July 11, 1984

DMB 016

Dockets Nos. 50-321
and 50-366

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Mr. J. T. Beckham, Jr.
Vice President - Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

The Commission has issued the enclosed Amendment Nos. 101 and 38 to Facility Operating License Nos. DPR-57 and NPF-5, respectively for the Edwin I. Hatch Nuclear Plant, Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your applications dated April 22, 1983 and September 9, 1983.

The amendments revise the TSs for Hatch Unit 1 to: reflect common reactor vessel water levels established in response to NUREG-0737, Item II.K.3.27 and for Hatch Unit 2 to reflect the modifications to pipe break detection circuitry made in response to NUREG-0737, Item II.K.3.15.

The amendments also revise the TSs for both Hatch Unit 1 and Unit 2 to: 1) in response to NUREG-0737, Item I.A.1.3, limit working hours of staff who perform safety-related functions; 2) in response to NUREG-0737, Item II.K.3.3, require reporting of safety/relief valve failures and challenges; 3) reflect RCIC automatic restart logic installed in response to NUREG-0737, Item II.K.3.13; and 4) reflect RCIC suction transfer logic installed in response to NUREG-0737, Item II.K.3.22.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next Monthly Notice.

Sincerely,

George W Rivenbark

George W. Rivenbark, Acting Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment Nos. 101 and 38
2. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	AD:OR:DL	OELD
~~RIngram~~ Norris	PKadambi;cf	GRivenbark	GLainas	
6/19/84	7/5/84	7/5/84	7/10/84	7/6/84

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PDR ADOCK 05000321
P PDR

Hatch 1/2
Georgia Power Company

50-321/366

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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 NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 101
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated April 22, 1983, 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 applications, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 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. 101 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Rivenbark

George W. Rivenbark, Acting Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 11, 1984

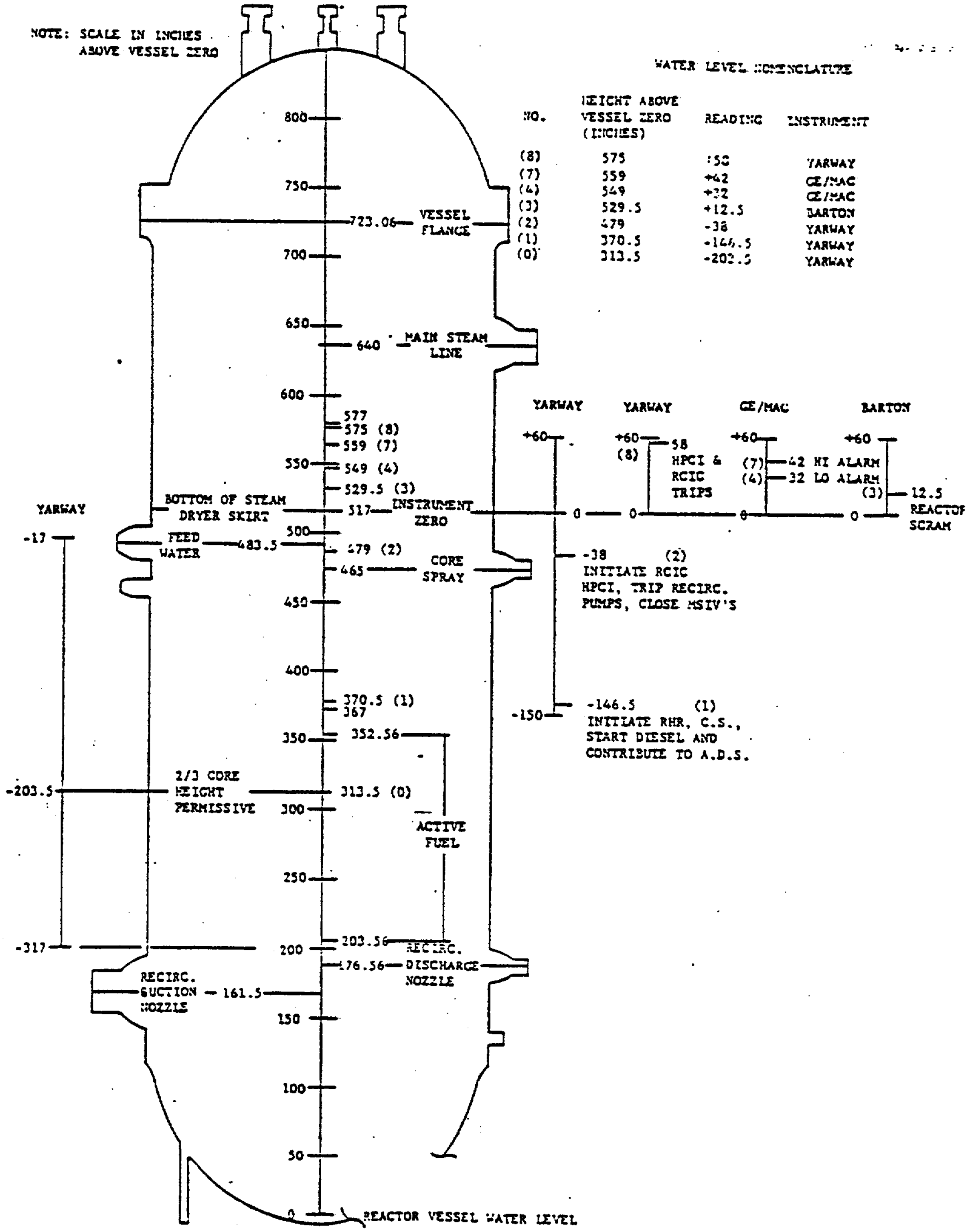
ATTACHMENT TO LICENSE AMENDMENT NO. 101

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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

Remove	Insert
Figure 2.1-1	Figure 2.1-1
3.2-9	3.2-9
--	3.2-9a
3.2-31	3.2-31
3.2-57	3.2-57
3.5-7	3.5-7
3.5-8	3.5-8
3.5-17	3.5-17
--	6-1a
6-15	6-15
6-17	6-17



WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	575	+58	YARWAY
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	529.5	+12.5	BARTON
(2)	479	-38	YARWAY
(1)	370.5	-146.5	YARWAY
(0)	313.5	-202.5	YARWAY

FIGURE 2.1-1

TABLE 3.2-3 (Continued)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
9.	RCIC Steam Line Pressure	Low	2	≥50 psig	Closes isolation valves in RCIC system, trips RCIC turbine.
10.	RCIC Steam Line Flow (Upstream and Downstream Elbow Taps)	High	1	≤300% Flow	Closes isolation valves in RCIC system, trips RCIC turbine.
11.	RCIC Turbine Exhaust Diaphragm Pressure	High	1	≤10 psig	Closes isolation valves in RCIC system, trips RCIC turbine.
12.	Suppression Chamber Area Air Temperature	High	1	≤175°F	Closes isolation valves in RCIC system, trips RCIC turbine.
13.	Suppression Chamber Area Differential Air Temperature	High	1	≤50°F	Closes isolation valves in RCIC system, trips RCIC turbine.
14.	RCIC Logic Power Failure Monitor		1	Not Applicable	Monitors availability of power to logic system.
15.	Condensate Storage Tank Water Level	Low	2	≥ 0"	Transfers suction from CST to suppression pool
16.	Suppression Pool Water Level	High	2	≤ 0"	Transfers suction from CST to suppression pool

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. Whenever 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, that system shall be repaired or the reactor shall be placed in the Cold Shutdown Condition within 24 hours after this trip system is made or found to be inoperable.

TABLE 4.2-3 (Continued)

Ref. No. (a)	Instrument	Instrument Check Minimum Frequency	Instrument Functional Test Minimum Frequency (b)	Instrument Calibration Minimum Frequency (c)
10	RCIC Steam Line ΔP (Flow)	None	(d)	Every 3 months
11	RCIC Turbine Exhaust Diaphragm Pressure	None	(d)	Every 3 months
12	Suppression Chamber Area Air Temperature	None	(d)	Every 3 months
13	Suppression Chamber Area Differential Air Temperature	None	(d)	Every 3 months
14	RCIC Logic Power Failure Monitor	None	Once/operating cycle	None
15	Condensate Storage Tank Level	None	Monthly	Every 3 months
16	Suppression Pool Water Level	None	Monthly	Every 3 months

Notes for Table 4.2-3

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

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

3.2.C.9. RCIC Steam Line Pressure Low (Continued)

lation setpoint of 50 psig is chosen at a pressure below that at which the RCIC turbine can effectively operate.

10. RCIC Steam Line Flow (High)

RCIC turbine high steam flow could indicate a break in the RCIC turbine steam line. The automatic closure of the RCIC steam line isolation valves prevents the excessive loss of reactor coolant and the release of significant amounts of radioactive materials from the nuclear system process barrier. Upon detection of RCIC turbine high steam flow the RCIC turbine steam line is isolated. The high steam flow trip setting of 300% flow was selected high enough to avoid spurious isolation, i.e., above the high steam flow rate encountered during turbine starts. The setting was selected low enough to provide timely detection of an RCIC turbine steam line break.

11. RCIC Turbine Exhaust Diaphragm Pressure High

High pressure in the RCIC turbine exhaust could indicate that the turbine rotor is not turning, thus allowing reactor pressure to act on the turbine exhaust line. The RCIC steam line isolation valves are automatically closed to prevent overpressurization of the turbine exhaust line. The turbine exhaust diaphragm pressure trip setting of 10 psig is selected high enough to avoid isolation of the RCIC if the turbine is operating, yet low enough to effect isolation before the turbine exhaust line is unduly pressurized.

12. Suppression Chamber Area Air Temperature High

As in the RCIC equipment room, and for the same reason, a temperature of 90 F + ambient will initiate a timer to isolate the RCIC turbine steam line.

13. Suppression Chamber Area Differential Air Temperature High

As for the RCIC equipment room differential temperature, and for the same reason, a high differential air temperature between the inlet and outlet ducts which ventilate the suppression chamber area will also initiate a timer to isolate the RCIC turbine steam line.

14. RCIC Logic Power Failure Monitor

The RCIC Logic Power Failure Monitor monitors the availability of power to the logic system. In the event of loss of availability of power to the logic system, an alarm is annunciated in the control room.

15. Condensate Storage Tank Level Low

The low CST level signal transfers RCIC suction from the CST to the suppression pool. The setpoint was chosen to ensure an uninterrupted supply of water during suction transfer.

16. Suppression Pool Water Level High

A high water level in the suppression chamber automatically switches RCIC suction from the CST to the suppression pool.

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 to perform the tests.

7. 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 113 psig or less within 24 hours.

4.5.D.1 Normal Operational Tests (Continued)

The HPCI pumps shall deliver at least 4250 gpm during each flow rate test.

d. Pump Operability	Once/month
e. Motor Operated valve operability	Once/month

2. Surveillance with Inoperable Components

When the HPCI system is inoperable, the ADS actuation logic, the RCIC system, the RHR system LPCI mode, and the CS system able immediately. The RCIC system and ADS logic shall be demonstrated to be operble daily thereafter until the HPCI system is returned to normal operation.

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 Automated Actuation (and restart*) Test	Once/Operating Cycle

*Automatic Restart on a Low Water Level Which is Subsequent to a High Level Trip.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.E.1. Normal System Availability (Continued)	4.5.E.1. Normal Operational Tests (Continued)
a.(2) when there is irradiated fuel in the reactor vessel and the reactor pressure is above 113 psig, except as stated in Specification 3.5.E.2.	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. Flow rate at normal reactor vessel operating pressure — Once/3 months and Flow rate at 150 psig reactor pressure — Once/Operating Cycle The RCIC pump shall deliver at least 400 gpm during each flow test.
	d. Pump Operability — Once/month
	e. Motor Operated valve operability — Once/month
2. Operation with Inoperable Components If the RCIC system is inoperable, the reactor may remain in operation for a period not to exceed seven (7) days if the HPCI system is operable during such time.	2. Surveillance with Inoperable Components When the RCIC system is inoperable, the HPCI system shall be demonstrated to be operable immediately and daily thereafter until the RCIC system is returned to normal operation.
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 113 psig within 24 hours.	

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. Both these systems are checked for operability if the HPCI system is determined to be inoperable. Considering the redundant systems, an allowable repair time of seven (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 core spray) are capable of providing all the cooling required for any plant event when nuclear system pressure is below 113 psig, the RCIC system is not required below this pressure. Between 113 psig and 150 psig the RCIC system need not provide its design flow, but reduced flow is required for certain events. 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 seven (7) days, provided that the HPCI system is operable during this period.

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 he nuclear system so that the low-pressure coolant injection (LPCI) and the core spray system can operate to protect the fission product barrier. Note that this Specification applies only to the automatic feature of the pressure relief system.

g. Administrative procedures shall be developed and implemented to limit the working hours of Unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, auxiliary operators, health physicists, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance, or major plant modifications, the following guidelines shall be followed on a temporary basis:

(1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

(2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

(3) A break of at least eight hours should be allowed between work periods, including shift turnover time.

(4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant General Manager or his deputy of higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant General Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

ANNUAL REPORTS (Continued)

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel, including contractors, receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[2] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. Documentation of all challenges to safety/relief valves.

c. Any other unit unique reports required on an annual basis.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office of Inspection and Enforcement no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specification 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

[2]This tabulation supplements the requirements of § 20.407 of 10 CFR Part 20.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components which requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Failure or malfunction of any safety/relief valve.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event*. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

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*All Type B and Type C Leakage Tests (i.e., Local Leak Rate Tests) that fail (i.e., test leakage is such that an LER would be required) during an outage shall be reported per one thirty-day written report and shall be submitted within 30 days of the end of such an outage.

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. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. NPF-5

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

   A. The applications for amendment by Georgia Power Company, et al., (the licensee) dated April 22, 1983, and September 9, 1983, comply 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;

   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. 38 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Rivenbark

George W. Rivenbark, Acting Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 11, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
3/4 3-20	3/4 3-20
3/4 3-34	3/4 3-34
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36
3/4 7-9	3/4 7-9
3/4 7-10	3/4 7-10
B3/4 3-2	B3/4 3-2
B3/4 7-1	B3/4 7-1
B3/4 7-2	B3/4 7-2
6-1a	6-1a
6-14	6-14
6-16	6-16

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	≤13*
b. Area Temperature - High	≤13*
c. Area Ventilation Temperature ΔT - High	≤13*
d. SLCS Initiation	NA
e. Reactor Vessel Water Level-Low	≤13*
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a. HPCI Steam Line Flow-High	3 ≤ Isolation Time ≤ 13*
b. HPCI Steam Supply Pressure - Low	≤13*
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA
d. HPCI Equipment Room Temperature - High	NA
e. Suppression Pool Area Ambient Temp. - High	NA
f. Suppression Pool Area ΔT - High	NA
g. Suppression Pool Area Temp. Timer Relays	NA
h. Emergency Area Cooler Temperature - High	NA
i. Drywell Pressure - High	≤13*
j. Logic Power Monitor	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	3 ≤ Isolation Time ≤ 13*
b. RCIC Steam Supply Pressure - Low	NA
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. Emergency Area Cooler Temperature - High	NA
e. Suppression Pool Area Ambient Temp. - High	NA
f. Suppression Pool Area ΔT - High	NA
g. Suppression Pool Area Temperature Timer Relays	NA
h. Drywell Pressure - High	≤13*
i. Logic Power Monitor	NA
6. SHUTDOWN COOLING SYSTEM ISOLATION	
a. Reactor Vessel Water Level - Low	NA
b. Reactor Steam Dome Pressure - High	NA

TABLE 3.3.4-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM NUMBER OF OPERABLE CHANNELS PER TRIP SYSTEM
a. Reactor Vessel Water Level - Low Low (2B21-N031 A, B, C, D)	2
b. Condensate Storage Tank Water Level - Low (2E51-N060, 2E51-N061)	2(a)
c. Suppression Pool Water Level-High (2E51-N062A, B)	2(a)

(a) Provides Signal to RCIC Pump Suction Valves Only

TABLE 3.3.4-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - Low Low	$\geq$ -38 inches*	$\geq$ -38 inches*
b. Condensate Storage Tank Level - Low	$\geq$ 0 inches**	$\geq$ 0 inches**
c. Suppression Pool Water Level-High	$\leq$ 151 inches	$\leq$ 151 inches

*See Bases Figure B 3/4 3-1

** This corresponds to a level of 131'-0" above mean sea level.

TABLE 4.3.4-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level- Low Low	D	M	Q
b. Condensate Storage Tank Level- Low	NA	M	Q
c. Suppression Pool Water Level- High	NA	M	Q

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 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.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation (and restart*) and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.

2. Verifying that the system will develop a flow of at least 400 gpm on recirculation flow when steam is supplied to the turbine at a pressure of 150 + 15, -0 psig.

3. Verifying that suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a low condensate storage tank level or a high suppression pool level signal.

*Automatic restart on a low water level signal which is subsequent to a high level trip.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY requirements, trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For DC operated valves a 3-second delay is assumed before the valve starts to move. For the AC operated valves it is assumed that the AC power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the DC operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 13 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13 second delay. It follows that checking the valve speeds and the 13 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

Minimum response times for isolation of HPCI or RCIC on high steam line flow prevent spurious isolations due to pressure spikes in the steam supply.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument is used to send the actuation signal to several systems at the same time.

3/4.7.1 SERVICE WATER SYSTEMS

The OPERABILITY of the service water systems ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.2 MAIN CONTROL ROOM ENVIRONMENTAL CONTROL SYSTEM

The OPERABILITY of the main control room environmental control system ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during the following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 10 of Appendix "A", 10 CFR Part 50.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the emergency core cooling equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 150 psig even though the residual heat removal (RHR) system provides adequate core cooling up to 350 psig.

The RCIC system specifications are applicable during CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

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.

With RCIC inoperable, adequate core cooling is assured by the demonstrated OPERABILITY of the HPCI system and justifies the specified 14 day out-of-service period.

The surveillance requirements provide adequate assurance that RCIC will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest moment.

3/4.7.4 HYDRAULIC SNUBBERS

The hydraulic snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. The only snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety related system.

The inspection frequency applicable to snubbers containing seals fabricated from materials which have been demonstrated compatible with their operating environment, only ethylene propylene compounds to date, is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during an inspection of these snubbers determines the time interval for the next required inspection of these snubbers. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed, nominal time less 25%, may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Observed failures of these sample snubbers will require functional testing of additional units. To minimize personnel exposures, snubbers installed in high radiation zones or in especially difficult to remove locations, as identified in Table 3.7.4-1, may be exempted from these functional testing requirements provided the OPERABILITY of these snubbers was demonstrated during functional testing at either the completion of their fabrication or at a subsequent date.

g. Administrative procedures shall be developed and implemented to limit the working hours of Unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, auxiliary operators, health physicists, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance, or major plant modifications, the following guidelines shall be followed on a temporary basis:

(1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

(2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

(3) A break of at least eight hours should be allowed between work periods, including shift turnover time.

(4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant General Manager or his deputy of higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant General Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

ANNUAL REPORTS (Continued)

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel, including contractors, receiving exposures greater than 100 mrme/yr and their associated man rem exposure according to work and job functions,[2] e.g., reactor operations and surveillance, inservice inspection routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. Documentation of all challenges to safety/relief valves.

c. Any other unit unique reports required on an annual basis.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office, of Inspection and Enforcement no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports shall be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

[2]This tabulation supplements the requirements of § 20.407 of 10 CFR Part 20.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components which requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Failure or malfunction of any safety/relief valve.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event*. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be—supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

*All Type B and Type C Leakage Tests (i.e., Local Leak Rate Tests) that fail (i.e., test leakage is such that an LER would be required) during an outage shall be reported per one thirty-day written report and shall be submitted within 30 days of the end of such an outage.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. DPR-57

AND AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. NPF-5

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNITS NOS. 1 AND 2

DOCKETS NOS. 50-321 AND 50-366

1. INTRODUCTION

In November 1980, the staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements" which included all TMI Action Plan items approved by the Commission for implementation at nuclear power reactors. NUREG-0737 identifies those items for which Technical Specifications are required. A number of items which require Technical Specifications were scheduled for implementation by December 31, 1981. The staff provided guidance on the scope of Technical Specifications for all of these items in Generic Letter 83-02. Generic Letter 83-02 was issued to all Boiling Water Reactor (BWR) licensees on January 10, 1983. In Generic Letter 83-02 the staff requested licensees to:

a. review their facility's Technical Specifications to determine if they were consistent with the guidance provided in the generic letter, and

b. submit an application for a license amendment where deviations or absence of Technical Specifications were found.

By letters dated April 22, 1983 and September 9, 1983, Georgia Power Company (the licensee) responded to Generic Letter 83-02 by submitting Technical Specification change requests for Edwin I. Hatch, Units 1 and 2. By letter dated May 2, 1983, the licensee withdrew its requests for some of the changes it had requested in its April 22, 1983 letter. This evaluation covers the balance of the change requests, which includes the following TMI Action Plan Items:

a. Limit Overtime (I.A.1.3)
b. Reporting of Safety and Relief Valve Failures Challenges (II.K.3.3)
c. RCIC Restart and RCIC Suction (II.K.3.13 and II.K.3.22)
d. HPCI and RCIC Modification (II.K.3.15)
e. Common Reference Level (II.K.3.27)

2. DISCUSSION AND EVALUATION

A. Limit Overtime (I.A.1.3)

On June 15, 1982, we transmitted to licensees of operating plants a revised version of the Commission's Policy Statement on nuclear power plant staff working hours (Generic Letter 82-12). This policy statement was also referenced in Generic Letter 83-02.

The licensee has proposed adequate Technical Specifications to establish administrative procedures to limit working hours of the staff who perform safety-related functions and establish guidelines on the use of overtime. We have reviewed the licensee's proposed Technical Specifications and find that they adequately satisfy the intent of the Commission's Policy Statement and the guidelines provided in Generic Letter 83-02. We find these Technical Specifications to be acceptable.

B. Reporting of Safety/Relief Valve Failures and Challenges (II.K.3.3)

In Generic Letter 83-02, the staff requested licensees to formalize the reporting requirements for safety/relief valve failures and challenges. The licensee has proposed Technical Specifications which will require the licensee to report the failures promptly with written follow-up, and the challenges in an annual report. This is consistent with our guidance provided in Generic Letter 83-02. Therefore we find it acceptable.

C. RCIC Restart and RCIC Suction (II.K.3.13 and II.K.3.22)

TMI Action Plan Items II.K.3.13 and II.K.3.22 recommend modifications to the Reactor Core Isolation Cooling System (RCIC) such that:

1. The system will restart on subsequent low water level after it has been terminated by a high water level in the reactor vessel, and

2. RCIC system suction will automatically switchover from the condensate storage tank to the suppression pool when the condensate storage tank level is low.

In Generic Letter 83-02, the staff provided the guidance on necessary changes in the Technical Specifications for implementation of the modifications. The proposed changes in Technical Specifications for RCIC are in response to Generic Letter 83-02. We have reviewed the proposed changes in the Technical Specifications and determined that the changes are consistent with the guidance provided in Generic Letter 83-02. We find the changes acceptable.

D. Isolation of HPCI and RCIC Modifications (II.K.3.15)

TMI Action Plan Item II.K.3.15 recommends that the pipe-break-detection circuitry should be modified so that pressure spikes resulting from high-pressure coolant injection (HPCI) and RCIC system initiation will not cause inadvertent system isolation. The licensee has completed the modification recommended by this item.

The staff provided guidance on the necessary changes in the Technical Specifications by Generic Letter 83-02. The licensee has proposed changes in the Technical Specifications for Hatch Unit 2. The licensee indicated that present Technical Specifications for Hatch Unit 1 do not include isolation system instrumentation response times. However, the surveillance requirements on time delay relay are assured by logic system functional tests which are required by present HPCI and RCIC specifications.

We have reviewed the current Technical Specifications for HPCI and RCIC systems for Hatch Unit 1 and proposed changes in the Technical Specifications for Hatch Unit 2. We have determined that specifications for Unit 1 adequately cover the surveillance requirements on time delay relay included in HPCI and RCIC systems. We have also determined that the proposed changes in Unit 2 are consistent with our guidance in Generic Letter 83-02. We find the proposed changes to be acceptable.

E. Common Reference Level (II.K.3.27)

The guidance provided in Generic Letter 83-02 recommended that the figure defining reactor vessel water levels should be changed to reflect the common reference level established by this Action Plan Item. A sample figure was provided in the guidance.

In response to Generic Letter 83-02, the licensee submitted a proposed change in Figure 2.1-1 for Hatch Unit 1. This figure for Hatch Unit 2 was revised by a previous amendment.

We have reviewed the revised figure which defines reactor vessel water levels. We find that it reflects the common reference level established by the Action Plan Item II.K.3.27. We find the proposed change to be acceptable.

3. ENVIRONMENTAL CONSIDERATIONS

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 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 these amendments.

4. CONCLUSION

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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 11, 1984

Principal Contributor: C. Patel