UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 176
License No. DPR-35

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Boston Edison Company (the licensee) dated February 20, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-35 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 176, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas

Cecil O. Thomas, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 28, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 176

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3/4.5-1	3/4.5-1
3/4.5-2	3/4.5-2
3/4.5-3	3/4.5-3
3/4.5-4	3/4.5-4
3/4.5-5	3/4-5-5
3/4.5-6	3/4.5-6
-	3/4.5-7
-	3/4.5-8
-	3/4.5-9
-	3/4.5-10
-	3/4.5-11
-	3/4.5-12
B3/4.5-1	B3/4.5-1
B3/4.5-2	B3/4.5-2
B3/4.5-3	B3/4.5-3
B3/4.5-4	B3/4.5-4
B3/4.5-5	B3/4.5-5
B3/4.5-6	B3/4.5-6
B3/4.5-7	B3/4.5-7
B3/4.5-8	B3/4.5-8
B3/4.5-9	B3/4.5-9
B3/4.5-10	B3/4.5-10
B3/4.5-11	B3/4.5-11
B3/4.5-12	B3/4.5-12
B3/4.5-13	B3/4.5-13
B3/4.5-14	B3/4.5-14
B3/4.5-15	B3/4.5-15
B3/4.5-16	B3/4.5-16
B3/4.5-17	B3/4.5-17
B3/4.5-18	B3/4.5-18
B3/4.5-19	B3/4.5-19
B3/4.5-20	B3/4.5-20
B3/4.5-21	B3/4.5-21
B3/4.5-22	B3/4.5-22
B3/4.5-23	B3/4.5-23

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and suppression pool cooling systems.

Objective

To assure the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification

A. Core Spray and LPCI Systems

1. Both core spray systems shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.2 below.

2. From and after the date that one of the core spray systems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days, provided that during such seven days all active components of the other core spray system and active components of the LPCI system and the diesel generators are operable

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the Surveillance Requirements of the core and suppression pool cooling systems which are required when the corresponding Limiting Condition for operation is in effect.

Objective

To verify the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification

A. Core Spray and LPCI Systems

1. Core Spray System Testing.

	Item	Frequency
a.	Simulated Automatic Actuation Test.	Once/ Operating Cycle
b.	Pump Operability.	When tested as specified in 3.13 verify that each core spray pump delivers at least 3300 GPM against a system head corresponding to a reactor vessel pressure of 104 psig

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray and LPCI Systems (Cont)

3. The LPCI system shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.4 and 3.5.F.5.

4. From and after the date that the LPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during such seven days the active components of both core spray systems and the diesel generators required for operation of such components, if no external source of power were available, shall be operable.

5. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray and LPCI Systems (Cont)

1. c. Motor Operated Valve Operability — As Specified in 3.13

d. Core Spray Header Δp Instrumentation

Check	Once/day
Calibrate	Once/3 months
Test Step	Once/3 months

2. This section intentionally left blank

3. LPCI system testing shall be as follows:

a.	Simulated Automatic Actuation Test.	Once/ Operating Cycle
b.	Pump Operability.	When tested as specified in 3.13, verify that each LPCI pump delivers 4800 GPM at a head across the pump of at least 380 ft.
c	Motor Operated Valve Operability	As Specified in 3.13

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

B.1 Residual Heat Removal (RHR) Suppression Pool Cooling

Specification:

Two RHR suppression pool cooling subsystems shall be OPERABLE.

Applicability:

Whenever irradiated fuel is in the reactor vessel, reactor coolant temperature is >212° F, and prior to startup from a cold condition.

Actions:

A. One RHR suppression pool cooling subsystem inoperable,

1. Restore the RHR suppression pool cooling subsystem to OPERABLE status within 7 days.

B. Required Action and associated Completion Time not met.

OR

More than two RHR pumps inoperable

OR

Two RHR suppression pool cooling subsystems inoperable.

1. Be in Cold Shutdown within 24 hours

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

B.1 Residual Heat Removal (RHR) Suppression Pool Cooling

1. Verify each RHR suppression pool cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position every 31 days.

2. Verify each RHR pump develops a flow rate ≥ 5100 GPM through the associated heat exchanger while operating in the suppression pool cooling mode as specified in Specification 3/4.13.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

B.2 Residual Heat Removal (RHR) Containment Spray

Specification:

Two RHR containment spray subsystems shall be OPERABLE.

Applicability:

Whenever irradiated fuel is in the reactor vessel, reactor coolant temperature is >212°F, and prior to startup from a cold condition.

Actions:

A. One RHR containment spray subsystem inoperable,

   1. Restore RHR containment spray subsystem to OPERABLE status within 7 days.

B. Required Action and associated Completion Time not met

   OR

   Two RHR containment spray subsystems inoperable,

   1. Be in Cold Shutdown within 24 hours

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

B.2 Residual Heat Removal (RHR) Containment Spray

1. Verify each RHR containment spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position every 31 days.

2. Air test drywell and suppression pool (torus) headers and nozzles once per 5 years.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

B.3 Reactor Building Closed Cooling Water (RBCCW) System

Specification:

Two RBCCW subsystems shall be OPERABLE.

Applicability:

Whenever irradiated fuel is in the reactor vessel, reactor coolant temperature is >212° F, and prior to startup from a cold condition.

Actions:

A. One required RBCCW Pump Inoperable.

    1. Restore the required RBCCW pump to OPERABLE status within 7 days.

B. One RBCCW subsystem inoperable for reasons other than Condition A.

    1. Restore the RBCCW subsystem to OPERABLE status within 72 hours.

C. Required Action and associated Completion Times of Condition A or B not met.

    OR

    Two RBCCW subsystems inoperable.

    1. Be in Cold Shutdown within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

B.3 Reactor Building Closed Cooling Water (RBCCW) System

1. ----------------------NOTE-----------------------
Isolation of flow to individual components does not render the RBCCW subsystem inoperable.

Verify each RBCCW manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position every 31 days.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

B.4 Salt Service Water (SSW)] System

Specification:

Two SSW subsystems shall be OPERABLE.

Applicability:

Whenever irradiated fuel is in the reactor vessel, reactor coolant temperature is >212° F, and prior to startup from a cold condition.

Actions:

A. One SSW subsystem inoperable.

1. Restore the SSW subsystem to OPERABLE status within 72 hours.

B. Required Action and associated Completion Time not met,

OR

Two SSW subsystems inoperable,

OR

UHS inoperable,

1. Be in Cold Shutdowr within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

B.4 Salt Service Water (SSW)] System

1. Verify the water level in each SSW pump well of the intake structure is ≥13 ft 9 in below mean sea level every 24 hours.

2. Verify the average sea water temperature is ≤ 75°F every 24 hours.

3. ------------------NOTE------------------
Isolation of flow to individual components does not render the SSW subsystem inoperable.

Verify each SSW subsystem manual, power operated, and automatic valve in the flow paths servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position every 31 days.

4. Verify each SSW subsystem actuates on an actual or simulated initiation signal every 2 years.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5 CORE AND CONTAINMENT COOLING SYSTEMS	4.5 CORE AND CONTAINMENT COOLING SYSTEMS
C. HPCI System	C. HPCI System
1. The HPCI system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig., and reactor coolant temperature is greater than 365°F, except as specified in 3.5.C.2 below.	1. HPCI system testing shall be as follows:
2. From and after the date that the HPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days all active components of the ADS system, the RCIC system, the LPCI system and both core spray systems are operable.	a. Simulated Automatic Actuation Test — Once/ Operating Cycle b. Pump Operability — When tested as specified in 3.13, verify that the HPCI pump delivers at least 4250 GPM for a system head corresponding to a reactor pressure of 1000 psig
3. If the requirements of 3.5.C cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.	c. Motor Operated Valve Operability — As Specified in 3.13 d. Flow Rate at 150 psig — Once/ operating cycle, verify that the HPCI pump delivers at least 4250 GPM for a system head corresponding to a reactor pressure of 150 psig The HPCI pump shall deliver at least 4250 GPM for a system head corresponding to a reactor pressure of 1000 to 150 psig.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

D. Reactor Core Isolation Cooling (RCIC) System

1. The RCIC system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and reactor coolant temperature is greater than 365°F, except as specified in 3.5.D.2 below.

2. From and after the date that the RCIC system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days the HPCIS is operable.

3. If the requirements of 3.5.D cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

D. Reactor Core Isolation Cooling (RCIC) System

1. HPCI system testing shall be as follows:

a.	Simulated Automatic Actuation Test	Once/ Operating Cycle
b.	Pump Operability	When tested as specified in 3.13, verify that the RCIC pump delivers at least 400 GPM at a system head corresponding to a reactor pressure of 1000 psig.
c.	Motor Operated Valve Operability	As Specified in 3.13
d.	Flow Rate at 150 psig	Once/ operating cycle verify that the RCIC pump delivers at least 400 GPM at a system head corresponding to a reactor pressure of 150 psig

The RCIC pump shall deliver at least 400 GPM for a system head corresponding to a reactor pressure of 1000 to 150 psig.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

E. Automatic Depressurization System (ADS)

1. The Automatic Depressurization System shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 104 psig and prior to a startup from a Cold Condition, except as specified in 3.5.E.2 below.

2. From and after the date that one valve in the Automatic Depressurization System is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such valve is sooner made operable, provided that during such 14 days the HPCI system is operable.

3. If the requirements of 3.5.E cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

E Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

   a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage.

      The ADS manual inhibit switch will be included in this test.

   b. With the reactor at pressure, each relief valve shall be manually opened until a corresponding change in reactor pressure or main turbine bypass valve positions indicate that steam is flowing from the valve.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

F. Minimum Low Pressure Cooling and Diesel Generator Availability

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless such diesel generator is sooner made operable, provided that all of the low pressure core and containment cooling systems and the remaining diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown condition, both core spray systems, and the LPCI system may be inoperable, provided no work is being done which has the potential for draining the reactor vessel.

4. During a refueling outage, for a period of 30 days, refueling operation may continue provided that one core spray system or the LPCI system is operable or Specification 3.5.F.5 is met.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

F. Minimum Low Pressure Cooling and Diesel Generator Availability

1. When it is determined that one diesel generator is inoperable, within 24 hours, determine that the operable diesel generator is not inoperable due to a common cause failure,

<u>or</u>

perform surveillance 4.9.A.1.a for the operable diesel generator,

<u>and</u>

within 1 hour and once every 8 hours thereafter, verify correct breaker alignment and indicated power for each offsite circuit.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

F Minimum Low Pressure Cooling and Diesel Generator Availability (Cont)

5. When irradiated fuel is in the reactor vessel and the reactor is in the Refueling Condition with the torus drained, a single control rod drive mechanism may be removed, if both of the following conditions are satisfied:

a) No work on the reactor vessel, in addition to CRD removal, will be performed which has the potential for exceeding the maximum leak rate from a single control blade seal if it became unseated.

b) i) the core spray systems are operable and aligned with a suction path from the condensate storage tanks. ii) the condensate storage tanks shall contain at least 200,000 gallons of usable water and the refueling cavity and dryer/separator pool shall be flooded to a least elevation 114'-0"

G. Deleted

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

F. Minimum Low Pressure Cooling and Diesel Generator Availability (Cont)

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

H. Maintenance of Filled Discharge Pipe

Whenever core spray systems, LPCI system, HPCI or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements assure that the discharge piping of the core spray systems, LPCI system, HPCI and RCIC are filled:

1. Every month the LPCI system and core spray system discharge piping shall be vented from the high point and water flow observed.

2. Following any period where the LPCI system or core spray systems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS
B 3/4.5.A Core Spray and LPCI System

BASES:

BACKGROUND

Each Core Spray system consists of one pump and associated piping and valves with all active components required to be operable. The LPCI system consists of four LPCI pumps and associated piping and valves with all active components required to be operable.

The LPCI system is not considered inoperable when the RHR System is operating in the shutdown cooling mode.

APPLICABLE SAFETY ANALYSES

Based on the loss of coolant analysis performed by General Electric in accordance with Section 50.46 and Appendix K of 10CFR50, the Pilgrim I Emergency Core Cooling Systems are adequate to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit calculated fuel clad temperature to less than 2200° F, to limit calculated local metal water reaction to less than or equal to 17%, and to limit calculated core wide metal water reaction to less than or equal to 1%. The detailed bases is described in NEDC-31852P and summarized in Section 6.5 of the PNPS FSAR.

The analyses discussed in NEDC-31852P calculated a peak clad fuel temperature of less than 2200°F with a Core Spray pump flow of 3200 gallons per minute (gpm). A flow rate of 3300 gpm ensures adequate flow for events involving degraded voltage.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Pilgrim, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis takes credit for core spray flow into the core at vessel pressure below 205 psig. However, the analysis is conservative in that no credit is taken for spray cooling heat transfer in the hottest fuel bundle until the pressure at rated flow for the core spray (104 psig vessel pressure) is reached.

The LPCI system is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI system and the core spray system provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high pressure emergency core cooling systems. The analyses in NEDC-31852P calculated a peak clad fuel temperature of less than 2200°F with LPCI pump flows of 4550 gpm, 4033 gpm, and 3450 gpm for two, three, and four pump combinations feeding into a single loop. A single pump flow rate at 4800 gpm ensures sufficient flow to meet or exceed the analyses' assumptions.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analyses of LOCA for PNPS demonstrated the combination of LPCS/LPCI systems are sufficient to provide core cooling even with a single failure of either an active or passive safety-related component. The analyses determined there were four significant single failures that challenge the Emergency Core Coolant Systems' capability to prevent fuel damage during the postulated LOCA. They are:

1) Battery Failure - Loss of a single battery train could leave only one LPCS pump, two LPCI pumps, and ADS to mitigate the LOCA. This is the most limiting single failure for all but the largest postulated recirculation line breaks and for all postulated non-recirculation line breaks.

2) LPCI Injection Valve Failure - Loss of the injection valve selected by LPCI Loop Selection Logic for the pathway for all LPCI pumps' flow leaves two core spray pumps, HPCI, and ADS for LOCA mitigation. This becomes the limiting single failure for the largest postulated recirculation line breaks.

3) Loss of one emergency diesel generator - This leaves one LPCS pump, two LPCI pumps, and ADS for LOCA mitigation.

4) HPCI Failure - This leaves all other ECCS resources available. It is a significant failure primarily for small line breaks.

In all cases above, the remaining ECCS resources are sufficient to prevent PCT from exceeding 2200°F and other criteria provided in Section 50.46 and Appendix K of 10CFR50.

ACTIONS

Should one Core Spray system become inoperable, the remaining Core Spray and the LPCI system are available should the need for core cooling arise. Based on judgments of the reliability of the remaining systems (i.e., the Core Spray and LPCI) a seven-day repair period was obtained.

SURVEILLANCE REQUIREMENTS

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated valves are tested in accordance with ASME B&PV Code, Section XI (IWP and IWV, except where specific relief is granted) to assure their operability. The frequency and methods of testing are described in the PNPS IST

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

program. The PNPS IST Program is used to assess the operational readiness of pumps and valves that are safety-related or important to safety. When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and valves is deemed to be adequate testing of these systems.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

B 3/4.5.B.1 Residual Heat Removal (RHR) Suppression Pool Cooling

BASES

BACKGROUND

Following a Design Basis Accident (DBA), the RHR suppression pool cooling subsystem removes heat from the suppression pool. The suppression pool is designed to absorb the sudden input of heat from the primary system. In the long term, the pool continues to absorb residual heat generated by fuel in the reactor core. Some means must be provided to remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. This function is provided by two redundant RHR suppression pool cooling subsystems. The purpose of this Specification is to ensure that both subsystems are OPERABLE in applicable MODES.

Each RHR suppression pool cooling subsystem contains two pumps and one heat exchanger and is manually initiated and independently controlled. The two subsystems perform the suppression pool cooling function by circulating water from the suppression pool through the RHR heat exchangers and returning it to the suppression pool. The RHR heat exchangers (tube side) are cooled by the reactor building closed cooling water system (Specification 3/4.5.B.3), which is in turn cooled by the salt water service system (Specification 3/4.5.B.4)

S/RV leakage and high pressure coolant injection or reactor core isolation cooling system testing increase suppression pool temperature more slowly. The RHR suppression pool cooling subsystem is also used to lower the suppression pool water bulk temperature following such events.

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses used to predict primary containment pressure and temperature following large and small break LOCAs. The intent of the analyses is to demonstrate that the heat removal capacity of the RHR suppression pool cooling subsystem is adequate to maintain the primary containment conditions within design limits. The suppression pool temperature is calculated to remain below the design limit

The RHR suppression pool cooling subsystem satisfies Criterion 3 of 10 CFR 50 36(c)(2)(ii)

(continued)

SPECIFICATION

During a DBA, a minimum of one RHR suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below design limits (Ref. 1). To ensure that these requirements are met, two redundant suppression pool cooling subsystems must be OPERABLE with power from two safety related independent power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR suppression pool cooling subsystem is OPERABLE when one of the RHR pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE. When the Low Pressure Cooling Injection (LPCI) System is inoperable, a minimum of two RHR pumps and both Core Spray (CS) Systems must be OPERABLE.

APPLICABILITY

When reactor coolant temperature is > 212° F and irradiated fuel is in the reactor vessel, a DBA could cause a release of radioactive material to primary containment and cause a heat up and pressurization of primary containment. When the reactor temperature is ≤ 212 °F , the probability and consequences of these events are less severe. Therefore, the RHR suppression pool cooling subsystem is not required to be OPERABLE when reactor coolant temperature is ≤ 212° F.

ACTIONS

A.1

With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single active failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day completion time is acceptable in light of the redundant RHR suppression pool cooling capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

B.1

If the inoperable RHR suppression pool cooling subsystem cannot be restored to OPERABLE status within the associated completion time, more than two RHR pumps are inoperable, or if two RHR suppression pool cooling subsystems are inoperable, the plant must be brought to a condition in which the specification does not apply. To achieve this status, the plant must be brought to Cold Shutdown within 24 hours. The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

BASES

SURVEILLANCE REQUIREMENTS

SR 4.5.B.1.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR suppression pool cooling mode flow path provides assurance that the proper flow path exists for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is a manually initiated system. This frequency has been shown to be acceptable based on operating experience.

SR 4.5.B.1.2

Verifying that each RHR pump develops a flow rate ≥ 5100 gpm (Ref. 1) while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by ASME Code, Section XI (Ref. 2). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The frequency of this SR is in accordance with the Inservice Testing Program, Specification 3/4.13.

REFERENCES

1. FSAR, Section 14.5.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

B 3/4.5.B.2 Residual Heat Removal (RHR) Containment Spray

BASES

BACKGROUND

The RHR containment spray subsystem provides containment spray capability as an alternate method for reducing containment pressure (temperature) following a LOCA. A portion of the water pumped through the RHR heat exchangers can be diverted to spray headers in the drywell and above the suppression pool. The portion of the RHR heat exchanger flow to the spray headers in the drywell condenses any steam that may exist in the drywell thereby lowering containment pressure (temperature). The remaining portion returns to the suppression pool via the suppression pool bypass line. The spray collects in the bottom of the drywell until the water level rises to the level of the pressure suppression vent pipes where it overflows and drains back to the suppression pool. Approximately 5 percent of the total flow may be directed to the suppression chamber spray ring to cool any noncondensable gases collected in the free volume above the suppression pool. The containment spray subsystem will remove energy from the drywell by condensing steam, thereby, making available the drywell volume to accommodate additional quantities of gases from any postulated metal water reactions above that which the containment can inherently accommodate without spray (Ref. 1)

The containment spray mode of the RHR cannot be operated unless the level inside the reactor vessel shroud is above the two thirds core height set point and the drywell pressure exceeds a setpoint greater than 1 but less than 2 psig.

Interlocks are provided to prevent LPCI flow from being diverted to the containment spray mode unless the core is flooded. A keylock switch in the control room permits the overriding of this interlock to reduce containment pressure if required.

Each of the two RHR containment spray subsystems contains two pumps and one heat exchanger, which are manually initiated and independently controlled. The RHR heat exchangers (tube side) are cooled by the reactor building closed cooling water system (Specification 3/4.5.B.3), which is in turn cooled by the salt service water system (Specification 3/4.5.B.4). Either RHR containment spray subsystem is sufficient to condense the steam from small bypass leaks from the drywell to the suppression chamber airspace during the postulated DBA.

(continued)

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS
BASES

APPLICABLE SAFETY ANALYSES

Reference 2 contains the results of analyses used to predict primary containment pressure and temperature following loss of coolant accidents. The intent of the analvses is to demonstrate the pressure (temperature) reduction capability of the RHR containment spray system.

The RHR containment spray subsystem satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

SPECIFICATION

In the event of a DBA, a minimum of one RHR containment spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak temperature below the design limits (Ref. 3). To ensure that these requirements are met, two redundant RHR containment spray subsystems must be OPERABLE with power from two safety-related independent power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR containment spray subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

APPLICABILITY

When reactor coolant temperature is > 212°F and irradiated fuel is in the reactor vessel, a DBA could cause a release of radioactive material to primary containment and cause a heat up and pressurization of primary containment. When the reactor temperature is ≤ 212 °F, the probability and consequences of these events are less severe. Therefore, the RHR containment spray system is not required to be OPERABLE when reactor coolant temperature is ≤ 212°F.

ACTIONS

A.1

With one RHR containment spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE RHR containment spray subsystem is adequate to perform the primary containment bypass leakage mitigation function. However, the overall reliability is reduced because a single active failure in the OPERABLE subsystem could result in reduced primary containment peak temperature control. The 7 day completion time was chosen in light of the redundant RHR containment spray capabilities afforded by the OPERABLE subsystem and the low probability of a LOCA occurring during this period.

(continued)

ACTIONS (continued)

B.1

If the inoperable RHR containment spray subsystem cannot be restored to OPERABLE status within the associated completion time or if two RHR containment spray subsystems are inoperable, the plant must be brought to a condition in which the specification does not apply. To achieve this status, the plant must be brought to Cold Shutdown within 24 hours. The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 4.5.B.2.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the RHR containment spray mode flow path provides assurance that the proper flow paths will exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is a manually initiated system. This frequency has been shown to be acceptable based on operating experience.

SR 4.5.B.2.2

Verifying that the drywell and suppression pool (torus) headers and nozzles are free of obstructions by blowing air through them ensures an open flow path. The frequency for performance of the spray nozzle obstruction surveillance test of 5 years is justified due to the passive design of the nozzles and has been shown acceptable through industry operating experience.

(continued)

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS
BASES

REFERENCES

1. FSAR, Section 4.8
2. FSAR, Section 14.5.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3/4.5.B.3 Reactor Building Closed Cooling Water(RBCCW) System
BASES

BACKGROUND

The RBCCW system is designed to provide a heat sink for the Residual Heat Removal (RHR) system heat exchangers and the removal of heat from emergency core cooling system (ECCS) equipment, such as RHR pump lube oil coolers, core spray pump motor thrust bearings, and room coolers, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient.

The RBCCW system consists of two full capacity closed loops. Each loop has three centrifugal pumps, rated 1,700 gal/min at 100 ft total dynamic head (TDH), taking suction from the reactor building cooling water heat exchanger and capable of delivering inhibited demineralized water to the associated equipment. In order to transfer the design RHR system heat load during postulated transient or accident conditions, a minimum of two pumps in one loop and a cooling water heat exchanger is required.

Motor-operated gate valves are provided in each loop to manually isolate non-essential cooling loads under accident conditions. The two independent loops have the capability to be interconnected through two 12 inch cross-ties. The valves in the crossties are normally closed. The RBCCW system is designed with sufficient redundancy so that no single active component failure can prevent it from achieving its design function. The RBCCW system is described in the FSAR, Section 10.5.5 (Ref. 1).

During normal power operation, one pump in each loop is operating providing coolant flow to all of the associated equipment except the RHR heat exchangers which are valved off. Following a postulated loss of coolant accident (LOCA) coincident with loss of the preferred (offsite) AC power source, the operating RBCCW pumps will trip. One RBCCW pump in each loop is automatically restarted on its respective diesel generator approximately 30 sec after AC power is restored to the emergency service bus.

APPLICABLE SAFETY ANALYSES

The RBCCW system provides adequate cooling of safety equipment required for safe reactor shutdown and removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its function of limiting the release of radioactive materials to the environment following a LOCA. The ability of the RBCCW system to support long term cooling of the reactor or primary containment is discussed in the FSAR, Section 10.5.5.3 and 14.5.3.1.2 (Refs. 2 and 3, respectively). These analyses explicitly assume that the RBCCW system will provide adequate cooling support to the equipment required for safe shutdown. These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The RBCCW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

SPECIFICATION

Two RBCCW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power.

An RBCCW subsystem is considered OPERABLE when:

a. Two pumps are OPERABLE; and

b. An OPERABLE flow path is capable of taking suction from the reactor building cooling water heat exchanger and transferring the water to the associated safety equipment and RHR heat exchangers at the assumed flow rate. Additionally, the RBCCW cross tie valves (which allow the two RBCCW loops to be connected) must be closed so that failure of one subsystem will not affect the OPERABILITY of the other subsystems.

The isolation of the RBCCW system to individual components may render those components inoperable but does not affect the OPERABILITY of the RBCCW system.

APPLICABILITY

In all Modes except Cold Shutdown, the RBCCW system is required to be OPERABLE to support the OPERABILITY of the components or systems serviced by the RBCCW system.

In Cold Shutdown, the OPERABILITY requirements of the RBCCW system are determined by the systems it supports.

ACTIONS

A.1

With one required RBCCW pump inoperable, the inoperable pump must be restored to OPERABLE status within 7 days. The remaining required pump in the affected loop is sufficient to handle the normal operation heat loads and the remaining OPERABLE loop (2 required pumps) is sufficient to perform the RBCCW heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced RBCCW capability. The 7 day completion time is based on the remaining RBCCW heat removal capability and the low probability of a DBA with concurrent worst case single failure.

(continued)

ACTIONS (continued)

B.1

With one RBCCW subsystem inoperable for reasons other than Condition A, the inoperable subsystem must be restored to OPERABLE status within 72 hours. With the unit in this condition, the remaining OPERABLE RBCCW subsystem is adequate to perform the RBCCW containment cooling heat removal function. However, the overall reliability is reduced because a single active failure in the OPERABLE subsystem could result in loss of RBCCW function. The completion time is based on the capabilities afforded by the redundant OPERABLE RBCCW subsystem, the long term dependency of the RHR and Core Spray pumps for core cooling capability, and the low probability of an event occurring during this period requiring RBCCW.

C.1

If the inoperable RBCCW subsystem cannot be restored to OPERABLE status within the associated completion time or two RBCCW subsystems are inoperable, the unit must be placed in a MODE in which the Specification does not apply. To achieve this status, the unit must be placed in Cold Shutdown within 24 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 4.5.B.3.1

The 31 day frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation and ensures correct valve positions.

REFERENCES

1. FSAR, Section 10.5.5.1.
2. FSAR, Chapter 10.5.5.3.
3. FSAR, Chapter 14.5.3.1.2.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3/4.5.B.4 Salt Service Water System (SSW) and Ultimate Heat Sink (UHS)
BASES

BACKGROUND

The SSW system is designed to function as the ultimate heat sink for all the systems cooled by the Reactor Building Closed Cooling Water (RBCCW) and Turbine Building Closed Cooling Water (TBCCW) system during all planned operations in all operating states by continuously providing adequate cooling water flow to the secondary sides of the RBCCW and TBCCW heat exchangers.

The SSW system consists of two open loops. Each loop has two pumps (plus a common spare) rated at 2,700 gal/min at 55 ft TDH, piping, valving, instrumentation, and controls as necessary to provide coolant to one RBCCW heat exchanger and one TBCCW heat exchanger on each loop. The pumps take a suction from Cape Cod Bay (UHS) and discharge to a common header from which independent piping supplies each of the two cooling water loops, each loop consisting of one reactor building and one turbine building cooling water heat exchangers. The water then returns to the bay from the outlet of the heat exchanger. Two division valves are included in the common discharge header to permit the SSW System to be operated as two independent loops. Either of the two subsystems is capable of providing the required cooling capacity (4500 gpm) to support the required systems with two pumps operating. The SSW system is described in FSAR, Section 10.7.5 (Ref. 1).

During normal power operations, 3 pumps are all that is necessary to supply cooling water to accommodate the heat load. To ensure that sufficient seawater flow is maintained through the RBCCW heat exchangers (minimum of 4500 gpm for each heat exchanger), motor-operated butterfly valves on the TBCCW heat exchanger outlets will automatically adjust to preset throttling positions and the RBCCW outlet valves will simultaneously open. Automatic adjustment of the outlet valves occur following a loss of coolant accident (LOCA) with a coincident Loss of Offsite Power (LOOP), or a LOCA with degraded voltage on the safety buses while being supplied from the startup transformer. If a LOCA occurs without a LOOP or degraded voltage condition, the heat exchanger outlet valves remain as-is. Manual adjustments of the outlet valves will be made by operators to achieve adequate cooling water flow

A loss of AC power will trip all service water pumps and will close one of the two division valves in the common pump discharge header, effectively dividing the service water system into two independent loops. A selector switch determines which division valve will close and to which train the "C" SSW pump will be dedicated on loss of AC power. Two pumps would be connected to each loop. The two division valves are arranged to permit the fifth (middle) pump to be operated on either loop

(continued)

APPLICABLE SAFETY ANALYSES

The SSW system provides a supply of cooling water to the secondary side of the RBCCW heat exchangers adequate for the requirements of the RBCCW under transient and accident conditions. The ability of the SSW system to support long term cooling of the reactor containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the FSAR, Section 10.5.5.3 and 14.5.3.1.2 (Refs. 2 and 3, respectively). These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The long term cooling capability of the RHR, core spray, and reactor building closed cooling water pumps is dependent on the cooling provided by the SSW system.

The SSW system, together with the UHS, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

SPECIFICATION

The SSW subsystems are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one subsystem of SSW is required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, two subsystems of SSW must be OPERABLE. At least one subsystem will operate, if the worst single active failure occurs coincident with the loss of offsite power.

A subsystem is considered OPERABLE when it has an OPERABLE UHS, two OPERABLE pumps with associated controls and instrumentation and the following valves on that subsystem operable.

1. One TBCCW heat exchanger outlet valve unless the valve is throttled.

2. One RBCCW heat exchanger outlet valve unless the valve is open.

3. One discharge header valve (i.e., the one opposite to the selected train) unless the valve is fully closed.

The OPERABILITY of the UHS is based on having a minimum water level in the pump well of the intake structure of > 13 ft 9 inches below mean sea level and a maximum water temperature of 75°F.

(continued)

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS
BASES

APPLICABILITY

In all Modes except Cold Shutdown, the SSW system and UHS are required to be OPERABLE to support OPERABILITY of the equipment serviced by the SSW system.

In the Cold Shutdown Mode, the OPERABILITY requirements of the SSW system and UHS are determined by the systems they support.

ACTIONS

A.1

With one SSW subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 72 hours. With the unit in this condition, the remaining OPERABLE SSW subsystem is adequate to perform the RBCCW heat removal function. However, the overall reliability is reduced because a single active failure in the OPERABLE subsystem could result in loss of the SSW function. The completion time is based on the capabilities afforded by the redundant, OPERABLE SSW subsystem, the low probability of an event occurring during this period requiring SSW, and is consistent with the allowed Completion Time for restoring an inoperable RBCCW subsystem.

B.1

If the inoperable SSW subsystem cannot be restored to OPERABLE status within the associated completion time or if two SSW subsystems are inoperable, or the UHS is inoperable, the unit must be placed in a MODE in which the Specification does not apply. To achieve this status, the unit must be in Cold Shutdown within 24 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 4.5.B.4.1

This SR verifies the water level in each pump well of the intake structure to be sufficient for the proper operation of the SSW pumps (net positive suction head and pump vortexing are considered in determining this limit). The 24 hour frequency is based on operating experience related to trending of the parameter and the availability of alarms to alert the operators prior to exceeding the limit.

SR 4.5.B.4.2

Verification of the sea water inlet temperature ensures that the heat removal capability of the SSW system is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter and the availability of alarms to alert the operators prior to exceeding the limit.

(continued)

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS
BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 4.5.B.4.3

Verifying the correct alignment for each manual, power-operated, and automatic valve in each SSW subsystem flow path provides assurance that the proper flow paths will exist for SSW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be automatically realigned to its accident position within the required time.

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation and ensures correct valve positions.

SR 4.5.B.4.4

This SR verifies the automatic adjustment of the motor-operated butterfly valves on the TBCCW heat exchanger outlets, simultaneous opening of the RBCCW outlet valves, and closure of one of the two division valves in the common SSW pump discharge header during an accident event. This is demonstrated by the use of an actual or simulated initiation signal. This SR also verifies the automatic start capability of one of the two SSW pumps in each subsystem.

The 24 month frequency of the Surveillance is based on engineering judgment taking into consideration the plant conditions required to perform the Surveillance, and is consistent with fuel cycle lengths.

REFERENCES

1. FSAR, Chapter 10.7.5
2. FSAR, Chapter 10.5.5.3
3. FSAR Section 14.5.3.1.2

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS
3/4.5.C. High Pressure Coolant Injection (HPCI) System
BASES

Background

HPCI is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss-of-coolant which does not result in rapid depressurization of the reactor vessel. HPCI permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. HPCI continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray System operation maintains core cooling.

The capacity of the system is selected to provide this required core cooling. The HPCI pump is designed to pump 4250 gpm at reactor pressures between 1100 and 150 psig. Two sources of water are available. Initially, demineralized water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI System. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reached equilibrium with the flow through the break. Continued depressurization causes the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

APPLICABLE SAFETY ANALYSIS

The limiting conditions for operating the HPCI System are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a detailed functional analysis of the HPCI System (FSAR Section 6).

SPECIFICATION

The requirement that HPCI be operable when reactor coolant temperature is greater that 365°F is included in Specification 3.5.C.1 to clarify that HPCI need not be operable during certain testing (e.g., reactor vessel hydro testing at high reactor pressure and low reactor coolant temperature). 365°F is approximately equal to the saturation steam temperature at 150 psig

ACTION

The analysis in FSAR Appendix G shows that the ADS provides a single failure proof path for depressurization for postulated transients and accidents. The RCIC is required as an alternate source of makeup to the HPCI only in the case of loss of all offsite AC power. Considering the HPCI and the ADS plus RCIC as redundant paths, and considering judgments of the reliability of the ADS and RCIC systems, a 14 day allowable repair time is specified

(continued)

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS
BASES

SURVEILLANCES

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated valves are tested in accordance with ASME B&PV Code, Section XI (IWP and IWV, except where specific relief is granted) to assure their operability. The frequency and methods of testing are described in the PNPS IST program. The PNPS IST Program is used to assess the operational readiness of pumps and valves that are safety-related or important to safety. When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and valves is deemed to be adequate testing of these systems.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3/4.5.D. Reactor Core Isolation Cooling (RCIC) System

BASES

BACKGROUND

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the normal heat sink is unavailable. The Station Nuclear Safety Operational Analysis, FSAR Appendix G, shows that RCIC also serves as redundant makeup system on total loss of all offsite power in the event that HPCI is unavailable. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI.

SPECIFICATION

The requirement that RCIC be operable when reactor coolant temperature is greater than 365°F is included in Specification 3.5.D.1 to clarify that RCIC need not be operable during certain testing (e.g., reactor vessel hydro testing at high reactor pressure and low reactor coolant temperature). 365°F is approximately equal to the saturation steam temperature at 150 psig.

ACTION

Based on this and judgments on the reliability of the HPCI system, an allowable repair time of 14 days is specified.

SURVEILLANCES

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated valves are tested in accordance with ASME B&PV Code, Section XI (IWP and IWV, except where specific relief is granted) to assure their operability. The frequency and methods of testing are described in the PNPS IST program. The PNPS IST Program is used to assess the operational readiness of pumps and valves that are safety-related or important to safety. When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and valves is deemed to be adequate testing of these systems.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3/4.5.E. Automatic Depressurization (ADS) System
BASES

BACKGROUND

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray systems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considered only with respect to its depressurizing effect in conjunction with LPCI or Core Spray. There are four valves provided and each has a capacity of 800,000 lb/hr at a reactor pressure of 1125 psig.

APPLICABLE SAFETY ANALYSIS

The limiting conditions for operating the ADS are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a detailed functional analysis of the ADS (FSAR Section 6).

ACTIONS

The allowable out of service time for one ADS valve is determined as 14 days because of the redundancy and because of HPCI operability; therefore, redundant protection for the core with a small break in the nuclear system is still available.

SURVEILLANCES

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and valves is deemed to be adequate testing of these systems. The ADS test circuit permits continued surveillance on the operable relief valves to assure that they will be available if required.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3/4.5.F. Minimum Low Pressure Cooling and Diesel Generator Availability
BASES

BACKGROUND

The purpose of Specification 3/4.5.F is to assure that adequate core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only 2 LCPI pumps would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system.

Specification 3.4.F.5 allows removal of one CRD mechanism while the torus is in a drained condition without compromising core cooling capability. The available core cooling capability for a potential draining of the reactor vessel while this work is performed is based on an estimated drain rate of 300 gpm if the control rod blade seal is unseated. Flooding the refuel cavity and dryer/separator pool to elevation 114'-0" corresponds to approximately 350,000 gallons of water and will provide core cooling capability in the event leakage from the control rod drive does occur. A potential draining of the reactor vessel (via control rod blade leakage) would allow this water to enter into the torus and after approximately 140,000 gallons have accumulated (needed to meet minimum NPSH requirements for the LPCI and/or core spray pumps), the torus would be able to serve as a common suction header. This would allow a closed loop operation of the LPCI system and the core spray system (once re-aligned) to the torus. In addition, the other core spray system is lined up to the condensate storage tanks which can supplement the refuel cavity and dryer/separator pool water to provide core flooding, if required.

Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

3/4.5.G. Deleted

B 3/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3/4.5.H. Maintenance of Filled Discharge Pipe
BASES

BACKGROUND

If the discharge piping of the core spray, LPCI system, HPCI, and RCIC are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. An analysis has been done which shows that if a water hammer were to occur at the time at which the system were required, the system would still perform its design function. However, to minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition.

SURVEILLANCE

An acceptable method of ensuring that the lines are full is to vent at the high points. The monthly frequency is based on the gradual nature of void buildup in the ECCS piping, the procedural controls, and operating experience.