October 1, 1998

Mr. H. L. Sumner, Jr.
Vice President - Nuclear
Hatch Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M99393 AND M99394)

Dear Mr. Sumner:

The Commission has requested the Office of the Federal Register to publish the enclosed "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for Hearing." This notice relates to your amendment application dated August 8, 1997, as supplemented by letters dated March 9, May 6, July 6, July 31, September 4, September 11, and September 30, 1998, and also includes advance information related to the application dated April 17, 1997. The amendments would revise the Technical Specifications to accommodate an increase in maximum license thermal power level from 2558 megawatts thermal (MWt) to 2736 MWt.

Sincerely,
Original signed by H. Berkow for:

Leonard N. Olshan, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosure: Notice

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Edwin I. Hatch Nuclear Plant

cc:

Mr. Ernest L. Blake, Jr.
Shaw, Pittman, Potts
and Trowbridge
2300 N Street, NW.
Washington, DC 20037

Mr. D. M. Crowe
Manager, Licensing
Southern Nuclear Operating
Company, Inc.
P. O. Box 1295
Birmingham, Alabama 35201-1295

Resident Inspector
Plant Hatch
11030 Hatch Parkway N.
Baxley, Georgia 31531

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303

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

Harold Reheis, Director
Department of Natural Resources
205 Butler Street, SE., Suite 1252
Atlanta, Georgia 30334

Steven M. Jackson
Senior Engineer - Power Supply
Municipal Electric Authority
of Georgia
1470 Riveredge Parkway, NW
Atlanta, Georgia 30328-4684

Charles A. Patrizia, Esquire
Paul, Hastings, Janofsky & Walker
10th Floor
1299 Pennsylvania Avenue
Washington, DC 20004-9500

Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

Mr. J. D. Woodard
Executive Vice President
Southern Nuclear Operating
Company, Inc.
P. O. Box 1295
Birmingham, Alabama 35201-1295

Mr. P. W. Wells
General Manager, Edwin I. Hatch
Nuclear Plant
Southern Nuclear Operating
Company, Inc.
U.S. Highway 1 North
P. O. Box 2010
Baxley, Georgia 31515

Mr. R. D. Barker
Program Manager
Fossil & Nuclear Operations
Oglethorpe Power Corporation
2100 East Exchange Place
P. O. Box 1349
Tucker, Georgia 30085-1349

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

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UNITED STATES NUCLEAR REGULATORY COMMISSION

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

DOCKET NOS. 50-321 AND 50-366

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION; AND OPPORTUNITY FOR A HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. DPR-57 and NFP-5 issued to Southern Nuclear Operating Company, Inc., et al. (the licensee) for operation of the Edwin I. Hatch Nuclear Plant, Units 1 and 2, located in Appling County, Georgia.

The proposed amendments would revise the Technical Specifications to accommodate an increase in maximum licensed thermal power level from 2558 megawatts thermal (MWt) to 2736 MWt.

The licensee submitted the proposed changes by letter dated August 8, 1997. In processing this request, the staff recognized on September 29, 1998, it inadvertently failed to publish a notice of proposed issuance of the amendments in the FEDERAL REGISTER. In the August 8, 1997, original application, the licensee requested that the proposed amendments be issued prior to startup from the fall 1998 refueling outage on Unit 2. Startup from the refueling outage is presently scheduled for October 18, 1998.

Upon being informed by the staff that a notice of proposed issuance of amendments inadvertently was not published, the licensee requested, by letter dated September 30, 1998, that the proposed amendments be processed on a exigent basis.

The need for exigency is based on the fact that the licensee would be required to postpone changes to procedures, instrumentation, and setpoints on Unit 2 until after startup and power ascension of the plant if the amendments were not issued prior to restart. The licensee would then be required to implement these changes while online which would increase the possibility of a plant scram and introduce a potential for unnecessary transients on the plant.

The licensee has evaluated the impact of the schedule change and the online implementation of the extended power uprate (EPU) and determined that receiving the amendments prior to startup will result in a net increase in plant safety and reliability. Reliability benefits include a reduced potential for an inadvertent reactor scram while adjusting instrumentation online and human performance issues associated with training and procedures. Implementation of the EPU requires adjustment of the direct scram from the turbine stop valve and the turbine control valve fast closure and the main steamline high flow isolation setpoints. These adjustments place the plant in a configuration that results in generation of a half scram signal and an increased potential for an unnecessary full scram of the plant. Implementation of the EPU also requires adjustment of the average power range monitor (APRM) setpoints, including the APRM simulated thermal power scram.

In addition, the licensee has identified approximately 20 instrumentation and controls and 30 operations procedures that would require revisions prior to and after the issuance of the uprate amendments if they are not issued prior to Unit 2 startup. This may result in human factor concerns associated with procedure revisions and operator training.

Therefore exigency is appropriate in order to allow implementation of these amendments and will result in a net benefit in plant safety and reliability.

Before issuance of the proposed license amendments, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

Pursuant to 10 CFR 50.91(a)(6) for amendments to be granted under exigent circumstances, the NRC staff must determine that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendments would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

I. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated based upon the following discussion:

A. Evaluation of the Probability of Previously Evaluated Accidents

The proposed extended power uprate imposes only minor increases in plant operating conditions. No changes to rated core flow, rated reactor pressure, or turbine throttle pressure are required. The higher power level will result in moderate flow increases in systems associated with the turbine cycle (e.g., condensate, feedwater, and main steam). The small increase in operating temperatures for BOP [balance of plant] support systems has no significant effect on LOCA [loss-of-coolant accident] or other accident probabilities. The extended power uprate evaluations confirm the higher power level has no significant effect on flow induced erosion/corrosion. The limiting feedwater and main steam piping flow increases were evaluated and shown to be approximately proportional to the power increase. The affected systems are currently monitored by the Plant Hatch erosion/corrosion program. Continued system monitoring provides a high level of confidence in the integrity of potentially susceptible high energy piping systems.

When required, the occurrence frequency of accident precursors and transients is addressed by applying the guidance of NRC-reviewed setpoint methodology to ensure acceptable trip avoidance is provided during operational transients subsequent to implementation of extended power uprate. The setpoint evaluation confirmed Plant Hatch extended power uprate does not increase the number of challenges to the protective instrumentation.

Plant systems, components, and structures were verified as capable of performing their intended functions under increased power conditions with a few minor exceptions.

That is, some components will be modified prior to implementation of the extended power uprate program to accommodate the revised operating conditions....The Plant Hatch extended power uprate does not significantly affect the reliability of plant equipment. In cases where plant availability could be impacted by BOP equipment performance, modifications and administrative controls will be implemented to adequately compensate. No new components or system interactions that could lead to an increase in accident probability are created due to operation at 2763 MWt [megawatts thermal].

The probability of design basis accidents (DBAS) occurring is not affected by the increased power level, since the applicable criteria established for plant equipment (e.g., ANSI Standard B3 1.1 and ASME [American Society of Mechanical Engineers] Code) will still be followed when the plant is operated at the new power level. The extended power uprate analysis basis assures the limits prescribed by the Code of Federal Regulations (CFR) (e.g., LOCA PCT [peak clad temperature], SLMPCR, 10 CFR 20) will be maintained by meeting the appropriate regulatory criteria. Similarly, factors of safety specified by application of the CFR design rules were demonstrated to be maintained, as have other margin-assuring acceptance criteria used to judge the acceptability of the plant. Established reactor scram setpoints are such that there should be no increase in scram frequency due to the increased power level. No new challenges to safety-related equipment will result. Therefore, the proposed Operating License and Technical Specifications changes do not involve a significant increase in the probability of an accident previously evaluated.

B. Evaluation of the Consequences of Previously Evaluated Accidents

ECCS-LOCA Analysis

The Plant Hatch emergency core cooling system loss-of-coolant accident (ECCS-LOCA) performance analysis was performed for extended power uprate using methodology approved by the NRC for analysis required by

10 CFR 50.46. This revised analysis utilizes the same methodology (SAFER/GESTR) as the existing ECCS-LOCA analysis. ECCS requirements assumed for extended power uprate are very similar to the existing 1986 analysis. In accordance with regulatory guidance, the Plant Hatch ECCS-LOCA analysis was performed at 102% of the new RTP of 2763 MWt, or 2818 MWt. The licensing peak clad temperature remains well below the 10 CFR 50.46 required limit of 2200°F. Therefore, the analysis demonstrates Plant Hatch will continue to comply with 10 CFR 50.46 and 10 CFR 50, Appendix K at extended power uprate conditions. Thus, the consequences of accidents are not significantly increased at the higher power level.

Abnormal Operating Transient Analysis

An evaluation of the Plant Hatch Unit I and Unit 2 Final Safety Analysis Reports (FSARs) and reload transients was performed for extended power uprate to demonstrate the proposed maximum power level will have no adverse effect on plant safety. The evaluation was performed for a power level of 2763 MWt, with the exception of certain event evaluations that were performed at 102% of 2763 MWt. The transient analysis performed to demonstrate the acceptability of Plant Hatch extended power uprate employed the same NRC-approved methods used today.

The limiting transient events at extended power uprate conditions, including events that establish the core thermal operating limits and events that bound other transient protection criteria, were evaluated. The limiting transients were benchmarked against the existing RTP [rated thermal power] level by performance of the event analysis at both the proposed power level and the current RTP level. In addition, an expanded group of transient events was evaluated to confirm these events remained less limiting than the most limiting transients. The transient events included in the expanded group were chosen based upon events demonstrated to be sensitive to initial power level. This evaluation confirmed the existing set of limiting transient events remains valid for the Plant Hatch extended power uprate. The evaluation was performed for a representative core and demonstrates the overall capability to meet all transient safety criteria. Cycle-specific analyses will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and establish cycle-specific operating limits.

The results of the limiting transients evaluation demonstrate extended power uprate can be accomplished without a significant increase in the consequences of the transients evaluated. The fuel thermal-mechanical limits at extended power uprate conditions are within the specific design criteria for the GE fuels currently loaded in the Plant Hatch cores. Also, the power-dependent and flow-dependent minimum critical power ratio (MCPR) and maximum average planar linear heat generation rate (MAPLHGR) limits utilized at Plant Hatch since the mid-1980s require only minor changes. The peak reactor pressure vessel (RPV) bottom head pressure remains within the ASME Code requirement for RPV overpressure protection. The effects of plant transients were evaluated by assessing disturbances caused by a malfunction or single failure of equipment, or operator error, consistent with the FSARs [Final Safety Analysis Reports]. Limiting transient events tend to be slightly more severe ([approximately equal to] 1%) when initiated from the new power level, assuming a 1.12 safety limit (SLMCPR) which was determined using the latest NRC-approved methods. However, for the most limiting transient, an evaluation of a representative core showed little or no change is required to the operating limit MCPR (OLMCPR) at extended power uprate and the integrity of SLMCPR is maintained. The margin of safety established by the SLMCPR is not affected and the event consequences are not significantly affected by the proposed extended power uprate to 2763 MWt. Cycle-specific analyses will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and establish cycle-specific operating limits.

The transient analysis results demonstrate the Plant Hatch core thermal power output can be safely increased to 2763 MWt without significantly affecting the consequences of previously evaluated postulated transient events. The results of the extended power uprate transient evaluation are summarized as follows:

1. Events Resulting in Nuclear System Pressure Increase

   a. Main Generator Load Rejection with No Steam Bypass

      At extended power uprate conditions, the fuel transient thermal and mechanical overpower results remain below the NRC-accepted design criteria.

   b. Main Turbine Trip with No Steam Bypass

      At extended power uprate conditions, the fuel transient thermal and mechanical overpower results remain below the NRC-accepted design criteria.

   c. Main Steam Isolation Valve (MSIV) Closure

      At extended power uprate conditions, this event (with a scram initiated by the valve closure) remains nonlimiting with respect to fuel thermal limits.

d. Pressure Regulator Failure - Closed and Slow Closure of a Single TCV [temperature control valve]

These transients remain nonlimiting as compared with other more severe pressurization events.

2. Event Resulting in a Reactor Vessel Water Temperature Decrease

a. Loss of Feedwater Heating

The consequences of this event at the extended power uprate conditions remain nonlimiting with regard to the cycle OLMCPR. The results at low core flow conditions are actually slightly higher than for the high core flow condition because of increased inlet coolant subcooling into the reactor core. The calculated thermal and mechanical overpower limits at extended power uprate conditions for this event also meet fuel design criteria.

b. Inadvertent High Pressure Coolant Injection (HPCI) Actuation

For the limiting condition analyzed, both the high water level setpoint and the high RPV steam dome pressure scram setpoints are not reached. Based upon the peak average fuel surface heat flux results, the HPCI actuation event will be bounded by the limiting pressurization event with respect to delta critical power ratio ([delta]CPR) considerations. In addition, the fuel transient thermal and mechanical overpower limits remain within the allowable NRC-accepted design values.

c. Shutdown Cooling Residual Heat Removal (RHR) Malfunction

This event is not affected by extended power uprate.

3. Event Resulting in a Positive Reactivity Insertion

Rod Withdrawal Error (RWE)

The current rod block monitor (RBM) system with power-dependent setpoints was analyzed for the RWE event at extended power uprate conditions using a statistical approach consistent with NRC approved methods. The analysis concluded the transient is slightly more severe with a greater [delta]CPR from the initial most limiting CPR. However, the fuel and mechanical overpower limits remain within the NRC accepted design criteria.

4. Event Resulting in a Reactor Vessel Coolant Inventory Decrease

   a. Pressure Regulator Failure to Full Open

      The results of this transient for extended power uprate remain nonlimiting as compared with other more severe pressurization events.

   b. Loss of Feedwater Flow

      This transient event does not pose any direct threat to the fuel in terms of a power increase from the initial conditions. Water level declines rapidly and a low water level causes a reactor scram. Actuation of HPCI and reactor core isolation cooling (RCIC) terminate the event. However, the loss of feedwater flow event is included in the extended power uprate evaluation to assure sufficient water makeup capability is available to keep the core well covered when all normal feedwater is lost. A plant-specific analysis performed in support of the extended power uprate program shows a large amount of water remains above the top of the active fuel. This sequence of events does not require any new operator actions or shorter operator response times. Therefore, operator actions for the event do not significantly change for extended power uprate.

   c. Inadvertent Opening of a Safety/Relief Valve (S/RV), Loss of Auxiliary Power, and Loss of One DC System

      These events remain less severe at extended power uprate conditions.

5. Event Resulting in Core Coolant Flow Decrease

   a. Recirculation Pump Seizure

      The recirculation pump seizure transient evaluation includes the assumption the pump motor shaft of one recirculation pump stops instantaneously. As a result, core flow decreases rapidly. The heat flux decline lags core power and flow, and could result in a degradation of core heat transfer. At extended power uprate conditions, the consequences of the pump seizure event remain nonlimiting. Note the Unit 2 FSAR classifies this event as an accident due to the low probability of occurrence.

b. RPT and Recirculation Flow Control Failure Decreasing Flow

These transients remain nonlimiting at extended power uprate conditions.

6. Event Resulting in Core Coolant Flow Increase

Recirculation Flow Controller Failure Increasing Flow

The results of this transient for extended power uprate remain nonlimiting as compared with other more severe pressurization events.

7. Event Resulting in Core Coolant Temperature Increase

Failure of RHR Shutdown Cooling

This event is not significantly affected by the increase in licensed thermal power.

8. Event Resulting in Excess of Coolant Inventory

Feedwater Controller Failure - Maximum Demand

The CPR calculated for this event at extended power uprate conditions is slightly higher than the corresponding value for the current rated power. However, the trend for the feedwater controller failure - maximum demand event is consistent -with the analysis for the current rated power level. The fuel thermal margin results are within the acceptable limits for the fuel types analyzed.

DBA Challenges to Containment

The primary containment's response to the limiting DBA was evaluated at 2763 MWt, plus a 2% adder. The effect of extended power uprate on the short-term containment response (peak values), as well as the long-term containment response for containment pressure and temperature confirms the suitability of the plant for operation at the new power level. Factors of safety provided in the ASME Code are maintained, and the safety margin is not altered by uprating power to 2763 MWt.

Short-term containment response analyses were performed for the limiting DBA LOCA, a double-ended guillotine break of a recirculation suction line, to demonstrate operation at a bounding reactor power will not result in exceeding the containment design limits. This limiting DBA LOCA event results in the highest short-term containment pressures and dynamic loads. The analysis determined, at the proposed reactor power level, the maximum drywell pressure values increase only [approximately equal to] 1 psi and remain well bounded by the containment design pressure. Extended power uprate has no adverse effect on the containment structural design pressure.

Because increasing RTP increases residual heat, the containment long-term response will have slightly higher temperatures. Long-term suppression chamber temperatures remain within the design temperature of the structure; thus, ASME Code factors of safety are maintained and the safety margin is not affected. An analysis confirmed ECCS pump net positive suction head (NPSH) is not adversely affected with this temperature response, and the long-term response does not adversely affect the containment structure or the environmental qualification (EQ) of equipment located in the drywell and torus. The drywell long-term temperature response is not adversely affected for the higher reactor power; thus, the containment long-term response for extended power uprate is acceptable.

The impact of a reactor power increase on containment dynamic loads was evaluated and found to have no adverse effect for conditions that bound the proposed power level. Thus, containment dynamic loads are acceptable for operation at 2763 MWt.

The Plant Hatch extended power uprate evaluation of the primary containment response to DBAs confirmed the proposed power level does not result in a significant increase in the consequences of a postulated accident for a reactor power level [approximately equal to] 2% greater than the proposed increase to 2763 MWt.

Radiological Consequences of DBAs

For Plant Hatch extended power uprate, the radiological consequences of the limiting DBAs were reevaluated. The evaluations included the effect of the proposed power level on the radiological consequences of accidents presented in the FSARs. Reference 3 provides information on a revised radiological dose analysis for the DBA LOCA and shows doses remain within 10 CFR 100 limits at the new power level.

This DBA LOCA radiological evaluation was performed using input and evaluation techniques consistent with current regulatory guidance and appropriate plant design basis. The inputs and analysis methods are different from those utilized in the current licensing basis evaluation presented in the FSARs and the Atomic Energy Commission safety evaluation report supporting the initial plant licensing. However, the input used in the extended power uprate radiological evaluation provides a conservative assessment of the potential radiological consequences. The conclusions of these evaluations are consistent with the original licensing basis evaluations. The radiological consequences of the limiting DBA remain within 10 CFR 100 guidelines for the proposed RTP level. For the purpose of analysis, the new RTP level was increased by an additional 2% in accordance with regulatory guidance.

To demonstrate the change in consequences, the evaluation of radiological consequences using the different analysis inputs and methods was performed for the existing licensed RTP level and the proposed RTP level.

The impact of the proposed licensed power level on the fuel handling accident, control rod drop accident, and main steam line break outside primary containment was evaluated. The radiological consequences remain well below regulatory limits.

The evaluation of DBA radiological consequences confirmed extended power uprate does not result in a significant increase in consequences at a power level of 2763 MWt. The results remain below 10 CFR 100 guideline values. Therefore, the postulated radiological consequences do not represent a significant change in accident consequences and are clearly within the regulatory guidelines for the proposed power level increase.

Other Evaluations

1. Performance Improvements

   The extended power uprate safety analysis was performed taking into account the implementation of the following previously approved special operational features.

   a. Single-Loop Operation (SLO)

      The safety analysis for extended power conditions shows the single-loop operating mode remains valid. The current trip setpoints determined for two-loop operation (TLO) were confirmed to be acceptable for SLO, with a correction applied to account for the actual effective drive flow applied when operating with a single loop. The SLO settings were conservatively established to be consistent with the TLO settings, while ensuring the appropriate corrections are applied to the MAPLHGR and the OLCPR to account for SLO.

b. Maximum Extended Load Line Limit (MELLL)

The safety analysis for new power conditions shows the operating domain as analyzed is valid for extended power uprate conditions, even with operation permitted on a slightly higher absolute rod line.

c. Increased Core Flow (ICF).

The safety analysis for extended power uprate shows that operation at ICF conditions remains acceptable.

d. Final Feedwater Temperature Reduction (FFWTR)

The safety analysis for extended power uprate shows operation at FFWTR conditions remains acceptable.

e. Average Power Range Monitor/Rod Block Monitor Technical Specification (ARTS) Improvements

The safety analysis for extended power uprate conditions shows the ARTS improvements remain valid for the extended power uprate conditions.

2. Effect of Extended Power Uprate on Support Systems

An evaluation was performed to address the effect of the extended power uprate on accident mitigation features, structures, systems, and components within the BOP. The evaluation results are as follows:

a. Auxiliary systems, such as building heating, ventilation, and air-conditioning (HVAC) systems, reactor building closed cooling water, plant service water, spent fuel pool cooling; process auxiliaries, such as instrument air and makeup water; and the post-accident sampling system were confirmed to operate acceptably under normal and accident conditions at the proposed power level.

b. Secondary containment and standby gas treatment system were confirmed to be adequate relative to containing, processing, and controlling the release of normal and post-accident levels of radioactivity.

c. Instrumentation was reviewed and confirmed capable of performing control and monitoring functions at the proposed power level. As required, analyses were performed to determine the need for setpoint changes for various functions (e.g., APRM simulated thermal power scram setpoints). In general, setpoints are to be changed only to maintain adequate difference between plant operating parameters and trip setpoints, while ensuring safety performance is demonstrated. The revised setpoints were established using NRC-reviewed methodology as guidance.

d. Electric power systems, including the main generator and switchgear components, were verified as being capable of providing the required electrical load as a result of the increased power level. An evaluation of the auxiliary power system confirmed the system has sufficient capacity to support all required loads for safe shutdown, maintain a safe shutdown condition, and operate the required engineered safeguards equipment following postulated accidents. No safety-related electrical loads were affected which would impact the emergency diesel generators.

e. Piping systems were evaluated for the effect of operation at higher power levels, including transient loading. The evaluation confirmed piping and supports are adequate to accommodate the increased loading resulting from operation at higher power conditions.

f. The effect of the higher power conditions on a high energy line break (HELB) was evaluated. The evaluation confirmed structures, systems, and components important to safety are capable of accommodating the effects of jet impingement, blowdown forces, and the environmental effects resulting from HELB events.

g. Control room habitability was evaluated. Post-accident control room and Technical Support Center doses at 2763 MWt were confirmed to be within the guidelines of General Design Criterion 19 of 10 CFR 50, Appendix A. (See Ref. 3.)

h. The EQ of equipment important to safety was evaluated for the effect of normal and accident operating conditions at the proposed power level. The equipment remains qualified for the new conditions. The preventive maintenance program will continue to provide equipment maintenance or replacement to ensure equipment EQ at extended power uprate conditions.

3. Effect on Special Events

The consequences of special events (i.e., anticipated transient without scram (ATWS); 10 CFR 50, Appendix R; and station blackout) remain within NRC-accepted criteria at 2763 MWt. Vessel overpressure protection was analyzed assuming a closure of the MSIVs with a neutron flux scram, Although the peak reactor vessel bottom head pressure increases slightly at extended power uprate conditions, it is well within the ASME Code overpressure limit of 1375 psig. The standby liquid control (SLC) system capability analysis illustrates the plant can still achieve cold shutdown without dependence upon the control rods. Core thermal-hydraulic stability was evaluated. The new power level and modified power-to-flow map will not affect the ability to detect and suppress limit-cycle oscillations. Extended power uprate also does not adversely affect other special events, because the available equipment is not changed and the input assumptions for the evaluations are not significantly changed. Concurrent malfunctions assumed to occur during accidents were accounted for in the safety analyses for the proposed power level increase. The consequences of these equipment malfunctions do not change with the implementation of the extended power uprate program.

Conclusion

The evaluation of ECCS performance demonstrated the criteria of 10 CFR 50.46 are satisfied, thus, the margin of safety established by the criteria is maintained. The analysis demonstrated the ECCS will function with the most limiting single failure to mitigate the consequences of the accident and maintain fuel integrity. Challenges to the containment were evaluated and the integrity of the fission product barrier was confirmed. The radiological consequences of DBAs were evaluated and it was found the effect of the proposed extended power uprate on postulated radiological consequences does not result in a significant increase in accident consequences. The evaluations provide conservative results for the proposed power level of 2763 MWt and demonstrate the proposed extended power uprate does not result in a significant increase in accident consequences.

The abnormal transients were analyzed under extended power uprate conditions, and the analysis confirms the power increase to 2763 MWt has only a minor effect upon MCPR and the SLMCPR results. Thus, the margin of safety as assured by the SLMCPR is maintained. The effect of extended power uprate on the consequences of abnormal transients that result from potential component malfunctions is acceptable; thus, operation at the new power level does not result in a significant increase in transient event consequences.

The spectrum of analyzed postulated accidents and transients was investigated and determined to meet current regulatory criteria. In the area of core design, the fuel operating limits will still be met at the requested power level, and fuel reload analyses will show plant transients meet NRC-accepted criteria. The evaluation of accident consequences was performed consistent with the proposed changes to the plant Technical Specifications. Therefore, the proposed Operating License and Technical Specifications changes will not cause a significant increase in the consequences of an accident previously evaluated for Plant Hatch Unit 1 and Unit 2.

II. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated based upon the following discussion:

The BWR [boiling water reactor] configuration, operation, and event response is unchanged by the higher power level. Analyses of transient events confirm the same transients remain limiting and no transient events will result in a new sequence of events that could lead to a new accident scenario. The extended power uprate analyses confirm the accident progression is basically unchanged.

An increase in power level does not create a new fission product release path, or result in a new fission product barrier failure mode. The same fission product barriers, such as the fuel cladding, the reactor coolant pressure boundary (RCPB), and the reactor containment, remain in place. Fuel rod cladding integrity is ensured by operating within thermal, mechanical, and exposure design limits, and is demonstrated by the extended power uprate transient and accident analyses. Similarly, analysis of the RCPB and primary containment demonstrates the increased power level has no adverse effect upon these fission product barriers. The proposed Technical Specifications changes in support of extended power uprate implementation are consistent with the analyses, and assure transient and accident mitigation capability in compliance with regulatory requirements.

The effect of Plant Hatch extended power uprate on plant equipment was evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode resulting from the increased power was identified. The full spectrum of accident considerations defined in the FSARs was evaluated, and no new or different kind of accident resulting from the extended power uprate was identified. Extended power uprate analyses were performed using developed technology which was applied assuming the capability of existing plant equipment in accordance with existing regulatory criteria, including accepted codes, standards, and methods. GE has analyzed BWRs, with higher power densities and no new power-dependent accidents were identified. In addition, this uprate does not create any new sequence of events or failure modes that lead to a new type of accident.

All necessary actions will be taken prior to implementation of this program to ensure safety-related structures, systems, and components remain within their design allowable values and also ensure they can perform their intended functions under higher power conditions. The extended power uprate does not increase or create any new challenges to safety-related equipment or other equipment whose failure could cause a different kind of accident from that previously evaluated.

III. The proposed changes do not involve a significant reduction in a margin of safety based upon the following discussion:

The transient and accident analyses, as well as a majority of the plant-specific evaluations, to support the extended power uprate were performed at 2763 MWt and increased by an additional 2% in accordance with regulatory guidance, when applicable, for the evaluation of accidents and transients. The analyses demonstrate sufficient margins of safety exist. The evaluation of transient events and instrument setpoints demonstrate sufficient margin when compared to criteria establishing margins of safety for the proposed increase in power level.

The Plant Hatch extended power uprate analysis basis assures the power-dependent safety margin criteria prescribed by the CFR will be maintained by meeting the appropriate regulatory criteria. Similarly, factors of safety specified by application of the ASME Code design rules are maintained, as are other margin-assuring acceptance criteria used to judge the acceptability of the plant.

A. Fuel Thermal Limits

No change in the basic fuel design is required to achieve the extended uprate power level or to maintain the margins as discussed above. No increase in the allowable peak rod power is requested. The abnormal transients were evaluated at the higher power level for a representative core configuration. The analysis confirms the extended power uprate has no significant effect upon the OLMCPR or the SLMCPR. The fuel operating limits, such as MAPLHGR and the OLMCPR, will still be met at the new power level. The analyses confirm the acceptability of these operating limits for extended power uprate without an adverse effect upon margins to safety. Cycle specific analyses for each fuel reload will continue to be performed to demonstrate compliance with the applicable transient criteria and establish cycle-specific operating limits.

B. DBA Challenges to Fuel

Evaluation of the ECCS performance demonstrates the criteria of 10 CFR 50.46 are satisfied; thus, the margin of safety established by the criteria is maintained. This evaluation was performed at 2763 MWt, and increased by an additional 2% in accordance with regulatory guidance.

The analysis demonstrates Plant Hatch will continue to comply with the guidance of 10 CFR 50.46 and the margin of safety established by the regulation will be maintained following the increase in power level.

C. DBA Challenges to Containment

The primary containment response to the limiting DBA was evaluated for extended power uprate. The effect of the increased power on the short-term containment response (peak values), as well as the long-term containment response, for containment pressure and temperature confirms the suitability of the plant for operation at the proposed power level of 2763 MWt. Factors of safety provided in the ASME Code are maintained and safety margin is not affected.

Short-term containment response analyses were performed for the limiting DBA LOCA, consisting of a double-ended guillotine break of a recirculation suction line, to demonstrate operation at the new reactor power will not result in exceeding containment design limits. The analyses determined the maximum drywell pressure increases only slightly and is bounded by the containment design pressure. Extended power uprate has no adverse effect on containment structural design pressure.

Long-term suppression chamber temperatures remain within the design temperature of the structure; thus, factors of safety provided in the ASME Code are maintained and the safety margin is not affected. Analyses confirm ECCS pump NPSH is not adversely affected with this temperature response, and the long-term response does not adversely affect the containment structure or the EQ of equipment located in the drywell and torus.

The impact of a reactor power increase on containment dynamic loads was evaluated and found to have no adverse effect for conditions that bound the proposed increase in power level. Thus, containment dynamic loads are acceptable for extended power uprate.

The Plant Hatch extended power uprate evaluation of the primary containment response to the DBA confirms the increased power level does not result in the reduction in a margin of safety.

D. DBA Radiological Consequences

The FSARs provide the radiological consequences for each DBA. The magnitude of the potential consequences is dependent upon the quantity of fission products released to the environment, the atmospheric dispersion factors, and the dose exposure pathways. For the case of extended power uprate, the atmospheric dispersion factors and the dose exposure pathways do not change. Therefore, the only factor that will influence the magnitude of the consequences is the quantity of activity released to the environment. This quantity is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point.

The radiological consequences of DBAs were evaluated and it was found there is not a significant increase in consequences. The results remain below 10 CFR 100 guideline values. Therefore, the postulated radiological consequences are clearly within the regulatory guidelines, and all radiological safety margins are maintained for the proposed power level of 2763 MWt.

E. Transient Evaluations

The effect of plant transients was evaluated by assessing a number of disturbances of process variables, and malfunctions or failures of equipment consistent with the FSARS. The transient events tend to be slightly more severe ([approximately equal to] 1%) when initiated from the new power level, assuming a 1.12 SLMCPR, which was determined using the latest GE methods approved by the NRC. However, for the most limiting transient, an evaluation of a representative core shows no significant change to the OLMCPR is required for the new power level and the integrity of the SLMCPR is maintained.

Cycle-specific analyses for each fuel reload will continue to be performed to demonstrate compliance with the applicable transient criteria and establish cycle-specific operating limits.

The fuel thermal-mechanical limits at extended power uprate conditions are within the specific design criteria for the GE fuels currently loaded in the Plant Hatch cores. Also, the power-dependent and flow-dependent MCPR and MAPLHGR methods remain applicable. The peak RPV bottom head pressure remains within the ASME Code requirement for RPV overpressure protection.

The margin of safety established by the SLMCPR is not affected by the proposed power level increase to 2763 MWt.

F. Special Events

The event acceptance limits for special events remain unchanged for extended power uprate. For example, the peak RPV bottom head pressure remains below the 1375 psig ASME Code requirement for RPV overpressure protection. Acceptance limits for ATWS, Appendix R, and station blackout also remain unchanged.

G. Technical Specifications Changes

The Technical Specifications ensure the plant and system performance parameters are maintained at the values assumed in the safety analysis. The Technical Specifications (setpoints, trip settings, etc.) are selected such that adequate margin exists. For instruments that initiate protective functions (e.g., reactor protection system, ECCS, and containment isolation), proper account is taken of inaccuracies introduced by instrument drift, instrument accuracy, and calibration accuracy. The Technical Specifications address equipment availability and limit equipment out-of-service to assure the plant will have at least the complement of equipment available to deal with plant transients as that assumed in the safety analysis. The evaluations and analyses performed to demonstrate the acceptability of extended power uprate were performed using input consistent with the proposed changes to the plant Technical Specifications.

The events (i.e., transients and accidents) that form the Technical Specifications Bases were evaluated for extended power uprate conditions using input and initial conditions consistent with the proposed Technical Specifications changes. Although some changes to the Technical Specifications are required, no NRC acceptance limit is exceeded. Therefore, the margins of safety assured by safety limits and other Technical Specifications limits are maintained. The proposed changes to the Bases are consistent with the evaluations demonstrating acceptability of the new licensed power level of 2763 MWt.

Conclusion

The spectrum of postulated accidents and transients was investigated and was determined to meet the current regulatory criteria for Plant Hatch at extended power uprate conditions. In the area of core design, fuel operating limits will still be met at the new power level, and fuel reload analyses will show plant transients meet the NRC-accepted criteria as specified in the plant Technical Specifications. Challenges to fuel and ECCS performance were evaluated and shown to meet the criteria of 10 CFR 50.46 and 10 CFR 50, Appendix K. Challenges to the containment were evaluated and the integrity of the fission product barrier was confirmed. Radiological release events were evaluated and shown to meet the guidelines of 10 CFR 100. The proposed Operating License and Technical Specifications changes are consistent with the Plant Hatch extended power uprate evaluations. The evaluations demonstrate compliance with the margin-assuring acceptance criteria contained in applicable codes and regulations. Therefore, the proposed Operating License and Technical Specifications changes do not involve a significant reduction in the margin of safety.

REFERENCES

1. NRC letter from D. M. Crutchfield to G. L. Sozzi (GE), "Staff Position Concerning GE BWR Extended Power Uprate Program," TAC No. M91680, dated February 8, 1996.

2. NRC letter from K. N. Jabbour to J. T. Beckham, Jr., "Issuance of Amendments - Edwin I. Hatch Nuclear Plant Units I and 2," (TAC Nos. M91077 and M91078), dated August 31, 1995.

3. SNC letter BL-5356 from H. L. Sumner, Jr., to the NRC, "Revised Post-LOCA Doses," dated April 17, 1997.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 14 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendments until the expiration of the 14-day notice period. However, should circumstances change during the notice period, such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendments before the expiration of the 14-day notice period, provided that its final determination is that the amendments involve no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the FEDERAL REGISTER a notice of issuance. The Commission expects that the need to take this action will occur very infrequently. Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory

Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this FEDERAL REGISTER notice. Written comments may also be delivered to Room 6D59, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By November 5, 1998, the licensee may file a request for a hearing with respect to issuance of the amendments to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendments under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If the amendments are issued before the expiration of the 30-day hearing period, the Commission will make a final determination on the issue of no significant hazards consideration. If a hearing is requested, the final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendments and make them immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendments.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the

Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendments dated August 8, 1997, as supplemented by letters dated March 9, May 6, July 6, July 31, September 4, September 11, and September 30, 1998, and also advanced information related to the application dated April 17, 1998, which are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room, located at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia.

Dated at Rockville, Maryland, this 1st day of October 1998.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation