March 17, 1994

Docket No. 50-366

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

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Dear Mr. Beckham:

SUBJECT: ISSUANCE OF AMENDMENT - EDWIN I. HATCH NUCLEAR PLANT, UNIT 2 (TAC NO. M87850)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 132 to Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit 2. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 1, 1993, as revised January 6, 1994, and supplemented February 3, 1994.

The amendment revises the TS to increase the allowable leakage rate specified in TS 3.6.1.2 from the current 11.5 standard cubic feet per hour (scfh) for any one main steam isolation valve (MSIV) to 100 scfh for any one MSIV with a total maximum pathway leakage of 250 scfh through all four main steam lines. The amendment also modifies the TS Index, TS 3/4.6.1.4, Table 3.6.3-1, and Basis 3/4.6.1.4 to permit the deletion of the MSIV Leakage Control System from the TS.

A copy of the related Safety Evaluation and a Notice of Issuance, which has been forwarded to the Office of the Federal Register for publication, are also enclosed.

Sincerely,

ORIGINAL SIGNED BY:

Kahtan N. Jabbour, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000366
P PDR

Enclosures:
1. Amendment No. 132 to NPF-5
2. Safety Evaluation
3. Notice of Issuance

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cc w/enclosures:
See next page

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JAN
3/16/94

W. Hodges
MWH w/ changes indicated 3/16/94

*PREVIOUS CONCURRENCES

OFFICE	LA:PD23:DRPE	PM:PD23:DRPE	PM:PD23:DRPE	BC:OTSB:DORS #94-046	D:PD23:DRPE
NAME	LGBerry	CECarpenter	KNJabbour KNJ	CIGrimes	DBMatthews
DATE	3/10/94	3/1/94	3/1/94	3/2/94	3/17/94
OFFICE	BC:SPLB:DSSA*	BC:EMEB:DE	BC:ECGB:DE	BC:PRPB:DRSS*	OGC
NAME	CEMcCracken	JANorberg	GBagchi	LCunningham	EHollen
DATE	3/1/94	3/2/94	3/2/94	3/2/94	3/16/94

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DF01

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 132
License No. NPF-5

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

A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Facility Operating License No. NPF-5 filed by the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated October 1, 1993, as revised January 6, 1994, and supplemented February 3, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and 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. 132 , 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 from the date 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:
Technical Specification
Changes

Date of Issuance: March 17, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 132

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 vertical lines indicating the areas of change.

Remove Pages	Insert Pages
VII	VII
XII	XII
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-7	3/4 6-7
3/4 6-24	3/4 6-24
B 3/4 6-2	B 3/4 6-2

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$, 1.2 percent by weight of the containment air per 24 hours at P_a, 57.5 psig, or

2. $\leq L_t$, 0.849 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 28.8 psig.

b. A combined leakage rate of:

1. $\leq 0.60\ L_a$ for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests when pressurized to P_a, and

2. $\leq 0.009\ L_a$ for the following penetrations*:

(a) Main steam condensate drain, penetration 8;

(b) Deleted

(c) Reactor water cleanup, penetration 14;

(d) Equipment drain sump discharge, penetration 18;

(e) Floor drain sump discharge, penetration 19; and

(f) Chemical drain sump discharge, penetration 55;

(g) Deleted

c. When tested at 28.8 psig**, 100 scf per hour for any one main steam isolation valve and a combined maximum pathway leakage rate of 250 scf per hour for all four main steam lines.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

*Potential bypass leakage paths.
**Exemption to Appendix J of 10 CFR 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

With:

a. the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or

b. the measured combined leakage rate for all penetrations and valves, except main steam isolation valves, subject to Type B and C tests exceeding 0.60 L_a or with the measured combined leakage rate for all specified potential bypass leakage path penetrations exceeding 0.009 L_a, or

c. the main steam isolation valve measured leak rate exceeding 100 scf per hour for any one MSIV or a total maximum pathway leakage rate of > 250 scf per hour for all four main steam lines,

Restore:

a. the overall integrated leakage rate(s) to < 0.75 L_a or < 0.75 L_t as applicable, and

b. the combined leakage rate for all penetrations and valves, except main steam isolation valves, subject to Type B and C tests to $\leq$ 0.60 L_a and the combined leakage rate for the specified potential bypass leakage path penetrations to $\leq$ 0.009 L_a, and

c. the leakage rate to $\leq$ 11.5 scf per hour for any main steam isolation valve that exceeds 100 scf per hour, and restore the combined maximum pathway leakage rate to $\leq$ 250 scf per hour,

Prior to increasing the reactor coolant temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - (1972):

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a, 57.5 psig or at P_t, 28.8 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Deleted

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

B. MANUAL ISOLATION VALVES[a]

1. Deleted

2. RHR return to recirculation loop isolation valves
   2E11-F015A, B

3. LOCA H_2 recombiner isolation valves
   2T49-F002 A, B
   2T49-F004 A, B

4. Core spray isolation valves
   2E21-F005A, B

5. Service air isolation valves
   2P51-F651
   2P51-F513

6. RBCCW supply and return isolation valves
   2P42-F051
   2P42-F052

[a] Includes power operated valves which do not isolate automatically.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Deleted

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the primary containment steel vessel will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 57.5 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations on primary containment internal pressure ensure that the containment peak pressure of 57.5 psig does not exceed the maximum allowable internal pressure of 62 psig during LOCA conditions or that the external pressure does not exceed the design maximum external pressure of 2 psig. The limit of 0.75 psig for initial positive containment pressure will limit the total pressure to 57.5 psig which is less than the maximum allowable internal pressure and is consistent with the accident analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the accident analysis.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 132 TO FACILITY OPERATING LICENSE NPF-5

GEORGIA POWER COMPANY, ET AL.

EDWIN I. HATCH NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-366

1.0 INTRODUCTION

By letter dated October 1, 1993, as revised January 6, 1994, and supplemented February 3, 1994, Georgia Power Company, et al. (GPC or the licensee), proposed a license amendment to change the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Unit 2 (Hatch or the facility), Facility Operating License No. NPF-5. The January 6 and February 3, 1994, letters provided additional and clarifying information that did not change the initial scope of the October 1, 1993, application and the initial proposed no significant hazards consideration determination.

The licensee proposed an alternative to Regulatory Guide (RG) 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," by utilizing the main steam lines, the drain lines, and the main condenser as an alternate method for main steam isolation valve (MSIV) leakage treatment. The proposed changes are the result of extensive work performed by the Boiling Water Reactor Owners Group (BWROG) in support of the resolution of NRC Generic Issue C-8, "Main Steam Isolation Valve Leakage and Leakage Control System Failure." In addition to the licensee's submittals, General Electric (GE) Report NEDC-31858P, Revision 2, "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems," dated September 1993, provided the technical justification for the proposed changes.

The proposed changes would:

1. increase the allowable leakage rate specified in TS 3.6.1.2 from the current 11.5 standard cubic feet per hour (scfh) for any one MSIV, to 100 scfh for any one MSIV with a total maximum pathway leakage of 250 scfh through all four main steam lines;

2. add a new requirement in TS 3.6.1.2 related to restoration of acceptable leak rates if any of the proposed limits are exceeded, such that if any MSIV exceeds 100 scfh, it will be repaired and retested to meet a leak rate limit of 11.5 scfh per valve (the current criterion for leakage);

3. modify TS 3.6.1.4, Table 3.6.3-1, and Basis 3/4.6.1.4 to delete the MSIV Leakage Control System (LCS) from the TS; and,

4. administratively modify the Index, TS 3.6.1.4 and 4.6.1.4, and Basis 3/4.6.1.4 to rearrange the sections and page numbering to reflect the above requested changes.

2.0 BACKGROUND

Each of the four main steam lines (MSL) contains two (inboard and outboard), quick-closing MSIVs. These valves function to isolate the reactor system in the event of a break in a steam line outside the primary containment, a design-basis loss-of-coolant accident (LOCA), or other events requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur. Operating experience at various BWR plants has indicated that degradation has occurred occasionally in the leak-tightness of MSIVs, and the specified low leakage has not always been maintained.

Due to recurring problems with excessive leakage of MSIVs, the staff issued RG 1.96, which recommends the installation of a supplemental LCS to ensure that the isolation function of the MSIVs complies with the limits specified. The licensee's safety-related MSIV LCS is designed to control the release of fission products. The LCS develops a negative pressure, by use of a series of blowers, in the sections of the MSL between the inboard and outboard MSIVs, and between the outboard MSIVs and the turbine stop valves. The leakage is discharged to the standby gas treatment system.

Due to design limitations, the LCS is ineffective when the MSIV leak rate is greatly in excess of the TS-allowable value. Hence, NRC Generic Issue C-8 was initiated in 1983 to assess: (1) the causes of MSIV failures, (2) the effectiveness of the LCS and alternative leakage paths, and (3) the need for regulatory action to limit public risk. The staff's resolution of Generic Issue C-8, published in NUREG-1372, "Regulatory Analysis for the Resolution of Generic Issue C-8, 'Main Steam Isolation Valve Leakage and LCS Failure,'" dated June 1990, concluded that no backfit was warranted to reduce public risk associated with MSIV leakage and that maintaining the current requirements, systems, and leakage treatment practices, should be adequate. Furthermore, the staff concluded that there was insufficient basis for a generic requirement to remove the LCS from operation, although plant-specific requests to remove the LCS may be justified.

The BWROG formed an MSIV Leakage Committee in 1982 to identify and resolve the causes of high MSIV leakage rates. The BWROG then formed an MSIV Leakage Closure Committee to address alternate actions to resolve ongoing but less severe MSIV leakage problems and to address the limited capability of the LCS. The results of these committee activities were submitted to the NRC in several General Electric (GE) proprietary reports. These reports are: NEDC-31643P, dated November 1988; NEDC-31858P, Revision 0, dated February 1991; NEDC-31858P, Revision 1, dated October 1991; and, NEDC-31858P, Revision 2, dated September 1993. The reports are all titled "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems."

The latest GE report concludes that the proposed increase of the MSIV leakage limit will reduce radiation exposures to maintenance personnel, reduce outage durations, and extend the effective service life of the MSIVs. The report also concludes that the proposed elimination of the LCS will similarly reduce exposures to maintenance personnel and reduce outage durations, and that the LCS can be replaced with an alternate method for MSIV leakage treatment using the MSL and condenser. The licensee referred to this report as a basis for deleting the TS requirements for the MSIV LCS, and requested a higher MSIV leak rate limit.

The proposed alternative treatment method recommended in the BWROG report, and proposed by the licensee, takes advantage of the large volume in the main steam lines and main condenser to provide holdup and plateout of fission products that may leak through closed MSIVs. This method uses the main steam drain lines to direct leakage to the main condenser. In this approach, the main steam piping, the drain piping, and the main condenser are used to mitigate the consequences of an accident that could lead to potential offsite exposures in excess of 10 CFR Part 100 limits. However, as required by General Design Criterion (GDC) 2 of Appendix A to 10 CFR Part 50, the components and piping systems used in the alternative treatment path must be capable of performing their function during and following a safe-shutdown earthquake (SSE). The BWROG report and the licensee's submittals provide the technical justification for the seismic capability of the alternate treatment path and also provide the dose calculations to demonstrate the acceptability of the system.

3.0 EVALUATION

This evaluation has been performed in four parts. Section 3.1 provides the radiological evaluation; Section 3.2 provides the evaluation for seismic analysis of piping, supports, and condenser; Section 3.3 provides the drain path functional design evaluation; and Section 3.4 provides the overall conclusions.

3.1 RADIOLOGICAL EVALUATION

To demonstrate the adequacy of the Hatch Unit 2 engineered safety features (ESFs), the licensee assessed the offsite radiological consequences that could result from the occurrence of design-basis-accidents (DBAs) with a total MSIV leak rate of 250 scfh from the four MSL and without the MSIV LCS. The licensee presented the results of the offsite dose calculations in their submittal. The Hatch Unit 2 ESFs are designed to mitigate the radiological consequences of the DBAs.

During its operating license review stage, the staff had assessed the offsite radiological consequences of a LOCA at Hatch Unit 2. The calculated results are shown in Table 15.1 of NUREG-0411, "Safety Evaluation Report related to the Operation of Edwin I. Hatch Nuclear Plant, Unit 2 (June 1978)" (OL-SER). In the OL-SER, the staff considered the following sources and radioactivity transport paths to the environment, following a LOCA:

(1) containment leakage;

(2) MSIV leakage; and

(3) post-LOCA leakage from the ESFs outside containment.

In this evaluation, the staff recalculated the radiological consequences resulting from the same radioactivity transport paths in the OL-SER. The procedures used in the staff's recalculation of offsite radiological consequences are based on the current TID-14844 source term, which is consistent with the guidelines in the applicable Standard Review Plan (SRP, NUREG-0800) sections, regulatory guides, and the Hatch Unit 2 OL-SER, except for the following two deviations:

(1) the staff has credited the removal of radioactive iodine in the MSL and the main condenser due to holdup for decay and deposition; and

(2) the staff has deleted the TS requirements for the MSIV LCS.

The staff's recalculated offsite and control room operator doses resulting from a postulated LOCA are shown in the revised OL-SER Table 15-1. The parameters used in the staff's recalculation are the same as those listed in OL-SER Table 15-2, "Assumptions Used to Calculate Loss-of-Coolant Accident Doses," except for the two deviations stated above. The licensee has not claimed any credit for airborne fission-product removal by the suppression pool following a LOCA.

3.1.1 <u>Iodine Release Pathways</u>

Following a LOCA, three potential release pathways exist for main steam leakage through the MSIVs:

(1) main steam drain lines to the condenser, with delayed release through the low-pressure turbine seals;

(2) turbine bypass lines to the condenser with delayed release through the low-pressure turbine seals; and

(3) MSL through the turbine stop and control valves and through high-pressure turbine seals.

The consequences of leakage from pathways 1 and 2 will be essentially the same, since the condenser will process the MSIV leakage. The condenser's iodine-removal efficiency will vary depending on the inlet location of the bypass or drainline piping; however, in either case, iodine will be removed. For pathway 3, MSIV leakage through the closed turbine stop and control valves will not be processed via the condenser. For this case, the high-pressure turbine (having a large internal surface area associated with the turbine blades and casing) will remove iodine.

Previous experience with these valves leads the staff to believe that, as long as either turbine bypass or drain line leakage pathway is available, MSIV leakage through the closed turbine stop and control valves (pathway 3) will be negligible. Essentially all of the releases will be through the main condenser because there will be no differential pressure in the MSL downstream of the MSIVs following the closure of the valves.

The licensee has selected pathway 1 to mitigate the radiological consequences of an accident that could result in potential offsite exposures comparable to the dose reference values specified in 10 CFR Part 100. The staff has accepted the licensee's proposed pathway. In the calculation of the contribution to the LOCA dose, the staff assumed that the inboard MSIV failed to close, thus allowing potentially contaminated steam to travel to the outboard MSIV. The total leak rate, from both this outboard MSIV combined with the other three MSL outboard MSIVs, was assumed to be 250 scfh.

3.1.2 Iodine Transport Model

Chemical and physical principles predict that gaseous iodine and airborne iodine particulate material will deposit on surfaces. Several laboratory and in-plant studies have demonstrated that gaseous iodine deposits by chemical adsorption and particulate iodine deposits through a combination of sedimentation, molecular diffusion, turbulent diffusion, and impaction. Gaseous iodine exists in nuclear power plants in several forms: elemental (I_2), hypoiodous acid (HOI), organic (CH_3I), and particulate. In accordance with RG 1.3, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," dated June 1974, the staff assumed that 91 percent of the reactor core's iodine inventory is in the elemental form (includes hypoiodous acid), 5 percent in the particulate form, and 4 percent in the form of organic iodides.

Each of these forms deposits on surfaces at a different rate, described by a parameter known as the deposition velocity. The elemental iodine form, being the most reactive, has the largest deposition velocity, and organic iodide has the smallest. Further, studies of in-plant airborne iodine show that iodine (elemental and particulate) deposited on the surface undergoes both physical and chemical changes and can either be resuspended as an airborne gas or become permanently fixed to the surface. The data also show that the iodine can change its form so that iodine deposited as one form (usually elemental) can be resuspended in the same or in another form (usually organic). Conversion can be described in terms of resuspension rates that are different for each iodine species. Chemical surface fixation similarly can be described in terms of a surface fixation rate constant.

The transport of gaseous iodine in elemental and particulate forms has been studied for many years, with several groups proposing different models to describe the observed phenomena (References 1 - 5). The staff used the model specifically developed by an NRC contractor (Reference 6) for iodine removal in BWR MSL and the main condenser following a LOCA.

The staff model treats the MSIV leakage pathway as a sequence of small segments for which instantaneous and homogeneous mixing is assumed, the mixing computed for each segment is passed along as input to the next segment. The number of segments depends upon the parameters of the line and flow rate, and can be as many as 100,000 for a long, large-diameter pipe with low flow. Each line segment is divided into five elements that represent the concentrations of the three airborne iodine species, the surface that contains iodine available for resuspension, and surface iodine that has reacted and is fixed on the surface.

The staff model considers three iodine species: elemental, particulate, and organic. For the purpose of the staff model a fourth species, hypoiodous acid, was considered to be a form of elemental iodine. All the iodine in a segment undergoes radioactive decay.

The GE model, as well as the one used by the staff, is based on time-dependent temperature adsorption phenomena with instantaneous and perfect mixing in a given volume. Both models use the same MSIV leakage pathways. They differ, however, in the treatment of buildup of iodine in the main steam lines and condenser. GE assumed steady-state iodine in equilibrium in a large volume, while the staff model assumed transient buildup of iodine in a finite number of small volumes. The staff does not consider these differences to be significant, because the staff finds that the resulting radiological consequences (calculated doses) are in good agreement.

The staff's transport model also assumed iodine transport through the condenser as dilution flow rather than the plug flow as in the steam lines. The staff assumed that the iodine input into the condenser mixes instantaneously with a volume of air in the condenser, and that the diluted air exhausts at the same time and same rate as the input air (MSIV leakage) flows into the condenser.

Using published data, the staff developed the equations for iodine deposition velocities, resuspension rates, and surface fixation rates as a function of temperature. The equations and data are contained in the contractor's report (Reference 6). The equation for the deposition velocity of elemental iodine is based on the least-squares fit to the available data. Deposition velocity equations for HOI and organic iodine are based on the values at 30°C. Due to the lack of data at elevated temperatures, the temperature dependence is assumed to be similar to elemental iodine. Resuspension and fixation equations as a function of temperatures are based on measurements available in the literature at ambient temperature. The staff assumed that resuspension and fixation rates will increase with increasing temperature.

The technical references, and the GE and staff models indicate that particulate and elemental iodine would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. Therefore, the staff believes that an appropriate credit for the removal of iodine in the MSL and main condenser should be given in the radiological consequence assessment following a design-basis accident. Consequently, the staff accepted the licensee's proposed elimination of the LCS and allowed a higher MSIV leakage.

The parameters used to evaluate iodine transport and removal in MSL and condenser are listed in Table 15.4. Calculated iodine releases from the condenser after holdup and plateout in the MSL and condenser are shown in Table 15.5.

For the purpose of giving credit for iodine holdup and plateout, the staff's model requires that the main steam piping (including its associated piping to the condenser) and the condenser remain structurally intact following an SSE, so they can act as a holdup volume for fission products. By the term "structurally intact," the staff assumes the steam line will retain sufficient structural integrity to transport the relatively low flow rate ($\leq$ 2 cubic feet per minute (cfm)) of MSIV bypass leakage throughout the steam lines and condenser. The staff considers, in its radiological consequence assessment, that the condenser is open to the atmosphere via leakage through the low-pressure turbine seals. Thus, it is only necessary to ensure that gross structural failure of the condenser will not occur.

3.1.3 Control Room Habitability

The control rooms for Hatch Units 1 and 2 are housed in a shared facility. The control room habitability systems are designed to serve the combined control room facility of both units. During normal operation, the control room is maintained at a slightly positive pressure with respect to the adjacent turbine building. During an emergency, the Hatch control room emergency filtration system supplies outside air to the control room to pressurize it. The system is designed to maintain the control room at slight positive pressure related to adjacent areas. The pressurization is accomplished by introducing 400 cfm of outside air, which is mixed with 2100 cfm of control room return air before entering the control room emergency filtration unit. The filtration unit is an engineered safety feature and is redundant. Both trains contain, among other things, a 2-inch deep charcoal adsorber.

The staff has evaluated the doses to the operators in the control room following a postulated LOCA and found that the calculated doses were within the guidelines of SRP Section 6.4 (OL-SER Section 6.4.2). The staff considered that the fission-product releases from the low-pressure turbine seal are due to the MSIV leakage (up to 250 scfh total) through the MSIV drain lines and the main condenser.

In its evaluation of airborne radioactivity concentrations in building wakes, the staff assumed a ground level release of airborne fission products from the turbine building as a diffusion source and the control room emergency air intake as a single-point receptor. The staff estimated the control room building wake atmospheric dispersion parameters (X/Q) in accordance with the guidelines provided in the SRP. The parameters used in the staff's assessment, and the recalculated control room operator doses following a postulated LOCA, are listed in Table 15.6.

The staff finds that the recalculated whole-body and equivalent organ doses (thyroid) are still within the guidelines of SRP Section 6.4.

3.1.4 Conclusion

Several technical references (References 1 - 5) including an NRC contractor's report (Reference 6) indicate that particulate and elemental iodine would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. The staff, therefore, concludes that an appropriate credit for the removal of iodine in the MSL and main condenser should be taken in the radiological consequence assessment following a DBA. The amount of iodine removal credit for Hatch MSL and the main condenser is shown in Table 15.5.

The staff has reviewed the licensee's analysis and has independently assessed the radiological consequences resulting from the MSIV leakage transport pathway described in this safety evaluation. The recalculated thyroid and whole-body doses are listed in revised OL-SER Table 15.1. Based on the above evaluation and the calculated radiological consequences shown in Table 15.1, the staff concludes that the MSIV leak rate limit of 250 scfh total from four MSL and the proposed deletion of the TS requirements for the MSIV LCS are acceptable.

The staff further concludes that the existing distance to the exclusion area and to the low-population-zone boundaries of the Hatch plant, in conjunction with the remaining ESFs provided in the Hatch plant remain sufficient to provide reasonable assurance that the radiological consequences of a postulated LOCA will be within the dose reference values stated in 10 CFR Part 100, and the dose limits specified in GDC 19 of Appendix A to 10 CFR Part 50.

3.2 EVALUATION FOR SEISMIC ANALYSIS OF PIPING, SUPPORTS, AND EQUIPMENT

GPC proposed to use the main steam piping, drain lines, and main condenser as an alternate method for MSIV leakage treatment. Because certain main steam piping and components were not designed as seismic Category I items, the licensee has performed detailed evaluations and seismic verification walkdowns to demonstrate that the main steam system piping and equipment that constitute the alternate treatment path are seismically rugged and meet GDC 2 of Appendix A to 10 CFR Part 50 with regard to seismic adequacy.

These proposed changes to the TS are supported by work performed by the BWROG, with the licensee's participation. This work, as documented in GE Report NEDC-31858P, Revision 2, serves as the generic basis of the acceptability of the above Hatch 2 proposal. The staff reviewed the report and found the BWROG approach of utilizing the earthquake experience data to demonstrate the seismic ruggedness of nonseismically analyzed main steam system piping and main condenser, as supplemented by plant-specific seismic walkdowns, to be generally acceptable for this amendment request.

The BWROG has retained Earthquake Engineering, Inc. (EQE) as a consultant to conduct a review of the earthquake experience data on the performance of facility piping and condenser. The study summarized the data on the performance of main steam system piping and condenser in non-nuclear applications that experienced strong motion earthquakes. EQE also compared these piping systems and condenser to the piping systems and condenser typically used in domestic BWR plants. The result of the comparison appears to support the BWROG contention that main steam piping and condenser employed in GE BWR would maintain their pressure-retention capability during a design-basis earthquake. EQE stated that, for welded steel piping and condenser designed and constructed to normal industrial practices (e.g., ANSI B31.1 and Heat Exchange Institute (HEI) standards, respectively), earthquake experience shows that welded steel piping and condenser are seismically rugged, contain some safety margin, and have not shown a primary collapse mode of failure. A relatively small number of seismically induced piping failures have occurred due to excessive relative support movements or seismic interactions.

The primary components to be relied upon for pressure boundary integrity in resolution of the Hatch Unit 2 MSIV leakage issue are: (1) the main turbine condenser, (2) the MSL from the turbine stop and bypass valves, and (3) the main steam turbine bypass and drain line piping to the condenser. The condenser forms the ultimate boundary of the leakage pathway. Boundaries upstream of the condenser were established by existing valves, and were used to limit the extent of the seismic verification walkdown. Specifically, normally closed valves will be assured to remain closed; normally open valves will be required to close and remain closed; and other valves that require operator action will be operated to ensure closure.

3.2.1 <u>Seismic Analysis of Piping and equipment</u>

To confirm the capability of the main steam piping and condenser to serve as an alternate leakage treatment system, the licensee has performed seismic verification walkdowns to assure that the MSL, the steam drain lines, the condenser, and interconnecting piping and equipment that are not seismically analyzed fall within the bounds of the design characteristics of the seismic experience database as discussed in Section 6.7 of the BWROG report. Specifically, the walkdowns were performed to (1) physically verify that Hatch plant features have the attributes similar to those in the earthquake experience database that have demonstrated good seismic performance, (2) verify general conformance of pipe support spans to the requirements of ANSI B31.1, and (3) identify potential seismic vulnerabilities considering those structural details and causal factors that resulted in component damages at database plants. These potential vulnerabilities were identified as "outliers" for subsequent resolution. The licensee's October 1, 1993, submittal presents a complete list of the outliers identified during the walkdowns. The licensee found these outliers to fall within one of the following five types:

(1) potential deficiency in anchorage or support capacity;

(2) potential valve malfunction and collapse of the masonry walls which support the piping;

(3) potential damaging interaction between piping and nearby components;

(4) differential displacement of piping supports or attachments; and

(5) valves with extended motor operators beyond screening guidelines.

These outliers have been either evaluated or analyzed by the licensee to demonstrate their acceptability as they currently exist, or plant modifications will be implemented to resolve the concerns. As a result of the walkdowns, GPC noted that 13 components needed minor modifications or repairs.

As stated in the licensee's October 1, 1993, submittal, portions of the Hatch Unit 2 main steam and drain piping systems were originally seismically analyzed in accordance with the ASME Code Section III, Class 2. The analyzed lines included the main steam piping (from the MSIV to the turbine stop valves), the main steam bypass (to the bypass valves), the drain line portion in the reactor building, and the portions of various main steam branch connections to the seismic anchors downstream of the isolation valves. Design methods for these analyzed lines are consistent with seismic Category I analysis methods for Hatch Unit 2.

According to the licensee, the remainder of the Hatch main steam system piping, including main steam drain to the condenser and interconnected systems, is made of welded steel piping and standard support components, and was designed by rule and approximate methods. This piping is similar in diameter, thickness, and material to those installed in the plants that are in the earthquake experience database. Each one of the outliers, identified in the walkdowns as a potential source of damage, was either evaluated to demonstrate its acceptability as it exists, or designated to be modified as previously noted. The licensee has provided reasonable assurance that upon completion of all necessary modifications, the supports will keep the piping in place, and the piping pressure boundary integrity will be maintained, under normal and earthquake loadings.

The October 1, 1993, submittal also stated that the overall size (in terms of heat transfer area) and weight of the Hatch Unit 2 main condenser is generally enveloped by the condenser in the earthquake experience database. The overall dimensions of the Hatch main condenser are represented by the experience database as well. It was also indicated that the anchorage capacity-to-seismic demand ratios for the Hatch Unit 2 main condenser are higher than those at the selected database sites. The licensee stated, therefore, that the design parameters of Hatch Unit 2 condenser are enveloped by the design parameters of condenser at the facilities that are in the database.

At the December 10, 1993, meeting at NRC headquarters, EQE presented the survey results for EQE data and open literature for 18 earthquakes that covered 29 sites and 96 power plants. The EQE database covers facilities with underlying foundations varying from soft soils to rock. Hundreds of structures with a wide diversity of structural types and design criteria are included that house thousands of pipe runs, cable trays, conduits, tubing, and related components. In addition, thousands of equipment installations, from

1930s vintages to new items, are in the database. The 18 strong-motion earthquakes range in Richter magnitude from 5.4 to 8.1. The average peak ground accelerations (PGAs) range from 0.1 g to 0.85 g, with strong motions lasting up to about 50 seconds. The survey found no precedent for failure of main steam piping pressure boundary or the condenser shell. The survey did, however, find damage to piping insulation, valve operators, piping supports, and condenser tubes.

On the basis of comparison of the database earthquakes and Hatch Unit 2 design horizontal ground motions, the staff found that the Hatch Unit 2 design ground motions are generally enveloped by the experience earthquakes, in the frequency range of interest. With the exception of piping of smaller diameters, which were not well represented in the BWROG report, the staff found that the nonanalyzed portions of the Hatch Unit 2 main steam piping and condenser are generally bounded by the earthquake experience database.

At the December 10, 1993, meeting, the staff requested that the licensee broaden the piping database beyond those presented in Table 4-4 of BWROG Report NEDC-31858P, especially for piping of smaller diameters. The licensee's submittal of January 6, 1994, provided such additional data for a wide range of both large- and small-bore piping which demonstrated good seismic performance during other strong motion earthquakes not covered in BWROG report NEDC-31858P, Rev. 2. This submittal provided a detailed database that included 24 earthquakes and about 126 sites, some of which were originally included in the BWROG report. The measured or estimated horizontal ground accelerations for these sites range from 0.15 g to 1.0 g, with the majority of the sites experiencing an acceleration of 0.3g or higher. The duration of strong motion (on the order of 0.10 g or higher) ranges from 5 seconds to more than 50 seconds. This provided further assurance to the staff that the design-basis ground motions of Hatch Unit 2 are enveloped by those of the experience earthquakes.

The staff conducted an engineering audit at Plant Hatch on January 12 and 13, 1994. During the audit, the licensee provided the staff detailed walkdown procedures for Hatch Unit 2. The procedures include detailed review criteria for piping and tubing, supports, and interaction effects. The staff found them acceptable. The staff also performed a detailed review of the entire list of outliers identified in the licensee's October 1, 1993, submittal, with the aid of the photographs taken for the piping and components, piping and instrumentation diagrams (P&ID), and isometric drawings for piping. Based on the walkdown procedures presented, the staff found the licensee's approach of identifying the outliers was acceptable. In addition, there is reasonable assurance that the outliers identified are accurate and complete to the extent practicable. Because of limited accessibility in the plant due to power operation, the staff was able to independently verify only certain outliers in a plant walkdown during the audit.

The staff found that the Hatch Unit 2 nonseismically analyzed main steam system piping and condenser to be used for the alternate MSIV leakage treatment path compared well with the earthquake experience database and that GPC has performed walkdowns to identify and evaluate any of the characteristics associated with the limited component damages observed at the database facilities. By taking the proposed measures to ensure resolution for all of the identified outliers, GPC has provided assurance that the damage reported for the database components should not occur to the Hatch Unit 2 main steam piping and condenser or to the associated support systems. The staff, therefore, concludes that the licensee has demonstrated that the proposed method for MSIV leakage treatment is seismically adequate to serve as an acceptable alternative to the currently installed LCS.

The licensee has committed to include the alternate LCS in the ASME Section XI inservice inspection program. The piping will be treated as ASME Code Section III, Class 2. Any repairs or replacement of this piping should also be performed in accordance with Section XI requirements.

3.2.2 Seismic Analysis of Piping Supports

Performance of the turbine building during a seismic event is of interest to the issue of MSIV leakage only to the extent that nonseismically designed structures and components should survive and not degrade the capabilities of the selected main steam and the condenser pathways. The turbine building was designed for tornado load, with a windspeed of 300 mph, and no yielding of materials was allowed for the tornado wind design. The licensee has calculated the shear force at different levels of the turbine building generated from tornado wind, Uniform Building Code (UBC) zone 1 earthquake intensity, and an estimated median-centered earthquake with a peak ground acceleration of 0.15 g. The staff compared these shear forces and found that the shear forces at all levels of the turbine building, which were generated from the tornado wind, have exceeded those generated from the UBC earthquake - or the median-centered earthquake - by a significant margin. This indicates that the turbine building is sufficiently strong, since it was designed using a very stringent criteria: a 300 mph wind speed coupled with no yielding in structural materials (beams and columns), that it would also withstand the plant design earthquake with no yielding in structural materials. Therefore, the turbine building will survive earthquake events although it was not specifically designed for them.

The licensee stated that it had performed a walkdown and selected a total of fifteen pipe supports for evaluation. For the determination of the seismic load demand, the licensee used a factor of 1.25 times the peak acceleration of median-centered floor response spectra. On the anchorage capacity estimation, the licensee used the methods and values provided in the "Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment" which the staff had previously reviewed and approved. This approach is acceptable. The licensee has submitted the evaluation results of the 15 pipe supports. The staff found that the evaluation results have demonstrated that the support and anchorage capacities have exceeded the seismic demands by a substantial margin.

3.2.3 Conclusion

On the basis of this evaluation, the staff concludes that, upon completion of the plant modifications necessary for the identified outliers, there is reasonable assurance that the Hatch Unit 2 MSL, main steam drain lines, condenser, associated interconnected piping and equipment, and their supports in the reactor and turbine buildings, will be seismically adequate to serve as an alternate MSIV leakage treatment system. This is based on the fact that portions of the main steam system piping and their associated supports have been seismically analyzed, and the remaining nonseismically analyzed piping and equipment (1) are well represented by those in the earthquake experience database that demonstrated good seismic performance, (2) are able to exhibit adequate resistance to damage from a design-basis earthquake, and (3) have been shown to have adequate margins for seismic capability. The supports for the nonseismically analyzed piping have been evaluated and the evaluation shows that they have sufficient margins. Therefore, the staff concludes that the licensee's proposed alternate leakage treatment system is seismically adequate to withstand the Hatch Unit 2 design-basis earthquake and maintain its pressure-retaining integrity, and hence, is in conformance with GDC 2 of Appendix A to 10 CFR Part 50.

It should be noted that the main steam system piping and equipment have been demonstrated to be seismically adequate, and thus meet GDC 2 requirements. However, the staff acceptance of the experience data methodology as presented by the licensee is applicable only for ensuring the pressure boundary integrity of the alternate leakage treatment path, and is not an endorsement that the experience-based methodology is applicable for other applications at Plant Hatch.

3.3 DRAIN PATH FUNCTIONAL DESIGN EVALUATION

The proposed primary drain path at Hatch Unit 2 employs an MSL drain downstream of the MSIVs. There are two motor-operated valves (MOV) in series in this line between the MSL and the main condenser. Both valves must be open to establish the required drain path. The first (upstream) MOV is normally open and will fail "as-is" on a loss of power. The second (downstream) MOV is normally closed (with a 0.1 inch bypass orifice around it to allow drainage during normal operation) and is required to be opened following the DBA LOCA to establish a large enough drain path to support the radiological analysis. The staff requested the licensee to address the single failure of this downstream valve to open on demand, due to a valve or power supply failure.

In its January 6, 1994, submittal, the licensee stated that the downstream valve can be powered from two separate emergency ac power sources. Therefore, a single failure of a power supply does not disable the safety-related function to open on demand. To address a failure of the valve itself, the licensee verified that an alternate drain path will be available to convey MSIV leakage to the isolated condenser if the downstream valve fails to open. The alternate drain path is located downstream of the primary drain path and originates from the MSL drain pots. This alternate drain path is included in the seismic verification scope. The alternate path has an 0.8-inch restricting orifice in a bypass line around a normally closed valve in the drain line. Consequently, if the primary downstream MOV (1-inch flow path) failed to open as required, the second drain path would be available to convey MSIV leakage to the main condenser. This second path will convey essentially all of the MSIV leakage to the main condenser via the 0.1-inch and 0.8-inch orifices. Consequently, the radiological dose assessment for this alternate path is essentially equivalent to the dose assessment for the primary path. To increase the reliability of the MOV in the primary flow path, the licensee will include the valve in the inservice testing program to perform a stroking surveillance on a quarterly basis. Additionally, the licensee has committed to update the Operating and/or Emergency Operating Procedures as necessary to address the applicable alternate leakage treatment methods.

The licensee further proposed new requirements in the Hatch Unit 2 TS Section 3.6.1.2 related to restoration of acceptable leak rates if any of the proposed limits are exceeded. The new requirements basically state that if any MSIV exceeds 100 scfh, it will be repaired and retested to meet a leak rate limit of 11.5 scfh per valve (the current criterion for leakage) and that the maximum total leak rate will be restored to less than or equal to 250 scfh. Therefore the staff concludes that this new requirement proposed by the licensee is acceptable.

3.3.1 Conclusion

On the basis of the above evaluation, the staff concludes that the proposed design provides a reliable leakage path that meets the single-failure criterion of GDC 41, "Containment Atmosphere Cleanup." Therefore, the staff concludes that the proposed design is acceptable.

3.4 OVERALL CONCLUSIONS

Based on its evaluation as described above, the staff concludes that:

(1) The proposed increase in allowable MSIV leakage rates should avoid exposing maintenance personnel to unnecessary doses, reduce outage durations, extend the effective service life of the MSIVs, and has the potential to significantly reduce recurring valve leakage caused by repairs. In addition, the proposed alternate treatment method will be able to handle larger leakage rates which could not be handled at all by the LCS because of design limitations, and the resulting doses remain well within the guidelines of 10 CFR Part 100 for the offsite doses and 10 CFR Part 50, Appendix A (GDC 19) for the control room doses.

(2) The design of the alternate treatment path, including piping and supports, structures, and components, meets GDC 2 of Appendix A to 10 CFR Part 50, with respect to performing its safety function following a design-basis seismic event, and

(3) The design of the alternate treatment also meets the requirements of GDC 41 with respect to performing its safety function with and without offsite power and assuming a single active failure.

The staff, therefore, concludes that the alternate leakage path design is acceptable and that the proposed changes to the Technical Specifications to increase MSIV leak rates limits and to eliminate the LCS are acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on March 1, 1994 (59 FR 9780).

Accordingly, based upon the environmental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

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Date: March 17, 1994

Table 15-1

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

	0-2 Hour Doses, Exclusion Area Boundary, rem		0-30 Day Doses, Low Population Zone, rem	
Postulated Accident	Thyroid	Whole Body	Thyroid	Whole Body
Loss-of-Coolant*	60**	less than 2	266**	less than 2
Fuel Handling	29	less than 1	29	less than 1
Control Rod Drop	1	less than 1	3	less than 1

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* Includes contribution from MSIV leakage

** Revised values

Table 15-2

ASSUMPTIONS USED TO CALCULATE LOSS-OF-COOLANT ACCIDENT DOSES

Power Level, magawatts thermal	2,537
Operating Time, years	3
Core Fraction Released to Drywell, percent	
Noble Gases	100
Iodine	25
Drywell Free Volume, cubic feet	146,266
Reactor Building Free Volume, cubic feet	1,275,000
Reactor Building Mixing Fraction, percent	0
Reactor Building Exhaust System Flow rate, cubic feet per minute	4,000
Standby Gas Treatment System Filter Efficiencies for Iodines, percent	
Elemental	99
Organic	99
Particulate	99
Main Steam Isolation Valve Leakage, standard cubic feet per hour (total)	250*
Minimum Exclusion Area Boundary, meters	1,250
Low Population Zone Distance, meters	1,250

3Atmospheric Diffusion Values, seconds per cubic meter

	Stack Release	Ground Level Release
0-1/2 hours	3.7×10^{-5}	1.4×10^{-4}
1/2-2 hours	1.1×10^{-5}	1.4×10^{-4}
0-8 hours	5.6×10^{-6}	7.0×10^{-5}
8-24 hours	3.8×10^{-6}	5.0×10^{-5}
24-96 hours	1.9×10^{-6}	2.3×10^{-5}
95-720 hours	6.4×10^{-7}	8.0×10^{-6}

* Revised values

Table 15.4
Parameters Used to Evaluate Iodine Transport and Removal in Main Steam Lines, Drain Lines, and Main Condenser

Source Term	Regulatory Guide 1.3		
MSIV Leakage Rate	250 SCFH		
Elemental and Particulate Iodine			
Temperature	Deposition Velocities (cm/sec)	Resuspension Rate (1/sec)	Fixation Rate (1/sec)
300°k	3.2E-02	3.14E-06	4E-06
400°k	5.0E-03	7.05E-06	8E-06
500°k	1.0E-03	8.10E-06	1E-05
560°k	6.2E-04	9.20E-06	2E-05
Organic Iodine			
300°k	1.4E-03	9.5E-08	4E-06
400°k	3.5E-04	2.0E-07	8E-06
500°k	1.0E-05	3.0E-07	1E-05
560°k	1.3E-05	3.6E-07	2E-05
Leakage Duration	30 days		
Condenser Volume	2.35E+09 cc		
Components			
	Diameter (cm)	Length (cm)	Thickness (cm)
Main Steam Line	53.2	9601	2.6
Drain Line	6.65	8077	1.1

Table 15.5
Iodine Releases

	Inlet to Main Steam Lines (curies)	Outlet from Main Condenser (curies)
0 - 2 hours	4.14E+03	1.27E+01
2 - 8 hours	1.22E+04	2.20E+01
8 - 24 hours	3.22E+04	1.38E+02
24 - 96 hours	1.22E+05	2.19E+03
96 - 720 hours	3.62E+05	8.86E+03

Table 15.6
Assumptions and Estimates of the Radiological
Consequences to Control Room Operators following a LOCA

Control room free volume	9.35 E+4 ft^3
Recirculation Rates Filtered Intake Unfiltered Intake Filtered Recirculation Filter Efficacy (2 inch charcoal)	 400 CFM 0 CFM 2100 CFM 95%
Unfiltered control room infiltration rate (assumed)	10 CFM
Duration of accident	30 days
Breathing rate of operators in control room for the course of the accident	3.47E-04 m^3/sec
Meteorology (wind speeds for all sectors: 0 - 8 hours 8 - 24 hours 24 - 96 hours 96 - 720 hours	 1.7E-03 sec/m^3 1.0E-03 sec/m^3 5.6E-04 sec/m^3 1.3E-04 sec/m^3
Iodine protection factor	100
Iodine Dose Conversion Factors*	ICRP-30
Control Room Operator Occupational Factors 0 - 8 hours 8 - 24 hours 24 - 96 hours 96 - 720 hours	 1 1 0.6 0.4
Doses to control room operators: Thyroid dose* Whole body dose**	 29 rem <1 rem

* unweighted dose equivalent

** unweighted dose equivalent (red bone marrow) due to immersion in an infinite cloud

UNITED STATES NUCLEAR REGULATORY COMMISSION

GEORGIA POWER COMPANY, ET AL.

DOCKET NO. 50-366

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 132 to Facility Operating License No. NPF-5 issued to Georgia Power Company, et al. (the licensee), which revised the Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit 2, located in Appling County, Georgia. The amendment is effective as of the date of issuance.

The amendment modified the Technical Specifications (TS) to permit an increase in the allowable leak rate for the main steam isolation valves (MSIVs) and deleted the TS requirements for the MSIV leakage control system.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on November 5, 1993 (58 FR 59081). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment (59 FR 9780).

For further details with respect to the action see (1) the application for amendment dated October 1, 1993, as revised January 6, 1994, and supplemented February 3, 1994, (2) Amendment No. 132 to License No. NPF-5, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC 20555, and at the local public document room located at Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513.

Dated at Rockville, Maryland, this 17th day of March 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

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