May 26, 1987

Docket No.: 50-321

Mr. James P. O'Reilly
Senior Vice President - Nuclear Operations
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. O'Reilly:

Subject: Issuance of Amendment No. 137 to Facility Operating License DPR-57 - Edwin I. Hatch Nuclear Plant, Unit 1 (TAC 64786)

The Commission has issued the enclosed Amendment No. 137 to Facility Operating License DPR-57 for the Edwin I. Hatch Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications in response to your application dated March 4, 1987.

The amendment modifies the Technical Specifications to permit hydrostatic and leak testing with a non-critical reactor core.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/s/

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

Enclosures:
1. Amendment No. 137 to DPR-57
2. Safety Evaluation

cc w/enclosures:
See next page

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	PD#II-3/DRP-I/II
MDuncan/rad	LCrocker	BJYoungblood
05/14/87	05/19/87	05/19/87

Mr. J. T. Beckham, Jr.
Georgia Power Company

Edwin I. Hatch Nuclear Plant,
Units Nos. 1 and 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 NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 137
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 4, 1987, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 137, 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

/S/

B. J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 26, 1987

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	OGC-Bethesda	PD#II-3/DRP-I/II
MDuncan/rad	LCrocker	M. Karman	BJYoungblood
05/14/87	05/19/87	05/21/87	05/22/87

ATTACHMENT TO LICENSE AMENDMENT NO. 137

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 vertical lines indicating the areas of change. The overleaf pages are provided for convenience.

Remove Page	Insert Page
3.5-6	3.5-6
3.5-8	3.5-8
3.5-9	3.5-9
3.6-9	3.6-9
Figure 3.6-1	Figure 3.6-1
3.7-2	3.7-2

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
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 seven (7) days provided all redundant active components in both of the RHR service water subsystems are operable.	4.5.C.3. Two Pumps Inoperable When two RHR service water pumps are inoperable, the remaining operable RHR service water subsystems and their associated diesel generators shall be demonstrated to be operable immediately and daily thereafter for seven (7) days or until the inoperable components are returned to normal operation.
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	D. High Pressure Coolant Injection (HPCI) System
1. Normal System Availability	1. Normal Operational Tests HPCI system testing shall be performed as follows:
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 pressure is greater than 150 psig, except as stated in Specification 3.5.D.2.*	**Item** / **Frequency** a. Simulated automatic actuation test — Once/Operating Cycle. b. Flow rate at normal reactor vessel operating pressure — Once/3 months and Flow rate at 150 psig reactor pressure — 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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.E.1. Normal System Availability (Cont.)	4.5.E.1. Normal Operational Tests (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.*	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 150 psig within 24 hours.	

* 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.

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 seven (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

When it is determined that one of the seven ADS valves is incapable of automatic operation, the HPCI system and the actuation logic of the other ADS valves shall be demonstrated to be operable immediately and daily thereafter until all seven ADS valves are capable of automatic operation.

* 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.

3.6.H.1. Relief/Safety Valves

a. When one or more relief/safety valve(s) is known to be failed an orderly shutdown shall be initiated and the reactor depressurized to less than 113 psig within 24 hours. Prior to reactor startup from a cold condition all relief/safety valves shall be operable.**

b. With one or more relief/safety valve(s) stuck open, place the reactor mode switch in the shutdown position.

c. With one or more safety/relief valve tailpipe pressure switches of a safety/relief valve declared inoperable and the associated safety/relief valve(s) otherwise indicated to be open, place the reactor mode switch in the Shutdown position.

d. With one safety/relief valve tailpipe pressure switch of a safety/relief valve declared inoperable and the associated safety/relief valve(s) otherwise indicated to be closed, plant operation may continue. Remove the function of that pressure switch from the low low set logic circuitry until the next COLD SHUTDOWN. Upon COLD SHUTDOWN, restore the pressure switch(es) to OPERABLE status before STARTUP.

e. With both safety/relief valve tailpipe pressure switches of a safety/relief valve declared inoperable and the associated safety/relief valve(s) otherwise indicated to be closed, restore at least one inoperable switch to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6.H.1. Relief/Safety Valves

a. End of Operating Cycle

Approximately one-half of all relief/safety valves shall be benchchecked or replaced with a benchchecked valve each refueling outage. All 11 valves will have been checked or replaced upon the completion of every second operating cycle.

b. Each Operating Cycle

Once during each operating cycle, at a reactor pressure > 100 psig each relief valve shall be manually opened until thermocouples downstream of the valve indicate steam is flowing from the valve.

c. Integrity of Relief Valve Bellows*

The integrity of the relief valve bellows shall be continuously monitored and the pressure switch calibrated once per operating cycle and the accumulators and air piping shall be inspected for leakage once per operating cycle.

d. Relief Valve Maintenance

At least one relief valve shall be disassembled and inspected each operating cycle.

e. Operability of Tail Pipe Pressure Switches

The tail pipe pressure switch of each relief/safety valve shall be demonstrated operable by performance of a:

1. Functional Test:

   a. At least once per 31 days, except that all portions of instrumentation inside the primary containment may be excluded from the functional test, and

** The Relief/Safety valves are not required to be operable for performance of inservice hydrostatic or pressure testing with reactor pressure greater than 113 psig and all control rods inserted. Overpressure protection will be provided as required by ASME code.

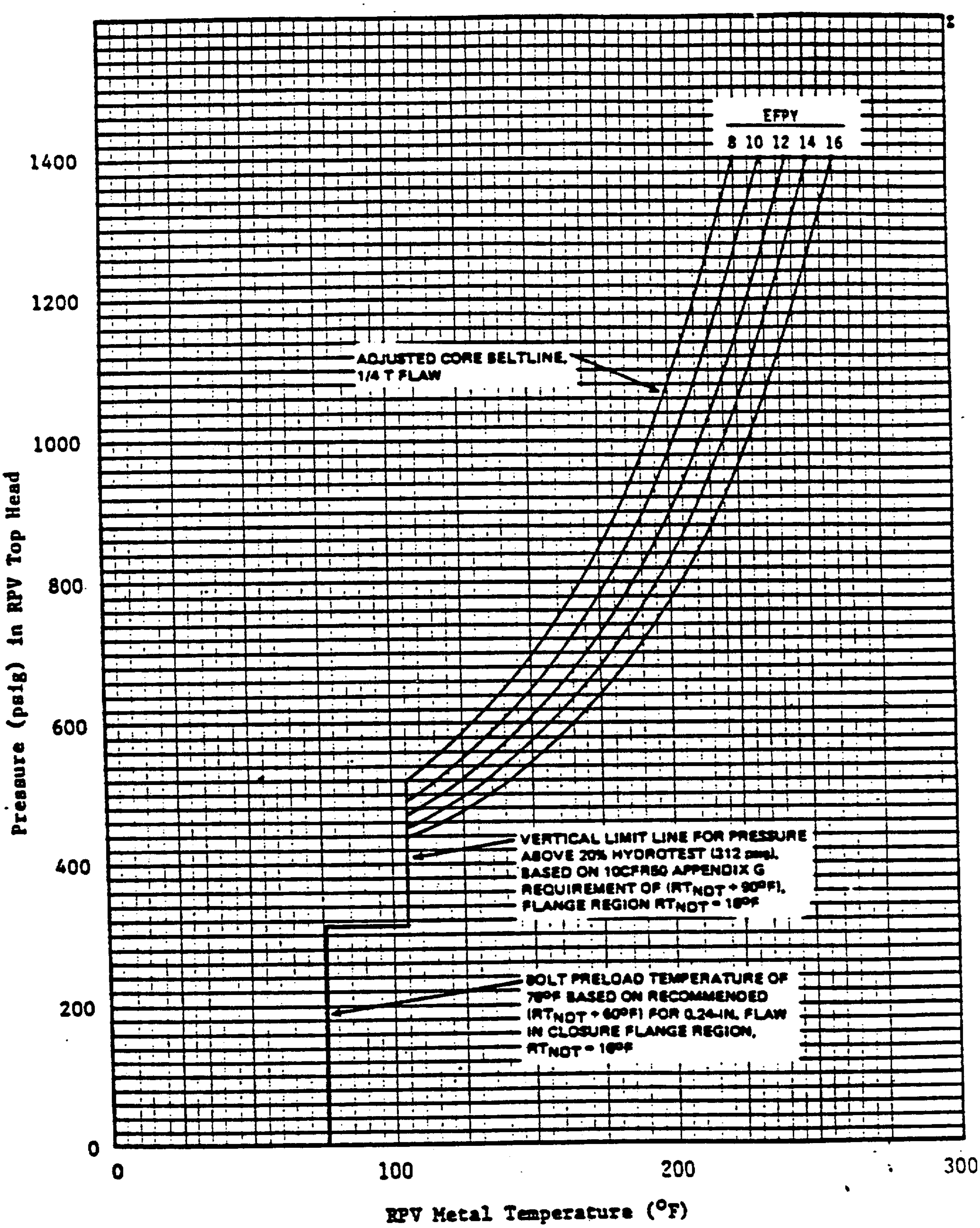

Figure 3.6-1 Pressure versus Minimum Temperature for Pressure Tests, Based on Surveillance Test Results

2. Primary Containment Integrity

Primary containment integrity is required:

a. Prior to withdrawing control rods for the purpose of going critical.

b. Whenever the reactor is critical.

c. Whenever the reactor water temperature is above 212°F and fuel is in the reactor vessel.**

An exception is made while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt, during which time primary containment integrity is not required.

4.7.A.2. Leak Testing to Verify Primary Containment Integrity

Primary containment integrity shall be demonstrated by the following test procedures:

a. Type A Tests - Integrated Leak Rate Test (ILRT)*

Primary containment integrity is confirmed if the leak rate does not exceed the maximum allowable leak rate, L_a, of 1.2 weight percent of the contained air per 24 hours at the peak test pressure.

(1) Type A tests shall be performed under the program established in Appendix J of 10 CFR Part 50. (Reference 1).

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* L_a - Maximum allowable peak pressure test leak rate - 1.2 weight percent per day
L_t - Maximum allowable reduced pressure test leak rate
L_{am} - Measured peak pressure test leak rate - values are subject to change with each ILRT performed
L_{tm} - Measured reduced pressure test leak rate - values are subject to change with each ILRT performed
L_{ao} - Allowable operational leak rate for peak pressure tests - values are subject to change with each ILRT performed
L_{to} - Allowable operational leak rate for reduced pressure tests - values are subject to change with each ILRT performed
(All leakage rates measured in weight percent of contained air per 24 hours)
P_a - Peak test pressure - 59 psig
P_t - Reduced test pressure - 29.5 psig

** Primary Containment Integrity is not required for performance of inservice hydrostatic or leak testing with reactor coolant temperature greater than 212°F and all control rods inserted.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.137 TO

FACILITY OPERATING LICENSE DPR-57

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

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-321

INTRODUCTION

In a letter to the Commission dated March 4, 1987 (Reference 1), the Georgia Power Company (the licensee) proposed changes to the Technical Specifications for the Edwin I. Hatch Nuclear Plant, Unit 1, Operating License DPR-57, to permit hydrostatic and leak testing with a non-critical reactor core. The licensee historically has used nuclear heat for performing inservice hydrostatic and leak testing at the Hatch Nuclear Plant. Section XI of the ASME Boiler and Pressure Vessel Code (Table IWB-2500-1) requires that the Systems Leakage Test be performed prior to startup following each refueling outage. The requirements are incorporated by reference as part of the Commission's regulations. The staff's position is that hydrostatic and leak testing are to be performed before the reactor goes critical from a refueling outage.

The licensee stated that the use of non-nuclear heat for the performance of hydrostatic and leak testing would necessitate changes to the Technical Specifications (TS) in three areas. The proposed changes are the following:

1. The first change would provide exception to the operability requirements for the Reactor Core Isolation Cooling (RCIC) system, the High Pressure Coolant Injection (HPCI) system, the automatic depressurization systems (ADS), and the safety/relief valves (S/RV) during performance of either hydrostatic or leak tests using non-nuclear means to heat the reactor coolant.

2. The second change would provide a refinement to TS Figure 3.6-1, providing pressure/temperature limit curves for fluence values expected for 8, 10, 12, 14 and 16 Effective Full Power Years (EFPY) of reactor operation.

3. The third change would allow either hydrostatic or leak tests to be performed with all control rods inserted at a coolant temperature greater than 212°F, without primary containment integrity.

2.0 EVALUATION

Each of the requested changes is evaluated separately.

2.1 Exemption to operability requirements for the RCIC, HPIC, ADS and S/RV systems during hydrostatic and system leakage testing.

The proposed use of non-nuclear heat for the performance of hydrostatic and leak testing in place of nuclear heating requires the use of recirculation pump operation and a water-solid reactor pressure vessel to achieve necessary temperatures (range 192 to 236 F) and pressures (range 1005 to 1086 psig). Under the proposed conditions, the lack of steam generation precludes operation of the HPCI and RCIC turbine driven pumps. The licensee's proposal is to add the following note to the TS for the two affected systems:

> "HPCI (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."

In a related matter, conduct of the tests at the proposed conditions results in test pressures greater than the lift pressures for the safety/relief valves (S/RVs). The licensee's solution is to gag the S/RVs, which may be allowed by the inclusion of the following note for the S/RVs and automatic depressurization system (ADS) in the TS:

> "The ADS (Relief/Safety) valves are not required to be operable for the performance of inservice hydrostatic or leak testing, with reactor pressure greater than 113 psig and all control rods inserted."

The current requirements for operability of the aforementioned systems when the reactor coolant temperature is greater than 212°F are to ensure the capability for makeup of reactor vessel water inventory for decay heat removal in the event of a small leak with feedwater capability lost and the main condenser not available. During hydrostatic and leak testing, control rods are fully inserted, the decay heat level is low following a refueling outage, and the reactor is maintained at or near cold shutdown conditions. (The reactor mode switch will be in either the shutdown or refueling position.) Therefore, the intended function of the systems is not required when the hydrostatic and leak tests are being performed. On this basis, the staff concludes that the proposed change to the TS Limiting Conditions for Operation (LCOs) which will eliminate the requirement for system operability during testing when the reactor coolant temperature is in excess of 212°F is acceptable.

2.2 Revise TS Figure 3.6-1 to provide pressure/temperature limit curves for fluence values expected for 8, 10, 12, 14 and 16 EFPY of reactor operation.

The licensee proposed to refine Figure 3.6-1, "Pressure Versus Minimum Temperature for Pressure Tests, such as Required by ASME Section XI," to provide pressure/temperature limit curves for fluence values for 8, 10, 12, 14 and 16 EFPY. In addition to Figure 3.6-1, Figures 3.6-2, "Pressure Versus Minimum Temperature for Non-Nuclear Heatup/Cooldown and Low Temperature Physics Tests," and 3.6-3, "Pressure Versus Minimum Temperature for Core Critical Operation (Includes 40°F Margin Required by 10 CFR 50, Appendix G)", curves valid for 16EFPY were reviewed and evaluated. Our evaluation was documented in a memorandum to Daniel R. Muller from Gus C. Lainas, March 10, 1986 (Reference 2). We concluded that the curves were conservative and met the requirements of 10 CFR 50, Appendix G and H, ASTM E-185, Regulatory Guide 1.99, Revision 1, and Appendix G, Section III of the ASME Boiler and Pressure Vessel Code.

Figure 3.6-1 in the Technical Specifications provides conservative pressure/temperature limits for the performance of hydrostatic and pressure tests based on the anticipated fracture toughness of the reactor vessel. Currently the figure contains one curve, which is valid for 16 EFPY, estimated from the damage received by irradiation surveillance specimens after 5.75 EFPY. Since the measured fracture toughness shift in the surveillance plate exceeded the predicted shift of RT_{NDT}, using the methodology of Regulatory Guide 1.99, Revision 1, by a factor of 2.76, the predicted damage was increased by that factor in preparing the 16 EFPY curve.

The Spring 1987 Hatch Unit 1 outage was initiated after approximately 7.12 EFPY of reactor operation. As proposed by the licensee, the refined Figure 3.6-1 consists of a series of curves in 2 EFPY increments from 8 to 16 EFPY operation. The proposed curve for 16 EFPY is the same pressure/temperature curve previously reviewed and evaluated by the staff (Reference 2).

The purpose of adding the series of curves from 8 to 16 EFPY is to eliminate conservatism in the pressure/temperature limits and permit hydrostatic and pressure tests to be performed at lower temperatures, which will require less recirculation pump heatup time. It is planned to use the 8 EFPY curve for the Spring 1987 test. No change is proposed for Figures 3.6-2 and 3.6-3 in the Technical Specifications.

The proposed change to Figure 3.6-1 of the Technical Specifications of Hatch Unit 1 is essentially administrative and allows hydrostatic and leak testing to be performed with the minimum amount of heat stored in the reactor coolant to meet the reactor vessel nil-ductility requirements. As the pressure/temperature limits meet the requirements of 10 CFR 50, Appendixes G and H, ASTM E185, Regulatory Guide 1.99, Revision 1, and Appendix G, Section III of the ASME Code, we find the proposed change to be acceptable.

2.3 Exception to the requirement to maintain primary system integrity during hydrostatic or leak testing with the reactor coolant greater than 212°F and fuel in the reactor vessel.

The proposed use of non-nuclear heating for the performance of hydrostatic or leak tests involves a non-critical core, water-solid conditions, low temperatures and low fuel decay heat values. Under these conditions primary containment integrity is not required since the secondary containment will be operable pursuant to TS 3.7.C.2 and capable of handling any airborne radiation or steam leaks that could occur. Under the proposed test conditions, the potential for failed fuel and subsequent increase in coolant activity above Technical Specification levels will be mitigated and the amount of stored energy in the primary system is small. The potential consequences become those resulting from release of steam to the secondary containment and subsequent use of the Standby Gas Treatment System (SGTS) to limit radioactive releases to the environment. The licensee has proposed the addition of the following note to the TS LCO for Primary Containment Integrity:

> "Primary Containment Integrity is not required for performance of inservice hydrostatic or leak testing with reactor coolant temperature greater than 212°F and all control rods inserted."

On the basis of the expected minimal consequences of a potential release under the proposed test conditions, the staff concludes that the proposed addition to the TS LCO for Primary Containment Integrity is acceptable.

ENVIRONMENTAL CONSIDERATIONS

The amendment involves a change in use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves 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 exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there have been no public comments on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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 amendment.

CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (52 FR 9568) on March 25, 1987, and 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Michael McCoy
Felix Litton

Dated: May 26, 1987

DATED May 26, 1987

AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE DPR-57, EDWIN I. HATCH, UNIT 1

DISTRIBUTION:
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