via the feedwater line on one loop and via the core cooling subsystem discharge return line on two loops). In the heat removal mode, pump suction may be taken either from the suppression pool or the reactor pressure vessel. With the pump suction being taken from the suppression pool, the pump discharge within these loops provides a flow path to the following points:

(1) Suppression pool

(2) Reactor pressure vessel (via feedwater on one loop and via the core cooling subsystem return lines on the other two loops)

(3) Wetwell and drywell spray spargers (on two loops only)

In the shutdown cooling mode, with the pump suction being taken from the reactor pressure vessel (via the shutdown cooling lines), the pump discharge within these loops provides a flow path back to the reactor vessel via the core cooling discharge return lines, and feedwater line, or to the upper reactor well via the fuel pool cooling system.

With the pump suction being taken from the skimmer surge tanks of the fuel pool cooling system, the pump discharge is returned to the fuel pool.

Each loop is in a single quadrant of the Reactor Building and receives its electric power from a bus separate from those serving the other two loops. Each bus is supplied from both onsite and offsite power sources.

For each loop, a full flow line is provided with discharge to the suppression pool to allow for full flow test of the system during normal operation. The valves in these lines are closed during normal operation.

For each loop, a minimum flow bypass line is also provided to return water to the suppression pool to prevent pump damage due to overheating when the injection valves on the main discharge lines are closed. The bypass line connects to the main discharge lines between the main pump and the discharge check valve. A motor-operated valve on the bypass line automatically closes when flow in the main discharge line is sufficient to provide the required pump cooling. A flow element in the main discharge line measures system flow rate during LOCA and test conditions and automatically controls the motor-operated valve on the bypass lines. The motor-operated valve does not receive automatic signals unless the associated pump indicates a high discharge pressure.

Each loop contains instruments necessary to maintain a ready condition, to evaluate loop performance, and to operate the minimum flow valve.

Each RHR pump discharge line is maintained in a filled condition to minimize the time lag between a starting signal and initiation of flow into the reactor vessel and to minimize momentum forces associated with accelerating fluid into an empty pipe.

Each division is provided with a keep-fill pump, which takes suction from the main pump’s suppression pool suction line. A check valve is located in the discharge line at an elevation lower than the suppression pool minimum water level line to prevent backflow from emptying the lines into the suppression pool.

Full flow functional tests of RHR can be performed during normal plant operation or during plant shutdown by manual operation of RHR from the control room. For plant testing during normal plant operation, the pump is started and the return line to the suppression pool is opened. A reverse sequence is used to terminate this test. Upon receipt of an automatic initiation signal while in the flow testing mode, the system is returned to automatic control.

6.3.2.2.5 ECCS Discharge Line Fill System

A requirement of the core cooling systems is that cooling water flow to the reactor vessel be initiated rapidly when the system is called on to perform its function. This quick-start system characteristic is provided by quick-opening valves, quick-start pumps, and standby AC power source. The lag between the signal to start the pump and the initiation of flow into the reactor vessel can be minimized by keeping the core cooling pump discharge lines full. Additionally, if these lines were empty when the systems were called for, the large momentum forces associated with accelerating fluid into an empty pipe could cause physical damage to the piping. Therefore, the ECCS discharge line fill system is designed to maintain the pump discharge lines in a filled condition.

Since the ECCS discharge lines are elevated above the suppression pool, check or stop-check valves are provided near the pumps to prevent back flow from emptying the lines into the suppression pool. Past experience has shown that these valves will leak slightly, producing a small back flow that will eventually empty the discharge piping. To ensure that this leakage from the discharge lines is replaced and the lines are always kept filled, a keep-fill pump is provided for each of the three RHR loops, as well as the RCIC loop. The power supply to these pumps is classified as essential when the main ECCS pumps are deactivated. Indication is provided in the control room as to whether these pumps are operating, and alarms indicate low discharge line level. The RCIC loop and the two HPCF loops are maintained full by connection to the makeup water (condensate).

6.3.2.3 Applicable Codes and Classifications

The applicable codes and classification of the ECCS are specified in Section 3.2. The edition of the codes applicable to the design are provided in Table 1.8-21. The piping and components of each ECCS within containment and out to and including the pressure retaining injection valve are Safety Class 1. The remaining piping and components are Safety Class 2, 3, or noncode, as indicated in Section 3.2 and on the individual system P&ID. The equipment and piping of the ECCS are designed to the requirements of Seismic Category I. This seismic designation applies to all structures and equipment essential to the core cooling function. IEEE codes applicable to the controls and power supply are specified in Section 7.1.

6.3.2.4 Materials Specifications and Compatibility

Materials specifications and compatibility for the ECCS are presented in Section 6.1. Nonmetallic materials such as lubricants, seals, packings, paints and primers, insulation, as well as metallic materials, etc., are selected as a result of an engineering review and evaluation for compatibility with other materials in the system and the surroundings with concern for chemical, radiolytic, mechanical and nuclear effects. Materials used are reviewed and evaluated with regard to radiolytic and pyrolytic decomposition and attendant effects on safe operation of the ECCS.

6.3.2.5 System Reliability

A single-failure analysis shows that no single failure prevents the starting of the ECCS, when required, or the delivery of coolant to the reactor vessel. No individual system of the ECCS is single-failure proof with the exception of the ADS; hence, it is expected that single failures will disable individual systems of the ECCS. The most severe effects of single failures with respect to loss of equipment occur if the LOCA occurs in combination with an ECCS pipe break coincident with a loss of offsite power. The consequences of the most severe single failures are shown in Table 6.3-3.

6.3.2.6 Protection Provisions

Protection provisions are included in the design of the ECCS. Protection is afforded against missiles, pipe whip and flooding. Also accounted for in the design are thermal stresses, loadings from a LOCA, and seismic effects.

The ECCS piping and components located outside the drywell are protected from internally and externally generated missiles by the reinforced concrete structure of the ECCS pump rooms. The ECCS pump rooms are provided with watertight doors to protect the equipment against flooding.

The ECCS is protected against the effects of pipe whip, which might result from piping failures up to and including the design basis event LOCA. This protection is provided by separation, pipe whip restraints, and energy absorbing materials. One of these three methods is applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level (see Section 3.6 for criteria on pipe whip).

The component supports which protect against damage from movement and from seismic events are discussed in Subsection 5.4.14. The methods used to provide assurance that thermal stresses do not cause damage to the ECCS are described in Subsection 3.9.3.

6.3.2.7 Provisions for Performance Testing

Periodic system and component testing provisions for the ECCS are described in Subsection 6.3.2.2 as part of the individual system description.

6.3.2.8 Manual Actions

The ECCS is actuated automatically and requires no operator action during the first 30 minutes following the accident, although operator action is not prevented. During the long-term cooling period (assume no cooling for the first 30 minutes) following a design basis accident, containment cooling occurs as a normal consequence of RHR operation in LPFL mode, because the RHR heat exchangers are in series with the pumps. Although not prevented from doing so earlier, the operator is not required to select another RHR mode, such as suppression pool cooling, until after the 30 minutes.

The operator has multiple instrumentation available in the control room to assist him in assessing the post-LOCA conditions. This instrumentation provides reactor vessel pressures, water levels, containment pressure, temperature and radiation levels, as well as indicating the operation of the ECCS. ECCS flow indication is the primary parameter available to assess proper operation of the system. Other indications, such as position of valves, status of circuit breakers, and essential power bus voltage, are available to assist him in determining system operating status. The electrical and instrumentation complement to the ECCS is discussed in detail in Section 7.3. Other available instrumentation is listed in the P&IDs for the individual systems.

6.3.3 ECCS Performance Evaluation

Performance of the ECCS is determined by evaluating the system response to an instantaneous break of a pipe. This evaluation is performed using models either approved by the USNRC or which have met the requirements stated in 10CFR50.46 and 10CFR50 Appendix K.

The analyses included in this subsection demonstrate the ECCS performance for the entire spectrum of postulated break sizes. The analyses are based upon the core loading shown in Figure 6.3-2 and were performed with the USNRC-approved LAMB/TASC and SAFER/GESTR models. The exposure-dependent MAPLHGR, peak cladding temperature, and oxidation fraction for each bundle design of Lungmen NPS, based on the limiting break size, will be supplied with the FSAR.

The accidents, as listed in Chapter 15, for which ECCS operation is required are:

Subsection	Title
15.2.8	Feedwater Line Break
15.6.4	Steam System Piping Failures Outside Containment
15.6.5	Loss-of-Coolant Accidents Inside Containment

6.3.3.1 ECCS Bases for Technical Specifications

The MAPLHGRs calculated in this performance analysis provide the basis for the Chapter 16 Technical Specifications designed to ensure conformance with the acceptance criteria of 10CFR50.46. Minimum ECCS functional requirements are specified in Subsections 6.3.3.4 and 6.3.3.5, and testing requirements are discussed in Subsection 6.3.4. Limits on minimum suppression pool water level are discussed in Section 6.2.

6.3.3.2 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria extracted from 10CFR50.46 are listed, and, for each criterion, applicable parts of Subsection 6.3.3 (where conformance is demonstrated) are indicated.

Criterion 1: Peak Cladding Temperature (PCT)

"The calculated maximum fuel element cladding temperature shall not exceed 1204°C (2200°F)." Conformance to Criterion 1 (≤1204°C) is shown for the system response analyses in Subsections 6.3.3.7.3 (Break Spectrum), 6.3.3.7.4 (Large Breaks), 6.3.3.7.5 (Intermediate Breaks), 6.3.3.7.6 (Small Breaks), 6.3.3.7.7 (Outside Containment Breaks), 6.3.3.7.8 (upper 95% Probability PCT) and, specifically, in Table 6.3-4 (Summary of LOCA Analysis Results). Conformance will be assured for the limiting break.

Criterion 2: Maximum Cladding Oxidation

"The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation." Conformance to Criterion 2 is shown in Table 6.3-4 (Summary of LOCA Analysis Results) for the system response analysis. This limit will be assured for the limiting break.

Criterion 3: Maximum Hydrogen Generation

"The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react." Conformance to Criterion 3 is shown in Table 6.3-4 (Summary of LOCA Analysis Results) for the system analysis.

Criterion 4: Coolable Geometry

"Calculated changes in core geometry shall be such that the core remains amenable to cooling." As described in Reference 6.3-1, Section III.A, conformance to Criterion 4 is demonstrated by conformance to Criteria 1 and 2.

Criterion 5: Long-Term Cooling

"After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." Conformance to Criterion 5 is demonstrated generically for GE BWRs in Reference 6.3-1, Section III.A. Briefly summarized, for any LOCA, the water level can be restored to a level above the top of the core and maintained there indefinitely.

6.3.3.3 Single-Failure Considerations

The functional consequences of potential operator errors and single failures (including those which might cause any manually controlled electrically operated valve in the ECCS to move to a position which could adversely affect the ECCS) and the potential for submergence of valve motors in the ECCS are discussed in Subsection 6.3.2. There it was shown that all potential single failures are no more severe than one of the single failures identified in Table 6.3-3.

It is therefore only necessary to consider each of these single failures in the ECCS performance analyses. The worst failure for any LOCA event is the failure of one of the diesel generators which provide electrical power to one HPCF and one RHR/LPFL. This failure results in the elimination of the greatest amount of flooding capability at both high and low reactor pressures.

6.3.3.4 System Performance During the Accident

In general, the system response to an accident can be described as:

(1) Receiving an initiation signal

(2) A small lag time (to open all valves and have the pumps up to rated speed)

(3) The ECCS flow entering the vessel

Key ECCS actuation setpoints and time delays for all the ECCS systems are provided in Table 6.3-1. The minimization of the delay from the receipt of signal until the ECCS pumps have reached rated speed is limited by the physical constraints on accelerating the diesel-generators and pumps. The delay time due to valve motion in the case of the high pressure system provides a suitably conservative allowance for valves available for this application. In the case of the low pressure system, the time delay for valve motion is such that the pumps are at rated speed prior to the time the vessel pressure reaches the pump shutoff pressure.

The ADS actuation logic includes a 29-second delay timer to confirm the presence of Low Water Level 1 (LWL 1) initiation signal. This timer is initiated upon receipt of a high drywell pressure signal (which is sealed-in) and a LWL 1 signal. The timer setting is consistent with the startup time of the ECCS which also must be running before ADS operation can occur. Once the ADS timer is initiated, it is automatically reset if the reactor water level is restored above the LWL 1 setpoint before ADS operation occurs. For defense-in-depth protection against inventory decreasing events where a high drywell pressure is not present, the ADS actuation logic also includes an 8-minute high drywell pressure bypass timer. This timer is initiated upon receipt of a LWL 1 signal and is automatically reset if the reactor water level is restored above the LWL 1. After this timer runs out, the need for a high drywell pressure signal to initiate the ADS 29-second delay timer is bypassed (i.e., the 29-second delay timer would require only a LWL 1 signal to initiate). The ADS control system also provides the operator with an ADS inhibit switch which can be used to prevent automatic ADS operation as covered by the engineering operating procedures (refer to Subsection 7.3).

The flow delivery rates analyzed in Subsection 6.3.3 can be determined from the vessel pressure versus system flow curves in Figures 6.3-4, 6.3-5 and 6.3-6 and the pressure versus time plots discussed in Subsection 6.3.3.7. Simplified piping and instrumentation and process flow diagrams for the ECCS are referenced in Subsection 6.3.2. The operational sequence of ECCS for the limiting case is shown in Table 6.3-2.

Operator action is not required, except as a monitoring function, during the short-term cooling period following the LOCA. During the long-term cooling period, the operator may need to take action as specified in Subsection 6.2.2.2 to place the containment cooling system into operation for some LOCA events.

6.3.3.5 Use of Dual Function Components for ECCS

With the exception of the RHR, the ECCS systems are designed to accomplish only one function: to cool the reactor core following a loss of reactor coolant. To this extent, components or portions of these systems (except for pressure relief) are not required for operation of other systems which have emergency core cooling functions, or vice versa. Because either the ADS initiating signal or the overpressure signal opens the safety-relief valve, no conflict exists.

The LPFL mode of RHR is configured from the RHR pumps and a portion of the RHR valves and piping. When the reactor water level is low, the LPFL mode (line up) has priority through the valve control logic over the other RHR mode of operation for containment cooling. Immediately following a LOCA, RHR is directed to the LPFL mode. When the RHR shutdown cooling mode is utilized, the transfer to the LPFL mode must be remote manually initiated.

6.3.3.6 Limits on ECCS Parameters

Limits on ECCS parameters are given in the sections and tables referenced in Subsections 6.3.3.1 and 6.3.3.7.1. Any number of components in any given system may be out of service, up to the entire system. The maximum allowable out-of-service time is a function of the level of redundancy and the specified test intervals.

6.3.3.7 ECCS Analyses for LOCA

The core design for Lungmen NPS will be based on GE 12 type fuel design which is different than the fuel design type used in the calculations presented in this section. ECCS performance analyses for core design based on GE 12 type fuel design will be performed and supplied with the FSAR.

Therefore, it is to be recognized that the ECCS analysis results presented and discussed in this section are based on previous GE 6/7 type fuel design data. The calculated PCT results, summarized in this section, show a large margin to the licensing PCT acceptance criteria. It is anticipated that ECCS analyses based on GE 12 type fuel design will exhibit the same trend as that exhibited by the results presented in this section, and the calculated results will show margins to the licensing acceptance criteria comparable to those shown with GE 6/7 type fuel design.

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

The methods used in the ECCS analyses have been approved by the USNRC or meet the requirement stated in 10CFR50.46. For the system response analysis, the LAMB/TASC and SAFER/GESTR models approved by the USNRC were used. The significant input variables used for the response analyses are listed in Table 6.3-1 and Figure 6.3-11.

6.3.3.7.2 Accident Description

The operation sequence of events for the limiting case is shown in Table 6.3-2.

6.3.3.7.3 Break Spectrum Calculations

A complete spectrum of postulated break sizes and their locations were evaluated to demonstrate ECCS performance. For ease of reference, a summary of figures presented in Subsection 6.3.3.7 is shown in Table 6.3-5.

A summary of results of the break spectrum calculations is shown in tabular form in Table 6.3-4 and graphically in Figure 6.3-10. Conformance to the acceptance criteria (PCT≤ 1204°C, local oxidation ≤ 17%, and core-wide metal-water reaction ≤ 1%) is demonstrated for the core loading in Figure 6.3-2. Results for the limiting break for each bundle design of Lungmen NPS will be provided for information to ROC-AEC. Details of calculations for specific breaks are included in subsequent paragraphs.

6.3.3.7.4 Large Line Breaks Inside Containment

Since the Lungmen NPS design has no recirculation lines, the maximum steamline break (985 cm^2), maximum feedwater line break (839 cm^2), and the maximum RHR shutdown suction line break (792 cm^2) become the large break cases. Important output variables from the sensitivity study of these events are shown in Figures 6.3-12 through 6.3-36.

These variables are:

(1) Core flow as a function of time

(2) Minimum critical power ratio as a function of time

(3) Water level in the fuel channels as a function of time

(4) Water level inside the shroud as a function of time

(5) Water level outside the shroud as a function of time

(6) Vessel pressure as a function of time

(7) Flows out of the vessel as a function of time

(8) Flows into the vessel as a function of time

(9) Peak cladding temperature as a function of time

A conservative assumption made in the analysis is that all offsite AC power is lost simultaneously with the initiation of the LOCA. As a further conservatism, all reactor internal pumps were assumed to trip at the start of LOCA event even though this, in itself, is considered to be an accident (Subsection 15.3.1). The resulting rapid core flow coastdown produces a calculated departure from nucleate boiling in the hot bundles within the first few seconds of the transient.

LOCA analyses using break areas less than the maximum values noted above were also considered for the steamline, feedwater line, and RHR shutdown suction line locations. The cases analyzed are indicated on the break spectrum plot (Figure 6.3-10). In general, the largest break at each location is the worst in terms of minimum transient water level in the downcomer.

6.3.3.7.5 Intermediate Line Breaks Inside Containment

For this case, the maximum RHR/LPFL injection line break (205 cm^2) was analyzed. Since the bottom head drain line ties into the RHR shutdown suction line, the total break flow for the maximum RHR shutdown suction line break includes flow from the vessel through RHR shutdown suction vessel nozzle, as well as through the bottom head drain line. Important variables from this analysis are shown in Figures 6.3-37 through 6.3-43.

6.3.3.7.6 Small Line Breaks Inside Containment

For these cases, the maximum high pressure core flooder line break (92 cm^2) and the maximum bottom head drain line break (20.3 cm^2) based on a 5.08 cm penetration in the vessel bottom head were analyzed. Since the bottom head drain line ties into the RHR shutdown suction line, the total break flow for the maximum bottom head drain line break includes flow from the vessel through the bottom head drain line penetration as well as through the RHR shutdown suction line. Important variables from these analyses are shown in Figures 6.3-44 through 6.3-59.

A break in a reactor internal pump would involve either the welds or the casing. If the weld from the pump casing to the RPV nozzle breaks, the stretch tube will prevent the pump casing from moving. The stretch tube clamps the diffuser to the pump casing, where its nut seats. In case the pump casing and the stretch tube break, the pump and motor will move downward until stopped by the casing vertical restraints. In either case, the break flow would be much less than the drain line break case. Therefore, the drainline break analysis is also bounding for any credible break within the reactor internal pump recirculation system and its associated motor housing and cover.

As expected, the high pressure core flooder line break is the worst break location in terms of minimum transient water level in the downcomer. In elevation it is the lowest break on the vessel except for the drainline break. Furthermore, the worst break/failure combination leaves the fewest number of ECC systems remaining and no high pressure core flooder systems. LOCA analyses using break areas less than the maximum values were also considered. The cases analyzed are indicated on the break spectrum plot (refer to Figure 6.3-10). From these results, it is clear that the overall most limiting break in terms of minimum transient water level in the downcomer, is the maximum high pressure core flooder line break case.

6.3.3.7.7 Line Breaks Outside Containment

This group of breaks is characterized by a rapid isolation of the break. Since a main steamline break outside the containment produces more vessel inventory loss before isolation than other breaks in this category, the results of this case are bounding for all breaks in this group. Important variables from these analyses are shown in Figures 6.3-60 through 6.3-66.

As discussed in Subsection 6.3.3.7.4, the trip of all reactor internal pumps at the start of the LOCA produces a calculated departure from nucleate boiling for all LOCA events. Furthermore, the high void content in the bundles following a large steamline break produces the earliest times of loss of nucleate boiling for any LOCA event. Thus, the summary of results in Table 6.3-4 shows that, though the PCTs for all break locations are similar, the steamline breaks result in higher calculated PCTs and the outside steamline break is the overall most limiting case in terms of the highest calculated PCT. As shown in Table 6.3-4, the outside steamline break results in a calculated PCT value slightly (1°C) higher than that resulting from the inside steamline break. This observed difference in the calculated PCT value can be explained qualitatively as follows.

The PCT value for both the inside and the outside steamline breaks occurs shortly after all MSIVs are fully closed. Up to the time when all MSIVs are closed, the vessel blowdown flow and the resulting depressurization of the vessel are essentially the same for both the breaks. During this period, the total vessel blowdown flow is comprised of flow from both sides of the break - flow from the vessel side, and flow from the other side of the break from the remaining three unbroken steam lines. Thus the total break flow is based on the flow area of the four steamline flow limiters. After the MSIVs are fully closed, the total vessel blowdown (from both sides of the break) for the outside steamline break drops to zero, whereas for the inside steamline break the flow from the other side of the break drops to zero, but the flow from the vessel side continues. As a result, for the outside steamline break the vessel pressure starts rising due to the continuing steam generation in the vessel from the core decay heat, and for the inside steamline break vessel depressurization continues. This expected rise in vessel pressure after closure of the MSIVs for the outside steamline break would result in collapse of some core voids which, in turn, is believed to result in a slightly higher PCT value than that for the inside steamline break.

Results of the analysis for this break case will be provided for each future Lungmen NPS bundle design for information to ROC-AEC.

6.3.3.7.8 Bounding Peak Cladding Temperature Calculations

Consistent with the SAFER application methodology in Reference 6.3-1, the Appendix K peak cladding temperatures calculated in the previous sections must be compared to a statistically calculated 95% probability value. Table 6.3-6 presents the significant plant variables which were considered in the determination of the 95% probability PCT or the sensitivity study. Again, since the LOCA results have a large margin to the acceptance criteria, a conservative PCT calculation was performed which bounds the 95% probability PCT. This bounding PCT was calculated by varying all plant variables in the conservative direction, representing upper 95% or higher probability values. The results of this calculation for the limiting case are given in Figures 6.3-67 through 6.3-75 and Table 6.3-4. Since the calculated results have large margins to the l0CFR50.46 licensing acceptance criteria, the licensing PCT (621°C) is well below the required 1204°C PCT limit.

6.3.3.8 LOCA Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of Subsection 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner, given operation at or below the MAPLHGRs in Table 6.3-7. For a core design other than the core loading shown in Figure 6.3-2, MAPLHGRs for each future Lungmen NPS bundle design will be calculated and provided to ROC-AEC for information.

6.3.3.9 LOCA Analyses to Support ECCS Technical Specifications for Allowable Outage Times

To be provided with the FSAR.

6.3.3.10 Severe Accident Considerations

In the unlikely event that the ECCS does not prevent core damage, its operation (recovery if necessary) can be beneficial in mitigating the consequences of core damage. The analysis of core damage events was performed using best-estimate methods rather than design basis codes such as SAFER/GESTR.

The primary injection path for RHR during a severe accident is into the vessel via the LPFL header. The operator actions will conform with the guidance provided by the Emergency Operating Procedures. For injection to occur, the RPV must be at low pressure.

If the LPFL mode is not initiated in time to prevent core damage, it can act to mitigate the accident by enhancing cooling and preventing radioactive heating from the core debris. If injection is initiated prior to vessel failure, melt progression may be arrested in-vessel. However, if vessel failure occurs, debris will relocate from the vessel breach into the lower drywell. Water flowing through the vessel into the lower drywell will cover the core debris and enhance debris cooling.

6.3.4 Tests and Inspections

6.3.4.1 ECCS Performance Tests

All systems of the ECCS are tested for their operational ECCS function during the preoperational and/or startup test program. Each component is tested for power source, range, direction of rotation, setpoint, limit switch setting, torque switch setting, etc. Each pump is tested for flow capacity for comparison with vendor data. (This test is also used to verify flow measuring capability). The flow tests involve the same suction and discharge source (i.e., suppression pool).

All logic elements are tested individually and then as a system to verify complete system response to emergency signals including the ability of valves to revert to the ECCS alignment from other positions.

Finally, the entire system is tested for response time and flow capacity taking suction from its normal source and delivering flow into the reactor vessel. This last series of tests is performed with power supplied from both offsite power and onsite emergency power.

See Chapter 14 for a description and discussion of preoperational testing for these systems. During every refueling, a test, in which each ECCS subsystem is actuated through the emergency operating sequence, will be performed in accordance with Technical Specifications.

6.3.4.2 Reliability Tests and Inspections

The average reliability of a standby (nonoperating) safety system is a function of the duration of the interval between periodic functional tests. The factors considered in determining the periodic test interval of the ECCS are:

(1) The desired system availability (average reliability)

(2) The number of redundant functional system success paths

(3) The failure rates of the individual components in the system

(4) The schedule of periodic tests (simultaneous versus uniformly staggered versus randomly staggered)

All of the active components of HPCF, ADS, RHR and RCICs are designed so that they may be tested during normal plant operation. Full flow test capability is provided by a test line back to the suction source. The full flow test is used to verify the capacity of each ECCS pump loop while the plant remains undisturbed in the power generation mode. In addition, each individual valve may be tested during normal plant operation.

All of the active components of ADS, except the SRVs and their associated solenoid valves, are designed so that they may be tested during normal plant operation. The SRVs and associated solenoid valves are all tested during plant initial power ascension per Regulatory Guide 1.68, Appendix A. SRVs are bench tested to establish lift settings.

Testing of the initiating instrumentation and controls portion of the ECCS is discussed in Subsection 7.3.1. The emergency power system, which supplies electrical power to the ECCS divisions in the event that offsite power is unavailable, is tested as described in Subsection 8.3.1. The frequency of testing is specified in Chapter 16 (Technical Specifications). Visual inspections of all the ECCS components located outside the drywell can be made at any time during power operation. Components inside the drywell can be visually inspected only during periods of access to the drywell. When the reactor vessel is open, the spargers and other internals can be inspected.

6.3.4.2.1 HPCF Testing

HPCF can be tested at full flow with suppression pool water at any time during plant operation except when a system initiation signal is present. If an initiation signal occurs while HPCF is being tested, the system returns automatically to the operating mode. The motor-operated valve in the suction line from the condensate storage tank is interlocked closed when the suction valve from the suppression pool is open.

A design flow functional test of HPCF over the operating pressure and flow range is performed by pumping water from the suppression pool through the full flow test return line and back to the suppression pool.

The suction valve from the condensate storage tank and the discharge valve to the reactor remain closed. These two valves are tested separately to ensure their operability.

The HPCF test conditions are tabulated on the HPCF process flow diagram (Figure 6.3-1).

6.3.4.2.2 ADS Testing

An ADS logic system functional test and simulated automatic operation of all ADS logic channels are to be performed. Instrumentation channels are demonstrated operable by the performance of a channel functional test and a trip unit calibration, at least once per month and a transmitter calibration. The frequency of these tests is specified in Chapter 16 (Technical Specifications).

All SRVs, which include those used for ADS, are bench tested to establish lift settings in compliance with ASME Code Section XI.

6.3.4.2.3 RHR Testing

The RHR pump and valves are tested periodically during reactor operation. With the injection valves closed and the return line open to the suppression pool, full flowing pump capability is demonstrated. The injection valve and the check valve are tested in a manner similar to that used for the HPCF valves. The system test conditions during reactor operation are shown on the RHR process flow diagram (Figure 5.4-11).

6.3.4.2.4 RCIC Testing

The RCIC loop can be tested during reactor operation. To test the RCIC pump at rated flow, the test bypass line valve to the suppression pool and the pump suction valve from the suppression pool are opened and the pump is started using the turbine controls in the control room. Correct operation is determined by observing the instruments in the control room.

If an initiation signal occurs during the test, RCIC returns to the operating mode. The valve in the test bypass line is closed automatically and the RCIC pump discharge valve is opened to assure flow is correctly routed to the vessel.

6.3.5 Instrumentation Requirements

Design details including redundancy and logic of the ECCS instrumentation are discussed in Section 7.3.

All instrumentation required for automatic and manual initiation of HPCF, RCIC, RHR and ADS is discussed in Subsection 7.3.1, and is designed to meet the requirements of IEEE-279 and other applicable regulatory requirements. HPCF, RCIC, RHR and ADS can be manually initiated from the control room.

RCIC, HPCF, and RHR are automatically initiated on low reactor water level or high drywell pressure. ADS is automatically actuated by sensed variables for reactor vessel low water level and drywell high pressure plus indication that at least one RHR or HPCF pump is operating. HPCF, RCIC, and RHR automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. The RHR LPFL mode injection into the RPV begins when reactor vessel pressure is less than 1.55 MPa above the drywell pressure.

HPCF injection begins as soon as the HPCF pump is up to speed and the injection valve is open, since HPCF is capable of injecting water into the RPV over a pressure difference range from 0.69 to 8.12 MPaD (pressure difference between the vessel and the air space of the compartment containing the water source for the pump).

6.3.6 Reference

6.3-1 *General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K*, NEDE-20566-P-A, September 1986, (Proprietary).

Table 6.3-1 Significant Input Variables Used in the Loss-of-Coolant Accident Analysis

Variable	Units	Value*
A. Plant Parameters*		
Core Thermal Power	MWt	4005
Vessel Steam Output	kg/h	7.82×10^6
Corresponding Percentage of Rated Steam Flow	%	102.4
Vessel Steam Dome Pressure	MPaA	7.28
B. Emergency Core Cooling Systems Parameters*		
B.1 Low Pressure Flooder System*		
Vessel Pressure at which Flow may Commence	MPaD	1.55
Minimum Rated Flow per system at Vessel Pressure	m^3/h MPaD (vessel to drywell)	954 0.275
Initiating signals Low Water Level (Level 1) or High Drywell Pressure	 cm above TAF MPaG	 ≤15.3 ≥0.014
Maximum Allowable Time Delay from Initiating Signal to Pumps at Rated Speed	s	29.0
Maximum Allowable Time Delay from Low Pressure Permissive Signal to Injection Valve Fully Open	s	36.0
B.2 Reactor Core Isolation Cooling System*		
Vessel Pressure at which flow may commence	MPaD	8.12
Minimum Rated Flow at Vessel Pressure	m^3/h MPaD (vessel to the air space of the compartment containing the water source for the pump suction)	182 1.035 to 8.12
Initiating signals Low Water Level (Level 2) or High Drywell Pressure	 cm above TAF MPaG	 ≤243.4 ≥0.014
Maximum Allowable Time Delay from Initiating Signal to Rated Flow Available and Injection Valve Fully Open	s	29.0

Table 6.3-1 Significant Input Variables Used in the Loss-of-Coolant Accident Analysis (Continued)

Variable	Units	Value*
B.3 High Pressure Core Flooder System*		
Vessel Pressure at which Flow May Commence	MPaD	8.12
Minimum Rated Flow per System Available at Vessel Pressure	m^3/h MPaD (vessel to the air space of the compartment containing the water source for the pump suction)	182 to 727 8.12 to 0.69
Initiating Signals		
Low Water Level (Level 1.5) or High Drywell Pressure	cm above TAF MPaG	$\leq$98.7 $\geq$0.014
Maximum Allowed Delay Time (Following the Receipt of Initiating Signal, 20 (closure of DG breaker) plus 16 (injection valve stroke time))	s	36.0
B.4 Automatic Depressurization System*		
Total Number of Relief Valves with ADS Function		8
Total Minimum Flow Capacity	kg/h	2.903×10^6
At Vessel Pressure	MPaG	7.76
Initiating Signals Low Water Level (Level 1) and High Drywell Pressure or High Drywell Pressure Bypass Timer Timed Out	 cm above TAF MPaG s	 $\leq$15.3 $\geq$0.014 $\leq$480
Delay Time from All Initiating Signals Completed to the Time Valves are Open	s	$\leq$29
C. Fuel Parameters*†		
Fuel Type	-------	GE 6/7
Fuel Bundle Geometry	-------	8x8
Lattice	-------	C
Number of Fueled Rods per Bundle	-------	62
Peak Technical Specification Linear Heat Generation Rate	kW/m	44.0
Initial Minimum Critical Power Ratio	-------	1.13

Table 6.3-1 Significant Input Variables Used in the Loss-of-Coolant Accident Analysis (Continued)

Variable	Units	Value*
Design Axial Peaking Factor	-------	1.40

* Estimate values. Final design values will be provided with the FSAR.

† The system response analysis is based upon the GE 6/7 type fuel and core loading in Figure 6.3-2. Fuel parameters based on the GE 12 type fuel and the associated results will be provided with the FSAR.

Table 6.3-2 Operational Sequence of Emergency Core Cooling System Maximum Core Flooder Line Break

Time (s)*	Events
0	Design basis LOCA assumed to start; normal auxiliary power assumed to be lost.†
~5	Reactor Low Water Level 3 is reached. Reactor scram occurs.
~10	Drywell high pressure is reached. All diesel-generators, RCIC, HPCF, RHR/LPFL signaled to start. ‡
~18	Reactor Low Water Level 2 is reached. RCIC System receives second signal to start.
~48	RCIC injection valve open and pump at design flow which completes RCIC startup.
~65	Reactor Low Water Level 1.5 is reached. All diesel-generators and HPCF receive second signal to start. Main steam isolation valves signaled to close.
~86	All diesel-generators ready to load; RHR/LPFL and HPCF loading sequence begins.
~102	HPCF injection valves open and pumps at design flow, which completes HPCF startup.ƒ
~118	Reactor Low Water Level 1 is reached. RHR/LPFL receives second signal to start. ADS delay timer initiated.
~148	ADS delay timer timed out. ADS valves actuated.
~344	Vessel pressure decreases below shutoff head of RHR/LPFL. RHR/LPFL injection valves open and flow into vessel begins.
See Figure 6.3-46	Core effectively reflooded assuming worst single failure; heatup terminated.

* Estimate values. Final design values will be provided with the FSAR.

† All ECCS equipment is assumed to function as designed. Performance analysis calculations consider the effects of single equipment failures (Subsection 6.3.3.3). Note: Figure 6.3-46 is based on worst failure.

‡ For the LOCA analysis, the ECCS initiation on high drywell pressure is not considered.

ƒ Analysis does not take credit for HPCF, a single failure. See Table 6.3-4, Systems available for High Pressure Core Flooder break.

Table 6.3-3 Single Failure Evaluation*

Assumed Failure	Systems Remaining†
Emergency Diesel Generator A	All ADS, RCIC, 2 HPCF, 2 RHR/LPFL
Emergency Diesel Generator B or C	All ADS, RCIC, 1 HPCF, 2 RHR/LPFL
RCIC Injection Valve	All ADS, 2 HPCF, 3 RHR/LPFL
One ADS Valve	All ADS minus one, RCIC, 2 HPCF, 3 RHR/LPFL

* Single, active failures are considered in the ECCS performance evaluation. Other postulated failures are not specifically considered because they all result in at least as much ECCS capacity as one of the above designed failures.

† Systems remaining, as identified in this table, are applicable to all non-ECCS line breaks. For the LOCA from an ECCS line break, the systems remaining are those listed, less the ECCS system in which the break is assumed.

Table 6.3-4 Summary of Results of LOCA Analysis *

Break Location	Break Size† (cm^2)	Systems Available	PCT* (°C)	Maximum Local Oxidation*
Based on Appendix K evaluation models:				
Steamline Inside Containment	985‡	1HPCF + RCIC +2 RHR/LPFL + 8 ADS	620	0.03%
Feedwater Line	839	1 HPCF + 2 RHR/LPFL + 8 ADS	542	0.03%
RHR Shutdown Cooling Suction Line	792	1 HPCF + RCIC + 2 RHR/LPFL+ 8 ADS	542	0.03%
RHR/LPFL Injection Line	205	1 HPCF + RCIC + 1RHR/LPFL + 8 ADS	542	0.03%
High Pressure Core Flooder	92	RCIC+2RHR/ LPFL + 8 ADS	542	0.03%
Bottom Head Drain Line	20.3	1HPCF + RCIC + 2 RHR/LPFL + 8 ADS	542	0.03%
Steamline Outside Containment	3939ƒ	1 HPCF + RCIC + 2 RHR/LPFL + 8 ADS	621	0.03%
Based on upper bound values:				
Steamline Outside Containment	3939	1 HPCF + RCIC + 2 RHR/LPFL + 8 ADS	619	0.03%

* These results are estimate values. Final design values will be provided with the FSAR.

† The most severe design basis LOCA calculations (Subsection 6.3.3.7.8) involve use of bounding worst-case values for key plant parameters - including an arbitrary 20% increase in the break flow rate. Even with these bounding assumptions, the LOCA analyses demonstrate that the design still retains large margins between predicted peak fuel clad temperatures and the criteria of 10 CFR 50, Appendix K.

Tolerances associated with fabrication and installation may result as-built break areas that could be 5% greater than these values. Based on the above conservatisms in the LOCA analyses, these as-built variations would not invalidate the plant safety analysis presented in Chapter 6 and Chapter 15.

‡ Combined flow area of the four steamline flow limiters up to MSIVs closure, thereafter flow area of one flow limiter.

ƒ Combined flow area of the four steamline flow limiters.

Note: The core-wide metal-water reaction for this analysis has been calculated using Method 1 described in Reference 6.3-1. This results in a core-wide metal-water reaction of 0.03%.

Table 6.3-5 Key to Figures

Appendix K Evaluation Models								
	Main Steamline Inside Contain-ment	**Feedwater Line**	**RHR Suction Line**	**LPFL Injection Line**	**Core Flood Line**	**Bottom Drain Line**	**Main Steamline Outside Contain-ment**	**Bounding Values Main Steamline Outside Contain-ment**
Core Flow	6.3-12	6.3-21	6.3-21	6.3-21	6.3-44	6.3-21	6.3-21	6.3-67
Minimum Critical Power Ratio	6.3-13	6.3-22	6.3-22	6.3-22	6.3-45	6.3-22	6.3-22	6.3-68
Water Level in Fuel Channel	6.3-14	6.3-23	6.3-30	6.3-37	6.3-46	6.3-53	6.3-60	6.3-69
Water Level Inside Shroud	6.3-15	6.3-24	6.3-31	6.3-38	6.3-47	6.3-54	6.3-61	6.3-70
Water Level Outside Shroud	6.3-16	6.3-25	6.3-32	6.3-39	6.3-48	6.3-55	6.3-62	6.3-71
Vessel Pressure	6.3-17	6.3-26	6.3-33	6.3-40	6.3-49	6.3-56	6.3-63	6.3-72
Flow out of Vessel	6.3-18	6.3-27	6.3-34	6.3-41	6.3-50	6.3-57	6.3-64	6.3-73
Flow into Vessel	6.3-19	6.3-28	6.3-35	6.3-42	6.3-51	6.3-58	6.3-65	6.3-74
Peak Cladding Temperature	6.3-20	6.3-29	6.3-36	6.3-43	6.3-52	6.3-59	6.3-66	6.3-75

Table 6.3-6 Plant Variables with Nominal and Sensitivity Study Values

(Proprietary information provided in a separate proprietary volume.)

Table 6.3-7 MAPLHGR Versus Exposure*†

	Exposure (MW·d/t)	MAPLHGR‡ kW/m
High Enrichment		
	200	39.4
	1,000	40.0
	5,000	41.7
	10,000	42.3
	15,000	42.3
	20,000	41.3
	25,000	38.4
	30,000	35.4
Medium Enrichment		
	200	39.0
	1,000	39.4
	5,000	39.7
	10,000	40.0
	15,000	40.4
	20,000	39.7
	25,000	38.1
	30,000	37.1
Low Enrichment		
	200	37.7
	1,000	37.4
	5,000	37.1
	10,000	37.7
	15,000	37.7
	20,000	36.1
	25,000	34.1

* For the core loading in Figure 6.3-2

† These tabulated values are estimate values. Final design values will be provided with the FSAR.

‡ These values are limited by the peak LHGR of 44.0 kw/m and not by ECCS performance

Table 6.3-8 Design Parameters* for HPCF (E22) Components

(1) Main Pumps (P-0001 B/C)*	
Number of Pumps	2
Pump Type	Centrifugal
Drive Unit	Constant speed induction motor
Flow Rate†	182 m^3/h @ 8.12 MPaD
	727 m^3/h @ 0.69 MPaD
Developed Head	890m @ 8.12 MPaD
	190m @ 0.69 MPaD
Maximum Runout Flow	890 m^3/h @ 0.0 MPaD
Minimum Bypass Flow	73 m^3/h
Water Temperature Range	10° to 100°C
NPSH Required	2.2m
(2) Strainer (STR-0002 B/C)*	
Location	Suppression Pool
Size	As required for insulation debris per Appendix 6C.
(3) Restricting Orifice (ORF-0001 B/C)*	
Location	Pump discharge line
Size	Limit pump flow to values specified
(4) Condensate Storage Tank *	570 m^3 reserve storage for HPCF and RCIC Systems combined
(5) Flow Elements (FE-0007B/C)*	
Location	Pump discharge—downstream of minimum flow bypass line
Head Loss	6.1m w.g. maximum @ 727 m^3/h
Accuracy	±2.5% combined element, transmitter and indicator at maximum rated
(6) Core Flooder Sparger*	
Flow Rate	727 m^3/h minimum @ 0.69 MPaD
Pressure Drop	50m w.g. maximum @ 727 m^3/h
(7) Piping and Valves*	
Design Pressures	0.8 MPaG—suction and discharge connected to suppression pool
	2.82 MPaG—pump suction
	10.8 MPaG—pump discharge

Table 6.3-8 Design Parameters* for HPCF (E22) Components (Continued)

Design Temperatures	66°C – condensate tank suction 180°C – pump suction and discharge 302°C – discharge to vessel
(8) Valve Operation*	
Pump Suction Valve, Suppression Pool **(MBV-0007 B/C)**	Normally closed, opens on low water level in condensate storage tank or high water level in the suppression pool.
Pump Suction Check Valve, Suppression Pool **(UV-0008 B/C)**	Prevents backflow into suppression pool.
Pump Suction Valve, Condensate Tank **(MBV-0001B/C)**	Normally open, closes when MBV-0007 B/C is fully open.
Pump Suction Check Valve, Condensate Tank **(UV-0002B/C)**	Prevents backflow into condensate storage tank.
Pump Discharge Valve, Reactor Injection Valve **(MBV-0004B/C)**	Normally closed, opens within 36 seconds after initiation signal including D/G loading sequence time.
Testable Check Valve, Reactor Injection Line **(AUV-0005B/C)**	Prevent loss of coolant outside drywell for line break.
Maintenance Valve, Reactor Injection Line **(BV-0006 B/C)**	Normally open, used to isolate system from reactor for maintenance purposes.
Pump Test Line Valve **(MCV-0009B/C)**	Normally closed, throttle valve used to test system flow at rated and runout conditions.
Pump Minimum Flow Line Valve **(MBV-0010B/C)**	Normally closed, opens on signal when pump discharge pressure is high or low flow through flow meter. Used to protect pump from overheating.

* Estimate values. Final design values will be provided with the FSAR.

† The HPCF System has the capability to deliver at least 50% of these flow rates with 171°C water at the pump suction.

Table 6.3-9 Design Parameters* for RHR (E11) Components

(1) Main Pumps (P-0001A/B/C)*	
Number of Pumps	3
Pump Type	Centrifugal
Drive Unit	Constant Speed Induction Motor
Rated Flow Rate	954 m^3/h
Developed Head	105 m
Maximum Runout Flow	1130 m^3/h
Maximum Value of Minimum Flow Mode	148 m^3/h
Minimum Shutoff Head	200 m
Maximum Pump Brake Horsepower	475 kW
Water Temperature Range	10° to 182°C
NPSH Required	2.4 m
Saturated Water in Suppression Pool Pumped Over Pressure Range	0 to 0.62 MPaG
(2) Heat Exchangers (HX-0001A/B/C)*	
Number of units	3
Seismic	Category I design and analysis
Types of exchangers	U-Tube/Shell
Maximum primary side pressure	3.80 MPaG
Design Point Function Cooling	Post-LOCA Containment
Primary side (tube side) performance data	
(1) Flow	954 m^3/h
(2) Inlet temperature	182°C maximum
(3) Allowable pressure drop (Max)	7.0 m w.g.
(4) Type water	Suppression Pool, Reactor Water, or Fuel Pools
(5) Fouling factor	0.0005
Secondary side (shell side) performance data	
(1) Flow	1200 m^3/h
(2) Inlet temperature	37.8°C maximum
(3) Allowable pressure drop (Max)	7.0m w.g.

Table 6.3-9 Design Parameters* for RHR (E11) Components (Continued)

(4) Type water	Reactor Building Cooling Water
(5) Fouling factor	0.0005
(3) Strainer (STR-0001A/B/C)*	
Location	Suppression Pool
Size	As required for insulation debris per Appendix 6C
(4) Restricting Orifices*	
Location **(ORF-0001A/B/C)**	Vessel return line
Size	Limit flow to vessel to 954 m^3/h
Location **(ORF-0004A/B/C)**	Suppression pool return line
Size	Limit flow during suppression pool cooling to 954 m^3/h
Location **(ORF-0006A/B/C)**	Fuel pool return line
Size	Limit flow during fuel pool cooling to 350 m^3/h
Location **(ORF-0003A/B/C)**	Pump minimum flow line
Size	Limit pump flow through the bypass line to 148 m^3/h
Location **(ORF-0005B/C)**	Discharge line to wetwell spray
Size	Limit wetwell spray sparger flow to 114 m^3/h
Location	Discharge line to drywell spray sparger
Size	Limit drywell spray sparger flow to 840 m^3/h
(5) Flow Elements (FE-0010A/B/C)*	
Location	Pump discharge line, downstream of heat exchanger bypass return
Rated Flow	954 m^3/h
Head Loss	6.1m w.g. maximum @ 954 m^3/h
Accuracy	±2.5% combined element, transmitter and indicator at rated flow
(6) Vessel Flooder Sparger*	
Flow Rate	954 m^3/h
Minimum Exit Velocity	11 m/s @ 954 m^3/h
(7) Wetwell Spray Sparger* (SPRG-0002)	
Flow Rate	114 m^3/h
(8) Drywell Spray Sparger* (SPRG-0001)	
Flow Rate	840 m^3/h

Table 6.3-9 Design Parameters* for RHR (E11) Components (Continued)

(9) Piping and Valves*	
Design Pressures	0.39 MPaG—discharge piping connected to suppression pool
	0.39 MPaG—suction piping connected to suppression pool
	2.65 MPaG—wetwell and drywell sparger piping
	2.82 MPaG—pump suction piping
	3.5 MPaG—pump discharge piping
	8.62 MPaG—vessel suction and return piping
Design Temperatures	122°C—suppression pool piping
	171°C—drywell sparger piping
	190°C —pump suction and discharge piping
	302°C—vessel suction and return piping
(10) Valve Operation*	
See Table 5.4-3, RHR Pump/Valve Logic	

* Estimate values. Final design values will be provided with the FSAR.

Figure 6.3-1 High Pressure Core Flooder System (HPCF) PFD

To be provided in FSAR.

NOTE: THIS LOADING PATTERN IS FOR GE 6/7 TYPE FUEL. LOADING PATTERN IS SHOWN FOR QUARTER CORE ONLY. ROTATIONAL SYMMETRY APPLIES.

J →	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1															4	4	4
2												4	4	4	2	2	1
3											4	2	2	1	1	2	2
4								4	4	4	1	2	1	1	2	1	2
5							4	2	2	1	2	1	2	3	1	2	1
6						4	2	1	1	2	1	2	2	1	2	3	2
7					4	2	2	1	2	1	2	3	1	2	3	1	3
8				4	2	1	1	2	1	1	2	1	2	3	1	2	1
9				4	2	1	2	1	3	3	1	2	3	1	2	1	3
10				4	1	2	1	1	3	3	1	3	1	2	2	1	3
11			4	1	2	1	2	2	1	1	2	1	2	3	1	2	1
12		4	2	2	1	2	3	1	2	3	1	2	1	1	2	3	2
13		4	2	1	2	2	1	2	3	1	2	1	3	3	2	1	3
14		4	1	1	3	1	2	3	1	2	3	1	3	3	1	2	1
15	4	2	1	2	1	2	3	1	2	2	1	2	2	1	2	3	2
16	4	1	2	1	2	3	1	2	1	1	2	3	1	2	3	1	2
17	4	2	1	2	3	1	2	2	3	3	2	1	2	3	1	2	3

BUNDLE TYPE	ENRICHMENT	NUMBER OF BUNDLES
1	HIGH ENRICHMENT, 3.18 WT %	308
2	MEDIUM ENRICHMENT, 2.18 WT %	324
3	LOW ENRICHMENT, 1.23 WT %	148
4	NATURAL URANIUM, 0.71 WT %	92

Figure 6.3-2 Core Loading Map Used for Response Analyses

Figure 6.3-3 Not Used

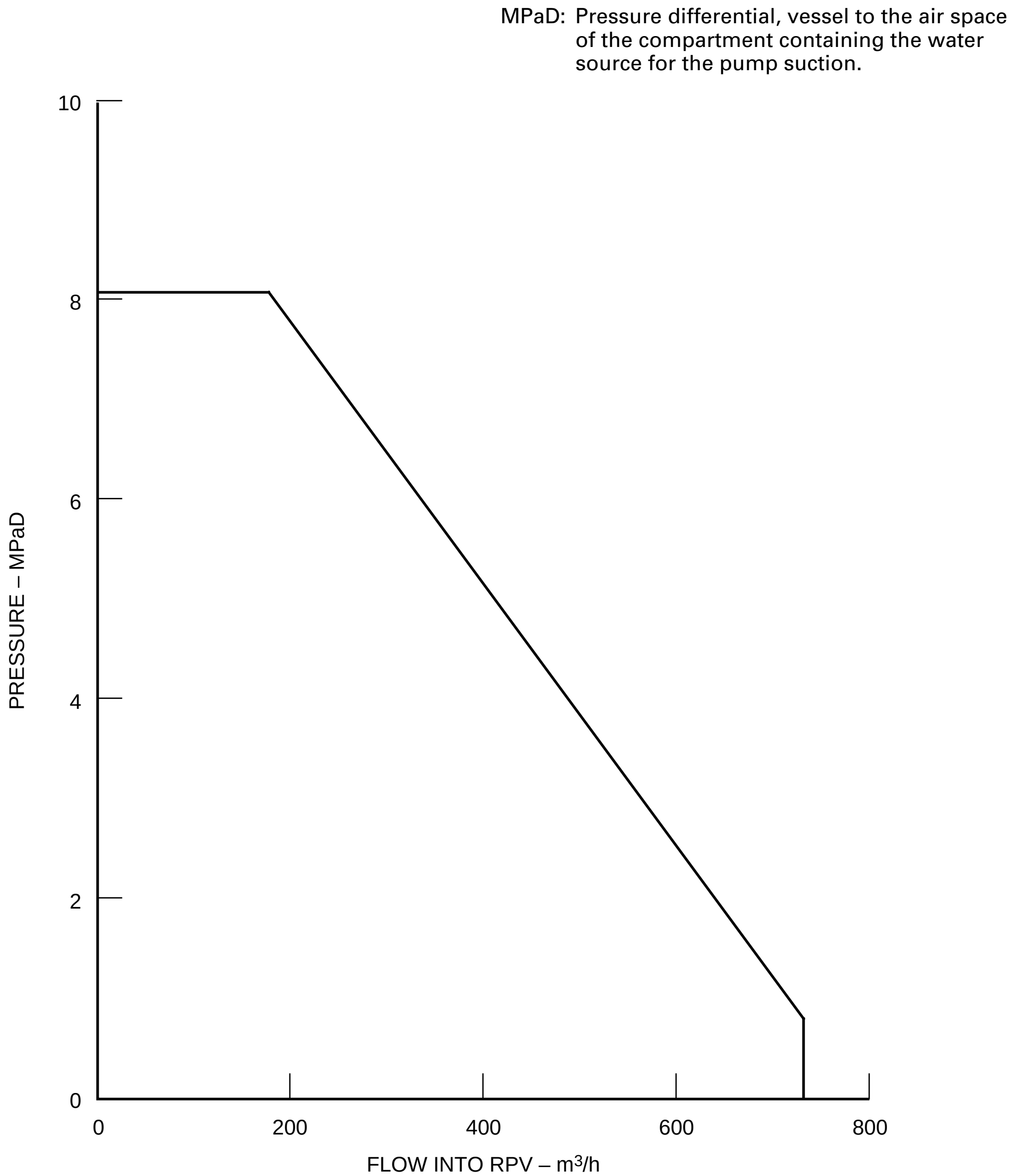

Figure 6.3-4 Pressure Versus High Pressure Core Flooder Flow (Per System) Used in LOCA Analysis

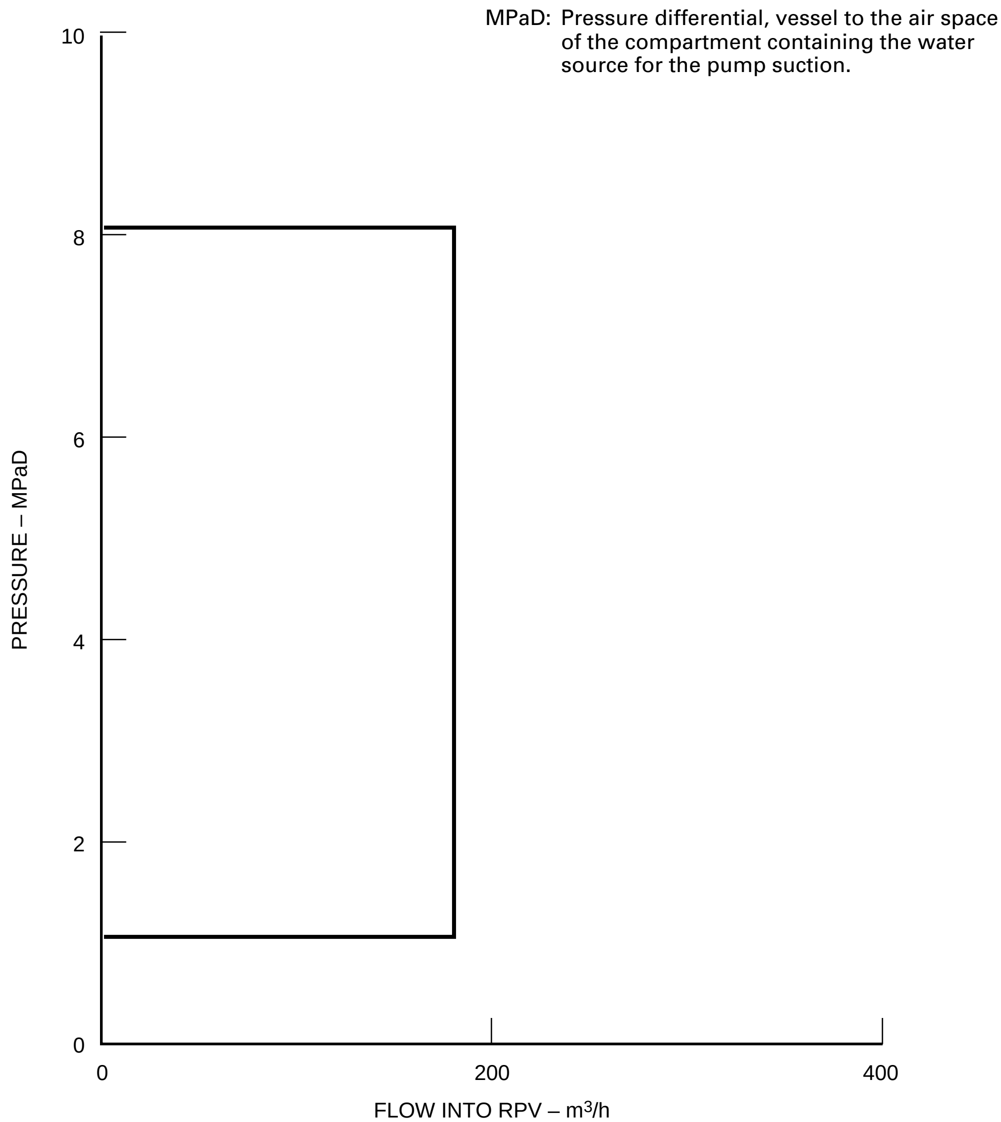

Figure 6.3-5 Pressure Versus Reactor Core Isolation Cooling Flow Used in LOCA Analysis

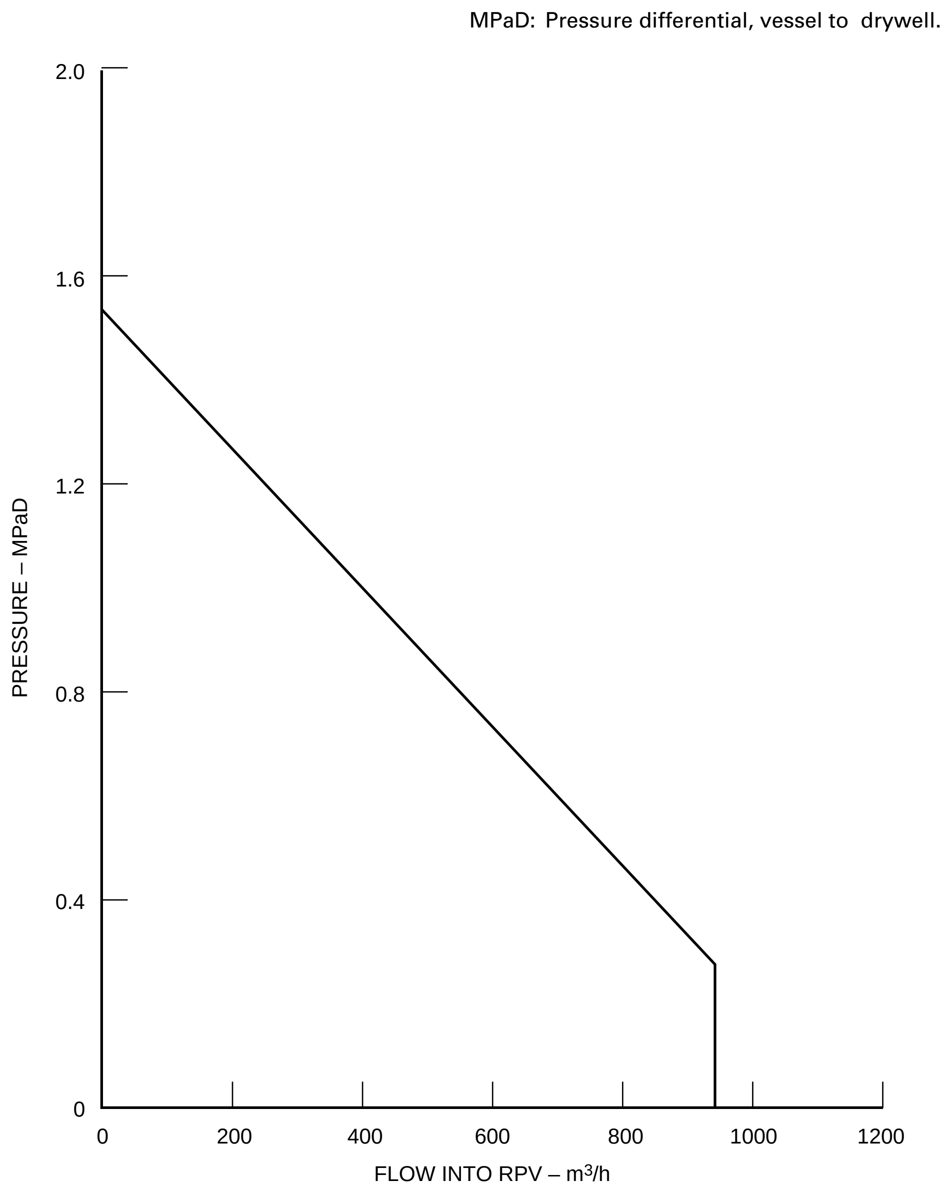

Figure 6.3-6 Pressure Versus Low Pressure Flooder Flow (Per System) Used in LOCA Analysis

Figure 6.3-7 High Pressure Core Flooder System (HPCF) P&ID (Sheets 1-2)

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-8 Not Used

Figure 6.3-9 Not Used

Figure 6.3-10 Minimum Water Level Outside Shroud Versus Break Area

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-11 Normalized Core Power Versus Time For Loss-of-Coolant Accident Analysis

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-12 Normalized Core Flow Following a Main Steamline Break Inside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-13 Minimum Critical Power Ratio Following a Main Steamline Break Inside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-14 Water Level in Fuel Channels Following a Main Steamline Break Inside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-15 Water Level Inside Shroud Following a Main Steamline Break Inside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-16 Water Level Outside Shroud Following a Main Steamline Break Inside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-17 Vessel Pressure Following a Main Steamline Break Inside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-18 Flow Out of Vessel Following a Main Steamline Break Inside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-19 Flow Into Vessel Following a Main Steamline Break Inside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-20 Peak Cladding Temperature Following a Main Steamline Break Inside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-21 Normalized Core Flow Following a Feedwater Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-22 Minimum Critical Power Ratio Following a Feedwater Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-23 Water Level in Fuel Channels Following a Feedwater Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-24 Water Level Inside Shroud Following a Feedwater Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-25 Water Level Outside Shroud Following a Feedwater Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-26 Vessel Pressure Following a Feedwater Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-27 Flow Out of Vessel Following a Feedwater Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-28 Flow Into Vessel Following a Feedwater Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-29 Peak Cladding Temperature Following a Feedwater Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-30 Water Level in Fuel Channels Following an RHR Shutdown Suction Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-31 Water Level Inside Shroud Following an RHR Shutdown Suction Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-32 Water Level Outside Shroud Following an RHR Shutdown Suction Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-33 Vessel Pressure Following an RHR Shutdown Suction Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-34 Flow Out of Vessel Following an RHR Shutdown Suction Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-35 Flow Into Vessel Following an RHR Shutdown Suction Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-36 Peak Cladding Temperature Following an RHR Shutdown Suction Line, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-37 Water Level in Fuel Channels Following an RHR/LPFL Injection Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-38 Water Level Inside Shroud Following an RHR/LPFL Injection Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-39 Water Level Outside Shroud Following an RHR/LPFL Injection Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-40 Vessel Pressure Following an RHR/LPFL Injection Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-41 Flow Out of Vessel Following an RHR/LPFL Injection Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-42 Flow Into Vessel Following an RHR/LPFL Injection Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-43 Peak Cladding Temperature Following an RHR/LPFL Injection Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-44 Normalized Core Flow Following a Core Flooder Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-45 Minimum Critical Power Ratio Following a Core Flooder Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-46 Water Level in Fuel Channels Following a Core Flooder Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-47 Water Level Inside Shroud Following a Core Flooder Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-48 Water Level Outside Shroud Following a Core Flooder Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-49 Vessel Pressure Following a Core Flooder Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-50 Flow Out of Vessel Following a Core Flooder Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-51 Flow Into Vessel Following a Core Flooder Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-52 Peak Cladding Temperature Following a Core Flooder Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-53 Water Level in Fuel Channels Following a Bottom Drain Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-54 Water Level Inside Shroud Following a Bottom Drain Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-55 Water Level Outside Shroud Following a Bottom Drain Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-56 Vessel Pressure Following a Bottom Drain Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-57 Vessel Pressure Following an RHR/LPFL Injection Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-58 Flow Into Vessel Following a Bottom Drain Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-59 Peak Cladding Temperature Following a Bottom Drain Line Break, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-60 Water Level in Fuel Channels Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-61 Water Level Inside Shroud Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-62 Water Outside Shroud Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-63 Vessel Pressure Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-64 Flow Out of Vessel Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-65 Flow Into Vessel Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-66 Peak Cladding Temperatures Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-67 Normalized Core Flow Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure (Based on Bounding Values)

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-68 Minimum Critical Power Ratio Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure (Based on Bounding Values)

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-69 Water Level in Fuel Channels Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure (Based on Bounding Values)

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-70 Water Level Inside Shroud Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure (Based on Bounding Values)

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-71 Water Level Outside Shroud Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure (Based on Bounding Values)

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-72 Vessel Pressure Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure (Based on Bounding Values)

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-73 Flow Out of Vessel Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure (Based on Bounding Values)

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-74 Flow Into Vessel Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure (Based in Bounding Values)

(Proprietary information provided in a separate proprietary volume.)

Figure 6.3-75 Peak Cladding Temperature Following a Main Steamline Break Outside Containment, HPCF Diesel Generator Failure (Based on Bounding Values)

(Proprietary information provided in a separate proprietary volume.)