August 30, 1990

Docket Nos. 50-321
and 50-366

Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
Georgia Power Company
P. O. Box 1295
Birmingham, AL 35201

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE DPR-57 AND AMENDMENT NO.108 TO FACILITY OPERATING LICENSE NPF-5, EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TACS 76461 AND 76462)

The Nuclear Regulatory Commission has issued the enclosed Amendment No.170 to Facility Operating License No. DPR-57 and Amendment No.108 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 2, 1990.

The amendments make a number of miscellaneous changes to the TSs for Units 1 and 2.

A copy of the related Safety Evaluation supporting the amendments is also enclosed. Notice of issuance of the amendments will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

Lawrence P. Crocker, Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 170 to DPR-57
2. Amendment No. 108 to NPF-5
3. Safety Evaluation

cc w/enclosures:
See next page
*See previous concurrence

OFC	*LA:PDII-3	*PM:PDII-3	PM:PDII-3	OGC	D:PDII-3	
NAME	RIngram	LCrocker	FRinaldi	L. Dewey	DMatthews	
DATE	7/26/90	7/26/90	8/14/90	8/15/90	8/30/90	

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Mr. W. G. Hairston, III
Georgia Power Company

Edwin I. Hatch Nuclear Plant,
Units Nos. 1 and 2

cc:
Mr. Ernest L. Blake, Jr.
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N.W.
Washington, D.C. 20037

Mr. J. T. Beckham
Vice President - Plant Hatch
Georgia Power Company
P.O. Box 1295
Birmingham, Alabama 35201

Mr. S. J. Bethay
Manager Licensing - Hatch
Georgia Power Company
P.O. Box 1295
Birmingham, Alabama 35201

Mr. H. C. Nix
General Manager, Nuclear Plant
Georgia Power Company
Route 1, Box 439
Baxley, Georgia 31513

Resident Inspector
U.S. Nuclear Regulatory Commission
Route 1, Box 725
Baxley, Georgia 31513

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 2900
Atlanta, Georgia 30323

Mr. Charles H. Badger
Office of Planning and Budget
Room 610
270 Washington Street, S.W.
Atlanta, Georgia 30334

Mr. J. Leonard Ledbetter, Director
Environmental Protection Division
Department of Natural Resources
205 Butler Street, S.E., Suite 1252
Atlanta, Georgia 30334

Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

Mr. R. P. McDonald
Executive Vice President -
Nuclear Operations
Georgia Power Company
P.O. Box 1295
Birmingham, Alabama 35201

Mr. Alan R. Herdt, Chief
Project Branch #3
U.S. Nuclear Regulatory Commission
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Mr. Dan Smith
Program Director of
Power Production
Oglethorpe Power Corporation
100 Crescent Centre
Tucker, Georgia 30085

Charles A. Patrizia, Esq.
Paul, Hastings, Janofsky & Walker
12th Floor
1050 Connecticut Avenue, N.W.
Washington, D.C. 20036

DATED August 30, 1990

AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE DPR-57, EDWIN I. HATCH, UNIT 1
AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NPF-5, EDWIN I. HATCH, UNIT 2

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

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. 170
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 Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, (the licensee) dated March 2, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-57 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 170, 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: August 30, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 170

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of Appendices "A" and "B" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Appendix A

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Appendix B

Remove Page	Insert Page
5-3	5-3

*Overleaf page provided to maintain document completeness.

GG. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

HH. Start & Hot Standby Mode - The reactor is in the Start & Hot Standby Mode when the Mode Switch is in the START & HOT STANDBY position. In this mode the reactor protection system is energized with IRM and APRM (Start & Hot Standby Mode) neutron monitoring system trips and control rod withdrawal inter-locks in service.

II. Surveillance Frequency - Periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. The operating cycle interval is defined as 18 months. In the case where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

JJ. Surveillance Requirements - The surveillance requirements are requirements established to ensure that the LCO stated in Section 3 of these Technical Specifications are met. Performance of a surveillance requirement within the specified surveillance interval shall constitute compliance with the operability requirement for an LCO. Surveillance requirements are not required on systems or parts of systems that are not required to be operable or are tripped. If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status.

KK. Total Peaking Factor (TPF) - The total peaking factor is the highest product of radial, axial, and local peaking factors simultaneously operative at any segment of fuel rod.

LL. Transition Boiling - Transition boiling is the boiling that occurs between nucleate and film boiling. Transition boiling is manifested by an unstable fuel cladding surface temperature, rising suddenly as steam blanketing of the heat transfer surface occurs, then dropping as the steam blanket is swept away by the coolant flow, then rising again.

3.1. REACTOR PROTECTION SYSTEM (RPS)

Applicability

The Limiting Conditions for Operation associated with the Reactor Protection System apply to the instrumentation and associated devices which initiate a reactor scram.

Objective

The objective of the Limiting Conditions for Operation is to assure the operability of the Reactor Protection System.

Specifications

A. Sources of a Trip Signal Which Initiate a Reactor Scram

The instrumentation requirements associated with each source of a scram signal shall be as given in Table 3.1-1.

The action to be taken if the number of operable channels is not met for both trip systems is also given in Table 3.1-1.

B. Core Maximum Fraction of Limiting Power Density (CMFLPD)

This section deleted.

4.1. REACTOR PROTECTION SYSTEM (RPS)

Applicability

The Surveillance Requirements associated with the Reactor Protection System apply to the instrumentation and associated devices which initiate a reactor scram.

Objective

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the protection instrumentation to assure operability.

Specifications

A. Test and Calibration Requirements for the RPS

RPS instrumentation systems and associated systems shall be functionally tested and calibrated as indicated in Table 4.1-1.

The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels.

B. Core Maximum Fraction of Limiting Power Density (CMFLPD)

This section deleted.

Table 4.1-1 (Cont.)

Scram Number (a)	Source of Scram Trip Signal	Group (b)	Instrument Check Minimum Frequency	Instrument Functional Test Minimum Frequency (c)	Instrument Calibration Minimum Frequency
9	Main Steam Line High Radiation	B	D	Every 3 months (e)	Every 3 months(i)
10	Main Steam Line Isolation Valve Closure	A	NA	Every 3 months	(h)
11	Turbine Control Valve Fast Closure	A	NA	Every 3 months (j)	Once/Operating Cycle (k)
12	Turbine Stop Valve Closure	A	NA	Every 3 months	(h)
	RPS Channel Switch	A	NA	Once/Operating Cycle	Not Applicable
	Turbine First Stage Pressure Permissive	A	NA	Every 3 months	Every 6 months

a. The column entitled "Scram Number" is for convenience so that a one-to-one relationship can be established between items in Table 4.1-1 and items in Table 3.1-1.

b. The definition for each of the four groups is as follows:

Group A. On-off sensors that provide a scram trip signal.
Group B. Analog devices coupled with bi-stable trips that provide a scram trip signal.
Group C. Devices which only serve a useful function during some restricted mode of operation, such as startup or shutdown, or for which the only practical test is one that can be performed at shutdown.
Group D. Analog transmitters and trip units that provide a scram trip function.

c. Functional tests are not required when the systems are not required to be operable or are tripped. However, if functional tests are missed, they shall be performed prior to returning the systems to an operable status.

d. Calibrations are not required when the systems are not required to be operable or are tripped. However, if calibrations are missed, they shall be performed prior to returning the system to an operable status.

e. This instrumentation is exempted from the instrument functional test definition. This instrument functional test will consist of injecting a simulated electrical signal into the measurement channels.

f. Deleted

g. The water level in the reactor will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every 3 months after completion of the functional test program.

h. Physical inspection and actuation of these position switches will be performed once per operating cycle.

i. Standard current source used which provides an instrument channel alignment. Calibration using a radiation source shall be made once per operating cycle.

j. Measure time interval from EHC pressure switch actuation to RPS relay K14 de-energization.

Notes for Table 3.2-2 (Cont.)

b. When any CCCS subsystem is required to be operable by Section 3.5, there shall be two operable trip systems. If the required number of operable channels cannot be met for one of the trip systems, place the inoperable channel in the tripped condition or declare the associated CCCS inoperable within 1 hour. If the required number of operable channels cannot be met for both trip systems, declare the associated CCCS inoperable within 1 hour.

NOTES FOR TABLE 3.2-3

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-3 and items in Table 4.2-3.

b. When any CCCS subsystem is required to be operable by Section 3.5, there shall be two operable trip systems. If the required number of operable channels cannot be met for one of the trip systems, place the inoperable channel in the tripped condition or declare the associated CCCS inoperable within 1 hour. If the required number of operable channels cannot be met for both trip systems, declare the associated CCCS inoperable within 1 hour.

Table 3.2-4

INSTRUMENTATION WHICH INITIATES OR CONTROLS ADS

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1.	Reactor Vessel Water Level	Low (Level 3)	1	≥10.0 inches	Confirms low level, ADS permissive
	Reactor Vessel Water Level	Low Low Low (Level 1)	2	≥-113 inches	Permissive signal to ADS timer
2.	Drywell Pressure	High	2	≤1.92 psig	Permissive signal to ADS timer
3.	RHR Pump Discharge Pressure	High	2	≥112 psig	Permissive signal to ADS timer
4.	CS Pump Discharge Pressure	High	2	≥137 psig	Permissive signal to ADS timer
5.	Auto Depressurization Low Water Level Timer		2	≤13 minutes	Bypasses high drywell pressure permissive upon sustained Level 1
6.	Auto Depressurization Timer		1	120 ± 12 seconds	With Level 3 and Level 1 and high drywell pressure and CS or RHR pump at pressure, timing sequence begins. If the ADS timer is not reset it will initiate ADS.
7.	Automatic Blowdown Control Power Failure Monitor		1	Not applicable	Monitors availability of power to logic system

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-4 and items in Table 4.2-4.

b. When any CCCS subsystem is required to be operable by Section 3.5, there shall be two operable trip systems. If the required number of operable channels cannot be met for one of the trip systems, place the inoperable channel in the tripped condition or declare the associated CCCS inoperable within 1 hour. If the required number of operable channels cannot be met for both trip systems, declare the associated CCCS inoperable within 1 hour.

Notes for Table 3.2-5

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-5 and item in Table 4.2-5.

b. When any CCCS subsystem is required to be operable by Section 3.5, there shall be two operable trip systems. If the required number of operable channels cannot be met for one of the trip systems, place the inoperable channel in the tripped condition or declare the associated CCCS inoperable within 1 hour. If the required number of operable channels cannot be met for both trip systems, declare the associated CCCS inoperable within 1 hour.

Table 3.2-6

INSTRUMENTATION WHICH INITIATES OR CONTROLS CORE SPRAY

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1.	Reactor Vessel Water Level	Low Low Low (Level 1)	2	≥-113 inches	Initiates CS.
2.	Drywell Pressure	High	2	≤1.92 psig	Initiates CS. Also initiates HPCI and LPCI mode of RHR and provides a permissive signal to ADS.
3.	Reactor Vessel Steam Dome Pressure	Low	2	≥422 psig*	Permissive to open CS injection valves.
4.	Core Spray Sparger Differential Pressure		1[c]	≤ 3.1 psid greater (less negative) than the normal indicated ΔP at rated core power and flow.	Monitors integrity of CS piping inside vessel (between the nozzle and core shroud).
5.	CS Pump Discharge Flow	Low	1	≥610 gpm (≥ 4.13 inches)	Minimum flow bypass line is closed when low flow signal is not present.
6.	Core Spray Logic Power Failure Monitor		1	Not Applicable	Monitors availability of power to logic system.

*This trip function shall be ≤500 psig.

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-6 and items in Table 4.2-6.

b. When any CCCS subsystem is required to be operable by Section 3.5, there shall be two operable trip systems. If the required number of operable channels cannot be met for one of the trip systems, place the inoperable channel in the tripped condition or declare the associated CCCS inoperable within 1 hour. If the required number of operable channels cannot be met for both trip systems, declare the associated CCCS inoperable within 1 hour.

c. Alarm only. When inoperable, verify that the core spray differential pressure is within limits at least once per 12 hours or, declare the associated core spray loop inoperable.

Table 3.2-8

RADIATION MONITORING SYSTEMS WHICH LIMIT RADIOACTIVITY RELEASE

Ref. No. (a)	Instrument	Trip Condition Nomencla-ture	Required Operable Channels per Trip System (b)	Trip Setting	Action to be taken if there are not two operable or tripped trip systems	Remarks
1.	Off-gas Post Treatment Radiation Monitors	Upscale/ Downscale	1	At a value not to exceed the equivalent of the stack re-lease limit indicated in Environmental Tech Specs	(c) (d)	2 upscales, or 1 downscale and 1 upscale, or 2 down-scales will isolate the SJAE off-gas
2.	Refueling Floor Exhaust Vent Radiation Monitors	Upscale	2	At a value not to exceed the equivalent of the stack re-lease limit indicated in Environmental Tech Specs	Cease refueling opera-tions, if in progress. Isolate the secondary containment and start the standby gas treat-ment system.	2 upscale will isolate the secondary containment and initiate the standby gas treatment system
3.	Reactor Bldg. Exhaust Vent Radiation Monitors	Upscale	2	≤20 mr/hr	Isolate the secondary containment, start stand-by gas treatment system, close primary contain-ment and vent valves.	2 upscale will isolate the secondary con-tainment and initiate the standby gas treatment system.
4.	Control Room Intake Radiation Monitors	Downscale Hi	1	≥0.015 mr/hr ≤1.0 mr/hr	Refer to Specifications 3.12.C. and 3.12.D.	1 upscale or 2 down-scales will actuate the MCRECS in the control room pres-surization mode.

Notes for Table 4.2-8 (Cont'd)

c. Instrument functional tests are not required when the instruments are not required to be operable or are tripped. However, if instrument functional tests are missed, they shall be performed prior to returning the instrument to an operable status.

d. Instrument calibrations are not required when the instruments are not required to be operable or are tripped. However, if instrument calibrations are missed, they shall be performed prior to returning the instrument to an operable status.

e. Initially once per month or according to Figure 4.1-1 with an interval of not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other BWR's for which the same design instrument operates in an environment similar to that of HNP-1. The failure rate date must be reviewed and approved by the AEC prior to any change in the once-a-month frequency.

f. This instrumentation is exempted from the instrument functional test definition. This instrument functional test will consist of injecting a simulated electrical signal into the measurement channels.

g. Standard current source used which provides an instrument channel alignment. Calibration using a radiation source shall be made once per operating cycle.

Logic system functional tests and simulated automatic actuation shall be performed once each operating cycle for the following:

1. Secondary Containment Actuation

Notes for Table 4.2-8 (Cont'd)

2. Standby Gas Treatment System Actuation
3. Steam Jet Air Ejector Off-gas Actuation
4. Primary Containment Purge and Vent Valve Closure
5. MCRECS Control Room Pressurization Mode Actuation
6. (Deleted)
7. Mechanical Vacuum Pump Isolation

The logic system functional test shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

4.4.A.2. Each Operating Cycle (Continued)

c. vessel. This test checks the explosive charge, proper operation of the associated valves and selected pump operability. The replacement charge to be installed will be selected from a manufactured batch which has been tested.

d. Both loops including both explosive valves should be tested in the course of two operating cycles.

e. Prior to startup, verify (by analysis) that the sodium pentaborate enrichment is within prescribed limits.

3.4.B. Operating with Inoperable Components

If one Standby Liquid Control redundant component is inoperable the reactor may remain in operation for a period not to exceed seven (7) days provided the redundant component is operable.

B. Surveillance with Inoperable Components

(Deleted)

C. Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be operable the following conditions shall be met:

1. Volume

The volume of the liquid control solution in the liquid control tank shall be maintained as required in Figure 3.4-1.

2. Concentration

The concentration of the liquid control tank shall be maintained as required in Figure 3.4-1.

C. Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the liquid control solution:

1. Volume

Check the standby liquid control tank volume at least once per day.

2. Concentration

Check the concentration of the liquid in the standby liquid control tank by chemical analysis:

3.5. CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

The Limiting Conditions for Operation apply to the operational status of the core and containment cooling systems.

Objective

The objective of the Limiting Conditions for Operation is to assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specifications

A. Core Spray (CS) System

1. Normal System Availability

a. The CS System shall be operable:

(1) Prior to reactor startup from a cold condition, or

(2) When irradiated fuel is in the reactor vessel and the reactor pressure is greater than atmospheric pressure, except as stated in Specification 3.5.A.2.

4.5. CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

The Surveillance Requirements apply to the core and containment cooling systems when the corresponding limiting conditions for operation are in effect.

Objective

The objective of the Surveillance Requirements is to verify the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specifications

A. Core Spray (CS) System

1. Normal Operational Tests

CS system testing shall be performed as follows:

	Item	Frequency
a.	Simulated Automatic Actuation Test	Once/Operating Cycle.
b.	System flow rate: Each loop can develop at least 4250 gpm against a system head corresponding to a reactor vessel pressure of at least 113 psig.	Once/3 months.
c.	Valve lineups: Verify that each valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.	Once/31 days.
d.	(Deleted)	

3.5.A.2. Operation with Inoperable Components

If one CS system loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CS system loop, the RHR system LPCI mode and the diesel generators (per Specification 4.9.A.2.a) are operable. When performing an inservice hydrostatic or leakage test with the reactor coolant temperature above or below 212°F the CS system is not required to be operable.

3. Shutdown Requirements

If Specification 3.5.A.1.a. or 3.5.A.2. cannot be met the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

B. Residual Heat Removal (RHR) System (LPCI and Containment Cooling Mode)

1. Normal System Availability

a. The RHR System shall be operable:

(1) Prior to reactor startup from a cold condition, or

(2) When irradiated fuel is in the reactor vessel and the reactor pressure is greater than atmospheric except as stated in Specification 3.5.B.2.

4.5.A.2. Surveillance with Inoperable Components

(Deleted)

B. Residual Heat Removal (RHR) System (LPCI and Containment Cooling Mode)

1. Normal Operational Tests

RHR system testing shall be performed as follows:

	Item	Frequency
a.	Air test on drywell headers and nozzles and air or water test on torus headers and nozzles	Once/10 years.

3.5.B.1. Normal System Availability (Cont.)

b. One RHR loop with two pumps or two loops with one pump per loop shall be operable in the shutdown cooling mode when irradiated fuel is in the reactor vessel and the reactor pressure is atmospheric except prior to a reactor startup as stated in Specification 3.5.B.1.a. During an inservice hydrostatic or leakage test, one RHR loop with two pumps or two loops with one pump per loop shall also be operable in the LPCI mode.

c. The reactor shall not be started up with the RHR system supplying cooling to the fuel pool.

d. During reactor power operation, the LPCI system discharge cross-tie valve, E11-F010, shall be in the closed position and the associated valve motor starter circuit breaker shall be locked in the off position. In addition, an annunciator which indicates that the cross-tie valve is not in the fully closed position shall be available in the control room.

e. Both recirculation pump discharge valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).

2. Operation with Inoperable Components

a. One LPCI Pump Inoperable

If one LPCI pump is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided that the remaining LPCI pumps, both LPCI subsystem flow paths, the CS system, and the associated diesel generators are operable (per Specification 4.9.A.2.a).

b. One LPCI Subsystem Inoperable

A LPCI subsystem is considered to be inoperable if (1) both of the LPCI pumps within that system are inoperable or (2) the active valves in the subsystem flow path are inoperable.

4.5.B.1. Normal Operational Tests

	Item	Frequency
b.	Simulated Automatic Actuation Test	Once/Operating Cycle.
c.	System flow rate: Each RHR pump shall deliver at least 7700 gpm against a system head corresponding to a reactor vessel pressure of at least 20 psig.	Once/3 months.
d.	Valve lineups: Verify that each valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.	Once/31 days.
e.	(Deleted)	

f. Both recirculation pump discharge valves shall be tested for operability during any outage exceeding 48 hours, if operability tests have not been performed during the preceding month.

2. Surveillance with Inoperable Components

a. (Deleted)

b. (Deleted)

3.5.B.2. Operation with Inoperable Components (Continued)

b. If one LPCI subsystem is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided that all active components of the remaining LPCI subsystem, the CS system, and the associated diesel generators are operable (per Specification 4.9.A.2.a).

c. When performing an inservice hydrostatic or leakage test with the reactor coolant temperature above or below 212°F, comply with Specification 3.5.B.1.b.

4.5.B.2. (Deleted)

3.5.B.3. Shutdown Requirements

If Specification 3.5.B.1.a. or 3.5.B.2. cannot be met, the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

C. RHR Service Water System

1. Normal System Availability

The RHR service water system shall be operable:

a. Prior to reactor startup from a Cold Shutdown Condition, or

b. When irradiated fuel is in the reactor vessel and the reactor vessel pressure is greater than atmospheric pressure except as stated in Specification 3.5.C.2, or

c. When irradiated fuel is in the reactor vessel and the reactor is depressurized at least one RHR service water loop shall be operable.

2. One Pump Inoperable

If one RHR service water pump is inoperable the reactor may remain in operation for a period not to exceed 30 days provided all other active components of both subsystems are operable. When performing an inservice hydrostatic or leakage test, comply with Specification 3.5.C.1.c.

4.5.C. RHR Service Water System

1. Normal Operational Tests

RHR service water system testing shall be performed as follows:

Item	Frequency
a. Valve lineups: Verfiy that each valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.	Once/31 days.
b. Pump Capacity Test: Each RHR service water pump shall deliver at least 4000 gpm at a system head of at least 847 feet.	Once/3 months.

2. One Pump Inoperable

(Deleted)

3.5.C.3. Two Pumps Inoperable

If two RHR service water pumps are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided all redundant active components in both of the RHR service water subsystems are operable.

4. Shutdown Requirements

If Specifications 3.5.C cannot be met, the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

D. High Pressure Coolant Injection (HPCI)System

1. Normal System Availability

a. The HPCI System shall be operable:

(1) Prior to reactor startup from a cold condition, or

(2) When irradiated fuel is in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except as stated in Specification 3.5.D.2.*

4.5.C.3. Two Pumps Inoperable

(Deleted)

D. High Pressure Coolant Injection (HPCI) System

1. Normal Operational Tests

HPCI system testing shall be performed as follows:

	Item	Frequency
a.	Simulated automatic actuation test	Once/Operating Cycle.
b.(1)	Flow rate for a system head corresponding to a reactor vessel pressure of $\geq$ 1000 psig when steam is being supplied to the turbine at $\leq$ 1000 psig, and	Once/3 months.
(2)	Flow rate for a system head corresponding to a reactor vessel pressure of $\geq$ 165 psig when steam is being supplied to the turbine at 165 $\pm$ 15 psig.	Once/Operating Cycle.

*HPCI is not required to be operable for performance of inservice hydrostatic or leak testing with reactor pressure greater than 150 psig and all control rods inserted.

4.5.D.1.b. Normal Operational Tests
The HPCI pumps shall deliver at least 4250 gpm during each flow rate test.

c. Valve lineups: Once/31 days. Verify that each valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.

3.5.D.2. Operation with Inoperable Components

If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed fourteen (14) days provided the ADS, CS system, RHR system LPCI mode, and RCIC system are operable.

With the surveillance requirements of Specification 4.5.D.1. not performed at the required frequencies due to low reactor steam pressure, reactor startup is permitted and the appropriate surveillance will be performed within 12 hours after reactor steam pressure is adequate (i.e., reactor pressure is such that the required steam pressure is maintained at the turbine for the duration of the test) to perform the tests.

2. Surveillance with Inoperable Components

(Deleted)

3. Shutdown Requirements

If Specification 3.5.D.1. or 3.5.D.2. cannot be met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

E. Reactor Core Isolation Cooling (RCIC) System

1. Normal System Availability

a. The RCIC system shall be operable with an operable flow path capable of (automatically) taking suction from the suppression pool and transferring the water to the reactor pressure vessel:

(1) Prior to reactor startup from a cold condition, or

E. Reactor Core Isolation Cooling (RCIC) System

1. Normal Operational Tests

RCIC system testing shall be performed as follows:

	Item	Frequency
a.	Simulated Automatic Actuation (and restart*) Test.	Once/Operating Cycle.

*Automatic Restart on a Low Water Level which is subsequent to a High Level Trip.

3.5.E.1. **Normal System Availability** (Cont.)

a.(2) When there is irradiated fuel in the reactor vessel and the reactor pressure is above 150 psig, except as stated in Specification 3.5.E.2.*

2. Operation with Inoperable Components

If the RCIC system is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCI system is operable during such time. With the surveillance requirements of Specification 4.5.E.1 not performed at the required frequencies due to low reactor steam pressure, reactor startup is permitted and the appropriate surveillance will be performed within 12 hours after reactor steam pressure is adequate (i.e., reactor pressure is such that the required steam pressure is maintained at the turbine for the duration of the test) to perform the test.

3. If Specification 3.5.E.1. or 3.5.E.2. is not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours.

4.5.E.1. Normal Operational Tests (Cont.)

b. Verifying that suction for the RCIC system is automatically transferred from the CST to the suppression pool on a simulated low CST level or high suppression pool level signal.	Once/ Operating Cycle.
c.(1) Flow rate when steam is being supplied to the turbine at normal reactor vessel operating pressure, 1000 + 20,-80 psig, and	Once/3 months.
(2) Flow rate when steam is being supplied to the turbine at a pressure of 150 + 15, -0 psig.	Once/ operating Cycle.

The RCIC pump shall deliver at least 400 gpm during each flow test.

d. Valve lineups: Verify that each valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.	Once/31 days.

e. (Deleted)

2. Surveillance with Inoperable Components

(Deleted)

*RCIC is not required to be operable for performance of inservice hydrostatic or leak testing with reactor pressure greater than 150 psig and all control rods inserted.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5.F. Automatic Depressurization System (ADS)

1. Normal System Availability

The seven valves of the Automatic Depressurization System shall be operable:

a. Prior to reactor startup from a cold shutdown, or

b. When there is irradiated fuel in the reactor vessel and the reactor is above 113 psig except as stated in Specification 3.5.F.2.*

2. Operation with Inoperable Components

If one of the seven ADS valves is known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is operable. (Note that the pressure relief function of these valves is assured by Specification 3.6.H.; Specification 3.5.F. only applies to the ADS function).

3. Shutdown Requirements

If Specification 3.5.F.1. or 3.5.F.2. cannot be met, an orderly shutdown will be initiated and the reactor pressure shall be reduced to 113 psig or less within 24 hours.

4.5.F. Automatic Depressurization System (ADS)

1. Normal Operational Tests

a. A simulated automatic actuation test shall be performed on the ADS prior to startup after each refueling outage. Surveillance of all relief valves is covered in Specification 4.6.H.

b. A leak rate test of each ADS valve accumulator, check valve, and actuator assembly shall be performed during each refueling outage at a pressure of 90 ± 18 psig. The leakage rate shall be verified to be ≤ 4.5 SCFH.

2. Surveillance with Inoperable Components

(Deleted)

*The ADS valves are not required to be operable for performance of inservice hydrostatic or leak testing with reactor pressure greater than 113 psig and all control rods inserted.

3.5.G. Minimum Core and Containment Cooling Systems Availability

During any period when one of the standby diesel generators is inoperable, continued reactor operation is limited to 7 days unless operability of the diesel generator is restored within this period. During such 7 days all of the components in the RHR system LPCI mode and containment cooling mode mode shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours. Specification 3.9. provides further guidance on electrical system availability.

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 core and containment cooling functions.

When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown Condition, both CS systems and the LPCI and containment cooling subsystems of the RHR system may be inoperable provided that the shutdown cooling subsystem of the RHR system is operable in accordance with Specification 3.5.B.1.b and that no work is being done which has the potential for draining the reactor vessel.

4.5.G. Surveillance of Core and Containment Cooling Systems

When it is determined that one of the standby diesel generators is inoperable, all of the components in the RHR system LPCI mode and containment cooling mode connected to the operable diesel generators shall be verified to be operable.

3.5.H. Maintenance of Filled Discharge Pipes

Whenever the CS system, LPCI, 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. The suction of the HPCI pumps shall be aligned to the condensate storage tank.

I. Minimum River Level

1. If the water level, as measured in the pump well, is less than 61.2 ft MSL, the discharge from each plant service water (PSW) pump will be throttled such that each pump does not exceed 7000 gpm.

2. If the water level, as measured in the pump well, decreases to less than 60.7 ft MSL, or if the level in the river* drops to a level equivalent to less

4.5.H. Maintenance of Filled Discharge Pipes

The following surveillance requirements shall be performed to assure that the discharge piping of the CS system, LPCI, HPCI, and RCIC are filled when required:

1. Every month, the discharge piping of the LPCI and CS systems shall be vented from the high point and water flow observed.

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

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, 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.

4. The level switches which monitor the discharge lines shall be functionally tested every month and calibrated every 3 months.

I. Minimum River Level

The water level as, measured in the pump well, and the level in the river* shall be verified with the following frequencies:

	Level (MSL)	Frequency
1.	> 61.7 ft	Biweekly.
2.	$\leq$ 61.7 ft	Every 12 hrs.

*Only pump well monitoring is required if a temporary weir is not in place.

than 60.7 ft in the pump well of the intake structure, an orderly shutdown of the reactor shall be initiated, and the reactor shall be in the Cold Shutdown Condition within 24 hours until the level in the river is greater than or equal to 60.7 ft MSL equivalent in the pump well.

3.5.J. Plant Service Water System

1. Normal Availability

The reactor shall not be made critical from the Cold Shutdown Condition unless the PSW System (including four PSW pumps and the standby service water pump) is operable.

2. Inoperable Components

a. The standby service water pump may be inoperable for a period not to exceed 60 days provided that an alternate Unit 1 PSW water cooling source to the 1B diesel generator is OPERABLE.

b. One PSW pump may be inoperable for a period not to exceed 30 days provided all other PSW pumps and the standby service water pump are operable.

c. One PSW pump and the standby service water pump may be inoperable for a period not to exceed 30 days provided all other PSW pumps are operable.

d. Two PSW pumps or one PSW division may be inoperable for a period not to exceed 7 days provided all other PSW pumps and the standby service water pump are operable.

4.5.J. Plant Service Water System

1. The automatic pump start functions and automatic isolation functions shall be tested once per operating cycle.

2. Inoperable Components

a. With the standby service water subsystem inoperable for up to 60 days, provide Unit 1 service water cooling to the 1B diesel generator by verifying OPERABILITY of an alternate Unit 1 service water cooling source within 8 hours. Otherwise, declare the 1B diesel generator inoperable and take the action required by Specification 3.9.B.2.

b. (Deleted)

c. (Deleted)

d. (Deleted)

LIMITING CONDITIONS FOR OPERATION

3.5.J. Plant Service Water System

2. Inoperable Components (Cont'd)

e. Two PSW pumps or one PSW division, and the standby service water pump may be inoperable for a period not to exceed 7 days provided all other PSW pumps are operable.

For each condition above in which the standby service water pump is inoperable, cooling water to diesel generator 1B shall be intertied with the PSW divisional piping supply.

3. Shutdown Requirements

If the requirements of Specifications 3.5.J.1. and 3.5.J.2. cannot be met the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

3.5.K. Equipment Area Coolers

1. The equipment area coolers serving the Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), Core Spray or Residual Heat Removal (RHR) pumps must be operable at all times when the pump or pumps served by that specific cooler is considered to be operable.

2. When an equipment area cooler is not operable, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

SURVEILLANCE REQUIREMENTS

4.5.J. Plant Service Water System

2. Inoperable Components (Cont'd)

e. When cooling water to diesel generator 1B is intertied with the PSW divisional piping supply, operability of the divisional interlock valves shall be demonstrated.

4.5.K. Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

3.5. CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray (CS) System

1. Normal System Availability

Analyses presented in Reference 1 demonstrated that the CS system provides adequate cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature to below 2200°F which assures that core geometry remains intact and to limit any clad metal-water reaction to less than one percent. CS distribution has been shown in tests of systems similar in design to HNP-1 to exceed the minimum requirements. 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 intent of the CS system specifications is to prevent operation above atmospheric pressure without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. The allowable repair times have been selected using engineering judgment based on experiences and supported by availability analysis. Assurance of the availability of the remaining systems is increased by demonstrating operability immediately and by requiring selected testing during the outage period.

When the reactor vessel pressure is atmospheric, the limiting conditions for operation are less restrictive. At atmospheric pressure, the minimum requirement is for one supply of makeup water to the core. Requiring two operable RHR pumps and one CS pump provides redundancy to ensure makeup water availability.

2. Operation with Inoperable Components

Should one CS loop become inoperable, the remaining CS loop and the RHR system are required to be operable to ensure their availability should the need for core cooling arise. The surveillance testing required by Specification 4.5.A, 4.5.H, and 4.6.K ensures the availability of the remaining CS loop. The surveillance testing required by Specifications 4.5.B, 4.5.H, and 4.6.K ensures the availability of the RHR system. These provide extensive margin over the operable equipment needed for adequate core cooling. With due regard for this margin, the allowable repair time of 7 days was chosen.

B. Residual Heat Removal (RHR) System (LPCI and Containment Cooling Mode)

1. Normal System Availability

The RHR system LPCI mode is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the CS system; however, it does function in combination with the CS system to prevent excessive fuel clad temperature. The LPCI mode of the RHR system and the CS 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.

3.5.B.1. Normal System Availability (Continued)

Observation of the stated requirements for the containment cooling mode assures that the suppression pool and the drywell will be sufficiently cooled, following a loss-of-coolant accident, to prevent primary containment over pressurization. The containment cooling function of the RHR system is permitted only after the core has reflooded to the two-thirds core height level. This prevents inadvertently diverting water needed for core flooding to the less urgent task of containment cooling. The two-thirds core height level interlock may be manually bypassed by a keylock switch.

The intent of the RHR system specifications is to prevent operation above atmospheric pressure without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. The allowable repair times have been selected using engineering judgment based on experiences and supported by availability analysis. Assurance of the availability of the remaining systems is increased by demonstrating operability immediately and by requiring selected testing during the outage period.

When the reactor vessel pressure is atmospheric, the limiting conditions for operation are less restrictive. At atmospheric pressure, the minimum requirement is for one supply of makeup water to the core.

2. Operation with Inoperable Components

With one LPCI pump inoperable or one LPCI subsystem inoperable, adequate core flooding is assured by the required operability of the redundant LPCI pumps and LPCI subsystem and the CS system. The surveillance testing required by Specifications 4.5.B, 4.5.H, and 4.6.K ensures the availability of the redundant LPCI pump and LPCI subsystem. The surveillance testing required by Specifications 4.5.A, 4.5.H, and 4.6.K ensures the availability of the CS system. The reduced redundancy justifies the specified 7 day out-of-service period.

3.5.D.2. Operation with Inoperable Components

The HPCI system serves as a backup to the RCIC system as a source of feedwater makeup during primary system isolation conditions. The ADS serves as a backup to the HPCI system for reactor depressurization for postulated transients and accidents. The ADS must be operable if the HPCI system is determined to be inoperable. In addition, the surveillance testing required by the specified Specifications ensures the availability of the following: CS (4.5.A, 4.5.H, and 4.6.K), LPCI (4.5.B, 4.5.H, and 4.6.K), RCIC (4.5.E, 4.5.H, and 4.6.K), and ADS (4.5.F and 4.6.K). Considering the redundant systems, an allowable repair time of 7 days was selected.

E. Reactor Core Isolation Cooling (RCIC) System

1. Normal System Availability

The various conditions under which the RCIC system plays an essential role in providing makeup water to the reactor vessel have been identified by evaluating the various plant events over the full range of planned operations. The specifications ensure that the function for which the RCIC system was designed will be available when needed.

Because the low-pressure cooling systems (LPCI and CS) are capable of providing all the cooling required for any plant event when nuclear system pressure is below 150 psig, the RCIC system is not required below this pressure. RCIC system design flow (400 gpm) is sufficient to maintain water level above the top of the active fuel for a complete loss of feedwater flow at the design power.

Two sources of water are available to the RCIC system. Suction is initially taken from the condensate storage tank and is automatically transferred to the suppression pool upon low CST level or high suppression pool level.

2. Operation With Inoperable Components

Consideration of the availability of the RCIC system reveals that the average risk associated with failure of the RCIC system to cool the core when required is not increased if the RCIC system is inoperable for no longer than 7 days, provided that the HPCI system is operable during this period. The surveillance testing required by Specifications 4.5.D, 4.5.H, and 4.6.K ensures the availability of the HPCI system.

F. Automatic Depressurization System (ADS)

1. Normal System Availability

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

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the LPCI and the CS systems can operate to protect the fission product barrier. Note that this Specification applies only to the automatic feature of the pressure relief system.

3.5.F.1. Normal System Availability (continued)

Specification 3.6. states the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures yet be fully capable of performing their pressure relief function.

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 Core Standby Cooling Systems.

The ADS valve accumulators are sized such that, following loss of the pneumatic supply, at least two valve actuations will be possible with the drywell at 70% of its design pressure. This drywell pressure results from the largest break which could lead to the need for rapid depressurization through the ADS valves. The allowable accumulator leakage criterion ensures the above capability for 30 minutes following loss of the pneumatic supply.

2. Operation with Inoperable Components

With one ADS valve known to be incapable of automatic operation six valves remain operable to perform their ADS function. However, since the ECCS Loss of Coolant Accident analysis for small line breaks assumed that all seven ADS valves were operable, reactor operation with one ADS valve inoperable is only allowed to continue for 7 days provided that the HPCI system is operable and that the (remaining) six ADS valves are operable. In addition, surveillance testing required by the specified Specifications ensures the availability of the following: HPCI (4.5.D, 4.5.H, and 4.6.K) and ADS (4.5.F and 4.6.K).

G. Minimum Core and Containment Cooling Systems Availability

The purpose of this Specification is to assure that adequate core cooling equipment is available at all times. If, for example, one CS loop were out of service and the diesel which powered the opposite CS were out of service, only 2 RHR pumps would be available. Specification 3.9. must also be consulted to determine other requirements for the diesel generators.

This specification establishes conditions for the performance of major maintenance, such as draining of the suppression pool. The availability of the shutdown cooling subsystem of the RHR system and the RHR service water system ensure adequate supplies of reactor cooling and emergency makeup water when the reactor is in the Cold Shutdown Condition. In addition this specification provides that, should major maintenance be performed, no work will be performed which could lead to draining the water from the reactor vessel.

3.5.H. Maintenance of Filled Discharge Pipes

If the discharge piping of the CS, LPCI, HPCI, and RCIC systems are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. 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. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be inoperable for Specification purposes.

The CS and LPCI discharge piping high point vents are visually checked for water flow once a month to ensure that the lines are filled.

Assurance that the HPCI and RCIC discharge piping remains filled is provided by observing water flow from these systems high points monthly.

I. Minimum River Flow

A very low-flow river stage-discharge relationship was developed at the Plant Hatch intake structure location. USGS rating data were available for flows above 1740 cfs at the Baxley gauge (at U.S. Highway No. 1 bridge, on the plant site). This data, which includes bathymetric surveys of the rating cross-section, were used to extend the USGS rating curve by computation. Since the USGS data used in these computations result in the highest flow for a given low-flow stage ever recorded at the location, the computed rating curve should give a conservative low stage for a given flow. The river rating curve at the Plant Hatch intake structure was developed by subtracting 0.1 ft from the USGS guage evaluation for a given discharge. The 0.1-ft adjustment was determined by level survey when the river level at the USGS guage was approximately 62 ft MSL. At the Plant Hatch site, the river level would be 61.3 ft MSL for 1200 cfs which is the low flow of record at Charlotte and 60.8 ft MSL for the hypothetical minimum low flow of 950 cfs.

The minimum low flow is important because of its effect on the operation of PSW and RHR service water pumps. The RHR service water pumps at rated-flow conditions require for net positive suction head (NPSH) a river stage of only 59.0 ft . Thus, no further consideration is required on river stage with regard to submergence of these pumps.

At the rated flow of 8500 gpm each for the PSW pumps, 4 ft of submergence will satisfy the NPSH and vortexing requirement. This corresponds to a stage in the pump well of 61.2 ft. Normal operation requires about 7840 gpm for each of three pumps. Shutdown or emergency conditions require only one pump with a discharge flow of 4428 gpm. This corresponds to a pump well level of 59.9 ft for safe shutdown. For a 0.1-ft-head loss through the trash rack and traveling screen, the corresponding river level would be 60.0 ft MSL, which corresponds to a flow of 660 cfs. Similarly, 3.5.J/4.5.J Plant Service Water System

The Plant Service Water (PSW) system consists of two subsystems (divisions) of two pumps each and a separate standby service water pump system for diesel generator 1B. During normal full power operation the two subsystems function as a 3 out of 4 pump cross connected system supplying cooling water to the turbine and reactor building cooling systems. In the event of an accident signal, nonsafety-related cooling loads are isolated and the PSW pumps in the two subsystems supply cooling water to diesel generators 1A and 1C, the reactor building cooling system and the control room air conditioners, while the standby service water pump is available to automatically supply cooling water to diesel generator 1B should it be needed. Additionally, diesel 1B has a manual backup water supply available from the Unit 1 Division 1 or Division 2 PSW subsystems so that during maintenance on the standby diesel service water pump, either division of the PSW system can manually be aligned to supply cooling water to the 1B diesel. The two subsystems and the standby service water pump system are split in the accident mode for greater reliability with one pump in each of the two subsystems automatically starting while a start signal from diesel generator 1B initiates standby service water pump operation. Only one of the Division 1 PSW pumps and one of the Division 2 PSW pumps are required for cooling diesel generators 1A and 1C, respectively, while the standby service water pump provides adequate cooling water to diesel generator 1B. In the event that the standby service water pump is inoperable, the HNP-1 Division 1-Division 2 intertie supply piping can be aligned to cool the 1B diesel. In this condition, one PSW pump is capable of supplying the cooling requirements for the reactor building cooling system, the control room air conditioners, and the 1A, 1B, and 1C diesel generators.

The PSW system can supply all power generation systems at full load and the diesel generators with redundancy if one PSW pump and/or the standby service water pump are inoperable. Hence, a 60-day outage time is justified if the standby service water pump is inoperable since all four PSW pumps are available (divisional intertie to 1B diesel required). In addition, a 30-day outage is justified if one PSW pump is inoperable, or if one PSW pump and the standby service water pump are inoperable (divisional intertie to 1B diesel required). Should two PSW pumps (or one subsystem) become inoperable, or should two PSW pumps (or one subsystem) and the standby service water pump become inoperable (division intertie to 1B diesel required) plant operation will probably only continue at less than full power. However, safety-related loads are still adequately powered for these conditions. Therefore, a 7-day outage time is justified for such events. The surveillance testing required by Specifications 4.5.J and 4.6.K ensures availability of the redundant pumps and subsystem.

K. Engineering Safety Features Equipment Area Coolers

The equipment area cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicate that the temperature rise in safeguard compartments without adequate ventilation flow or cooling is such that continued operation of the safeguard equipment or associated auxiliary equipment cannot be assured.

The surveillance and testing of the equipment area coolers in each of their various modes is accomplished during the testing of the equipment served by these coolers. The testing is adequate to assure the operability of the equipment area coolers.

L. References

1. "Edwin I. Hatch Nuclear Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDC-31376-P, December 1986.

3.6.J. Recirculation System

1. Core thermal power shall not exceed 1% of rated thermal power without forced recirculation.

2. Whenever the reactor is in the START & HOT STANDBY or RUN modes, at least one recirculation loop shall be in operation.

3. The requirements applicable to single-loop operation as identified in Sections 1.1.A, 2.1.A, 3.1.A, 3.2.G, 3.11.A, and 3.11.C shall be in effect within 24 hours following the removal of one recirculation loop from service, or the unit shall be placed in the HOT SHUTDOWN Condition within 12 hours and in COLD SHUTDOWN within the following 12 hours.

4. With only one recirculation loop in operation and the unit in the Operation Not Allowed Region, specified in Figure 3.6-5, initiate action within 15 minutes to place the unit in the Operation Allowed Region, identified in Figure 3.6-5, within 2 hours. Otherwise, place the reactor in the HOT SHUTDOWN Condition within 12 hours.

5. Following one pump operation the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

4.6.J. Recirculation System

1. Recirculation pump speeds shall be recorded at least once per day.

2. With only one recirculation loop in operation, verify that the reactor operating conditions are outside the Operation Not Allowed Region in Figure 3.6-5:

   (a) At least once per 24 hours,

   (b) Whenever thermal power has been changed by at least 5% of rated thermal power and steady-state conditions have been reached.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Health Physics Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager of Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 The Fire Protection Program, except training, is maintained under the direction of the Manager-Engineering Support. The Fire Protection Program meets or exceeds the guidelines of NFPA Code 27, 1975.

Fire Protection Training is maintained under the direction of the Training and Emergency Preparedness Manager. Fire Protection Training meets or exceeds the guidelines of NFPA Code 27, 1975, except retraining frequency. Fire Brigade and Fire Emergency Support Group (FB/FESG) members are required to attend retraining once per calendar quarter.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW BOARD (PRB)

FUNCTION

6.5.1.1 The PRB shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PRB shall be composed of, as a minimum, a supervisor or higher level individual from each of the departments listed below:

Operations
Maintenance
Quality Control (QC)
Health Physics
Nuclear Safety and Compliance
Engineering Support

The Chairman, his alternate, and other members of the PRB shall be designated by the Plant Manager. The Chairman and his designated alternate shall both be managers of one of the six above listed departments or a higher level onsite manager.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PRB activities at any one time.

5.3.2 Audit Responsibility

5.3.2.1 The General Manager-Quality Assurance is responsible for an audit, conducted annually, of the activities of the Plant Manager and the Manager-Environmental Affairs, related to compliance with ETS.

5.3.2.2 Audits of facility activities shall be performed annually under the cognizance of the SRB to ensure conformance of facility operation to provisions of the ETS.

5.4 State and Federal Permit and Certificates

Section 401 of PL 92-500, the Federal Water Pollution Control Act Amendments of 1972 (FWPCA), requires any applicant for a Federal license or permit to conduct any activity that may result in any discharge into provisions of Sections 301, 302, 306, and 307 of the FWPCA. Section 401 of PL 92-500 further requires that any certification provided under this section shall set any effluent limitations and other limitations and monitoring requirements necessary to assure that any applicant for a Federal license or permit will comply with the applicable limitations. Certifications provided in accordance with Section 401 set forth conditions on the Federal license or permit for which the certification is provided. Accordingly, the licensee shall comply with the requirements set forth in the currently applicable 401 certification and amendments thereto issued to the licensee by the Georgia Environmental Protection Division. In accordance with the provisions of the Georgia Water Quality Control Act, the FWPCA and the rules and regulations promulgated pursuant to each of these acts, the Georgia Environmental Protection Division, under authority delegated by the U.S. EPA, issued NPDES permit No. GA 0004120 to the licensee. The NPDES permit authorizes the licensee to discharge from HNP Units 1 and 2 to the Altamaha River in accordance with effluent limitations, monitoring requirements, and other conditions stipulated in the permit.

Subsequent revisions to the certifications will be accommodated in accordance with the provisions of section 5.6.3.

5.5 Procedures

Detailed written procedures, including applicable checklists and instructions, shall be prepared and followed for all activities involved in implementing the ETS. All procedures shall be maintained in a manner convenient for review and inspection. Procedures that are the responsibility of the Plant Manager shall be kept at the plant. Procedures that are the responsibility of the Manager-Environmental Affairs shall be kept at the Georgia Power Company General Office.

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. 108
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 Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, (the licensee) dated March 2, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-5 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 108 , 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of issuance: August 30, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 108

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of Appendices "A" and "B" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Appendix A

Remove Page	Insert Page
3/4 5-1	3/4 5-1
3/4 6-46	3/4 6-46
3/4 7-9	3/4 7-9
3/4 8-17	3/4 8-17
3/4 8-18*	3/4 8-18
3/4 8-21	3/4 8-21
3/4 8-22*	3/4 8-22
6-5	6-5

Appendix B

Remove Page	Insert Page
5-3	5-3

*Overleaf page provided to maintain document completeness.

3/4.5.1 HIGH PRESSURE COOLANT INJECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.1 The High Pressure Coolant Injection (HPCI) system shall be OPERABLE with:

a. One OPERABLE HPCI pump, and

b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor pressure vessel.

APPLICABILITY: CONDITIONS 1*, 2* and 3* with reactor vessel steam dome pressure > 150 psig.

ACTION:

a. With the HPCI system inoperable, POWER OPERATION may continue and the provisions of 3.0.4 do not apply*, provided the RCIC system, ADS, CSS, and LPCI system are OPERABLE; restore the inoperable HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 150 psig within the following 24 hours.

b. With the surveillance requirements of Specification 4.5.1 not performed at the required frequencies due to low reactor steam pressure, the provisions of Specification 4.0.4 are not applicable provided the appropriate surveillance is performed within 12 hours after reactor steam pressure is adequate (i.e., reactor pressure is such that the required steam pressure is maintained at the turbine for the duration of the test) to perform the tests.

SURVEILLANCE REQUIREMENTS

4.5.1 The HPCI shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water, and

*See Special Test Exception 3.10.5

PRIMARY CONTAINMENT PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.5.1 The drywell and suppression chamber 18-inch purge supply and exhaust isolation valves shall be OPERABLE with:

a. Each valve closed except for purge system operation for inerting, deinerting, and pressure control.

b. A leakage rate such that the provisions of Specification 3.6.1.2 are met.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With an 18-inch drywell and suppression chamber purge supply and/or exhaust isolation valve(s) inoperable or open for other than inerting, deinerting or pressure control, close the open 18-inch valve(s) or otherwise isolate the penetrations(s) within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.5.1 The primary containment purge system shall be demonstrated OPERABLE:

a. In addition to the requirements of Specification 3.6.3, at least once per 31 days, when not PURGING and VENTING, by verifying that each 18-inch drywell and suppression chamber isolation valve is closed.

b. At least once per 18 months by replacing the valve seat of each 18-inch drywell and suppression chamber purge supply and exhaust isolation valve having a resilient material seat and verifying that the leakage rate is within its limit.

PLANT SYSTEM

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The Reactor Core Isolation Cooling (RCIC) System shall be OPERABLE with an OPERABLE flow path capable of (AUTOMATICALLY) taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: CONDITIONS 1, 2, and 3 with reactor steam dome pressure > 150 psig.

ACTION:

a. With the RCIC system inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to < 150 psig within the following 24 hours.

b. With the surveillance requirements of Specification 4.7.3 not performed at the required intervals due to low reactor steam pressure, the provisions of Specification 4.0.4 are not applicable provided the appropriate surveillance is performed within 12 hours after reactor steam pressure is adequate (i.e., reactor pressure is such that the required steam pressure is maintained at the turbine for the duration of the test) to perform the tests.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water, and

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 92 days by verifying that the RCIC pump develops a flow of 400 gpm on recirculation flow when steam is being supplied to the turbine at normal reactor vessel operating pressure, 1000 + 20, - 80 psig.

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITIONS FOR OPERATION

3.8.2.5 The following A.C. circuits inside primary containment shall be de-energized*:

a. Breaker Numbers 2, 4, 6, 8, 10, 12, 14, 40 and 42 in panel 2T51-S003,

b. Breaker Numbers 2, 4, 6, 8, 10, 12, 40 and 42 in panel 2T51-S004,

c. Breaker Numbers 28 and 34 in panel 2R25-S105, and

d. Compartment 1EL on MCC 2R24-S014.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.2.5 Each of the above required A.C. circuits shall be determined to be de-energized at least once per 24 hours by verifying that the associated circuit breakers in the specified panels are in the tripped condition.

*Except during entry into the drywell.

ELECTRICAL POWER SYSTEMS

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.2.6 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.2.6-1 shall be OPERABLE.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

With one or more of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.2.6-1 inoperable;

a. De-energize the circuit(s) by tripping the associated circuit breaker(s) within 72 hours and the provisions of Specification 3.0.4 are not applicable, or

b. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.6.1 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.2.6-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. For at least one 4 KV reactor recirculation pump circuit, such that both recirculation pump circuits are demonstrated OPERABLE at least once per 36 months, by performance of;

(a) A CHANNEL CALIBRATION of the associated protective relays, and

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

2. For molded case circuit breakers, by performance of a functional test of at least one circuit breaker of each type, such that all circuit breakers of each type are demonstrated OPERABLE at least once per N x 18 months, where N is the number of circuit breakers of each type. The functional test shall consist of injecting a current input as specified by NEMA AB2-1980 to the circuit breaker and verifying that the circuit breaker functions as designed. Should any circuit breaker fail to function as designed, all other circuit breakers of that type shall be tested.

TABLE 3.8.2.6-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION*	SYSTEM/COMPONENT POWERED
c. Type 3:	
1. 600 VAC, MCB, T.M. 2R24-S014, COMPT. 5E	RECIRC. PUMP MOTOR HEATER 2B31-C001B
2. 600 VAC, MCB, T.M. 2R24-S013, COMPT. 5B	REACTOR RECIRC. PUMP MOTOR HEATER 2B31-C001A
3. 600 VAC, MCB, T.M. 2R24-S013, COMPT. 3B	DRYWELL COOLING UNIT 2T47-B010A
4. 600 VAC, MCB, T.M. 2R24-S014, COMPT. 8A	DRYWELL COOLING UNIT 2T47-B010B
d. Type 4:	
1. 120 VAC, MCB, T.M. 2R25-S102, CKT. 10	CABLE BHX808C05
2. 120 VAC, MCB, T.M. 2R25-S101, CKT. 10	CABLE BGX708C05
e. Type 5:	
1. 600 VAC, MCB, M.O. 2R24-S014, COMPT. 2A	DRYWELL EQUIP. DR. SUMP DISCH. MOV 2G11-F018
2. 600 VAC, MCB, M.O. 2R24-S014, COMPT. 6C	DRYWELL EQUIP. DRAIN SUMP RECIRC. MOV 2G11-F015
3. 600 VAC, MCB, M.O. 2R24-S012B, COMPT. 4A	RCIC STEAMLINE INBOARD ISO. MOV. 2E51-F007
4. 600 VAC, MCB, M.O. 2R24-S011, COMPT. 9A	RHR HEAD SPRAY ISOLATION MOV. 2E11-F022
5. 600 VAC, MCB, M.O. 2R24-S011A, COMPT. 4A	HPCI STEAM LINE INBOARD ISOLATION MOV. 2E41-F002
6. 600 VAC, MCB, M.O. 2R24-S011, COMPT. 14C	RWCU INBOARD ISOLATION MOV. 2G31-F001
7. 600 VAC, MCB. M.O. 2R24-S011, COMPT. 15B	MAIN STEAM LINE DRAIN MOV. 2B21-F016

*M.C.B. - molded case circuit breaker
M.O. - magnetic only
T.M. - thermal magnetic

TABLE 3.8.2.6-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION*	SYSTEM/COMPONENT POWERED
c. Type 6:	
1. 600 VAC, MCB, M.O. 2R24-S018A, COMPT. 2A	LOOP 'A' PUMP SUCTION MOV 2B31-F023A
2. 600 VAC, MCB, M.O. 2R24-S018A, COMPT. 2B	LOOP 'A' PUMP DISCH. MOV 2B31-F031A
3. 600 VAC, MCB, M.O. 2R24-S018B, COMPT. 3A	LOOP 'B' PUMP SUCTION MOV 2B31-F023B
4. 600 VAC, MCB, M.O. 2R24-S018B, COMPT. 3B	LOOP 'B' PUMP DISCH. MOV 2B31-F031B
5. 600 VAC, MCB, M.O. 2R24-S014, COMPT. 1B	DRYWELL EQUIP. DRAIN PUMP B 2G11-C006B
6. 600 VAC, MCB, M.O. 2R24-S014, COMPT. 7D	DRYWELL FLOOR DRAIN SUMP PUMP 'B' 2G11-C001B
7. 600 VAC, MCB, M.O. 2R24-3013, COMPT. 4A	DRYWELL FLOOR DRAIN SUMP PUMP 1A 2G11-C001A
8. 600 VAC, MCB, M.O. 2R24-S013, COMPT. 4B	DRYWELL EQUIP. DRAIN SUMP PUMP A 2G11-C006A
9. 600 VAC, MCB, M.O. 2R24-S012, COMPT. 18B	DRYWELL COOLING UNIT 2T47-B007B
10. 600 VAC, MCB, M.O. 2R24-S012, COMPT. 19A	DRYWELL COOLING UNIT 2T47-C001B
11. 600 VAC, MCB, M.O. 2R24-S011, COMPT. 6C	RHR SHUTDOWN COOLING ISO. MOV 2E11-F009
12. 600 VAC, MCB, M.O. 2R24-S011, COMPT. 18A	DRYWELL COOLING UNIT 2T47-B007A
13. 600 VAC, MCB, M.O. 2R24-S011, COMPT. 18C	DRYWELL COOLING RETURN AIR FAN 2T47-C001A.

*M.C.B. - molded case circuit breaker
M.O. - magnetic only
T.M. - thermal magnetic

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Health Physics Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager of Training and shall meet or exceed the requirements and recommendations of section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR part 55.

6.4.2 The Fire Protection Program, except training, is maintained under the direction of the Manager-Engineering Support. The Fire Protection Program meets or exceeds the guidelines of NFPA Code 27, 1975.

Fire Protection Training is maintained under the direction of the Training and Emergency Preparedness Manager. Fire Protection Training meets or exceeds the guidelines of NFPA Code 27, 1975, except retraining frequency. Fire Brigade and Fire Emergency Support Group (FB/FESG) members are required to attend retraining once per calendar quarter.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW BOARD (PRB)

FUNCTION

6.5.1.1 The PRB shall function to advise the Plant Manager on all matters related to nuclear safety.

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6.5.1.2 The PRB shall be composed of, as a minimum, a supervisor or higher level individual from each of the departments listed below:

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The Chairman, his alternate, and other members of the PRB shall be designated by the Plant Manager. The Chairman and his designated alternate shall both be managers of one of the six above listed departments or a higher level onsite manager.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PRB activities at any one time.

5.3.2 Audit Responsibility

5.3.2.1 The General Manager-Quality Assurance is responsible for an audit, conducted annually, of the activities of the Plant Manager and the Manager-Environmental Affairs, related to compliance with ETS.

5.3.2.2 Audits of facility activities shall be performed annually under the cognizance of the SRB to ensure conformance of facility operation to provisions of the ETS.

5.4 State and Federal Permit and Certificates

Section 401 of PL 92-500, the Federal Water Pollution Control Act Amendments of 1972 (FWPCA), requires any applicant for a Federal license or permit to conduct any activity that may result in any discharge into provisions of Sections 301, 302, 306, and 307 of the FWPCA. Section 401 of PL 92-500 further requires that any certification provided under this section shall set any effluent limitations and other limitations and monitoring requirements necessary to assure that any applicant for a Federal license or permit will comply with the applicable limitations. Certifications provided in accordance with Section 401 set forth conditions on the Federal license or permit for which the certification is provided. Accordingly, the licensee shall comply with the requirements set forth in the currently applicable 401 certification and amendments thereto issued to the licensee by the Georgia Environmental Protection Division. In accordance with the provisions of the Georgia Water Quality Control Act, the FWPCA and the rules and regulations promulgated pursuant to each of these acts, the Georgia Environmental Protection Division, under authority delegated by the U.S. EPA, issued NPDES permit No. GA 0004120 to the licensee. The NPDES permit authorizes the licensee to discharge from HNP Units 1 and 2 to the Altamaha River in accordance with effluent limitations, monitoring requirements, and other conditions stipulated in the permit.

Subsequent revisions to the certifications will be accommodated in accordance with the provisions of section 5.6.3.

5.5 Procedures

Detailed written procedures, including applicable checklists and instructions, shall be prepared and followed for all activities involved in implementing the ETS. All procedures shall be maintained in a manner convenient for review and inspection. Procedures that are the responsibility of the Plant Manager shall be kept at the plant. Procedures that are the responsibility of the Manager-Environmental Affairs shall be kept at the Georgia Power Company General Office.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 170 AND 108 TO

FACILITY OPERATING LICENSES DPR-57 AND NPF-5

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated March 2, 1990, Georgia Power Company, the licensee for the Edwin I. Hatch Nuclear Plant, proposed changes to the Technical Specifications (TSs) for Units 1 and 2. The ten proposed changes would modify TSs for Units 1 and 2 as follows:

1. Allow a 24-hour period of time for Unit 1 to meet the requirements for single-loop operation (SLO) before entering a 12-hour shutdown limiting condition for operation (LCO).

2. Allow placing an inoperable channel of a required Unit 1 Core and Containment Cooling System (CCCS) subsystem in the tripped condition or declaring the associated CCCS inoperable within one hour.

3. Change the Unit 1 definition of Surveillance Requirement to indicate that the performance of a surveillance requirement within the specified surveillance interval constitutes compliance with the operability requirement for an LCO.

4. Change a number of Unit 1 TSs and associated Bases to delete the requirement to perform additional surveillances when it is determined that the associated redundant components and/or subsystems are operable.

5. (a) Change Unit 1 TS Table 3.2-8 to specify the requirement for one operable channel per trip system instead of two channels, as now specified.

   (b) Modify the "Remarks" section to indicate that a trip signal will result in actuation of the Main Control Room Environmental Control System (MCRECS) in the pressurization mode and not in the isolation mode.

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(c) Modify TS Table 4.2-8 to delete Logic System Functional Test (LSFT) 6 and change LSFT 5 to MCRECS Control Room Pressurization Mode Actuation.

6. Delete a number of individual surveillance requirements for pumps and valves, and change testing frequencies and post-maintenance testing requirements in accordance with the testing requirements of the American Society of Mechanical Engineers (ASME) Section XI pursuant to 10 CFR 50.55a(g).

7. Administrative editorial changes to Unit 1 TSs and associated Bases.

8. Modify Unit 1 TSs 3.5.J.2.b through 3.5.J.2.e to delete references to diesel generators having to be operable and add information requiring that other Plant Service Water (PSW) components be operable.

9. Modify Unit 1 TSs 4.5.D.1.b and 4.5.E.1.c to clarify the location for measurement of steam pressure supply for the testing of the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) systems, and to specify the acceptable range of test pressures.

10. Administrative editorial changes to Unit 2 TSs.

2.0 EVALUATION

2.1 Proposed Change 1

This proposed change would modify Unit 1 TS 3.6.J.3 to allow a 24-hour period of time to meet the requirements for single-loop operation (SLO) before entering the 12-hour shutdown limiting condition for operation (LCO).

The change would make the Unit 1 requirements for extended SLO consistent with the Unit 2 requirements as approved by the NRC on June 10, 1987, in Amendment 77 to the Unit 2 TSs (specifically, Specification 3.4.1.1). The 24-hour period is mainly to allow time to adjust the flow-biased average power range monitor rod block and the simulated thermal power range monitor to account for minor changes in the core flow to drive flow relationship in SLO versus two-loop operation. Prior to approval of continuous SLO, the Unit 1 and Unit 2 TSs allowed operation with a single pump for up to 24 hours without taking any compensatory measures (an analyzed condition). Without the 24-hour time period, operation with one pump immediately puts Unit 1 in a 12-hour shutdown LCO.

Since the proposed change would make the Unit 1 TSs consistent with both the Unit 2 TSs and the BWR/4 Standard Technical Specifications, and since the additional time allowed before entry into the 12-hour shutdown LCO would not place the unit in an unanalyzed condition, we find this proposed change acceptable.

2.2 Proposed Change 2

This proposed change would revise Note "b" to TS Tables 3.2-2, 3.2-3, 3.2-4, 3.2-5, and 3.2-6 to allow an inoperable channel of a required Unit 1 Core and Containment Cooling System (CCCS) subsystem to be placed in the tripped condition without declaring the associated CCCS subsystem inoperable, provided at least one trip system is maintained with the minimum number of channels operable.

This proposed change would make the Unit 1 requirements for operating required CCCS subsystems with inoperable channels in the trip systems consistent with LCO requirements for Emergency Core Cooling System Actuation Instrumentation and Reactor Core Isolation Cooling System Actuation Instrumentation in Unit 2 TSs 3/4.3.3 and 3/4.3.4, respectively. Further, the LCO in Unit 1 TS 3.5 (Core and Containment Cooling Systems) assures the operability of CCCS subsystems under all conditions for which this cooling capability is required. Also, by maintaining the minimum number of operable channels on at least one trip system with the other system placed in the tripped condition, the single failure criterion as defined in 10 CFR Part 30, Appendix A, will still be satisfied. Accordingly, the NRC staff finds this proposed change acceptable.

2.3 Proposed Change 3

This proposed change would revise the Unit 1 TS Definition JJ, "Surveillance Requirements," to indicate that performance of a surveillance requirement within the specified surveillance interval constitutes compliance with the operability requirement for an LCO. The change also would revise TS 4.1.A to delete the requirement to functionally test other Reactor Protection System channels that monitor the same variable before the trip system containing the failed channel is tripped. The change would allow credit to be taken for the normal periodic surveillance as a demonstration of operability and availability of the redundant components and subsystems and is consistent with both the Unit 2 TSs and the BWR/4 STS. Implementation of the change would eliminate unnecessary challenges and wear to the redundant components and subsystems. However, by their submittal of March 17, 1989, the Licensee has requested modification of definition JJ in accordance with GL 87-09. The combination of the previously requested change and the change proposed by this action address the staff position as stated in GL 87-09. Accordingly, the NRC staff finds this change to be acceptable.

2.4 Proposed Change 4

This proposed change to the Unit 1 TSs would delete the requirements from a number of individual specifications to perform additional surveillances when it is determined the associated redundant components and/or subsystems have been found to be inoperable. Specific specifications and changes are as follows:

- Specification 4.4.B - Delete the requirement to demonstrate operability of redundant Standby Liquid Control System components when an associated component is found to be inoperable.

- Specification 4.5.A.2 - Delete the requirement to demonstrate operability of the other Core Spray (CS) loop and the Residual Heat Removal (RHR) system Low Pressure Coolant Injection (LPCI) mode when it is determined one CS loop is inoperable.

- Specification 4.5.B.2.a - Delete the requirement to demonstrate operability of the remaining LPCI pumps, associated flow paths, and the CS System when one LPCI pump is inoperable.

- Specification 4.5.B.2.b - Delete the requirement to demonstrate operability of all active components of the remaining LPCI subsystem and the CS System when one LPCI subsystem is inoperable.

- Specification 4.5.C.2 - Delete the requirement to demonstrate operability of the remaining active components of both RHR service water subsystems when one RHR service water pump is inoperable.

- Specification 4.5.C.3 - Delete the requirement to demonstrate operability of the remaining operable RHR service water subsystems when two RHR service water pumps are inoperable.

- Specification 4.5.D.2 - Delete the requirement to demonstrate operability of the ADS actuation logic, the RCIC System, the RHR System LPCI mode, and the CS System when the HPCI System is inoperable.

- Specification 4.5.E.2 - Delete the requirement to demonstrate operability of the HPCI System when the RCIC System is inoperable.

- Specification 4.5.F.2 - Delete the requirement to demonstrate operability of the HPCI System and the actuation logic of the other Automatic Depressurization System (ADS) valves when it is determined one of the seven ADS valves is incapable of automatic operation.

- Specification 4.5.G - The licensee proposed to delete the requirement to demonstrate operability of all the components in the RHR System LPCI mode and containment cooling mode connected to the operable diesel generator when it is determined one of the standby diesel generators is inoperable. The staff does not agree with the licensee deletion request. However, we accept deletion of the phrase "immediately and daily thereafter," and the replacement of the word, "demonstrated" with, "verfied."

- Specifications 4.5.J.2.b, 4.5.J.2.c, 4.5.J.2.d, 4.5.J.2.e - Delete the requirements to demonstrate operability of various Plant Service Water (PSW) components and systems when it is determined associated redundant PSW components and/or systems are inoperable.

- Bases for Specification 3.5.A.2 - Revise the wording to indicate additional surveillance is not required to be performed when CS components are inoperable. The change also provides additional information indicating that operability and availability of redundant components and subsystems of the CS System and the RHR System are demonstrated in Specifications 4.5.A, 4.5.B, 4.5.H, and 4.6.K.

- Bases for Specification 3.5.B.2 - Revise the wording to indicate additional surveillance is not required to be performed when LPCI components are inoperable. The change also provides additional information indicating that operability and availability of redundant components and subsystems of the LPCI mode and the CS System are demonstrated in Specifications 4.5.A, 4.5.B, 4.5.H, and 4.6.K.

- Bases for Specification 3.5.D.2 - Revise the wording to indicate additional surveillance is not required to be performed when HPCI components are inoperable. The change also provides additional information indicating that operability and availability of redundant components and subsystems of the CS System, LPCI System, RCIC System, and ADS are demonstrated in Specifications 4.5.A, 4.5.B, 4.5.E, 4.5.F, 4.5.H, and 4.6.K.

- Bases for Specification 3.5.E.2 - Revise the wording to indicate additional surveillance is not required to be performed when RCIC components are inoperable. The change also provides additional information indicating that operability and availability of redundant components and subsystems of the HPCI System are demonstrated in Specifications 4.5.D, 4.5.H, and 4.6.K.

- Bases for Specification 3.5.F.2 - Revise the wording to indicate additional surveillance is not required to be performed when ADS components are inoperable. The change also provides additional information indicating that operability and availability of redundant components and subsystems of HPCI and ADS are demonstrated in Specifications 4.5.D, 4.5.F, 4.5.H, and 4.6.K.

- Bases for Specification 3.5.J/4.5.J - The change provides additional information indicating that the operability and availability of redundant components and subsystems of the PSW System are demonstrated in Specifications 4.5.J and 4.6.K.

These changes are consistent with Proposed Change 3 and would eliminate unnecessary testing of, and challenges to, redundant components and subsystems. The changes would make the Unit 1 TSs consistent with the Unit 2 TSs and the BWR/4 STSs in this regard.

Except for RCIC System testing, the proposed surveillance requirements are those normally performed in accordance with ASME Section XI, pursuant to 10 CFR 50.55a(g). Normal testing required by ASME Section XI is significantly more comprehensive than the present testing requirements identified in the existing Unit 1 TSs. Therefore, the STS approach can be judged to be an equivalent or more reliable testing program.

The RCIC System is not currently in the ASME Section XI inservice testing program for Unit 1. However, existing Technical Specification 4.5.E.1.c requires pump flow tests every 3 and 18 months and provides adequate assurance the RCIC system is operable.

The NRC staff has reviewed these proposed changes and has found them to result in TSs that are consistent with both the Unit 2 TSs and with the BWR/4 STSs. They are, therefore, acceptable.

2.5 Proposed Change 5

This proposed change to the Unit 1 TSs would revise Table 3.2-8 to change the required operable channels per trip system from 2 to 1 for the Control Room Intake Radiation Monitors and would revise the "Remarks" section to indicate that actuation of the Main Control Room Environmental Control System (MCRECS) would be in the control room pressurization mode. In TS Table 4.2-8, the title of Logic System Functional Test (LSFT) 5 would be changed to "MCRECS Control Room Pressurization Mode Actuation," and LSFT 6 would be deleted.

The installed design has only two channels, one per trip system. The instrument channels are arranged in a one upscale, two downscale trip logic with the trip settings based on limiting the radioactivity from entering the control room from the outside. In the pressurization mode, the MCRECS stops the normal control room exhaust fan and takes in outside air through charcoal filters. The change from 2 to 1 operable channels per trip system corrects the TSs to reflect the actual installed design and is an editorial type correction. The change to the "Remarks" section to indicate that actuation of the MCRECS is in the pressurization mode should have been made at the time Amendment 156 was issued, which deleted a previously installed alternate operational mode (recirculation to avoid the possibility of chlorine gas entering the control room). This alternate operational mode was removed by Amendment 156 in conjunction with the licensee's elimination of the potential for release of chlorine gas on site. This change, therefore, is an administrative change to make the TSs conform to the present design. The changes to Table 4.2-8 also should have been made at the time Amendment 156 was issued to recognize the deletion of the recirculation mode of operation of the MCRECS. They are, thus, administrative in nature.

The NRC staff has reviewed proposed Change 5 and has concluded that the changes are administrative/editorial-type changes and that they are acceptable.

2.6 Proposed Change 6

This proposed change to the Unit 1 TSs would delete individual surveillance requirements for certain pumps and valves, and change testing frequencies and post-maintenance testing requirements in accordance with testing required by ASME Section XI pursuant to 10 CFR 50.55a(g). Specific specifications and changes are as follows:

- Specification 4.5.A.1 - Delete existing Sureveillance Requirements 4.5.A.1.c (Monthly Pump Operability) and 4.5.A.1.d (Monthly Motor-Operated Valve Operability), and add proposed Surveillance Requirement 4.5.A.1.c to verify valve lineups once per 31 days.

- Specification 4.5.B.1.a - Change the frequency of the test from "once per 5 years" to "once per 10 years."

- Specification 4.5.B.1 - Delete existing Surveillance Requirements 4.5.B.1.d (Monthly Pump Operability) and 4.5.B.1.e (Monthly Motor-Operated Valve Operability), and add proposed Surveillance Requirement 4.5.B.1.d to verify valve lineups once per 31 days.

- Specification 4.5.C.1 - Delete existing Surveillance Requirement 4.5.C.1.a (once per 3 months pump and valve operability), and add proposed Surveillance Requirement 4.5.C.1.a to verify valve lineup every 31 days.

- Specification 4.5.C.1 - Delete the requirement to perform a pump capacity test after pump maintenance.

- Specification 4.5.D.1 - Delete existing Surveillance Requirements 4.5.D.1.d (Monthly Pump Operability) and 4.5.D.1.e (Monthly-Motor Operated Valve Operability Surveillance Requirement), and add proposed Surveillance Requirement 4.5.D.1.c to verify proper valve lineup every 31 days.

- Specification 4.5.E.1 - Delete existing Surveillance Requirements 4.5.E.1.d (Monthly Pump Operability) and 4.5.E.1.e (Monthly Motor-Operated Surveillance Requirement), and add proposed Surveillance Requirement 4.5.E.1.d to verify proper valve lineup every 31 days.

The deletion of individual surveillance requirements for pumps and valves, and changes to test frequency and post-maintenance testing requirements are allowed due to the inservice testing provisions of TSs 3.6.K and 4.6.K. These specifications require inservice inspection/testing of ASME Code Class 1, 2 and 3 pumps and valves in accordance with Section XI of the ASME Boiler and Pressure Vessel code, pursuant to 10 CFR 50.55a(g). The testing required by ASME Section XI is significantly more comprehensive than the testing requirements now specified in the existing TSs.

The revised requirements for the RHR, CS, RHR Service Water, HPCI and RCIC systems are similar to the requirements in the Unit 2 TSs and the BWR/4 STSs and are acceptable.

2.7 Proposed Change 7

This proposed change to the Unit 1 TSs would make revisions that are purely editorial in nature. Specific administrative changes made for clarity are as follows:

- Specification 4.1, Table 4.1-1 - Delete the requirement for "Reactor Pressure Permissive." The basic function of these instruments listed in Table 4.1-1 was deleted from Specification 2.1.A.5 by Amendment 103. This function required an automatic scram on MSIV closure when the reactor was in the startup/hot standby mode, and the pressure was above 1045 psig. Since the high-pressure scram is operable for all modes of operation, this reactor permissive scram was not required. However, in Amendment 103, the requirement to perform the surveillance was not deleted from Table 4.1-1. This change corrects an inadvertent omission, is administrative in nature, and has no impact on plant operations or safety.

- Specification 4.5.B.1.c - Clarify that the discharge against which the test must be performed is "...a system head corresponding to a reactor vessel pressure of at least 20 psig." Clarifying what discharge pressure is being considered will eliminate any confusion that may arise over the point of pressure measurement.

- Specification 3.5.C.1.b - Add the word "or" at the end of the section.

- Specification 3.5.D.1.a - Change "1." to "(1)" and "2." to "(2)" to make the section numbering consistent with other sections.

- Specification 3.5.D.2 - Clarify what is meant by "adequate reactor steam pressure." The proposed change reads as follows: "...pressure is adequate (i.e., reactor pressure is such that the required steam pressure is maintained at the turbine for the duration of the test) to perform the tests." This change will eliminate any possible confusion as to when the 12-hour limitation must start.

- Specification 4.5.F.1 - Indent subsections 4.5.F.1.a and 4.5.F.1.b to make the section formatting consistent with other sections.

- Specification 4.5.H.2 - After the word "inoperable", add a comma (,) to correct a typographical error.

- Specification 4.5.H.1 - Change the time from when the high point vents of the Core Spray and LPCI Systems must be checked from "Every month prior to the testing of the LPCI and core spray systems," to "Every month, the discharge piping of the LPCI and core spray systems..." This is an editorial-type change to accommodate the changes made to delete monthly surveillance requirements of valves and pumps in the Core Spray and LPCI Systems.

- Bases for Specification 3.5.H - Change wording in the Bases to reflect the editorial changes necessary to accommodate changes made to delete monthly surveillance requirements of valves and pumps in the Core Spray and LPCI Systems.

- Specification 6.4.2 - Revise this specification to show who is responsible for each aspect of fire protection. This change is administrative in nature and clarifies the responsibilities for fire protection within the Plant Hatch management organization.

- Specification 5.4 of the Unit 1 Environmental Technical Specifications - Delete the words "effective August 1, 1983, through December 5, 1987" at the end of the first paragraph. This change eliminates the need to periodically revise the Environmental Technical Specifications when the effective dates of the NPDES permit change. Since Plant Hatch is required to maintain a current NPDES permit, deleting the effective dates is administrative in nature and has no impact on plant operation or safety.

The NRC staff has reviewed these proposed changes and agrees that they are editorial in nature. Accordingly, we find them acceptable.

2.8 Proposed Change 8

This proposed change would delete references to the diesel generators having to be operable from Unit 1 TSs 3.5.J.2.b through 3.5.J.2.e, and would add information to require operability of the other PSW components.

The diesel generator operability is assured by TS 3/4.9 and PSW component operability is assured by TS 3.5.J.2. Equipment operation is not affected, no new mode of failure is created, and equipment performance or safety analysis assumptions are not changed.

The proposed changes would make the Unit 1 TSs consistent with both the Unit 2 TSs and the BWR/4 STSs, and they are, therefore, acceptable.

2.9 Proposed Change 9

This proposed change would clarify where the pressure of the steam supply is to be measured for testing the HPCI and RCIC turbines, and would specify a range of pressures required for conducting the tests.

This change eliminates possible confusion as to the pressure to be measured. It would make the Unit 1 TSs consistent with both the Unit 2 TSs and the BWR/4 STSs and it is, therefore, acceptable.

2.10 Proposed Change 10

This proposed change to the Unit 2 TSs would make revisions that are purely editorial in nature. Specific administrative changes made for clarity are as follows:

- Specification 3.8.2.5 - In Specifications 3.8.2.5.a, 3.8.2.5.b, and 3.8.2.5.c, change the word "Circuit" to "Breaker." The change is a correction of terminology and has no impact on the breaker function.

- Specification 3.8.2.5 - In Specification 3.8.2.5.c, change "26" to "28" and "32" to "34." This change is due to a renumbering of breakers within distribution panels to provide a consistent breaker numbering scheme throughout the plant. There were no physical hardware changes or changes in breaker function; only breaker numbers were changed.

- Table 3.8.2.6-1 - For type 4 breakers, correct the cable numbers actually powered by the circuits which require primary containment penetration conductor overcurrent protection. No physical plant changes were made.

- Specification 6.4.2 - Revise the specification to show who is responsible for each aspect of fire protection. This change clarifies the responsibilities for fire protection within the Plant Hatch management organization.

- Specification 5.4 of the Unit 2 Environmental Technical Specifications - Delete the words "effective August 1, 1983 through December 5, 1987" at the end of the first paragraph. This change eliminates the need to periodically revise the Environmental Technical Specifications when the effective dates of the NPDES permit change. Plant Hatch is required to maintain a current NPDES permit at all times.

- Specifications 3.5.1 and 3.7.3 - In ACTION B, clarify what is adequate reactor steam pressure. The proposed change reads as follows: "...pressure is adequate (i.e., reactor pressure is such that the required steam pressure is maintained at the turbine for the duration of the test) to perform the tests." This change eliminates any possible confusion as to when the 12-hour limitation must start. Also, in Specification 3.5.1.a, the acronym "HPCI" has been substituted for "high pressure coolant injection."

- Specification 4.6.6.5.1.a - Add the word "isolation" after "...each 18 inch drywell and suppression chamber." In Amendment 58, a new subsection (3/4.6.6.5.1) was added to the TS; however, the word "isolation" was inadvertently omitted.

The staff has reviewed these proposed changes and agrees that they are merely editorial or clarifying in nature. Accordingly, we find them acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in the 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

4.0 CONCLUSION

The Commission's proposed determination that the amendments involve no significant hazards consideration was published in the Federal Register on May 16, 1990 (55 FR 20356). The Commission consulted with the State of Georgia. No public comments were received, and the State of Georgia did not have any comments.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Lawrence P. Crocker, PDII-3, DPR-I/II, NRR
Frank Rinaldi, PDII-3, DRP-I/II, NRR

Dated: August 30, 1990