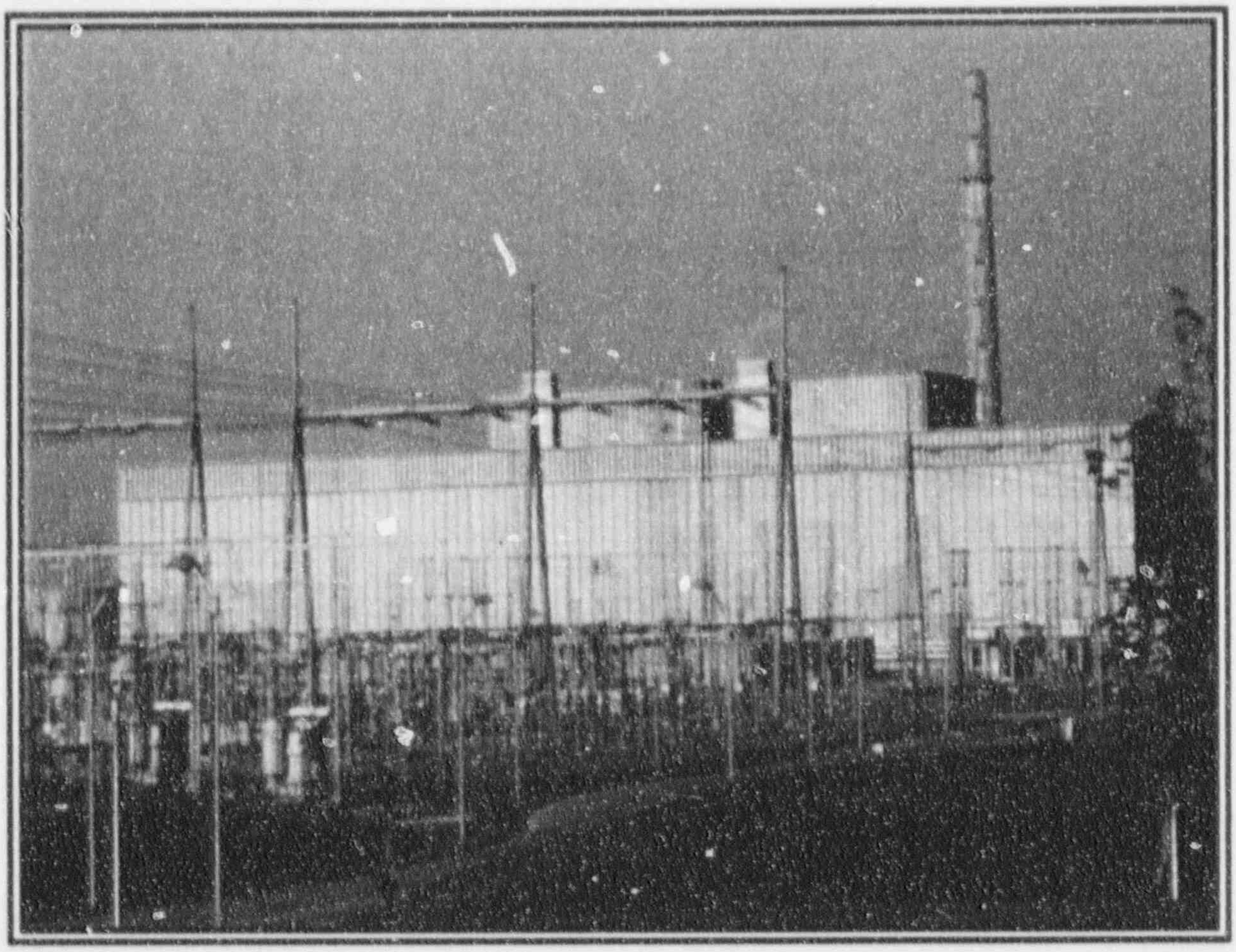
EDWIN I. HATCH NUCLEAR PLANT

EXTENDED POWER UPRATE LICENSING SUBMITTAL



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Southern Nuclear Operating Company

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Enclosure 1

Edwin I. Hatch Nuclear Plant
Request for License Amendment
Extended Power Uprate Operation

<u>Bases for Change Request</u>

Enclosure 1

Edwin I. Hatch Nuclear Plant
Request for License Amendment
Extended Power Uprate Operation

Bases for Change Request

This proposed amendment consists of a number of changes that will permit power operation up to 2763 MWt for Plant Hatch Units 1 and 2. This power level is 8% above the current maximum rated thermal power (RTP) of 2558 MWt. The proposed power level of 2763 MWt was selected based upon limitations (and modification costs) of balance-of-plant (BOP) equipment, not upon design limitations within the nuclear steam supply system (NSSS). This is a second phase of power uprate (extended power uprate), with the first phase approved by the NRC approximately 2 years ago (Ref. 1). The original power uprate increased the licensed RTP 5% from 2436 MWt to 2558 MWt.

By the time the second phase of power uprate is completed, Plant Hatch will have acquired 6 reactor years of operating experience at 2558 MWt. The analysis and startup testing for the original power uprate allowed Southern Nuclear Operating Company (SNC) to gather data on plant systems and equipment. This experience aided in determining the target RTP level and defining the scope of required BOP modifications.

The analyses and evaluations supporting the requested changes were completed using the guidelines presented in General Electric Nuclear Energy (GENE) Licensing Topical Report (LTR) NEDC-32424P (ELTR1, Ref. 2), dated February 1995. This LTR was accepted by the NRC by letter dated February 8, 1996 (Ref. 3). SNC is taking exception to one GENE startup test recommendation listed in ELTR1, Appendix L. Section 10.4 of Enclosure 6 provides more information. Generic evaluations performed in support of extended power uprate are addressed in GENE LTR NEDC-32523P (ELTR2, Ref. 4). This LTR is currently under NRC review.

The safety analyses supporting this license amendment are primarily an extension of the analysis reviewed and approved by the NRC for the original power uprate (Ref. 1). In almost all cases, the analyses use the same codes and methodology as the original power uprate submittal. Also, no change in reactor pressure is requested. (The previous power uprate submittal involved a 30 psi increase in reactor operating pressure.) Because reactor pressure is not changing, it is not necessary to modify many Technical Specifications setpoints.

An increase in electrical output is accomplished primarily by the generation and supply of higher steam flow to the turbine generator. Continuing improvements in analytical techniques (i.e., computer codes and data) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core design have resulted in a significant increase in the margin between calculated safety analysis results and licensing limits. This available safety analysis margin, combined with the excess capability of as-designed equipment, systems, and components, provides the potential for an increase of up to 8% in the full power rating of Plant Hatch without the need to perform NSSS or major BOP hardware modifications. Power level can be increased safely, and the installed systems and equipment are capable of performing their required functions at uprated conditions. The method for achieving higher power is to expand or raise the reactor core power-to-flow map by increasing reactor core flow along equivalent flow control load lines. The maximum core flow limit will not exceed the pre-uprate value.

The plant-specific safety analysis performed by GENE to support the requested change is documented in Enclosure 6. This report demonstrates Plant Hatch can operate safely with an 8% increase in maximum reactor thermal power. This includes the corresponding increase in main turbine inlet steam flow and the corresponding increases in flow, temperature, pressure, and capacity in supporting systems and components. The GENE analysis included the following performance improvements that are currently licensed for use at Plant Hatch:

- Single-Loop Operation (SLO).
- Maximum Extended Load Line Limit (ELLL).
- Increased Core Flow (ICF).
- Final Feedwater Temperature Reduction.
- APRM/RBM/Technical Specifications (ARTS) Improvements.

Tables E1-1 and E1-2 summarize the Technical Specifications and Bases changes needed to support the extended power uprate effort.

PROPOSED CHANGES

Each Operating License and Technical Specifications change and the justification for the change is provided below. Unless noted otherwise, the proposed changes are identical for both Unit 1 and 2. Table 11-1 of Enclosure 6 also provides a list of these changes.

Proposed Change One

On page 3 of Unit 1 Facility Operating License DPR-57 and page 4 of Unit 2 Facility Operating License NFP-5 rated thermal power (RTP) is increased from "2558 megawatts" to "2763 megawatts." In section 1.1 (Definitions) of the Technical Specifications, the definition of RATED THERMAL POWER is changed to reflect the uprated power level of 2763 MWt.

Justification for Proposed Change One

This increase and redefinition of RTP for Plant Hatch follows the generic guidelines of GENE LTR NEDC-32424P (Ref. 2) which provides generic licensing criteria, clarified methodology, and a defined scope of analytical evaluations and equipment review to be performed to demonstrate safe operation at the uprated power level. Technical Specifications parameter values which are expressed as a percentage of RTP or steam flow, are not changed since the uprated values were used in the bounding analyses and evaluations. The analyses and methods used are the same as those used for the original power uprate license amendments (Ref. 1), unless otherwise specified in this submittal. GENE LTR NEDC-32749P, provided in Enclosure 6, provides the results of the evaluations supporting the proposed extended uprated power operation consistent with the methodology presented in NEDC-32424P. NEDC-32749P concludes a new licensed thermal power level of 2763 MWt can be achieved without a significant impact to safety-related equipment or safety analyses.

Proposed Change Two

In Technical Specification 3.3.1.1, RPS Instrumentation, Required Action E.1, Surveillance Requirement (SR) 3.3.1.1.11, and Functions 8 and 9 of Table 3.3.1.1-1, the power level at which the direct scram is bypassed on turbine stop valve (TSV) closure and turbine control valve (TCV) fast closure is reduced from 30% to 28%. In Technical Specification 3.3.4.1, EOC-RPT Instrumentation, the Applicability for Limiting Condition of Operation (LCO) 3.3.4.1, Required Action C.2, and SR 3.3.4.1.2, the operability requirements for the end-of-cycle recirculation pump trip (EOC-RPT), are also changed from 30% to 28%.

Justification for Proposed Change Two

Currently, the direct scrams on TSV closure and TCV fast closure are bypassed at ≤ 30% RTP. The effect of bypassing the direct scrams is explicitly analyzed in transient evaluations and impacts power-dependent fuel thermal limits; i.e., minimum critical power ratio (MCPR) and kW/ft. The transient evaluations may also be analyzed with EOC-RPT in service (to reduce the calculated MCPR) and take credit for the vessel high water level feedwater pump turbine trip.

With extended power uprate, 100% power will be 8% higher than the current power level. Reducing the scram bypass power level from 30% to 28% keeps the absolute thermal power at approximately the same level as it is today. The analysis and methods used to evaluate the scram bypass setpoints are consistent with those used for the original power uprate.

Proposed Change Three

In Technical Specification Table 3.3.1.1-1, Function 2.b., the allowable value for the average power range monitor (APRM) simulated thermal power - high scram is changed from "≤ 0.58 W + 62% RTP" to "≤ 0.58 W + 58% RTP." Footnote (b) is also revised to reflect this change.

Justification for Proposed Change Three

As shown in Figure 2-1 of NEDC-32749P (Enclosure 6) the power-to-flow map is modified for extended power uprate operation. The APRM signals are recalibrated to the new licensed thermal power. The APRM rod block line (which is not in the Technical Specifications) and the APRM simulated thermal power scram are changed to be consistent with the maximum rod (flow control) line upon which operation is permitted. Resetting the scram function assures adequate margin is available to prevent spurious trips while operating on a high rod line.

For extended power uprate, the trip is actually lowered 4% from its current value. However, since 100% power is redefined to be 8% higher than the current value, the absolute value (i.e., MWt at a given core flow) is approximately 4% higher. Enclosure 6 provides results of all the safety analyses that support operation on the higher absolute rod lines. The transient analyses in Section 9 of Enclosure 6 were performed with the proposed APRM setpoints. The APRM simulated thermal power scram is not a critical scram function in the mitigation of anticipated operational transients. The analyses and methods used for extended power uprate are consistent with those used for original uprate.

Proposed Change Four

Reactor coolant system pressure and temperature (P/T) limits are modified slightly to account for extended power uprate conditions. The proposed changes affect Technical Specifications Figure 3.4.9-1, Pressure/Temperature Limits for Inservice Hydrostatic and Inservice Leakage Tests; Figure 3.4.9-2, Pressure/Temperature Limits for Non-Nuclear Heatup, Low Power Physics Tests, and Cooldown Following a Shutdown; and Figure 3.4.9-3, Pressure/Temperature Limits for Criticality.

Justification for Proposed Change Four

The calculated neutron fluence values, which increase slightly in the vicinity of the core beltline region at the increased power level, cause a slight change in the integrated fluence over the remaining plant life. Section 3.3.1 of Enclosure 6 provides additional information and documents the analysis and revised curves to demonstrate continued conformance to 10 CFR 50, Appendix G and accordance with Regulatory Guide 1.99, Revision 2. The changes to the limits are not significant enough to affect operation. The analyses and methods used to recalculate the P/T limits are consistent with those used for original uprate. A recent Unit 1 Technical Specifications amendment request providing P/T curve updates was submitted after a surveillance capsule was pulled (Ref. 5).

Proposed Change Five

In Technical Specifications Section 5.5, Programs and Manuals, Subsection 5.5.12, the peak calculated primary containment internal pressure (P_a) for Unit 1 is increased from "49.6 psig" to "50.5 psig," and for Unit 2 is increased from "45.5 psig" to "46.9 psig."

Justification for Proposed Change Five

Section 4.1.1.3 of NEDC-32749P (Enclosure 6) discusses the peak short-term containment pressure response recalculated for extended power uprate. The slight increase in the absolute rod (flow control) line increases vessel subcooling, causing an increase in break flow following a design basis accident and a corresponding increase in P_a. The value for extended power uprate is well below the design pressure of containment. Changing P_a assures integrated and local leak rate testing is performed at the maximum calculated containment pressure. These analyses and methods are consistent with those used for original power uprate.

REFERENCES:

1. NRC letter from K. N. Jabbour to J. T. Beckham, Jr. (GPC), "Issuance of Amendments - Edwin I. Hatch Nuclear Plant Units 1 and 2," TAC Nos. M91077 and M91078 dated August 31, 1995.

2. NEDC-32424P, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," February 1995.

3. 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.

4. NEDC-32523P, "Generic Evaluations for General Electric Boiling Water Reactor Extended Power Uprate," Supplement 1, Volumes 1 and 2, March 1996 and June 1996.

5. SNC letter HL-5376 from H. L. Sumner, Jr. to the NRC, "Technical Specifications Revision Request for: Pressure/Temperature Limits," dated April 29, 1997.

TABLE E1-1

TECHNICAL SPECIFICATIONS CHANGES
FOR EXTENDED POWER UPRATE

Parameter	Change	Comments
RTP	Increased to 2763 MWt	Redefined.
TSV and TCV direct scram bypass and EOC-RPT	Decreased from 30% RTP to 28% RTP	Essentially same "absolute" power since 100% is redefined.
APRM simulated thermal power scram	Reduced 4%	Flow-biased slope unchanged. Slightly higher "absolute" power since 100% is redefined.
Reactor coolant system pressure/temperature limits	More restrictive	Slight increase in vessel fluence at higher power.
Peak containment pressure (post accident)	Unit 1: Increased 0.9 psi Unit 2: Increased 1.4 psi	Section 5.5, "Programs and Manuals."

Legend:

APRM - Average power range monitor
EOC-RPT - End-of-cycle recirculation pump trip
RTP - Rated thermal power
TSV - Turbine stop valve
TCV - Turbine control valve

TABLE E1-2

BASES CHANGES FOR EXTENDED POWER UPRATE

Parameter	Change	Comments
Main steam line high flow	Differential pressure changes vary with instrument.	AV in % of rated does not change.
Peak containment pressure	Unit 1: Increased 0.9 psi Unit 2: Increased 1.4 psi	Recalculated for power uprate conditions.
TSV and TCV direct scram and EOC-RPT	Decreased from 30% RTP to 28%	Technical Specifications changed.
Operability requirement for Level 8 main turbine and reactor feed pump turbine trip	Decreased from 30% RTP to 28% RTP	Consistent with TSV, TCV, and EOC-RPT requirements.
Main condenser offgas LCO	Note new RTP limit	No change to offgas limit.
Turbine bypass capacity	Decreased from 25% RTP to 21% RTP	Slightly affects transient analysis.

Legend:

AV - Allowable value
EOC-RPT - End-of-cycle recirculation pump trip
LCO - Limiting Condition for Operation
RTP - Rated thermal power
TSV - Turbine stop valve
TCV - Turbine control valve

Enclosure 2

Edwin I. Hatch Nuclear Plant
Request for License Amendment
Extended Power Uprate Operation

<u>10 CFR 50.92 Evaluation</u>

Enclosure 2

Edwin I. Hatch Nuclear Plant
Request for License Amendment
Extended Power Uprate Operation

10 CFR 50.92 Evaluation

The Nuclear Regulatory Commission (NRC) provides standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists in a proposed license amendment. A proposed license amendment does not involve a significant hazards consideration "if operation of the facility in accordance with the proposed amendment 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."

Southern Nuclear Operating Company (SNC) reviewed the proposed license amendments and determined their adoption does not involve a significant hazards consideration. The basis for this determination is given below.

DESCRIPTION OF PROPOSED CHANGES

The proposed amendment increases the licensed core rated thermal power (RTP) from 2558 MWt to 2763 MWt, which is an increase of 8% over the current licensed power level. This request is in accordance with the generic boiling water reactor (BWR) extended power uprate program established by General Electric Nuclear Energy (GENE) and approved by the NRC by letter dated February 8, 1996. (Ref. 1). The proposed amendment is very similar in format and content to the Plant Hatch original power uprate amendment which was reviewed and approved by the NRC by letter dated August 31, 1995 (Ref. 2). The proposed amendment addresses the following changes:

The power level at which the direct scram is bypassed on turbine stop valve (TSV) closure and turbine control valve (TCV) fast closure is reduced from 30% to 28%. The end-of-cycle recirculation pump trip (EOC-RPT) and the high water level turbine trip instrumentation are required above 28% power rather than the current 30% power. The average power range monitor (APRM) simulated thermal power scram equation is reduced 4%. The reactor coolant system pressure and temperature (P/T) limits and the peak calculated primary containment accident pressure are increased slightly.

Implementation of the proposed extended power uprate at Plant Hatch increases steam flow to ≈ 109% of the current value but requires no changes to the basic fuel design. Core reload design and fuel parameters are modified, since extended power uprate is implemented to support the current 18-month reload cycle. The higher power level is achieved by expanding the power-to-flow map. Vessel operating pressure is not increased. The maximum core flow limit is not increased over the pre-uprate value. Implementation of the proposed extended power uprate requires modifications in some balance-of-plant (BOP) systems, as well as calibration of plant instrumentation to reflect the uprated power. The appropriate plant operating, emergency, and other procedure changes will be made to support operation at the new rated power.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

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 support systems has no significant effect on LOCA 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. (Enclosure 5 provides a list of plant modifications.) 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.

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 B31.1 and ASME 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, 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 1 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 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. Limiting transient events tend to be slightly more severe ($\approx$ 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

    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 (ΔCPR) 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 ΔCPR 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 ≈ 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 ≈ 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 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 (≈ 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 1 and 2," (TAC Nos. M91077 and M91078), dated August 31, 1995.

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

Enclosure 3

Edwin I. Hatch Nuclear Plant
Request for License Amendment
Extended Power Uprate Operation

Environmental Assessment

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Enclosure 3

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Environmental Assessment

1.0 SUMMARY AND CONCLUSIONS

This enclosure provides an evaluation of the proposed Plant Hatch extended thermal power uprate from a rated thermal power (RTP) level of 2558 MWt to an RTP level of 2763 MWt. The intent is to provide information for the NRC to evaluate the environmental impact of extended power uprate in accordance with the requirements of 10 CFR 51.

The environmental impact of extended power uprate is identified and compared with the environmental impact previously evaluated by the AEC/NRC in the Final Environmental Statement (FES) for Plant Hatch (Refs. 1 and 2). The results of this comparison show that, in all cases, the conclusions of the FES remain valid for plant operation at 2763 MWt. Since the FES is the baseline for the assessment, SNC chose to evaluate from the original licensing basis of 2436 MWt to the proposed extended power uprate value of 2763 MWt.

In a few cases, a specific input parameter or assumption in the FES is no longer valid, or the FES and related documentation do not contain sufficient detail for comparing the impact of extended power uprate operation. In these instances, more detail is presented, or comparisons and conclusions are made, using other appropriate criteria established by the NRC.

Extended power uprate can be implemented at Plant Hatch without making extensive changes to plant systems that directly or indirectly interface with the environment. (Enclosure 5 contains a summary of the necessary modifications.) No changes to State permits are required.

This evaluation demonstrates the environmental impact of plant operation at 2763 MWt is insignificant. The impact of extended power uprate is bounded either by the previously reviewed FES or other appropriate regulatory criteria.

The Plant Hatch extended power uprate involves no significant environmental impacts as delineated by 10 CFR 51.22(a) and (c)(9). Southern Nuclear Operating Company (SNC) believes this evaluation provides sufficient justification for a categorical exclusion as provided by 10 CFR 51.21. The conclusions of the FES remain valid for the proposed amendment allowing extended power uprate operation.

2.0 OVERVIEW OF OPERATIONAL AND EQUIPMENT CHANGES FOR EXTENDED POWER UPRATE

Plant Hatch is a boiling water reactor (BWR) that operates in a direct thermodynamic cycle between the reactor and the turbine. Extended power uprate is accomplished by increasing the heat output of the reactor, thereby increasing steam flow to the turbine, as well as feedwater flow to the vessel. To support the extended power uprate to 2763 MWt, the reactor core operating range will be expanded by increasing reactor power. No changes in operating pressure, core flow, or turbine throttle pressure are necessary to support extended power uprate.

The increase in steam flow does increase the duty on the main condenser. However, no increase in circulating flow rate is required. Since Plant Hatch operates with cooling towers in the closed-cycle mode, the increase in the heatup of and water makeup requirements from the Altamaha River is small. The increase in condensate/feedwater flow will potentially increase the depletion of the resins in the condensate demineralizers. However, as shown in Enclosure 5, modifications to the demineralizers were made to reduce resin consumption.

Because of design and safety margins in the plant equipment, the proposed extended power uprate can be accomplished with relatively few modifications. Enclosure 5 provides a summary of the expected plant hardware changes. The single most significant change is the replacement of two to three stages of the Unit 2 high-pressure turbine to accommodate the higher rated steam flow. (The Unit 1 high-pressure turbine has more flow margin, thus the increase in flow area can be met by machining existing components.) Other modifications to support extended power uprate are routine in nature and are being conducted within the plant boundary using normal maintenance and modification processes. The majority of plant systems do not require any significant changes.

3.0 PROPOSED ACTION

With the operational goal of increasing electrical generating capacity, SNC, in conjunction with the nuclear steam supply system (NSSS) supplier, General Electric, comprehensively evaluated the effects of extended power uprate at Plant Hatch. The evaluation results indicate sufficient safety and design margins exist such that a prudent increase in the core RTP level from 2558 MWt to 2763 MWt can be accomplished without any adverse impact upon the health and safety of the public and the environment. Accordingly, SNC is proposing an amendment to the Plant Hatch Operating Licenses to allow for an increase in the licensed core RTP level to 2763 MWt.

The Southern Company forecasts the increase in electrical generation to allow prudent planning for adding power capacity. Large baseload plants are not required for several years. However, expected increases in customer peak demand will be met by either increasing the number of combustion turbines or purchasing electrical power from other sources.

The proposed extended power uprate will supply electrical power to the grid and displace the need for two 50-MWe gas turbines. The cost of adding this nuclear generating capacity roughly equals the cost of constructing combustion turbines. However, the fuel cost of nuclear power is approximately one-tenth that of natural gas. The proposed power uprate reduces the ¢/kWh cost at Plant Hatch and reduces the cost of electricity to Southern Company customers.

4.0 SOCIOECONOMIC IMPACT

Extended power uprate does not significantly affect the size of the Plant Hatch workforce and does not have a material effect upon the labor force required for future outages. Implementation of the program will result in a slight increase in sales tax and additional revenue to local and national businesses. However, this impact will be slight relative to the large tax revenues currently generated by Plant Hatch largely because plant modifications are minor.

Extended power uprate is an important step in improving the economic performance of Plant Hatch during and after utility deregulation. The improved performance is accomplished by cost reductions in production and total bus bar cost per kWh. Therefore, extended power uprate should enhance the value of Plant Hatch as a generating asset and reduce the probability of early plant retirement. Even in the unlikely event deregulation does not occur, extended power uprate reduces costs to the ratepayers because the additional energy is produced for < 1¢ per kWh.

The impact of not implementing extended power uprate on the probability of early plant retirement is difficult to quantify. Since this probability is greater than zero, it is prudent to outline the socioeconomic impacts of early retirement. The Southern Company is a major employer in the community and the largest single contributor to the local tax base. SNC personnel also contribute to the tax base by payment of sales and property tax. Many SNC personnel are involved in volunteer work within the community. Early plant retirement would have a significant negative impact upon the local economy and the community as a whole. The ability of the local economy to provide substitute tax revenues and similar employment opportunities for SNC employees is limited. Serious reductions in public services, employment, income, business revenues, tax revenues, and property values would result. These reductions may be mitigated by decommissioning activities in the short term.

5.0 COST-BENEFIT ANALYSIS

The direct benefit of extended power uprate to The Southern Company's residential and commercial customers is that the program will supply an additional 80 to 120 MW of reliable electrical generating capacity. A cost-benefit analysis for extended power uprate concluded the installation cost of the generating capacity ($/kWe) is comparable to a peaking unit, such as a natural gas combustion turbine. However, the energy production cost (¢/kWh) is roughly one-tenth the cost of a combustion turbine. Energy can be produced for 1¢/kWh, which is less than any other generating capacity.

Detailed Southern Company computer models that forecast system energy and generation costs were used to compare cases with and without extended power uprate. The cost of increasing energy from Plant Hatch (and deferring construction of new capacity) and obtaining the power from the generating unit with the next lower cost were estimated. The actual costs and benefits projected for power uprate are commercially sensitive; however, the benefit of the savings (present-worth to 1997 dollars) is over 100M dollars.

Although a quantitative study of environmental costs of alternatives was not performed, it is apparent significant environmental benefits can be derived from extended power uprate when compared to other options of adding capacity. As demonstrated herein, significant environmental costs are not associated with extended power uprate. Unlike fossil fuel plants, Plant Hatch does not routinely emit SO_2, NO_x, CO_2, or other atmospheric pollutants during normal operation. Routine operation of Plant Hatch at extended power uprate conditions will not contribute to greenhouse gases or acid rain.

Based upon the discussion above, it is reasonable to conclude the Plant Hatch extended power uprate project provides an economic advantage to other alternatives for added generation. Extended power uprate involves effective utilization of an existing asset with no significant additional environmental impact and is the preferable power replacement option.

6.0 NONRADIOLOGICAL ENVIRONMENTAL IMPACT

6.1 Terrestrial Effects

6.1.1 Impact on Land Use

The proposed extended power uprate for Plant Hatch will not result in any activity which will change or otherwise affect present or future requirements for land use at the site or in the surrounding area. Neither construction of new facilities nor the modification of existing facilities, including buildings, roads, parking, equipment storage areas, or transmission facilities is required to support the extended power uprate effort. Extended power uprate will not significantly affect material storage, including chemicals, fuels, and other materials stored in aboveground and/or underground storage tanks. No changes to aesthetic resources or lands with historical or archeological significance will result from extended power uprate. The conclusions stated in the FES (NUREG-0417) (Ref. 2), relative to land use remain valid for extended power uprate.

6.1.2 Transmission Facilities

A. Transmission Lines

No changes in existing transmission line design and operation will result from extended power uprate. No new requirements or changes to onsite transmission equipment, operating transmission voltages, or offsite power systems will result from implementation of extended power uprate.

B. Electromagnetic Fields (EMFs)

According to sections 4.5.4.2.3 and 4.5.6.3.4 of NUREG 1437 (Ref. 3), a review of studies on the effects of EMFs indicates chronic effects to humans are unqualified at this time, and no significant effects of EMFs on terrestrial biota have been identified. The FES (Ref. 2) determined there are no significant biological effects attributable to EMFs from high voltage transmission lines associated with Plant Hatch. The rise in generator output associated with extended power uprate will produce a slight increase in current and the corresponding EMF. However, this increase is not significant and the conclusions of the FES relative to effects of EMFs remain valid for extended power uprate.

C. Electric Shock Hazards

The FES concluded proper grounding of all structures located along high voltage power lines associated with Plant Hatch provides adequate protection against hazards from electric shock. Plant Hatch transmission lines are designed and constructed in accordance with applicable shock prevention provisions of the National Electric Safety Code. Extended power uprate does not increase the probability of electric shock associated with high voltage power lines; thus, the conclusions of the FES relative to electric shock remain valid.

6.1.3 Miscellaneous Wastes

Permits issued by the State of Georgia Department of Natural Resources - Environmental Protection Division govern the management of sanitary waste, solid waste, and air emissions. Plant Hatch utilizes a waste minimization program to further reduce waste generation. Neither the quantity nor the quality of waste generated as a result of extended power uprate will change significantly. In addition, extended power uprate will not significantly reduce the margin to limits established in the referenced environmental permits.

6.1.4 Noise

The FES concluded noise levels at the site boundary resulting from operation of the cooling towers do not adversely affect surrounding properties. Additionally, the impact of noise generated by the recently constructed helper cooling towers was reviewed and determined to be insignificant. No significant increase in ambient noise levels within the plant or at the site boundary will occur due to extended power uprate. Thus, the conclusions of the FES relative to noise levels remain bounding for extended power uprate.

6.1.5 Terrestrial Biota

Extended power uprate will not change land use evaluated in the FES and will not disturb the habitat of any terrestrial plant or animal species. The conclusions of the FES relative to impact on terrestrial ecology, including endangered or threatened plant or animal species, remain valid for extended power uprate.

6.1.6 Cooling Tower Drift, Fog

In the FES for operation of Unit 1 and construction of Unit 2, dated October 1972 (Ref. 1), the NRC concluded approximately 600 tons of minerals resulting from two-unit operation of Plant Hatch would be deposited within the site boundaries and would not result in a significant concentration of minerals on the land. This conclusion was based upon the similarity of the deposited minerals to those that occur in agricultural fertilizer, and the location of Plant Hatch in an area that receives moderately heavy, high-intensity rainfall. The NRC further concluded most of the minerals would leach into the soil and eventually reach the river, and operation of the Plant Hatch cooling towers would not be detrimental to either the land or the vegetation. To confirm the adequacy of the NRC's conclusions, a vegetative monitoring program was required for Unit 1.

In the FES for operation of Unit 2, the NRC noted that, although advanced models that could be applied to the cooling tower drift issue were available, it was more appropriate to evaluate the effects of cooling tower drift using the actual data collected from the Unit 1 monitoring program. Based upon review of the actual data, no vegetative effects attributable to salt deposition from cooling tower drift were observed. A monitoring program, which included low altitude true and false color photography, was required for Unit 2 for a minimum of 4 years. The NRC reviewed the results of the Unit 2 monitoring program and determined them to be acceptable. Thus, the Unit 2 monitoring requirement was eliminated.

A small increase in fogging potential due to operation of the cooling towers was noted in the FES but was determined to be insignificant.

The proposed extended power uprate will produce a small increase in cooling tower drift due to the increase in circulating water flow provided to accommodate the increased heat load on the cooling towers. The increase in cooling tower drift is estimated at 10 gal/min. Based upon the relative magnitude of the increase and the conservatism associated with the initial conclusions of the FES, no significant environmental impact due to the slight increase in cooling tower drift resulting from extended power uprate will occur.

6.2 Hydrology

6.2.1 Groundwater

The FES states that a minimal quantity of groundwater (327 gal/min) will be withdrawn from two wells in the regional aquifer for normal two-unit operation. The FES concluded groundwater use at the site is not expected to significantly impact the regional aquifer and is not expected to affect offsite use. A permit issued by the State of Georgia Department of Natural Resources - Environmental Protection Division governs groundwater use, with limits for withdrawal significantly above the 327 gal/min withdrawal rate associated with two-unit operation.

The proposed extended power uprate will not result in a significant increase in the use of groundwater resources and will not significantly reduce the margin to limits contained in the referenced permit. The conclusions of the FES relative to groundwater use remain valid for extended power uprate.

6.2.2 Surface Water

The FES states that Plant Hatch is expected to have a negligible effect upon surface water supplies. The 22,550 gal/min/unit withdrawal rate and the corresponding 21,800 gal/min two-unit consumptive use constitute < 0.4% of the average river flow and ≈ 4% of historical low flow. A permit issued by the State of Georgia Department of Natural Resources - Environmental Protection Division governs the use of surface water at Plant Hatch. The permit authorizes withdrawal of surface water at amounts well in excess of the rates discussed in the FES.

The proposed extended power uprate at Plant Hatch will not result in an increase in surface water withdrawal but will result in an increase in consumptive surface water use due to increased evaporation and drift. The increase in evaporation and drift will be offset by a decrease in the cooling tower blowdown discharge rate such that the 22,550 gal/min withdrawal rate upon which the conclusions of the FES were based will remain valid.

6.2.3 Intake Effects

The FES evaluated the effect of the two-unit operation of Plant Hatch upon fish impingement and the entrainment of phytoplankton, periphyton, free drifting macroinvertabrates, and fish eggs and larvae, resulting from intake system operation. The evaluation considered a 22,550 gal/min (50 ft^3/s) withdrawal rate per unit, an average river flow of 13,000 ft^3/s and a minimum river flow of 1250 ft^3/s. The evaluation concluded no significant effect upon fish impingement will occur. Additionally, the FES concluded no significant entrainment losses to riverine populations of phytoplankton, periphyton, drifting macroinvertabrates, and fish eggs and larvae will occur.

For the proposed extended power uprate, the 22,550 gal/min (50 ft^3/s) withdrawal rate considered in the FES remains unchanged, since the slight increase evaporation and drift will be balanced by a decrease in blowdown discharge such that no increase in withdrawal is anticipated. However, in the event additional withdrawal is desired, any increase in withdrawal to accommodate the slight increase in evaporation and drift (consumptive use) will be of insignificant impact to the surface water resource, and based upon the FES, will not significantly impact previous conclusions concerning impingement and entrainment.

6.2.4 Discharge Effects

Normal two-unit operation of Plant Hatch requires withdrawal of ≈ 100 ft^3/s of water from the Altamaha River to provide cooling for certain once-through loads and makeup water to the cooling towers. Based upon the FES, ≈ 49 ft^3/s of this water is lost to cooling tower evaporation and drift, while the remaining 51 ft^3/s are returned to the river via discharge pipes which extend 120 ft into the river at a depth of 4 ft. Under extended power uprate conditions, the slight increase in evaporation and drift will be offset by a decrease in the discharge of cooling tower blowdown. While withdrawal rates will not be affected, a slight decrease in the amount of water discharged will occur.

As discussed in section 6.2.3, the increase in consumptive loss of water resulting from extended power uprate will not significantly impact water resources or adversely affect the environment. The slight change in cooling tower cycles of concentration due to decreased blowdown flow will not significantly alter the chemical nature of the Plant Hatch discharge.

To evaluate the thermal impact of the two-unit operation of Plant Hatch in the FES, the NRC utilized the Motz-Benedict model for horizontal jet discharges. Based upon the results of the model, the NRC determined the least likely and most thermally severe cases for winter and summer occur only when the following conditions exist:

- River flow is equal to lowest seasonal flow of record;
- Both units are in cold shutdown condition.
- Loss of offsite power renders cooling towers inoperable.

Based upon the NRC's evaluation of the model results for the 1°F, 3°F, and 5°F isotherms generated by the Motz-Benedict model for normal and conservative meteorological and riverine conditions, the FES concluded thermal impacts associated with normal operation of Plant Hatch are acceptable. The NRC also concluded the area of the thermal plume, even under extreme conditions, was acceptably small in that the effect upon phytoplankton and periphyton, as well as drifting macroinvertabrates, will not result in significant acute or chronic mortality to populations

The effect upon fish populations due to thermal shock, the disruption of migratory routes due to thermal blockage, cold shock resulting from rapid shutdown, and other thermal-related concerns were evaluated and determined to be insignificant. The NRC concluded the small size of the thermal plume, even under extreme conservative conditions, the demonstrated ability of fish to avoid elevated temperature, and the lack of thermal blockage of the Altamaha River will prevent adverse effects on the fish population.

To evaluate the effect of extended power uprate on discharge temperature, Southern Company Services (SCS) performed a study and determined extended power uprate will increase the duty on the circulating water system, as well as the cold water temperature from the cooling towers. Since cooling tower blowdown is comprised mainly of cold water from the cooling tower, the blowdown temperature will also increase. A decrease in blowdown flow to compensate for increased evaporation will also occur. No significant increase in service water heat load beyond the current 10°F is expected from extended power uprate.

The impact of extended power uprate on river water temperature was determined based upon the following information and assumptions:

- River water flow of 1250 ft^3/s.
- River water temperature of 87.8°F.
- Service water flow of 22,550 gal/min.
- Service water temperature rise of 10°F.
- Ambient wetbulb temperature of 78°F.
- Condenser/cooling tower flow of 566,000 gal/min.
- Tower evaporation of 0.9% of tower range and flow.
- Tower drift of 0.2% of tower flow.

Condenser duty was calculated utilizing heat balance methodology. Based upon past experience, condenser duty can be 5% higher than calculated with the heat balance method. An additional 5% of duty was added for conservatism. Based upon the above information, blowdown temperature will increase an estimated 0.3°F from 89.8°F to 90.1°F. Blowdown flow to the river will decrease by an estimated 626 gal/min. River water temperature, after complete mixing, will not change significantly (< 0.1°F) with extended power uprate.

The current Plant Hatch National Pollutant Discharge Elimination System (NPDES) Permit contains monitoring requirements for discharge temperature but does not contain thermal limits for the outfall. This is based, in part, upon information relative to characterization of the thermal plume obtained from post operation studies conducted in accordance with the requirements of the FES and the Environmental Technical Specifications (currently the

Environmental Protection Plan). These studies confirmed the predictions of the Motz-Benedict model and provided a technical basis to support the State of Georgia's determination that temperature limits in the NPDES Permit for Plant Hatch were not necessary. The thermal effect of extended power uprate does not significantly alter the findings of the referenced studies or other information which supports the current NPDES Permit. No changes to thermal monitoring requirements or other NPDES Permit parameters are necessary due to extended power uprate.

Based upon the original conclusions of the FES relative to thermal impacts and the results of SCS's study, the approximate 0.1°F increase in river temperature, after mixing, under extreme temperature and flow conditions will not result in a significant adverse environmental impact. The conclusions of the FES relative to thermal impact remain valid for the extended power uprate condition.

7.0 RADIOLOGICAL ENVIRONMENTAL IMPACT

7.1 Radioactive Waste Streams

The radioactive waste systems at Plant Hatch are designed to collect, process, and dispose of radioactive wastes in a controlled and safe manner. The design bases for these systems during normal operation are to limit discharges in accordance with 10 CFR 20 and satisfy the design objectives outlined in Appendix I to 10 CFR 50. These limits and objectives will continue to be adhered to under extended power uprate.

Operation at extended power uprate conditions will not result in any changes in either the operation or the design of equipment in the liquid waste, gaseous waste, or solid waste systems. The safety and reliability of these systems are unaffected by extended power uprate. Extended power uprate does not affect the environmental monitoring of any of these waste streams or the radiological monitoring requirements outlined in the Plant Hatch Technical Specifications and the Offsite Dose Calculation Manual (ODCM). Extended power uprate does not introduce any new or different radiological release pathways, and does not increase the probability of an operator error or equipment malfunction that would result in an uncontrolled radioactive release. The specific effects of extended power uprate on each of the radioactive waste streams are evaluated in the following sections.

7.1.1 Liquid Wastes

The liquid radwaste system is designed to process, and recycle to the extent practicable, the liquid waste collected. During normal plant operation, the annual radiation doses to individuals from each reactor on the site resulting from routine liquid waste discharges are maintained below the guidelines set forth in 10 CFR 50, Appendix I. The design further ensures the short-term liquid releases from the plant resulting from equipment malfunctions or operational transients are within the limits of 10 CFR 20.1 - 20.601 (found in 10 CFR published before January 1994). System operation ensures liquid releases from the plant are within the 10 CFR 20.1001 - 20.2401 limits, as modified by Technical Specification 5.5.4.b. Liquid effluents are continuously monitored, and discharges are terminated if effluents exceed preset radioactivity levels.

Extended power uprate conditions will not result in significant increases in the volume of fluid from the various sources to the liquid radwaste system. The single largest source of liquid and wet solid waste is the backwash of the condensate demineralizers. With extended power uprate, the average time between backwash and precoat will be reduced slightly. The floor drain collection subsystem and the waste collector subsystem both receive periodic inputs from a variety of sources. Neither subsystem is expected to experience a significant increase in the total volume of liquid radwaste due to operation at extended power uprate conditions.

During normal operation, treated high-purity radwastes are normally routed to condensate storage for reuse. Treated floor drain wastes can also be routed to condensate storage, to the extent practical, consistent with reactor water inventory and reactor water quality requirements. Treated floor drain and chemical wastes are discharged into the cooling tower blowdown discharge pipe after being sampled to ensure discharge pipe concentrations after dilution are within ODCM limits. The liquid radwaste effluent from the plant to the discharge pipe, all of which must pass through a sample tank, is monitored by taking batch samples. Records of the volumes and concentration levels are retained. A process monitoring system is provided to indicate high-radiation levels in the release to the discharge pipe. Upon the annunciation of the high-radiation level alarm, the release of the liquid radwaste is terminated.

The activated corrosion products in liquid wastes are expected to increase proportionally to extended power uprate. However, the total volume of processed waste is not expected to increase appreciably, since the only significant increase is due to the more frequent backwashes of the condensate demineralizers. As noted in Table E3-1, there is no significant dose increase in the liquid pathway as shown by the comparison of analytical results from the current Final Safety Analysis Repor (FSARs) and the extended power uprate liquid effluent dose analysis.

7.1.2 Gaseous Wastes

During normal operation, the gaseous effluent treatment systems process and control the release of gaseous radioactive effluents to the site environs. These effluents include small quantities of noble gases, halogens, particulates, and tritium. The gaseous effluent treatment systems are designed to limit offsite concentrations from routine station releases to significantly less than the airborne limits specified in 10 CFR 20.1 - 20.601 (found in 10 CFR published before January 1994) and satisfy the gaseous effluent design objectives outlined in Appendix I to 10 CFR 50. System operation ensures gaseous effluent releases from the plant are within 10 CFR 20.1001 - 20.2401 limits.

The gaseous radioactivity of the reactor coolant is, in part, a function of the extent of fuel defects -- the causes of which are independent of extended power uprate.

The gaseous waste management systems include the offgas system and various building ventilation systems. Gaseous effluents are only released from three points: the main stack and each unit's reactor building vent plenum. All gaseous effluents, other than ventilation releases, are released from the main stack. The main stack release is composed of effluents from the offgas system and includes input from the steam jet air ejectors, mechanical vacuum pumps, and gland-seal systems. Assuming noble gas generation rates and the radioactivity contribution from halogens, particulates, and tritium are approximately proportional to the power increase, it is reasonable to conclude a small increase in gaseous effluents due to extended power uprate will occur.

Reactor, turbine, and radwaste building ventilation releases are discharged from the reactor building vent plenums. Gaseous releases through this pathway are dependent upon the radioactivity and the concentration of airborne particulates and gases from leakage of contaminated systems. However, system leakage is independent of extended power uprate.

As noted in Table E3-1, the estimated extended power uprate dose values are still below Appendix I requirements. Furthermore, the dose impact is a very small increase (< 8%) for the gaseous pathway, based upon the comparison of the analytical results from the FSARs (updated for original power uprate) and the extended power uprate gaseous effluent dose impact evaluation.

7.1.3 Solid Wastes

The solid radwaste system collects, monitors, processes, packages, and provides temporary storage facilities for radioactive solid wastes prior to offsite shipment and permanent disposal. The Edwin I. Hatch Nuclear Plant Solid Radioactive Waste Process Control Program (PCP) describes this objective. The PCP is implemented by procedures which contain formulas, sampling requirements, analyses, tests, and determinations performed to ensure the processing and packaging of solid radioactive waste, is accomplished in compliance with 10 CFR 20, 61, and 71, as well as State regulations and burial ground requirements governing the disposal of solid radioactive waste.

A. Wet Wastes

The wet solid radwaste system is a continuous part of the liquid radwaste system. Wet wastes, consisting primarily of spent demineralizer resins and filter sludges, are accumulated in phase separators and waste sludge tanks. These tanks serve as storage and batching tanks for the wet solid radwaste system.

B. Dry Wastes

Dry wastes consist of air filters; miscellaneous paper and rags from contaminated areas; contaminated clothing, tools, and equipment parts that cannot be effectively decontaminated; and solid laboratory wastes. The activity of much of this waste is low enough to permit manual handling.

Dry wastes are collected in containers located throughout the plant, as dictated by the volume of wastes generated during operation and maintenance. The filled containers are sealed and moved to a controlled-access enclosed area for temporary storage. Compressible wastes are compacted, as needed, into appropriate containers in a hydraulic press to reduce volume. Noncompressible wastes are packaged in similar containers. Because of its low activity, dry waste can be stored until enough is accumulated to permit economical transportation to an offsite processing facility or a burial ground for final disposal.

C. Irradiated Reactor Components

This waste consists primarily of spent control blades, fuel channels, incore ion chambers, and large pieces of equipment. Because of the high activation and contamination levels, reactor equipment waste is stored in the spent-fuel storage pool to allow for sufficient radioactive decay before removal to inplant or offsite storage and final disposal in shielded containers or casks.

The largest volume contributors to radioactive solid waste are the spent resin and filter sludges from the process wastes. Equipment wastes from operation and maintenance activities, chemical wastes, and reactor system wastes also contribute to solid waste generation. Extended power uprate conditions may involve a slight increase in the process wastes generated from the operation of the reactor water cleanup (RWC) filter demineralizers, fuel pool filter demineralizers, and the condensate filter demineralizers. The exchange limits for the RWC filter demineralizers, the fuel pool filter demineralizers, and the condensate filter demineralizers are based upon differential pressure and effluent water chemistry. More frequent RWC backwashes are expected to occur under extended power uprate conditions due to water chemistry limits. Extended power uprate will not involve changes in either RWC flow rates or filter performance.

The principle effect of extended power uprate upon the condensate demineralizer system (CDS) is increased condensate flow. A consequent result of increased condensate flow is that the condensate vessel differential pressure exchange limit will be reached more frequently, resulting in reduced run times. Without any modification, the spent resin generation from the condensate demineralizers is expected to increase. To offset these reduced run times, in parallel with other uprate activities, Plant Hatch is adopting the use of pleated filter elements in the condensate demineralizer vessels. With the current demineralizer flow rates (3400 gal/min), the estimated run times for vessels utilizing pleated filter elements or septa are on the order of 50 days, as compared to 25 days using conventional filters.

The use of the pleated filters allows precoating with less resin. Conventional filters used at Plant Hatch prior to the adoption of pleated filters required an average of 18 ft^3 to 20 ft^3 of resin per demineralizer precoat. The pleated filters are expected to require 9 ft^3 to 12 ft^3, thereby resulting in a significant reduction in resin usage. Other types of filters may be used depending on the performance of the pleated filters.

In conjunction with the adoption of pleated filters, Plant Hatch is installing an air surge system which increases the energy of the backwash, enhancing the ability to flush material out of the filters. Air surge backwash will extend the life of the demineralizer filters and reduce the deterioration in run length normally seen over the life of a set of filters. Current long-range plans call for utilizing pleated filters in all 14 condensate demineralizer vessels (7/unit); however, the short-term plans to support extended power uprate call for using pleated filters in four to five vessels/unit. Based upon previous operational experience, slight increases in solid wastes from the RWC/CDS processes under extended power uprate conditions are not expected to result in waste volumes substantially above current levels.

An insignificant impact upon the amount of irradiated reactor components is also anticipated. Because of the mitigating effects of extended burnup and increased U-235 enrichment under extended power uprate conditions, the number of irradiated fuel assemblies and other components discharged from the reactor should not increase substantially.

7.2 Process and Effluent Monitoring

The process and effluent radiological monitoring systems, including the primary containment radiation monitoring system, are contained in the process radiation monitoring system (PRMS) which is designed to:

- Measure and record radioactivity levels.
- Alarm on high radioactivity levels.
- Control, as required, the release of radioactive liquids, gases, and particulates produced in the operation of the plant.
- Comply with the requirements of 10 CFR 20.1 - 20.601 (found in 10 CFR published before January 1994) and the appropriate General Design Criteria (GDC) in 10 CFR 50, Appendix A.

The PRMS is operated to ensure compliance with the requirements of 10 CFR 20.1001 - 20.2401.

The PRMS furnishes information to operations personnel regarding radioactivity levels in principal plant process and effluent streams to assist in maintaining radiation levels as low as reasonably achievable (ALARA). The information is also used to verify compliance with applicable governmental regulations for the containment, control, and release of radioactive liquids, gases, and particulates generated as a result of normal or emergency plant operations. The PRMS monitors serve this function in conjunction with a comprehensive sampling program which is the primary method for quantitatively evaluating system and effluent activity levels. No changes to the PRMS are required to accomplish extended power uprate.

7.3 Inplant Radiation Levels and Offsite Dose

7.3.1 Operating and Shutdown Inplant Radiation

Extended power uprate will involve potential increases in radiation sources. Plant Hatch was conservatively designed with respect to shielding and radiation sources. This conservatism, coupled with plant physical improvements and administrative controls, compensates for the potential increase in radiation sources.

In a shielding analysis, the reactor water fission and corrosion product activity was conservatively assumed to be 2.1 μCi/g. During typical plant operating conditions, total reactor water activity is limited to a much lower level. Based upon weekly surveillance data from 1990 to present, normal values of reactor water fission and corrosion product activities are 0.12 μCi/g and 0.08 μCi/g on Unit 1 and Unit 2, respectively.

The original design value for N-16 concentration was 50 μCi/g. To support the injection of hydrogen into the feedwater, plant shielding reviews were performed to evaluate the potential impact of injection rates of up to 65 sft^3/min. This injection rate theoretically yields an increase in N-16 dose rates of up to a factor of approximately 5.5, which far exceeds the potential impact of increased N-16 generation due to extended power uprate. The results of these reviews identified target areas of the plant which were modified to maintain radiation levels within acceptable limits. Therefore, extended power uprate will not have an impact upon the acceptability of the shielding design.

Plant Hatch implemented hydrogen injection in 1987 and 1991 for Units 1 and 2 respectively. As expected, there was an increase in personnel occupational dose after implementation. To offset this increase and to improve personnel dose rates, cobalt removal and zinc injection dose reduction programs were implemented. The cobalt reduction program was initiated in 1993 and zinc injection was initiated in 1990. Unit 1 chemical decontaminations were performed in 1991 and 1996 to reduce radiation fields in the reactor auxiliary systems. Since May 1993, the Unit 1 and Unit 2 reactor water cobalt-60 and zinc-65 concentrations (cycle mean values) have decreased in value as shown below:

	Unit 1		Unit 2	
	1993	1997	1993	1997
Co-60	0.315	0.088	0.564	0.122
Zn-65	2.186	0.439	2.513	0.649

(units µCi/kg)

The cobalt reduction and zinc injection programs, and the Unit 1 chemical decontamination have helped reduce overall radiation dose rates (i.e., BWR Radiation Assessment and Control (BRAC) Points Unit 1 EOC 16 from 268 mR/hr to 153 mR/hr and BRAC Points Unit 2 from 250 mR/hr EOC 11 to 193 mR/hr EOC 13.) These programs will adequately compensate for possible dose rate increases due to extended power uprate.

The activity concentration of corrosion products in the reactor water and on piping surfaces may increase due to the increased transport of corrosion products from the feedwater system into the primary system, and the increased core average flux. This potential is mitigated by Plant Hatch's programs involving zinc injection and cobalt reduction that have reduced normal concentrations of corrosion products, with consequent reductions in inplant doses, and can be expected to maintain corrosion product levels below design values.

Compared to assumptions in the FES, the amount of fuel required for extended power uprate has been reduced. This reduction was achieved through the use of higher enrichments, more efficient fuel designs, and the achievement of higher burnups. Potential environmental effects of these fuel cycle changes are discussed in section 8.0. No significant increases in inplant dose rates attributable to the original power uprate have been observed, and a minimal change in dose rate is anticipated with implementation of extended power uprate.

The plant radiation protection program will maintain individual doses consistent with ALARA requirements and below the limits of 10 CFR 20. Routine plant radiation surveys required by the radiation protection program will continue to identify any increased radiation levels in accessible areas of the plant and radiation postings will be adjusted as necessary. Current administrative dose control limits are established below applicable regulatory limits to ensure compliance.

7.3.2 Offsite Doses at Extended Power Uprate Conditions

An offsite dose compliance analysis is typically provided as a licensing tool to demonstrate the plant, as designed, can meet the offsite effluent release requirements of ALARA as outlined in 10 CFR 50.34a and 50.36a. The actual operational requirements to conform with the 10 CFR 50 ALARA and Appendix I commitments for offsite dose, as well as the requirements of 10 CFR 20.1001 - 20.2401 as modified by Technical Specification 5.5.4.b, are based upon the application of ODCM methodology. This is augmented with actual liquid and gaseous effluent release data, in conjunction with current dispersion/deposition data and periodic land/population/biota usage survey information. The basis for the monitoring and surveillance program is contained in the radiological effluent technical specifications (RETS), which were incorporated into the ODCM, as permitted by the implementation of NRC Generic Letter 89-01.

Table E3-1 provides the results of calculations performed to assess the potential offsite dose impact due to extended power uprate. The estimated doses from both the liquid and gaseous release pathways are well within the design objectives outlined in Appendix I to 10 CFR 50. Furthermore, the doses to offsite individuals due to normal operational liquid effluent releases do not exceed the estimated liquid effluent dose values currently outlined in the FSARs.

Table E3-1 also compares actual data from the Plant Hatch 1996 Radioactive Effluent Release Report (RERRR) with the calculated estimates in the FSARs for the original power uprate. The doses associated with the actual data indicate a very small dose impact (< 7%) relative to Appendix I criteria and a large dose decrease (77.1% less) when compared to the estimated dose values in the FSARs. Based upon this comparison, it is reasonable to assume similar results will be seen when comparing actual extended power uprate effluent release results with calculated extended power uprate values. Therefore, an insignificant increase in the dose to offsite individuals due to gaseous effluent releases under extended power uprate conditions is expected. This increase is ≈ 2.4% greater than the total body dose and ≈ 7.3% greater than the individual organ dose presented in the FSARs.

Based upon this assessment, and the Biological Effects of Ionizing Radiation (BEIR) Report referenced in the FES, which concluded that no other living organisms are very much more radiosensitive than man (such that control of dose to man adequately controls dose to other biota), it is expected that there will be an insignificant dose impact to the hydrosphere. The hydrosphere typically includes the general public, area population, and terrestrial and aquatic biota other than man.

Since extended power uprate does not create any new or different sources of offsite dose from plant operation and does not involve significant increases in present radiation levels, it is reasonable to conclude under extended power uprate conditions, offsite doses will remain well within regulatory requirements and pose no significant environmental impact.

7.4 Radiological Consequences of Accidents

Georgia Power Company's (GPC) November 1971 Supplement 1 to the Plant Hatch Environmental Report (ER) considered the radiological environmental impact of the plant as required by Appendix D to 10 CFR 50, "Interim Statement of General Policy and Procedure: Implementation of the National Environmental Policy Act of 1969 (PL91-190)," dated September 9, 1971. Section 6.0 of Supplement 1 focused upon the environmental effects of accidents. The information provided was based upon the guidance provided by the Atomic Energy Commission (AEC) in "Scope of Applicants' Environmental Reports with Respect to Transportation, Transmission Lines, and Accidents," dated September 1, 1971, and "Draft -- Guide to the Preparation of Environmental Reports for Nuclear Power Plants," issued for comments and interim use, and dated February 1971.

In the ER, GPC used a population of 246,300 within a 50-mile radius of the site and a combined annual background due to natural and manmade sources of ≈ 0.24 rem/year. GPC calculated total annual man-rem values due to natural and manmade backgrounds of 34,500 and 24,600 respectively. These man-rem values were used in the evaluation of accident consequences.

In the Plant Hatch FES, the AEC provided estimates of the doses that might be received by an assumed individual standing at the site boundary in the downwind direction resulting from nine classes of proposed accidents using the assumptions in the proposed Annex to Appendix D of 10 CFR 50. These estimates are presented in Section VI of the FES. Estimates of the integrated exposure in man-rem that might be delivered to the population within 50 miles of the site were also included in the FES. The man-rem estimates were based upon the projected population around the site for the year 2012 (≈ 270,000 people). The AEC calculated that the integrated exposure due to a natural background radiation level of 0.1 rem/year would be ≈ 27,000 man-rem.

As provided in Annex A of the Georgia Radiological Emergency Plan, census results from 1990 showed ≈ 355,000 people within a 50-mile radius of the site. The numbers estimated by GPC in the ER were updated to reflect the 1990 census results by adjusting the background levels to account for the 1990 population. Multiplying by a factor of ≈ 1.44 (i.e., 355,000/246,300) yields 49,726 man-rem and 35,457 man-rem, respectively. Also, the accident analysis results and estimates of cumulative doses provided by the AEC and reported in the FES were increased by a factor of ≈ 1.31 (i.e., 355,000/270,000) to account for the 1990 population. The AEC estimates and accident analysis results as presented in Table V1-2 of the FES are revised to reflect extended power uprate using the 1990 census information and provided in Table E3-2.

In the Plant Hatch FES, the AEC noted the examples selected by GPC in the ER were reasonably homogeneous in terms of probability within each class, with the exception of the failure of the offgas holdup system or a liquid radwaste tank rupture. Although these accidents were evaluated as Class 8 accidents (Accident Initiation Events Considered in the Design-Basis Evaluation in the Safety Analysis Report), the AEC believed both of these events were more appropriately evaluated as Class 3 accidents (Radwaste System Failures).

7.4.1 Class 1 - Trivial Incidents - Small Leaks Inside Containment:

The ER defines Class 1 occurrences as primary coolant leaks, below or just above allowable Technical Specifications limits, within the primary containment or the secondary containment (reactor building). However, the ER states, "Class 1 events are not considered herein, in keeping with the guide[*sic*] Ref. 2 ('Scope of Applicants Environmental Reports With Respect to Transportation, Transmission Lines, and Accidents')." In the FES, the AEC concluded releases due to trivial incidents (i.e., small leaks inside containment) were comparable to the design objectives indicated in proposed Appendix I to 10 CFR 50 for routine effluents (i.e., 5 mrem/year to an individual from either liquid or gaseous effluents).

The ER defines the source for Class 1 accidents as primary coolant with a leak rate below or just above the allowable Technical Specifications leak rate. Because the design stress limits of plant piping systems and pressure-retaining components are not exceeded under extended power uprate conditions and extended power uprate does not change the limits on allowable reactor coolant system operational leakage given in Technical Specification 3.4.4, extended power uprate does not increase the frequency of occurrence of trivial events. Technical Specification 3.4.6 limits reactor coolant activity, and no change to this Technical Specifications limit will result from extended power uprate. Thus, the dose consequences to the environment due to small leaks inside containment are not increased by extended power uprate, and the AEC's conclusions in the FES remain valid.

7.4.2 Class 2 - Miscellaneous Small Leaks Outside Containment

The ER defines this class of accidents as reactor coolant leaks, below or just above the allowable Technical Specifications limits, located outside the primary and secondary containments. Effluents or sources of activities in this category include turbine building effluents:

- Gaseous, anywhere in the turbine building.
- Liquid, anywhere in the turbine building.

Since Class 2 events were assumed to occur within the turbine building, the ER concluded the leaks must be released to the building ventilation system rather than to the building drains (which would not result in a release to the environment).

The AEC concluded in the FES releases due to miscellaneous small leaks outside containment were comparable to the design objectives indicated in proposed Appendix I to 10 CFR 50 for routine effluents (i.e., 5 mrem/year to an individual from either liquid or gaseous effluents).

The ER defines the source for the small releases outside containment as primary coolant. Since design stress limits of plant piping systems and components are not exceeded under plant extended power uprate conditions, and no new pathways for release to the environment are created by extended power uprate, extended power uprate does not increase the frequency of occurrence of these events. Technical Specification 3.4.6 limits reactor coolant specific activity. No change to this Technical Specification limit will result from extended power uprate. Furthermore, since initial operation, improvements in fuel performance were developed and implemented, resulting in lower coolant gaseous activity. Thus, the dose consequences to the environment due to small leaks outside containment are not increased by extended power uprate, and the AEC's conclusions in the FES remain valid under extended power uprate.

7.4.3 Class 3 - Radwaste System Failures

The ER restricts events in this category to high-probability single functional system or equipment failures, or single operator error occurrences. The ER considered low-probability radwaste system failures (i.e., liquid radwaste tank accidents and offgas system accidents) Class 8 events; however, the FES considered these failures were Class 3 accidents. Sources of effluents in Class 3 accidents are defined as:

a. Single functional equipment failures:
   Gaseous release from offgas system.
   Liquid leakage through valves.

b. Single operator error:
   Liquid discharge without batch testing.
   Gaseous release of holdup system via purge valve operation.

Since extended power uprate does not involve changes to radwaste system design and operation, the frequency of occurrence of radwaste system failures is not increased. Extended power uprate does not change the Technical Specifications limit on reactor coolant activity, nor does it change the Technical Specifications limit on the gross gamma activity rate of the noble gases measured at the main condenser evacuation system pretreatment monitor station (Technical Specification 3.7.6).

The following discussion provides the effect of extended power uprate on radwaste system failures involving liquid discharge, gaseous discharge, offgas system accidents, and liquid radwaste tank failures.

Liquid Discharge

The ER discusses doses due to operator and equipment failure for the liquid and gaseous radwaste systems. For the liquid radwaste system, the area of primary significance for liquid radwaste releases is the consumption of water by the general public. Since the river downstream of Plant Hatch is not used for municipal water supply, any event that results in the release of radioactivity into the river is of minimal concern to the general population for exposure. Thus, this event was determined to be not applicable for doses to the general public.

Since river usage has not changed, the conclusions of the ER regarding doses due to liquid radwaste system malfunctions remain valid.

Gaseous Discharge

For gaseous releases, the ER states that the source of potential release is via the drain lines, caused by a failure of the water seal to prevent gaseous leakage.

Since extended power uprate does not involve changes to radwaste system design and operation, the frequency of occurrence of this type of accident is not increased by extended power uprate. The ER concluded the integrated dose to the population within a 50-mile radius of the plant due to a gaseous release was not significant (i.e., < 0.01 man-rem). Considering a potential increase in activity of 13% to account for extended power uprate and adjusting for the 1990 population, the potential dose increases to < 0.016 man-rem, which is still insignificant when compared to annual background radiation.

The FES listed the estimated fraction of the 10 CFR 20 limit (as it existed in 1971) of 0.5 rem at the site boundary attributable to radwaste equipment leakage or malfunction as 0.037, and the integrated dose to the population within a 50-mile radius of the plant as 0.98 man-rem. Considering a potential increase in activity of 13% to account for the power uprating, the estimated fraction increases to 0.042 (still a small fraction of the 10 CFR 20 limit). To account for the increased activity and the increased population projections based upon the 1990 census, the 50-mile integrated population dose is revised to 1.46 man-rem. This population dose is insignificant when compared to that from background radiation.

Since the doses due to this postulated accident remain a small fraction of the dose due to naturally occurring radiation, the AEC's conclusion that the environmental risks due to postulated radiological accidents at Plant Hatch are exceedingly small remains valid.

Offgas System Accident

Although the ER considers the offgas system accident a Class 8 accident, the FES considers it a Class 3 accident. The Bases for Technical Specification 3.7.6 states that this limit ensures compliance with the assumptions of the offgas system failure evaluated in the FSARs. Extended power uprate does not change the Technical Specifications limit on the gross gamma activity rate of the noble gases measured at the main condenser evacuation system pretreatment monitor station, and thus, does not increase the dose resulting from the offgas system accident.

Since the doses due to this postulated accident are not changed and remain a small fraction of the dose due to naturally occurring radiation, the AEC's conclusion that the environmental risks due to postulated radiological accidents at Plant Hatch are exceedingly small remains valid.

Liquid Radwaste Tank Failure

Although the ER considers liquid radwaste tank failures Class 8 accidents, the FES considers these failures Class 3 accidents. The AEC concluded the estimated fraction of the 10 CFR 20 limit at the site boundary due to a release of a liquid waste storage tanks contents was < 0.001. The resulting estimated dose to the population in a 50-mile radius of the plant was < 0.1 man-rem. Increasing these results to account for the increase in thermal power, the fraction of the Part 20 limit is increased to < 0.0011. The estimated population dose, adjusted for the thermal power increase and the 1990 population, is < 0.15 man-rem, which, when compared to the estimated dose to the same population due to background radiation, remains small.

Since the doses due to the postulated accident remain a very small fraction of the dose due to naturally occurring radiation, the AEC's conclusion that the environmental risks due to postulated radiological accidents at Plant Hatch are exceedingly small remains valid.

7.4.4 Class 4 - Events that Release Radioactivity into Primary System

The ER states that the design basis for the BWR precludes fuel defects from operational transients. Fuel defects occurring during normal operation are addressed in a separate section of the ER (i.e., under the discussion of Normal Reactor Facility Operation). Although the ER does not identify any Class 4 events, the section of the FES entitled "Summary of Radiological Accidents Determined by the AEC," identifies two Class 4 events:

- Fuel cladding defects.
- Off-design transients that induce fuel failures above those expected.

The AEC concluded, for fuel cladding defects, the releases will be comparable to the design objectives indicated in proposed Appendix I to 10 CFR 50, for routine effluents (i.e., 5 mrem/year to an individual from either liquid or gaseous effluents). The AEC reported an estimated fraction of 10 CFR 20 limits of 0.002 for the off-design transients inducing fuel failures. The estimated dose to the population within a 50-mile radius of Plant Hatch was 0.1 man-rem.

Since initial operation, fuel cladding defects have been significantly reduced due to industry improvements in fuel cladding performance (Ref. 4). In addition, operational limits that include significant margin from fuel failure are calculated for each cycle to prevent transients from inducing fuel damage.

These operational limits will continue to be calculated after extended power uprate is implemented to ensure plant transients do not result in fuel damage.

Fuel damage can be monitored indirectly by monitoring reactor coolant activity. As previously indicated, the Technical Specifications limit reactor coolant activity, and extended power uprate will not increase the allowable reactor coolant activity above that currently allowed by the Technical Specifications.

Ignoring the improvements in fuels technology and reliability, and considering only the increase in core inventory associated with extended power uprate, in reviewing the consequences of fuel failures induced by off-design transients, a multiplier of 1.13 is used to account for the 13% increase in core inventory. Multiplying the fraction of Part 20 limits estimated for this accident by this factor results in an estimated fraction of 0.0023. Multiplying the estimated dose to the population by 1.13 and adjusting for the 1990 population yields an estimated dose to the population within a 50-mile radius of ≈ 0.15 man-rem. The estimated fraction of Part 20 limits remains small. The dose to the population within a 50-mile radius is extremely small when compared to the integrated dose to the same population that the AEC calculated due to normal annual background radiation.

Since the doses due to the postulated accident remain a small fraction of the dose due to naturally occurring radiation, the AEC's conclusion that the environmental risks due to postulated radiological accidents at Plant Hatch are exceedingly small remains valid.

7.4.5 Class 5 - Events that Release Radioactivity into the Secondary System

The ER interprets "secondary system" to mean the secondary side of heat exchangers whose primary side contains reactor water. The following Plant Hatch heat exchangers fit within this category:

- Main condensers.*
- RHR heat exchangers.
- RHR pump coolers.*
- Fuel pool heat exchangers.
- RWC nonregenerative heat exchanger.
- RWC pump coolers.
- Reactor recirculation pump seal coolers.
- Miscellaneous radwaste sump coolers.

- Sample coolers.
- Post-accident sample coolers.*

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* Prevented from leaking reactor water into the secondary side by operating with the secondary side pressure higher than the primary side pressure. To preclude leakage to the environment, the remaining heat exchangers are serviced by an intermediate closed-loop cooling system; i.e., (reactor building closed cooling water (RBCCW) system.

Considering the effects of extended power uprate, the secondary system operating pressure will remain above the primary side pressure. No new secondary systems are required and no changes to the RBCCW system that would affect the prevention of leakage to the environment are required for extended power uprate. Thus, extended power uprate does not affect the previously drawn conclusions as to the applicability of Class 5 events at Plant Hatch.

7.4.6 Class 6 - Refueling Accidents Inside Containment

The ER states that refueling accidents are of two essential types:

A. Dropping a Heavy Object onto the Core

The accident chosen as typical of this category was the design basis refueling accident, wherein an equipment failure allows a fuel bundle to drop onto the core from the maximum permissible height, resulting in perforation of a maximum of 49 rods (i.e., all of the fuel rods of one 7x7 assembly). This event was chosen because the fuel assembly is the only heavy object that is routinely suspended over the core, and if dropped, could damage the core. The refueling accident is assumed to result in perforation of all rods in the damaged assembly; however, the plant no longer uses 7x7 fuel. Although Plant Hatch no longer uses 7x7 fuel, for this discussion (i.e., to highlight the changes in radiological effects on the environment), it is assumed the percent of failed fuel resulting from the drop of a heavy object onto the core does not change with the use of a different fuel type. Based upon the assumption that the fraction of the core released does not change and the power density of the fuel is decreased with the use of 8x8 and 9x9 fuel, a change in fuel type does not affect the previously calculated doses. Extended power uprate does affect the previously calculated doses due to the increase in core inventory.

Per the ER, the only parameters used in calculating the consequences of the refueling accident, potentially changed by extended power uprate, are the type and fractional activity released. The ER references APED-5756, "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor." To account for the 13% increase in core inventory associated with extended power uprate, a multiplier of 1.13 is used on the previously calculated doses. For the refueling accident, the ER estimates the integrated dose to the 50-mile population to be 0.13 man-rem.

Adjusting for the increased core inventory and the 1990 population, the dose is estimated to be 0.21 man-rem. This dose is extremely small when compared to the cumulative man-rem due to naturally occurring and manmade background radiation, which, when adjusted for the 1990 population becomes 49,726 man-rem and 35,457 man-rem, respectively. Thus, the ER conclusion that this event is not significant with regard to environmental effects remains valid.

In the "Summary of Radiological Consequences of Postulated Accidents Determined by the AEC" as given in the FES, the AEC estimated the fraction of 10 CFR 20 limits at the site boundary resulting from this event as < 0.001, and the estimated dose to the population within a 50-mile radius was estimated to be < 0.1 man-rem. Using the multiplier of 1.13 to account for the increased core inventory due to extended power uprate results in a site boundary (10 CFR 20 limits) fraction of < 0.0011.

Adjusting the estimated 50-mile population dose for increased core inventory and the 1990 population yields ≈ 0.15 man-rem. Again, this dose is extremely small when compared to the cumulative annual man-rem due to naturally occurring background radiation. The FES conclusion that the environmental risks due to postulated radiological accidents at Plant Hatch are exceedingly small remains valid for extended power uprate.

B. Dropping a Spent Fuel Cask

A fully loaded spent fuel cask is assumed to be dropped while being lowered to a waiting flatcar. This event was chosen to represent this category because it demonstrates the potential for dropping a fuel cask from the maximum height in the plant; potentially compromising cask integrity. If the cask was dropped inside the fuel pool, the reactor building would be damaged but cask integrity would be assured, and no release from the cask would occur. The cask is considered as dropping from a height of ≤ 99 ft to a yielding surface (the flatcar and points below), resulting in a release within the limits of 10 CFR 71. The ER lists the consequences of this event as negligible (i.e., < 0.01 man-rem).

Extended power uprate does not modify any equipment used to handle spent fuel casks, and thus, cannot affect the frequency of occurrence of this event. However, since extended power uprate increases core inventory the potential to increase the consequences of the event exists. Again, when using a multiplier of 1.13 to account for the increased core inventory due to extended power uprate and adjusting for the 1990 population, the resulting 50-mile cumulative dose will be < 0.016 man-rem. This dose remains negligible when compared to the 1990 adjusted cumulative annual man-rem, out to 50 miles, resulting from naturally occurring and manmade background radiation (49,726 man-rem and 35,457 man-rem, respectively). Thus, the ER conclusion that the accident does not have a significant impact upon the environment remains valid when considering the effects of extended power uprate.

The AEC's evaluation of the fuel cask drop event is given as a Class 7 event (spent fuel handing accident). The AEC concluded the estimated fraction of the 10 CFR 20 limits at the site boundary was 0.055. The estimated dose to the population in a 50-mile radius of the plant was estimated to be 1.4 man-rem. When using the 1.13 multiplier to account for the increased core inventory due to extended power uprate, the resulting fraction of the Part 20 limits is increased to 0.062. The resulting estimated dose to the population within a 50-mile radius, considering the increased core inventory and the 1990 population, is 2.08 man-rem. The increase in the 10 CFR 20 limits fraction due to extended power uprate is small, and the resulting dose to the 50-mile population resulting from this event remains small when compared to the dose to the same population resulting from background radiation (35,500 man-rem/year). The FES conclusion that the environmental risks due to postulated radiological accidents at Plant Hatch are exceedingly small remains valid for extended power uprate.

7.4.7 Class 7 - Accidents to Spent Fuel Outside Containment

The ER indicates the Class 7 accident applies to the movement of a spent fuel cask on a railroad flatcar from the time the flatcar leaves the reactor building until it reaches the site boundary. Spent fuel movement outside containment is always done with the fuel inside a cask. It was concluded engineering and procedural cautions pertaining to the movement of spent fuel on site essentially preclude the possibility of a cask dropping due to instability, improper attachment to the bat of the flatcar, or a derailment. These cautions are not affected by extended power uprate. If a shipping cask were dropped, it would be from such a height that the cask would easily sustain the drop.

Although a cask could conceivably be damaged by fire, the ER states that the site arrangement precludes movement of the car in areas of appreciable fire hazard. Extended power uprate introduces no changes to this conclusion.

Fires due to wheel bearing overheating were discounted as unlikely, given the low velocity at which cask movement occurs. Extended power uprate does not affect the procedures for cask handling. The ER concluded doses to the public due to onsite movement of spent fuel outside containment are not expected. Extended power uprate does not affect the assumptions used to reach these conclusions. The ER determined doses due to Class 7 accidents are negligible. Extended power uprate does not affect this determination.

The FES considered a fuel assembly drop in the fuel storage pool as a Class 7 accident. The radiological consequences of this accident were identical to those reported under refueling (Class 6) accidents, and the effects of extended power uprate will increase the dose consequences in a manner identical to that for the Class 6 accidents. Because the resulting doses, considering increased core inventory and the 1990 population, will remain exceedingly small when compared to the doses to the population due to background radiation, the conclusion that the environmental risks due to this postulated accident at Plant Hatch were exceedingly small remains valid.

7.4.8 Class 8 - Accident Initiation Events Considered in the Design Basis Evaluation in the FSARs

The ER includes the following DBAs:

A. LOCA Inside Primary Containment - Recirculation Loop Pipe Break Accident

The ER states that a sudden circumferential break is assumed to occur in a recirculation line, permitting the discharge of coolant into the primary containment from both sides of the break. Concurrent with this failure, the worst single active component failure is assumed to occur, thus producing the maximum damage to the core. Extended power uprate does not increase the frequency of occurrence of the event, since the design stress limits of plant piping systems and components are not exceeded under plant extended power uprate conditions. Extended power uprate does not affect ECCS operation or the frequency of occurrence of the assumed single failure.

The ER uses fission products available for release as given in APED-5756, "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor." Considering the increase in core inventory associated with extended power uprate, a multiplier of 1.13 is used to account for the increased fission products for postulated fuel failures. The ER states that the cumulative man-rem to the population within a 50-mile radius of the plant is negligible (i.e., < 0.01 man-rem). Considering the 1990 population and the fission product increase, the resulting cumulative dose is 0.016 man-rem.

The "Summary of Radiological Consequences of Postulated Accidents Determined by the AEC," provided in the FES, gives the estimated fraction of the Part 20 limits at the site boundary as < 0.001 for both small- and large-break loss-of-coolant accidents (LOCAs), and estimated the dose to the population in a 50-mile radius to be < 0.1 man-rem for small-break LOCAs and 1.2 man-rem for large-break LOCAs. Using the 1.13 multiplier to account for the effects of extended power uprate, the resulting estimated fraction of the Part 20 limits remains < 0.0011. Using the 1.13 multiplier and adjusting for the 1990 population, the estimated dose to the population increases to < 0.15 man-rem for small break LOCAs and 1.78 man-rem for large break LOCAs. Again, when compared to the estimated doses to the same population from background radiation (35,500 man-rem/year), these doses remain insignificant.

B. LOCA Outside Primary Containment - Main Steam Line Break Accident

The ER states that a main steam line break (MSLB) is a sudden, complete severance of one main steam line outside the drywell, with subsequent release of steam and water containing fission and corrosion products to the pipe tunnel and the turbine building. The ER further states that this accident does not result in any fuel damage and the environmental effects are limited to radiological doses that may be received as a consequence of exposure to the activity associated with the primary coolant. It is noted that, currently, the accident analysis for the main steam line break accident assumes the break occurs at hot standby (the most limiting case). However, a main steam line break at full rated power is considered to be more probable and is used in this discussion to highlight the increase in radiological effects on the environment due to extended power uprate, since the original analysis in the ER assumed this break occurs at power.

Taking into account the effects of extended power uprate on core inventory (factor of 1.13) and the effects of increased reactor vessel pressure resulting from the original power uprate (factor of 1.05), the doses resulting from a MSLB are assumed to increase by a factor of ≈ 1.19. The ER gives the cumulative man-rem doses to the population as negligible, with the exception of the integrated man-rem thyroid dose, which is estimated to be 0.08 man-rem for the population at 50-miles. Increasing this dose by a factor of ≈ 1.19, and accounting for the 1990 population, the dose increases to 0.14 man-rem to the thyroid. This resulting dose remains negligible when compared to the 1990 adjusted cumulative annual man-rem, out to 50-miles, resulting from naturally occurring and manmade background radiation (49,726 man-rem and 35,457 man-rem, respectively). Thus, the ER's conclusion that the accident does not have a significant impact upon the environment remains valid when considering the effects of extended power uprate.

The AEC evaluated both small and large MSLBs and calculated the estimated fraction of Part 20 limits at the site boundary to be 0.001 for the small-break case and 0.007 for the large-break case. Increasing each case by a factor of ≈ 1.19 to account for increased core inventory and pressure, the small-break fraction is 0.0012 and the large-break fraction is 0.008. Both cases remain very small fractions of the Part 20 limits.

Evaluating the effects of increased core inventory and pressure on the estimated dose to the population within a 50-mile radius, accounting for the change in population, the dose for the small MSLB increases to 0.16 man-rem (previously estimated at 0.1), and the large-break MSLB dose increases to 0.27 man-rem (previously estimated at 0.17 man-rem). Both of these estimated doses are much smaller than the estimated dose to the same population due to annual background radiation; thus, the AEC's conclusion that the environmental risks are exceedingly small remains valid.

C. Control Rod Drop Accident (CRDA)

The postulated accident is a reactivity excursion caused by the accidental removal of a control rod from the core at a rate more rapid than can be achieved using the control rod drive (CRD) mechanism. In a postulated CRDA, a fully inserted control rod is assumed to fall out of the core after becoming disconnected from its drive after the drive has been removed to the fully withdrawn position. The CRDA is assumed to result in the perforation of < 10 rods (based upon a 7x7 assembly), but with a high probability that none will actually fail.

Although the plant no longer uses 7x7 fuel, it is assumed for the purposes of this discussion (i.e., to highlight the changes in radiological effects on the environment) that the % of failed fuel resulting from a CRDA does not change with the use of different fuel. Based on the assumption the fraction of the core released by a CRDA does not change, and the power density of the fuel is decreased with the use of different fuel, a change in fuel type does not affect the previously calculated doses. Extended power uprate, does affect the previously calculated doses because of increased core inventory.

The inventory of the failed rods is assumed to be increased by a factor of 1.13 to account for the increased core inventory resulting from extended power uprate. The ER concludes that the doses due to this event are negligible (i.e., < 0.01 man-rem). Using the factor of 1.13 and accounting for the 1990 population, the dose resulting from extended power uprate is < 0.016 man-rem for the population within a 50-mile radius. This dose remains significantly lower than the annual cumulative man-rem for the same population due to naturally occurring and manmade background radiation.

Applying the 1.13 multiplier to the AEC's estimated fraction of the Part 20 limits at the site boundary to account for the core inventory increase, the fraction increases from 0.002 to 0.0023. This fraction remains a very small fraction of the limits. The AEC's estimated dose to the population within a 50-mile radius increases from 0.12 man-rem to 0.18 man-rem when adjusted to reflect the core inventory increase and the 1990 population. Again, this dose is very small when compared to the annual cumulative dose to the same population from background radiation (35,500 man-rem/year).

D. Liquid Radwaste Tank Accident

This accident is postulated as a failure of a high-level radwaste tank containing its Technical Specifications storage limit. As the high-level radwaste tanks are located below grade in the radwaste building, spills will be contained and no dose consequences to the public will result. Extended power uprate does not affect these conclusions, since it does not result in changes to either the location of high-level radwaste tanks or to the containment of the tank contents inside the building.

E. Offgas System Accident

The ER treats the offgas system accident as a Class 8 accident, whereas the FES treats waste gas storage releases as Class 3 accidents. The difference in categorizing this event is based primarily upon the frequency of occurrence, which is not affected by extended power uprate. The assumptions for the source term are based upon the same source as is currently used in the Unit 2 FSAR chapter 15 analysis (Unit 1 chapter 9 analysis) of the offgas system accident. Extended power uprate does not affect the source assumed, since uprate will not affect Technical Specifications limits relative to offgas reactor coolant activity. Thus, the conclusions in the ER and the FES remain valid.

7.4.9 Class 9 - Hypothetical Sequences of Failures More Severe Than Class 8

The ER identifies no Class 9 events for Plant Hatch. The FES concluded the postulated occurrences in Class 9 involve sequences of successive failures more severe than those required to be considered for the design basis of protection systems and engineered safety features. The AEC concluded, although the consequences could be severe, the probability of occurrence of Class 9 accidents is so small that the environmental risk is extremely low. The following factors provide and maintain the required high degree of assurance potential accidents in this class are, and will remain, sufficiently small in probability such that the environmental risk is extremely low:

- Defense-in-depth (multiple physical barriers).
- Quality assurance for design, manufacture, and operation.
- Conservative design.
- Continued surveillance and testing.

The factors remain inherent in the Plant Hatch design considering extended power uprate, and the conclusion that the environmental risk due to these types of accidents is extremely low remains valid (Refs. 1 and 2).

7.4.10 Radwaste Transportation Accidents

A. Solid Waste

The ER concludes the only exposures received in a transportation accident involving solid radwaste (e.g., dry, active waste and resin) are to individuals involved in the necessary cleanup. The effect to the population was judged to be insignificant. Extended power uprate does not change this conclusion.

B. Spent Fuel

The ER concludes the principle environmental effect from a transportation accident involving spent fuel is whole-body exposure due to increased radiation levels from the release of noble gases. Considering the dose attenuation effects with distance, the direct radiation dose effects to the general population will be orders of magnitude below normal background. Extended power uprate does not change this conclusion.

8.0 ENVIRONMENTAL EFFECTS OF URANIUM FUEL CYCLE ACTIVITIES

Extended power uprate at Plant Hatch is expected to result in an increase in the bundle average enrichment of the fuel. The environmental impact of a fuel cycle and fuel transportation is described in Tables S-3 and S-4 in 10 CFR 51.51 and 51.52, respectively. An additional NRC assessment evaluated the applicability of the tables to higher burnup fuel cycles and concluded there is no significant environmental impact for fuel cycles with uranium enrichments up to 5 weight % U-235 and discharge exposures up to 60 Gwd/MTU (Ref. 5). The fuel enrichment will not exceed 5 weight % U-235, nor will the rod average discharge exposure exceed 60 Gwd/MTU. Therefore, the environmental impact of the fuel cycle with extended power uprate is conservatively described in Tables S-3 and S-4.

REFERENCES

1. Final Environmental Statement for the Edwin I. Hatch Nuclear Plant, Unit 1 and Unit 2, Docket Nos. 50-321 and 50-366, October 1972.

2. "Final Environmental Statement Related to Operation of Edwin I. Hatch Nuclear Plant Unit No. 2," NUREG-0417, Docket No. 50-366, March 1978.

3. "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," NUREG-1437, Final Report, Volume 1, May 1996.

4. "Proceedings of the 1997 International Topical Meeting on LWR Fuel Performance," American Nuclear Society, March 1997.

5. Federal Register, Volume 53, Number 39, pages 640-643, dated February 28, 1988.

TABLE E3-1 (SHEET 1 OF 2)

UNIT 1 AND UNIT 2 RADIOLOGICAL EFFLUENT COMFARISON

	Radioactive Effluent		
	Noble Gases	Iodines and Particulates	Liquids
10 CFR 50 Appendix I Criteria	10 mRads/yr/unit - {γ} 20 mRads/yr/unit - {β} 5 mR/yr/unit - Total Body 15 mR/yr/unit - Skin	15 mR/yr/unit - Any Organ	3 mR/yr/unit - Total Body 10 mR/yr/unit - Any Organ
Plant Hatch Unit 1 FES (10/72) [Ref. 1]	No Data- {γ} No Data- {β} 1.2 mR/yr/unit - Total Body No Data - Skin	16.5 mR/yr/unit - Adult Thyroid	0.3 mR/yr/unit - Total Body 0.3 mR/yr/unit - Individual Thyroid
Plant Hatch Unit 2 FES (4/78) [Ref. 2] Note: Doses for both units would be twice these values since both units are identical.	0.5 mRad/yr/unit - {γ} 1.5 mRad/yr/unit - {β} 0.9 mR/yr/unit - Total Body 1.8 mR/yr/unit - Skin	3.1 mR/yr/unit - Infant Thyroid	1.1 mR/yr/unit - Total Body 2.0 mR/yr/unit - Individual Bone
Plant Hatch Unit 1 SER, dated 5/11/73	~ 1.0 mR/yr/unit with no specific reference to 10 CFR 50, Appendix I criteria.	~ 4.0 mR/yr/unit with no specific reference to 10 CFR 50, Appendix I criteria.	No specific dose reference to 10 CFR 50, Appendix I criteria
Plant Hatch Unit 2 SER, NUREG-0411, dated 6/13/78	References Plant Hatch Unit 2 FES Results	References Plant Hatch Unit 2 FES Results	References Plant Hatch Unit 2 FES Results
Plant Hatch Units 1&2 FSARs (Includes Original Power Uprate)	6.4 mRads/yr/unit-{γ} 7.1 mRads/yr/unit-{β} 4.2 mR/yr/unit - Total Body 11.4 mR/yr/unit - Skin	4.1 mR/yr/unit - Child's Thyroid	1.3 mR/yr/unit - Total Body 1.9 mR/yr/unit - Teenager's Liver
Extended Power Uprate Dose Impact	6.4 mRads/yr/unit- {γ} 7.1 mRads/yr/unit- {β} 4.3 mR/yr/unit - Total Body 11.4 mR/yr/unit - Skin	4.4 mR/yr/unit - Child's Thyroid	1.3 mR/yr/unit - Total Body 1.9 mR/yr/unit - Teenager's Liver
Extended Power Uprate Dose Impact as % of Appendix I	64.0 % - {γ} 35.5 % - {β} 86.0 % - Total Body 76.0 % - Skin	29.3 % - Any Organ (Thyroid)	43.3 % - Total Body 19.0 % - Any Organ (Liver)
Extended Power Uprate Dose Impact as % Change to FSARs (Original Power Uprate) (Note 1)	0% - {γ} 0% - {β} + 2.4% - Total Body 0% - Skin	+ 7.3% - Child's Thyroid	0% - Total Body 0% - Teenager's Liver

TABLE E3-1 (SHEET 2 OF 2)

UNIT 1 AND UNIT 2 RADIOLOGICAL EFFLUENT COMPARISON

	Radioactive Effluent		
	Noble Gases	Iodines and Particulates	Liquids
Plant Hatch Radioactive Effluent Release Report (1/96-12/96) (Note 2)	0.18 mRads/yr/unit- {γ}(Note 3) 0.19 mRads/yr/unit- {β}(Note 3) N/A - Total Body (Note 4) N/A - Skin (Note 4)	0 .94 mR/yr/unit - Thyroid (Note 5)	0.08 mR/yr/unit - Total Body (Note 6) 0.12 mR/yr/unit - Liver (Note 6)
Plant Hatch Radioactive Effluent Release Report (1/96-12/96) as % of Appendix I	1.8 % - {γ} 0.95 % - {β} N/A - Total Body (Note 4) N/A - Skin (Note 4)	6.3 % - Any Organ (Thyroid)	2.6 % - Total Body 1.2 % - Any Organ (Liver)
Plant Hatch Radioactive Effluent Release Report (1/96-12/96) as % Change to FSARs (Original Power Uprate Values) (Note 7)	-97.2% - {γ} -97.3% - {β} N/A - Total Body (Note 4) N/A - Skin (Note 4)	-77.1% - Thyroid	-93.8% - Total Body -93.7% - Liver

NOTES:

(1) For example, FSARs thyroid dose = 4.1 mR/yr/unit and extended power uprate thyroid dose = 4.4 mR/yr/unit. % change to FSARs thyroid dose = ((4.4 - 4.1) /4.1) x 100 = + 7.3%; or a 7.3% increase in the estimated thyroid dose.

(2) These data were derived from information presented in the Plant Hatch Annual Radioactive Effluent Release Report which contains actual liquid and gaseous effluent release dose data for Plant Hatch Units 1 & 2. Plant Hatch Unit 2 has operated at uprate conditions for 1 yr and Plant Hatch Unit 1 has operated at uprate conditions for 6 months. To envelope both units for an assumed 1 yr operation at uprated conditions, the highest (worst-case) cumulative dose values were doubled to represent the total annual release for each unit.

(3) Based upon 9.04E-02 mRads/yr/Unit 1 - {γ}, and 9.43E-02 mRads/yr/Unit 1 - {β}.

(4) Noble gas dose data were not available in the 1/96 - 12/96 Annual Report.

(5) Based upon 4.70E-01 mR/yr/Unit 2 - thyroid.

(6) Based upon 3.96E-02 mR/yr/Unit 1 - total body, and 5.92E-02 mR/yr/Unit 1 - liver.

(7) For example, FSARs thyroid dose = 4.1 mR/yr/unit and Plant Hatch Annual Report thyroid dose = 0.94 mR/yr/unit.

% change to FSARs thyroid dose = (0.94 - 4.1)/4.1) x 100 = -77.1%; or a 77.1% decrease in the estimated thyroid dose.

TABLE E3-2 (SHEET 1 OF 2)

SUMMARY OF RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS DETERMINED BY AEC (ADJUSTED FOR EXTENDED POWER UPRATE)

Class	Event	Estimated Fraction of 10 CFR Part 20 Limit at Site Boundary (1)	Estimated Dose to 1990 Population in 50-Mile Radius (man-rem)
1.0	Trivial incidents	(2)	(2)
2.0	Small releases outside containment	(2)	(2)
3.0	Radwaste system failures		
3.1	Equipment leakage or malfunction	0.0418	1.46
3.2	Release of waste gas storage tank contents	0.1695	5.79
3.3	Release of liquid waste storage tank contents	0.0011	0.15
4.0	Fission products to primary system (BWR)		
4.1	Fuel cladding defects	(2)	(2)
4.2	Off-design transients that induce fuel failures above those expected	0.0023	0.15
5.0	Fission products to primary and secondary systems (PWR)	N/A	N/A
6.0	Refueling accidents		
6.1	Fuel bundle drop	0.0011	0.15
6.2	Heavy object drop onto fuel in core	0.0011	0.15
7.0	Spent fuel handling accident		

TABLE E3-2 (SHEET 2 OF 2)

SUMMARY OF RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS DETERMINED BY AEC (ADJUSTED FOR EXTENDED POWER UPRATE)

Class	Event	Estimated Fraction of 10 CFR Part 20 Limit at Site Boundary (1)	Estimated Dose to 1990 Population in 50-Mile Radius (man-rem)
7.1	Fuel assembly drop in fuel storage pool	0.0011	0.15
7.3	Fuel cask drop	0.0622	2.08
8.0	Accident initiation events considered in design basis evaluation in the safety analysis report		
8.1(a)	Loss-of-coolant accident		
	Small break	0.0011	0.15
	Large break	0.0011	1.78
8.1(b)	Break in instrument line from primary system that penetrates the containment	0.0011	0.15
8.2(a)	Rod ejection accident (PWR)	N/A	N/A
8.2(b)	Rod drop accident (BWR)	0.0023	0.18
8.3(a)	Steamline breaks (PWRs outside containment)	N/A	N/A
8.3(b)	Steamline breaks (BWR)		
	Small break	0.0012	0.16
	Large break	0.0083	0.26

1. Represents the calculated fraction of a whole body dose body of 500 mrem or the equivalent dose to an organ.

2. These releases will be comparable to the design objectives indicated in the proposed Appendix I to 10 CFR 50 for routine effluents (i.e., 5 mrem/yr to an individual from either liquid or gaseous effluents).