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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 197
License No. DPR-57

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility) Facility Operating License No. DPR-57 filed by the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated January 13, 1995, as supplemented by letters dated Arpil 5 and June 20, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(1) and 2.C.(2) of Facility Operating License No. DPR-57 are hereby amended to read as follows:

   (1) Maximum Power Level

   The Georgia Power Company is authorized to operate the facility at steady state reactor core power levels not in excess of 2558 megawatts thermal.

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 197 , 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 prior to startup in Cycle 17.

FOR THE NUCLEAR REGULATORY COMMISSION

W. Russell

William T. Russell, Director
Office of Nuclear Reactor Regulation

Attachment:
1. Technical Specification Changes
2. License Changes

Date of Issuance: August 31, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 197

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the Appendix "A" Technical Specifications and Bases with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. Also replace the following page of the Operating License (OL).

Remove Pages	Insert Pages
1.1-5	1.1-5
3.1-23	3.1-23
3.3-7	3.3-7
3.3-32	3.3-32
3.3-66	3.3-66
3.4-4	3.4-4
3.4-8	3.4-8
3.4-28	3.4-28
3.5-5	3.5-5
3.5-12	3.5-12
B 3.1-44	B 3.1-44
B 3.3-154	B 3.3-154
B 3.3-159	B 3.3-159
B 3.4-4	B 3.4-4
B 3.4-53	B 3.4-53
B 3.4-54	B 3.4-54
B 3.5-3	B 3.5-3
B 3.5-23	B 3.5-23
B 3.6-2	B 3.6-2
B 3.6-7	B 3.6-7
B 3.6-28	B 3.6-28
B 3.7-33	B 3.7-33
B 3.10-1	B 3.10-1
Page 3 (OL)	Page 3 (OL)

(2) Pursuant to the Act and 10 CFR Part 70, Georgia Power Company to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70 Georgia Power Company to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, Georgia Power Company to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, Georgia Power Company to posses, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50-54 and 50-59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Georgia Power Company is authorized to operate the facility at steady state reactor core power levels not in excess of 2558 megawatts thermal.

1.1 Definitions (continued)

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Section 13.6, Startup and Power Test Program, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2558 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.5	Verify the concentration of sodium pentaborate in solution is within the Region A limits of Figure 3.1.7-1.	31 days AND Once within 24 hours after water or sodium pentaborate is added to solution AND Once within 24 hours after solution temperature is restored within the Region A limits of Figure 3.1.7-2
SR 3.1.7.6	Verify each SLC subsystem manual and power operated 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.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate $\geq$ 41.2 gpm at a discharge pressure $\geq$ 1201 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	18 months on a STAGGERED TEST BASIS

(continued)

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.15	≤ 120% RTP
d. Downscale	1	2	F	SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.15	≥ 4.2% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.15	NA
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1085 psig
4. Reactor Vessel Water Level – Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≥ 0 inches
5. Main Steam Isolation Valve – Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 10% closed
6. Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.92 psig

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level — ATWS-RPT Level: ≥ -73 inches; and b. Reactor Steam Dome Pressure — High: ≤ 1175 psig.	18 months
SR 3.3.4.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

Table 3.3.6.3-1 (page 1 of 1)
Low-Low Set Instrumentation

FUNCTION	REQUIRED CHANNELS PER FUNCTION	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Steam Dome Pressure – High	1 per LLS valve	SR 3.3.6.3.1 SR 3.3.6.3.4 SR 3.3.6.3.5 SR 3.3.6.3.6	≤ 1085 psig
2. Low-Low Set Pressure Setpoints	2 per LLS valve	SR 3.3.6.3.1 SR 3.3.6.3.4 SR 3.3.6.3.5 SR 3.3.6.3.6	Low: Open ≤ 1005 psig Close ≤ 857 psig Medium-Low: Open ≤ 1020 psig Close ≤ 872 psig Medium-High: Open ≤ 1035 psig Close ≤ 887 psig High: Open ≤ 1045 psig Close ≤ 897 psig
3. Tailpipe Pressure Switch	2 per S/RV	SR 3.3.6.3.2 SR 3.3.6.3.3 SR 3.3.6.3.5 SR 3.3.6.3.6	≥ 80 psig and ≤ 100 psig

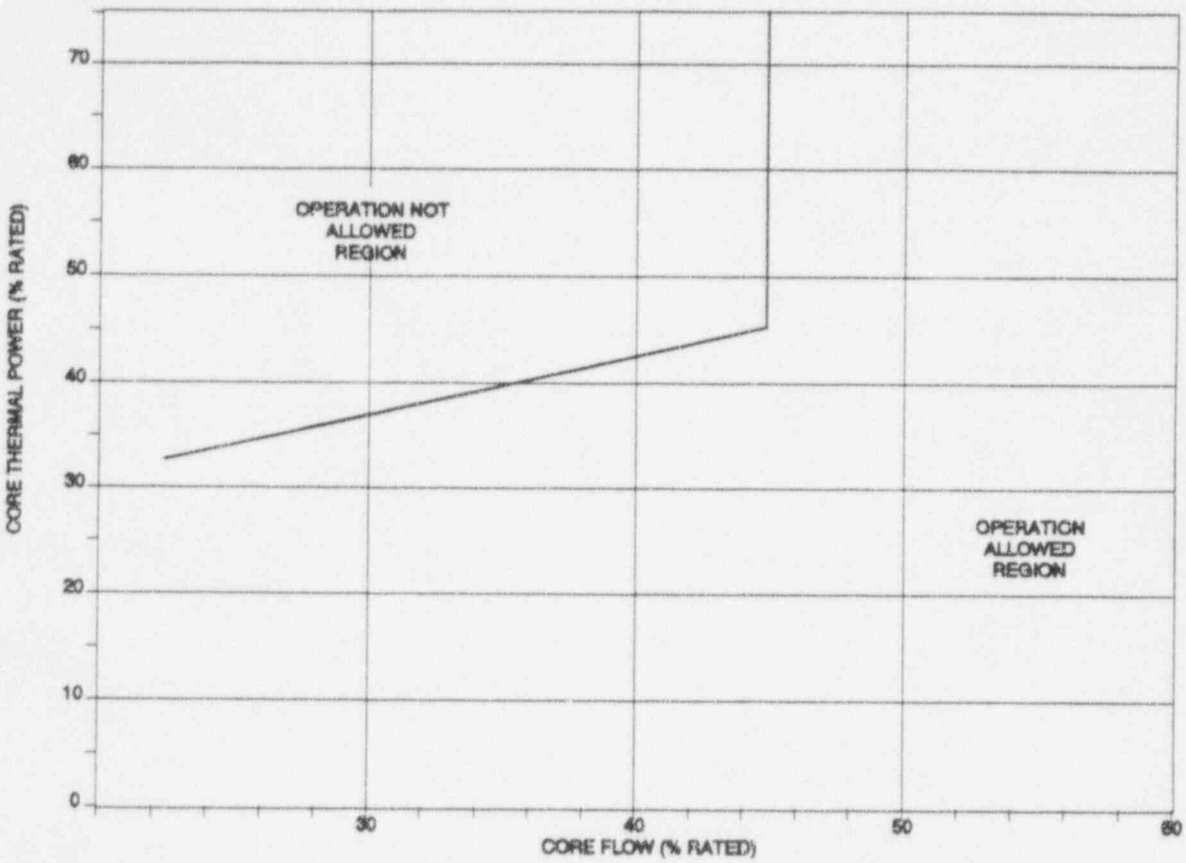

Figure 3.4.1-1 (Page 1 of 1)
Power-Flow Operating Map with One Reactor
Coolant System Recirculation Loop in Operation

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the S/RVs are as follows: Number of S/RVs — Setpoint (psig) 4 — 1110 ± 33.3 4 — 1120 ± 33.6 3 — 1130 ± 33.9 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each S/RV opens when manually actuated.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1058 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1	Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1058 psig.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	-------------------NOTE------------------- Only required to be performed prior to entering MODE 2 from MODE 3 or 4, when in MODE 4 > 48 hours. --- Verify each recirculation pump discharge valve cycles through one complete cycle of full travel or is de-energized in the closed position.	31 days
SR 3.5.1.7	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF CS / ≥ 4250 gpm / 1 / ≥ 113 psig LPCI / ≥ 17,000 gpm / 2 / ≥ 20 psig	In accordance with the Inservice Testing Program
SR 3.5.1.8	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1058 psig and ≥ 920 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.	92 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System 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.	31 days
SR 3.5.3.3	--------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1058 psig and ≥ 920 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	--------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.	18 months

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.4 and SR 3.1.7.6 (continued)

in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This Surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment 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 and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

SR 3.1.7.5

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank (within Region A limits of Figures 3.1.7-1 and 3.1.7-2). SR 3.1.7.5 must be performed anytime sodium pentaborate or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. SR 3.1.7.5 must also be performed any time the temperature is restored to within the Region A limits of Figure 3.1.7-2, to ensure that no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate $\geq$ 41.2 gpm at a discharge pressure $\geq$ 1201 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c. Main Steam Line Flow — High (continued)

detect the high flow. Four channels of Main Steam Line Flow — High Function for each unisolated MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break. The Allowable Value corresponds to ≤ 116 psid, which is the parameter monitored on control room instruments.

This Function isolates the Group 1 valves.

1.d. Condenser Vacuum — Low

The Condenser Vacuum — Low Function is provided to prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. Since the integrity of the condenser is an assumption in offsite dose calculations, the Condenser Vacuum — Low Function is assumed to be OPERABLE and capable of initiating closure of the MSIVs. The closure of the MSIVs is initiated to prevent the addition of steam that would lead to additional condenser pressurization and possible rupture of the diaphragm installed to protect the turbine exhaust hood, thereby preventing a potential radiation leakage path following an accident.

Condenser vacuum pressure signals are derived from four pressure transmitters that sense the pressure in the condenser. Four channels of Condenser Vacuum — Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to prevent damage to the condenser due to pressurization, thereby ensuring its integrity for offsite dose analysis. As noted (footnote (a) to Table 3.3.6.1-1), the channels are not required to be OPERABLE in MODES 2 and 3 when all turbine stop valves (TSVs) are closed, since the potential for condenser

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.a., 4.a. HPCI and RCIC Steam Line Flow — High (continued)

recirculation and MSL breaks. However, these instruments prevent the RCIC or HPCI steam line breaks from becoming bounding.

The HPCI and RCIC Steam Line Flow — High signals are initiated from transmitters (two for HPCI and two for RCIC) that are connected to the system steam lines. Two channels of both HPCI and RCIC Steam Line Flow — High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to be low enough to ensure that the trip occurs to prevent fuel damage and maintains the MSLB event as the bounding event. The Allowable Values correspond to ≤ 228 inches water column for HPCI and ≤ 209 inches water column for RCIC, which are the parameters monitored on control room instruments.

These Functions isolate the Group 3 and 4 valves, as appropriate.

3.b., 4.b. HPCI and RCIC Steam Supply Line Pressure — Low

Low MSL pressure indicates that the pressure of the steam in the HPCI or RCIC turbine may be too low to continue operation of the associated system's turbine. These isolations are for equipment protection and are not assumed in any transient or accident analysis in the FSAR. However, they also provide a diverse signal to indicate a possible system break. These instruments are included in Technical Specifications (TS) because of the potential for risk due to possible failure of the instruments preventing HPCI and RCIC initiations. Therefore, they meet Criterion 4 of the NRC Policy Statement (Ref. 6).

The HPCI and RCIC Steam Supply Line Pressure — Low signals are initiated from transmitters (four for HPCI and four for RCIC) that are connected to the system steam line. Four channels of both HPCI and RCIC Steam Supply Line Pressure — Low Functions are available and are required to

(continued)

BASES

LCO (continued)

and APRM Flow Biased Simulated Thermal Power — High setpoint (LCO 3.3.1.1) must be applied to allow continued operation consistent with the assumptions of Reference 3. In addition, core flow as a function of core thermal power must be in the "Operation Allowed Region" of Figure 3.4.1-1 to ensure core thermal-hydraulic oscillations do not occur.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1 and B.1

Due to thermal-hydraulic stability concerns, operation of the plant with one recirculation loop is controlled by restricting the core flow to ≥ 45% of rated core flow when THERMAL POWER is greater than the 76% rod line. This requirement is based on the recommendations contained in GE SIL-380, Revision 1 (Reference 4), which defines the region where the limit cycle oscillations are more likely to occur. If the core flow as a function of core thermal power is in the "Operation Not Allowed Region" of Figure 3.4.1-1, prompt action should be initiated to restore the flow-power combination to within the Operation Allowed Region. The 2 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing core oscillations to be quickly detected. An immediate reactor scram is also required with no recirculation pumps in operation, since all forced circulation has been lost and the probability of thermal-hydraulic oscillations is greater.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Reactor Steam Dome Pressure

BASES

BACKGROUND	The reactor steam dome pressure is an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria and is also an assumed initial condition of design basis accidents and transients.
APPLICABLE SAFETY ANALYSES	The reactor steam dome pressure of ≤ 1058 psig is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety/relief valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved. Reference 2 also assumes an initial reactor steam dome pressure for the analysis of design basis accidents and transients used to determine the limits for fuel cladding integrity (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"). Reactor steam dome pressure satisfies the requirements of Criterion 2 of the NRC Policy Statement (Ref. 3).
LCO	The specified reactor steam dome pressure limit of ≤ 1058 psig ensures the plant is operated within the assumptions of the overpressure protection analysis. Operation above the limit may result in a response more severe than analyzed.
APPLICABILITY	In MODES 1 and 2, the reactor steam dome pressure is required to be less than or equal to the limit. In these

(continued)

BASES

APPLICABILITY (continued)

MODES, the reactor may be generating significant steam and events which may challenge the overpressure limits are possible.

In MODES 3, 4, and 5, the limit is not applicable because the reactor is shut down. In these MODES, the reactor pressure is well below the required limit, and no anticipated events will challenge the overpressure limits.

ACTIONS

A.1

With the reactor steam dome pressure greater than the limit, prompt action should be taken to reduce pressure to below the limit and return the reactor to operation within the bounds of the analyses. The 15 minute Completion Time is reasonable considering the importance of maintaining the pressure within limits. This Completion Time also ensures that the probability of an accident occurring while pressure is greater than the limit is minimized.

B.1

If the reactor steam dome pressure cannot be restored to within the limit within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Verification that reactor steam dome pressure is ≤ 1058 psig ensures that the initial conditions of the vessel overpressure protection analysis is met. Operating experience has shown the 12 hour Frequency to be sufficient for identifying trends and verifying operation within safety analyses assumptions.

(continued)

BASES

BACKGROUND (continued)	pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling." Two LPCI inverters (one per subsystem) are designed to provide the power to various LPCI subsystem valves (e.g., inboard injection valves). This will ensure that a postulated worst case single active component failure, during a design basis loss of coolant accident (which includes loss of offsite power), would not result in the low pressure ECCS subsystems failing to meet their design function. (While an alternate power supply is available, the low pressure ECCS subsystems may not be capable of meeting their design function if the alternate power supply is in service.)

The HPCI System (Ref. 3) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for HPCI is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1154 psig). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open simultaneously and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in

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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the High Pressure Coolant Injection (HPCI) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from a main steam line upstream of the associated inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1154 psig). Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_a) is 1.2% by weight of the containment air per 24 hours at the maximum peak containment pressure (P_a) of 49.6 psig (Ref. 1).

Primary containment satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).

LCO

Primary containment OPERABILITY is maintained by limiting leakage to less than L_a, except prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test. At this time, the combined Type B and C leakage must be $< 0.6\ L_a$, and the overall Type A leakage must be $< 0.75\ L_a$. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

(continued)

BASES

BACKGROUND (continued)	containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.
APPLICABLE SAFETY ANALYSES	The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 1.2% by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (P_a) of 49.6 psig (Ref. 2). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock. Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment. The primary containment air lock satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).
LCO	As part of primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event. The primary containment air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be opened at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Pressure

BASES

BACKGROUND

The drywell pressure is limited during normal operations to preserve the initial conditions assumed in the accident analysis for a Design Basis Accident (DBA) or loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs (Ref. 1). Among the inputs to the DBA is the initial primary containment internal pressure (Ref. 1). Analyses assume an initial drywell pressure of 1.75 psig. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell internal pressure does not exceed the maximum allowable of 62 psig.

The maximum calculated drywell pressure occurs during the reactor blowdown phase of the DBA, which assumes an instantaneous recirculation line break. The calculated peak drywell pressure for this limiting event is 49.6 psig (Ref. 1).

Drywell pressure satisfies Criterion 2 of the NRC Policy Statement (Ref. 2).

LCO

In the event of a DBA, with an initial drywell pressure ≤ 1.75 psig, the resultant peak drywell accident pressure will be maintained below the drywell design pressure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell pressure within limits is not required in MODE 4 or 5.

(continued)

BASES

LCO (continued)	with this requirement (2436 MWt x 100 μCi/MWt-second = 240 mCi/second). The 240 mCi/second limit is conservative for a rated core thermal power of 2558 MWt.

APPLICABILITY

The LCO is applicable when steam is being exhausted to the main condenser and the resulting noncondensibles are being processed via the Main Condenser Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, steam is not being exhausted to the main condenser and the requirements are not applicable.

ACTIONS

A.1

If the offgas radioactivity rate limit is exceeded, 72 hours is allowed to restore the gross gamma activity rate to within the limit. The 72 hour Completion Time is reasonable, based on engineering judgment, the time required to complete the Required Action, the large margins associated with permissible dose and exposure limits, and the low probability of a Main Condenser Offgas System rupture.

B.1, B.2, B.3.1, and B.3.2

If the gross gamma activity rate is not restored to within the limits in the associated Completion Time, all main steam lines or the SJAE must be isolated. This isolates the Main Condenser Offgas System from the source of the radioactive steam. The main steam lines are considered isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain valve in the drain line is closed. The 12 hour Completion Time is reasonable, based on operating experience, to perform the actions from full power conditions in an orderly manner and without challenging unit systems.

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The

(continued)

B 3.10 SPECIAL OPERATIONS

B 3.10.1 Inservice Leak and Hydrostatic Testing Operation

BASES

BACKGROUND

The purpose of this Special Operations LCO is to allow certain reactor coolant pressure tests to be performed in MODE 4 when the metallurgical characteristics of the reactor pressure vessel (RPV) require the pressure testing at temperatures > 212°F (normally corresponding to MODE 3).

System hydrostatic testing and system leakage (same as inservice leakage tests) pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 1) are performed prior to the reactor going critical after a refueling outage. Inservice system leakage tests are performed at the end of each refueling outage with the system set for normal power operation. Some parts of the Class 1 boundary are not pressurized during these system tests. System hydrostatic tests are required once per interval and include all the Class 1 boundary unless the test is broken into smaller portions. Recirculation pump operation and a water solid RPV (except for an air bubble for pressure control) are used to achieve the necessary temperatures and pressures required for these tests. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.9, "Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits." These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. The hydrostatic test requires increasing pressure to approximately 1139 psig. The system leakage test requires increasing pressure to approximately 1035 psig.

With increased reactor vessel fluence over time, the minimum allowable vessel temperature increases at a given pressure. Periodic updates to the RCS P/T limit curves are performed as necessary, based upon the results of analyses of irradiated surveillance specimens removed from the vessel.

(continued)

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 138
License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Facility Operating License No. NPF-5 filed by the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated January 13, 1995, as supplemented by letters dated April 5 and June 20, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2. C.(1) and 2.C.(2) of Facility Operating License No. NPF-5 are hereby amended to read as follows:

   (1) Maximum Power Level

   The Georgia Power Company is authorized to operate the facility at steady state reactor core power levels not in excess of 2558 megawatts thermal.

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 138 , 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 prior to startup in Cycle 13.

FOR THE NUCLEAR REGULATORY COMMISSION

W. Russell

William T. Russell, Director
Office of Nuclear Reactor Regulation

Attachment:
1. Technical Specification
   Changes
2. License Changes

Date of Issuance: August 31, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 138

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the Appendix "A" Technical Specifications and Bases with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. Also replace the following page of the Operating License (OL).

Remove Pages	Insert Pages
1.1-5	1.1-5
3.1-23	3.1-23
3.3-8	3.3-8
3.3-33	3.3-33
3.3-67	3.3-67
3.4-4	3.4-4
3.4-8	3.4-8
3.4-28	3.4-28
3.5-5	3.5-5
3.5-12	3.5-12
B 3.1-44	B 3.1-44
B 3.3-154	B 3.3-154
B 3.3-159	B 3.3-159
B 3.4-4	B 3.4-4
B 3.4-53	B 3.4-53
B 3.4-54	B 3.4-54
B 3.5-3	B 3.5-3
B 3.5-23	B 3.5-23
B 3.6-2	B 3.6-2
B 3.6-7	B 3.6-7
B 3.6-29	B 3.6-29
B 3.7-33	B 3.7-33
B 3.10-1	B 3.10-1
Page 4 (OL)	Page 4 (OL)

(1) Maximum Power Level

Georgia Power Company is authorized to operate the facility at steady state reactor core power levels not in excess of 2558 megawatts thermal in accordance with the conditions specified herein and in Attachment 2 to this license. Attachment 2 is an integral part of this license.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 138 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

The Surveillance Requirements (SRs) contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment No. 135. The SRs listed below shall be successfully demonstrated prior to the time and condition specified below for each:

a) SRs 3.3.2.2.2, 3.3.2.2.3, 3.3.3.2.2, 3.3.8.1.4, 3.6.2.4.2, 3.7.7.2, and 3.7.7.3 shall be successfully demonstrated prior to entering MODE 2 on the first plant startup following the twelfth refueling outage;

b) SRs 3.8.1.8, 3.8.1.9 (for DG 2C), 3.8.1.10, 3.8.1.12, 3.8.1.13, 3.8.1.17 (for DG 2C), and 3.8.1.18 shall be successfully demonstrated at their next regularly scheduled performance;

c) SRs 3.6.4.1.3 and 3.6.4.1.4 will be met at implementation for the second containment configuration in effect at that time. The SRs shall be successfully demonstrated for the other secondary containment configurations prior to the plant entering the LCO applicability for that configuration.

1.1 Definitions

MINIMUM CRITICAL POWER RATIO (MCPR) (continued)	appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE — OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Tests and Operation, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2558 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.5	Verify the concentration of sodium pentaborate in solution is within the Region A limits of Figure 3.1.7-1.	31 days AND Once within 24 hours after water or sodium pentaborate is added to solution AND Once within 24 hours after solution temperature is restored within the Region A limits of Figure 3.1.7-2
SR 3.1.7.6	Verify each SLC subsystem manual and power operated 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.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate $\geq$ 41.2 gpm at a discharge pressure $\geq$ 1201 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	18 months on a STAGGERED TEST BASIS

(continued)

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.15 SR 3.3.1.1.16	≤ 120% RTP
d. Downscale	1	2	F	SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.15	≥ 4.2% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.15	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16	≤ 1085 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16	≥ 0 inches
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.92 psig

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level — ATWS-RPT Level: ≥ -73 inches; and b. Reactor Steam Dome Pressure — High: ≤ 1175 psig.	18 months
SR 3.3.4.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

Table 3.3.6.3-1 (page 1 of 1)
Low-Low Set Instrumentation

FUNCTION	REQUIRED CHANNELS PER FUNCTION	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Steam Dome Pressure — High	1 per LLS valve	SR 3.3.6.3.1 SR 3.3.6.3.4 SR 3.3.6.3.5 SR 3.3.6.3.6	≤ 1085 psig
2. Low-Low Set Pressure Setpoints	2 per LLS valve	SR 3.3.6.3.1 SR 3.3.6.3.4 SR 3.3.6.3.5 SR 3.3.6.3.6	Low: Open ≤ 1010 psig Close ≤ 860 psig Medium-Low: Open ≤ 1025 psig Close ≤ 875 psig Medium-High: Open ≤ 1040 psig Close ≤ 890 psig High: Open ≤ 1050 psig Close ≤ 900 psig
3. Tailpipe Pressure Switch	2 per S/RV	SR 3.3.6.3.2 SR 3.3.6.3.3 SR 3.3.6.3.5 SR 3.3.6.3.6	≥ 80 psig and ≤ 100 psig

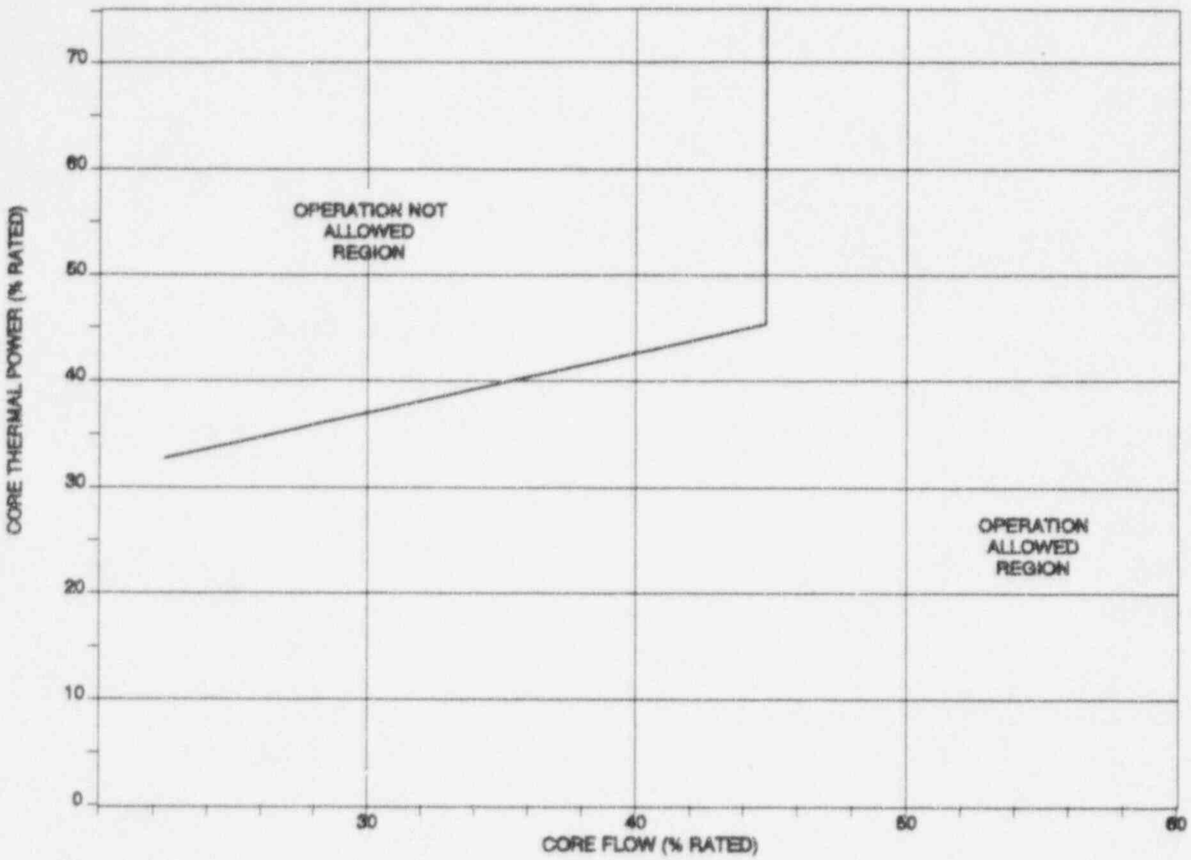

Figure 3.4.1-1 (Page 1 of 1)
Power-Flow Operating Map with One Reactor
Coolant System Recirculation Loop in Operation

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the S/RVs are as follows: Number of S/RVs — Setpoint (psig) 4 — 1120 ± 33.6 4 — 1130 ± 33.9 3 — 1140 ± 34.2 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	--------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify each S/RV opens when manually actuated.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1020 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1	Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1058 psig.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	-------------------NOTE------------------- Only required to be performed prior to entering MODE 2 from MODE 3 or 4, when in MODE 4 > 48 hours. -- Verify each recirculation pump discharge valve cycles through one complete cycle of full travel or is de-energized in the closed position.	31 days
SR 3.5.1.7	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF CS / $\geq$ 4250 gpm / 1 / $\geq$ 113 psig LPCI / $\geq$ 17,000 gpm / 2 / $\geq$ 20 psig	In accordance with the Inservice Testing Program
SR 3.5.1.8	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure $\leq$ 1058 psig and $\geq$ 920 psig, the HPCI pump can develop a flow rate $\geq$ 4250 gpm against a system head corresponding to reactor pressure.	92 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System 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.	31 days
SR 3.5.3.3	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 1058 psig and ≥ 920 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.	18 months

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.4 and SR 3.1.7.6 (continued)

in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This Surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment 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 and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

SR 3.1.7.5

This Surveillance requires an ex mination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank (within Region A limits of Figures 3.1.7-1 and 3.1.7-2). SR 3.1.7.5 must be performed anytime sodium pentaborate or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. SR 3.1.7.5 must also be performed any time the temperature is restored to within the Region A limits of Figure 3.1.7-2, to ensure that no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1201 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c. Main Steam Line Flow — High (continued)

detect the high flow. Four channels of Main Steam Line Flow — High Function for each unisolated MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break. The Allowable Value corresponds to ≤ 145 psid, which is the parameter monitored on control room instruments.

This Function isolates the Group 1 valves.

1.d. Condenser Vacuum — Low

The Condenser Vacuum — Low Function is provided to prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. Since the integrity of the condenser is an assumption in offsite dose calculations, the Condenser Vacuum — Low Function is assumed to be OPERABLE and capable of initiating closure of the MSIVs. The closure of the MSIVs is initiated to prevent the addition of steam that would lead to additional condenser pressurization and possible rupture of the diaphragm installed to protect the turbine exhaust hood, thereby preventing a potential radiation leakage path following an accident.

Condenser vacuum pressure signals are derived from four pressure transmitters that sense the pressure in the condenser. Four channels of Condenser Vacuum — Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to prevent damage to the condenser due to pressurization, thereby ensuring its integrity for offsite dose analysis. As noted (footnote (a) to Table 3.3.6.1-1), the channels are not required to be OPERABLE in MODES 2 and 3 when all turbine stop valves (TSVs) are closed, since the potential for condenser

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.a., 4.a. HPCI and RCIC Steam Line Flow — High (continued)

recirculation and MSL breaks. However, these instruments prevent the RCIC or HPCI steam line breaks from becoming bounding.

The HPCI and RCIC Steam Line Flow — High signals are initiated from transmitters (two for HPCI and two for RCIC) that are connected to the system steam lines. Two channels of both HPCI and RCIC Steam Line Flow — High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to be low enough to ensure that the trip occurs to prevent fuel damage and maintains the MSLB event as the bounding event. The Allowable Values correspond to ≤ 212 inches water column for HPCI and ≤ 153 inches water column for RCIC, which are the parameters monitored on control room instruments.

These Functions isolate the Group 3 and 4 valves, as appropriate.

3.b., 4.b. HPCI and RCIC Steam Supply Line Pressure — Low

Low MSL pressure indicates that the pressure of the steam in the HPCI or RCIC turbine may be too low to continue operation of the associated system's turbine. These isolations are for equipment protection and are not assumed in any transient or accident analysis in the FSAR. However, they also provide a diverse signal to indicate a possible system break. These instruments are included in Technical Specifications (TS) because of the potential for risk due to possible failure of the instruments preventing HPCI and RCIC initiations. Therefore, they meet Criterion 4 of the NRC Policy Statement (Ref. 7).

The HPCI and RCIC Steam Supply Line Pressure — Low signals are initiated from transmitters (four for HPCI and four for RCIC) that are connected to the system steam line. Four channels of both HPCI and RCIC Steam Supply Line Pressure — Low Functions are available and are required to

(continued)

BASES

LCO (continued)

and APRM Flow Biased Simulated Thermal Power — High setpoint (LCO 3.3.1.1) must be applied to allow continued operation consistent with the assumptions of Reference 3. In addition, core flow as a function of core thermal power must be in the "Operation Allowed Region" of Figure 3.4.1-1 to ensure core thermal-hydraulic oscillations do not occur.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1 and B.1

Due to thermal-hydraulic stability concerns, operation of the plant with one recirculation loop is controlled by restricting the core flow to ≥ 45% of rated core flow when THERMAL POWER is greater than the 76% rod line. This requirement is based on the recommendations contained in GE SIL-380, Revision 1 (Reference 4), which defines the region where the limit cycle oscillations are more likely to occur. If the core flow as a function of core thermal power is in the "Operation Not Allowed Region" of Figure 3.4.1-1, prompt action should be initiated to restore the flow-power combination to within the Operation Allowed Region. The 2 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing core oscillations to be quickly detected. An immediate reactor scram is also required with no recirculation pumps in operation, since all forced circulation has been lost and the probability of thermal-hydraulic oscillations is greater.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Reactor Steam Dome Pressure

BASES

BACKGROUND	The reactor steam dome pressure is an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria and is also an assumed initial condition of design basis accidents and transients.
APPLICABLE SAFETY ANALYSES	The reactor steam dome pressure of ≤ 1058 psig is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety/relief valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved. Reference 2 also assumes an initial reactor steam dome pressure for the analysis of design basis accidents and transients used to determine the limits for fuel cladding integrity (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"). Reactor steam dome pressure satisfies the requirements of Criterion 2 of the NRC Policy Statement (Ref. 3).
LCO	The specified reactor steam dome pressure limit of ≤ 1058 psig ensures the plant is operated within the assumptions of the overpressure protection analysis. Operation above the limit may result in a response more severe than analyzed.
APPLICABILITY	In MODES 1 and 2, the reactor steam dome pressure is required to be less than or equal to the limit. In these

(continued)

BASES

APPLICABILITY (continued)

MODES, the reactor may be generating significant steam and events which may challenge the overpressure limits are possible.

In MODES 3, 4, and 5, the limit is not applicable because the reactor is shut down. In these MODES, the reactor pressure is well below the required limit, and no anticipated events will challenge the overpressure limits.

ACTIONS

A.1

With the reactor steam dome pressure greater than the limit, prompt action should be taken to reduce pressure to below the limit and return the reactor to operation within the bounds of the analyses. The 15 minute Completion Time is reasonable considering the importance of maintaining the pressure within limits. This Completion Time also ensures that the probability of an accident occurring while pressure is greater than the limit is minimized.

B.1

If the reactor steam dome pressure cannot be restored to within the limit within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Verification that reactor steam dome pressure is ≤ 1058 psig ensures that the initial conditions of the vessel overpressure protection analysis is met. Operating experience has shown the 12 hour Frequency to be sufficient for identifying trends and verifying operation within safety analyses assumptions.

(continued)

BACKGROUND (continued)

pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling." Two LPCI inverters (one per subsystem) are designed to provide the power to various LPCI subsystem valves (e.g., inboard injection valves). This will ensure that a postulated worst case single active component failure, during a design basis loss of coolant accident (which includes loss of offsite power), would not result in the low pressure ECCS subsystems failing to meet their design function. (While an alternate power supply is available, the low pressure ECCS subsystems may not be capable of meeting their design function if the alternate power supply is in service.)

The HPCI System (Ref. 3) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for HPCI is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (162 psid to 1169 psid, vessel to pump suction). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open simultaneously and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in

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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the High Pressure Coolant Injection (HPCI) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from a main steam line upstream of the associated inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1154 psig). Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_a) is 1.2% by weight of the containment air per 24 hours at the maximum peak containment pressure (P_a) of 45.5 psig (Ref. 1).

Primary containment satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).

LCO

Primary containment OPERABILITY is maintained by limiting leakage to less than L_a, except prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test. At this time, the combined Type B and C leakage must be < 0.6 L_a, and the overall Type A leakage must be < 0.75 L_a. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

(continued)

BASES

BACKGROUND (continued)	containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.
APPLICABLE SAFETY ANALYSES	The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 1.2% by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (P_a) of 45.5 psig (Ref. 2). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock. Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment. The primary containment air lock satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).
LCO	As part of primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event. The primary containment air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be opened at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Pressure

BASES

BACKGROUND	The drywell pressure is limited during normal operations to preserve the initial conditions assumed in the accident analysis for a Design Basis Accident (DBA) or loss of coolant accident (LOCA).
APPLICABLE SAFETY ANALYSES	Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs (Ref. 1). Among the inputs to the DBA is the initial primary containment internal pressure (Ref. 1). Analyses assume an initial drywell pressure of 1.75 psig. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell internal pressure does not exceed the maximum allowable of 62 psig. The maximum calculated drywell pressure occurs during the reactor blowdown phase of the DBA, which assumes an instantaneous recirculation line break. The calculated peak drywell pressure for this limiting event is 45.5 psig (Ref. 1). Drywell pressure satisfies Criterion 2 of the NRC Policy Statement (Ref. 2).
LCO	In the event of a DBA, with an initial drywell pressure ≤ 1.75 psig, the resultant peak drywell accident pressure will be maintained below the drywell design pressure.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell pressure within limits is not required in MODE 4 or 5.

(continued)

BASES

LCO (continued)

with this requirement (2436 MWt x 100 μCi/MWt-second = 240 mCi/second). The 240 mCi/second limit is conservative for a rated core thermal power of 2558 MWt.

APPLICABILITY

The LCO is applicable when steam is being exhausted to the main condenser and the resulting noncondensibles are being processed via the Main Condenser Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, steam is not being exhausted to the main condenser and the requirements are not applicable.

ACTIONS

A.1

If the offgas radioactivity rate limit is exceeded, 72 hours is allowed to restore the gross gamma activity rate to within the limit. The 72 hour Completion Time is reasonable, based on engineering judgment, the time required to complete the Required Action, the large margins associated with permissible dose and exposure limits, and the low probability of a Main Condenser Offgas System rupture.

B.1, B.2, B.3.1, and B.3.2

If the gross gamma activity rate is not restored to within the limits in the associated Completion Time, all main steam lines or the SJAE must be isolated. This isolates the Main Condenser Offgas System from the source of the radioactive steam. The main steam lines are considered isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain valve in the drain line is closed. The 12 hour Completion Time is reasonable, based on operating experience, to perform the actions from full power conditions in an orderly manner and without challenging unit systems.

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The

(continued)

B 3.10 SPECIAL OPERATIONS

B 3.10.1 Inservice Leak and Hydrostatic Testing Operation

BASES

BACKGROUND

The purpose of this Special Operations LCO is to allow certain reactor coolant pressure tests to be performed in MODE 4 when the metallurgical characteristics of the reactor pressure vessel (RPV) require the pressure testing at temperatures > 212°F (normally corresponding to MODE 3).

System hydrostatic testing and system leakage (same as inservice leakage tests) pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 1) are performed prior to the reactor going critical after a refueling outage. Inservice system leakage tests are performed at the end of each refueling outage with the system set for normal power operation. Some parts of the Class 1 boundary are not pressurized during these system tests. System hydrostatic tests are required once per interval and include all the Class 1 boundary unless the test is broken into smaller portions. Recirculation pump operation and a water solid RPV (except for an air bubble for pressure control) are used to achieve the necessary temperatures and pressures required for these tests. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.9, "Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits." These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. The hydrostatic test requires increasing pressure to approximately 1139 psig. The system leakage test requires increasing pressure to approximately 1035 psig.

With increased reactor vessel fluence over time, the minimum allowable vessel temperature increases at a given pressure. Periodic updates to the RCS P/T limit curves are performed as necessary, based upon the results of analyses of irradiated surveillance specimens removed from the vessel.

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