September 23, 2003

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

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING APPENDIX K MEASUREMENT UNCERTAINTY RECOVERY (TAC NOS. MB7026 AND MB7027)

Dear Mr. Sumner:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 238 to Renewed Facility Operating License DPR-57 and Amendment No. 180 to Renewed Facility Operating License NPF-5 for the Edwin I. Hatch (Hatch) Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 19, 2002, as supplemented by letters dated April 7, May 21, May 30, June 4, September 4, and September 12, 2003.

The amendments revise the Renewed Facility Operating License and the TSs to reflect the increased licensed power level for Hatch, Units 1 and 2 by 1.5 percent from 2763 megawatts thermal (MWt) to 2804 MWt. The change is based on the installation of the Advanced Measurement Analysis Group, Inc./Westinghouse Crossflow ultrasonic flow measurement instrumentation, resulting in improved feedwater flow measurement accuracy. The associated TS Bases will also be revised.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Steve Bloom, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 238 to DPR-57
2. Amendment No. 180 to NPF-5
3. Safety Evaluation

cc w/encls: See next page

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The amendments revise the Renewed Facility Operating License and the TSs to reflect the increased licensed power level for Hatch, Units 1 and 2 by 1.5 percent from 2763 megawatts thermal (MWt) to 2804 MWt. The change is based on the installation of the Advanced Measurement Analysis Group, Inc./Westinghouse Crossflow ultrasonic flow measurement instrumentation, resulting in improved feedwater flow measurement accuracy. The associated TS Bases will also be revised.

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1. Amendment No. 238 to DPR-57
2. Amendment No. 180 to NPF-5
3. Safety Evaluation

cc w/encls: See next page

<u>DISTRIBUTION</u>:

PUBLIC	OGC
PDII-1 R/F	ACRS
BBonser, RII	GHill(4)
WBeckner	

ML032590944

*SE input provided - no major changes made. ** See previous concurrence.

OFFICE	PDI-2/PM	PDI-2/LA	EMCB/SC*	EMCB/SC*	EMCB/SC*	EMEB/SC*	SRXB/SC*	SPSB/SC*
NAME	SBloom	CHawes	LLund	SCoffin	TChan	KManoly	FAkstulewicz	MCaruso
DATE	9/16/03	9/16/03	4/23/03	5/14/03	6/9/03	5/13/03	4/14/03	2/28/03

OFFICE	EEIB/SC*	EEIB/SC*	IOLB/SC**	SPLB/SC**	SPLB/SC**	OGC**	PDII-1/SC	PDII/PD	DLPM/AD
NAME	CHolden	EMarinos	DTrimble	SWeerakkody	DSolorio	RWeisman	JNakoski	EHackett	TMarsh
DATE	5/13/03	6/13/03	6/19/03	6 /19/03	6 /19/03	6 /20/03	9 /17/03	9/22/03	9/22/03

OFFICIAL RECORD COPY

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 238
Renewed License No. DPR-57

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility) Renewed Facility Operating License No. DPR-57 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated December 19, 2002, as supplemented by letters dated April 7, May 21, May 30, June 4, September 4, and September 12, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-57 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 238, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Before operating at the proposed power of 2804 megawatts thermal, the licensee shall:

   As described in the December 19, 2002, application, as supplemented by letters dated April 7, May 21, May 30, June 4, September 4, and September 12, 2003, and the September 23, 2003, NRC staff Safety Evaluation (SE): rescale the core power from the average power range monitors (APRMs) to the proposed power uprate level prior to exceeding the current licensed power level and make any necessary adjustments to the APRM alarm and trip settings. During power ascension at each steady-state heat balance power level (95 percent and 100 percent of the current licensed power level and 100 percent of the proposed power uprate level), the licensee will demonstrate an acceptable fuel thermal margin. Fuel thermal margin will be projected to the proposed rated thermal power point after the measurement at 95 percent and 100 percent of current licensed power level are taken to show the estimated margin. The demonstration of core and fuel conditions will be performed using current Hatch methods. In preparation for operation at the proposed power uprate level, routine measurements of reactor and system pressures, flows, and selected major rotating equipment vibration will be taken near 95 percent and 100 percent of the current licensed power level and at 100 percent of the proposed power uprate level. The operational aspect of the power uprate will be demonstrated by performing turbine pressure regulator controller and feedwater controller testing during power ascension testing. Reactor pressure control system testing, consistent with the guidelines of NEDC-33085P, "Safety Analysis Report for Edwin I. Hatch Units 1 and 2 THERMAL POWER OPTIMIZATION," dated December 2002, will be performed during power ascension testing. During these tests, a water level change of $\pm$ 3 inches and pressure setpoint changes of $\pm$ 3 psi will be used. If necessary, the controllers and actuator elements will be adjusted. The performance of the feedwater level control system will be recorded at 95 percent and 100 percent of the current licensed power level and confirmed at the proposed power uprate level during power ascension. The turbine pressure controller setpoint will be readjusted at 95 percent current licensed power level and held constant. Adjusting the pressure setpoint prior to recording the baseline power ascension data establishes a consistent basis for measuring the performance of the reactor and the turbine control valves.

4. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance. Prior to implementation of the license amendment, the licensee shall:

   A. Install the Advanced Measurement Analysis Group, Inc./Westinghouse Crossflow ultrasonic flow measurement (UFM) system in accordance with the provisions of the manufacturer's topical report CENPD-397-P-A, Rev. 1, "Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology," dated May 2000, Section 8, Crossflow Field Implementation, as described in the December 19, 2002, application, and the September 23, 2003, NRC staff SE. The assumed Crossflow system measurement uncertainty validation will be performed as described in the December 19, 2002, application, and the September 23, 2003, NRC staff SE.

   B. Revise the necessary maintenance and operational procedures, which will include the actions specified during UFM outages, including consideration of the uncertainties in the accuracy of the core thermal power and feedwater flow measurements using other instrumentation, and the allowed outage time for the UFM, including actions to reduce power, as described in the December 19, 2002, application, as supplemented by letters dated April 7, May 21, May 30, and June 4, 2003, and the September 23, 2003, NRC staff SE. These procedures shall be reflected in the update of the Final Safety Analysis Report following implementation of the amendment submitted in accordance with 10 CFR 50.71(e).

   C. Make and validate simulator changes for the power uprate in accordance with American National Standards Institute/American Nuclear Society 3.5-1985, "Nuclear Power Plant Simulators for Use in Operator Training." Minor changes (e.g. power/flow map and flow-reference setpoint changes) will be communicated through normal operator training.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by E. Leeds for/

Ledyard B. Marsh, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and
Technical Specifications

Date of Issuance: September 23, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 238

RENEWED FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following page of the Renewed Facility Operating License No. DPR-57 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
4	4

Replace the following pages of the Appendix A Technical Specifications and associated Bases with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
1.1-4	1.1-4
2.0-1	2.0-1
3.2-1	3.2-1
3.2-2	3.2-2
3.3-3	3.3-3
3.3-5	3.3-5
3.3-7	3.3-7
3.3-9	3.3-9
3.3-20	3.3-20
3.3-27	3.3-27
3.3-28	3.3-28
3.7-18	3.3-17
B 2.0-2	B 2.0-2
B 3.2-1	B 3.2-1
B 3.2-3	B 3.2-3
B 3.2-4	B 3.2-4
B 3.2-6	B 3.2-6
B 3.2-7	B 3.2-7
B 3.3-3	B 3.3-3
B 3.3-7	B 3.3-7
B 3.3-16	B 3.3-16
B 3.3-17	B 3.3-17
B 3.3-24	B 3.3-24
B 3.3-28	B 3.3-28
B 3.3-53	B 3.3-53
B 3.3-54	B 3.3-54
B 3.3-56	B 3.3-56
B 3.3-76	B 3.3-76
B 3.3-77	B 3.3-77

Remove	Insert
B 3.3-78	B 3.3-78
B 3.3-80	B 3.3-80
B 3.3-81	B 3.3-81
B 3.3-141	B 3.3-141
B 3.4-10	B 3.4-10
B 3.7-31	B 3.7-31
B 3.7-35	B 3.7-35
B 3.7-36	B 3.7-36

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 180
Renewed License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Renewed Facility Operating License No. NPF-5 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated December 19, 2002, as supplemented by letters dated April 7, May 21, May 30, June 4, September 4, and September 12, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-57 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 180, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Before operating at the proposed uprate of 2804 megawatts thermal, the licensee shall:

   As described in the December 19, 2002, application, as supplemented by letters dated April 7, May 21, May 30, June 4, September 4, and September 12, 2003, and the September 23, 2003, NRC staff Safety Evaluation (SE): rescale the core power from the average power range monitors (APRMs) to the proposed power uprate level prior to exceeding the current licensed power level and make any necessary adjustments to the APRM alarm and trip settings. During power ascension at each steady-state heat balance power level (95 percent and 100 percent of the current licensed power level and 100 percent of the proposed power uprate level), the licensee will demonstrate an acceptable fuel thermal margin. Fuel thermal margin will be projected to the proposed rated thermal power point after the measurement at 95 percent and 100 percent of current licensed power level are taken to show the estimated margin. The demonstration of core and fuel conditions will be performed using current Hatch methods. In preparation for operation at the proposed power uprate level, routine measurements of reactor and system pressures, flows, and selected major rotating equipment vibration will be taken near 95 percent and 100 percent of the current licensed power level and at 100 percent of the proposed power uprate level. The operational aspect of the power uprate will be demonstrated by performing turbine pressure regulator controller and feedwater controller testing during power ascension testing. Reactor pressure control system testing, consistent with the guidelines of NEDC-33085P, “Safety Analysis Report for Edwin I. Hatch Units 1 and 2 THERMAL POWER OPTIMIZATION,” dated December 2002, will be performed during power ascension testing. During these tests, a water level change of $\pm$ 3 inches and pressure setpoint changes of $\pm$ 3 psi will be used. If necessary, the controllers and actuator elements will be adjusted. The performance of the feedwater level control system will be recorded at 95 percent and 100 percent of the current licensed power level and confirmed at the proposed power uprate level during power ascension. The turbine pressure controller setpoint will be readjusted at 95 percent current licensed power level and held constant. Adjusting the pressure setpoint prior to recording the baseline power ascension data establishes a consistent basis for measuring the performance of the reactor and the turbine control valves.

4. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance. Prior to implementation of the license amendment, the licensee shall:

   A. Install the Advanced Measurement Analysis Group, Inc./Westinghouse Crossflow ultrasonic flow measurement (UFM) system in accordance with the provisions of the manufacturer's topical report CENPD-397-P-A, Rev. 1, "Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology," dated May 2000, Section 8, Crossflow Field Implementation, as described in the December 19, 2002, application, and the September 23, 2003, NRC staff SE. The assumed Crossflow system measurement uncertainty validation will be performed as described in the December 19, 2002, application, and the September 23, 2003, NRC staff SE.

   B. Revise the necessary maintenance and operational procedures, which will include the actions specified during UFM outages, including consideration of the uncertainties in the accuracy of the core thermal power and feedwater flow measurements using other instrumentation, and the allowed outage time for the UFM, including actions to reduce power, as described in the December 19, 2002, application, as supplemented by letters dated April 7, May 21, May 30, June 4, September 4, and September 12, 2003, and the September 23, 2003, NRC staff SE. These procedures shall be reflected in the update of the Final Safety Analysis Report following implementation of the amendment submitted in accordance with 10 CFR 50.71(e).

   C. Make and validate simulator changes for the power uprate in accordance with American National Standards Institute/American Nuclear Society 3.5-1985, "Nuclear Power Plant Simulators for Use in Operator Training." Minor changes (e.g. power/flow map and flow-reference setpoint changes) will be communicated through normal operator training.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by E.Leeds for/

Ledyard B. Marsh, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and
Technical Specifications

Date of Issuance: September 23, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 180

RENEWED FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following page of the Renewed Facility Operating License No. NPF-5 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
4	4

Replace the following pages of the Appendix A Technical Specifications and associated Bases with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
1.1-5	1.1-5
2.0-1	2.0-1
3.2-1	3.2-1
3.2-2	3.2-2
3.3-3	3.3-3
3.3-5	3.3-5
3.3-7	3.3-7
3.3-9	3.3-9
3.3-20	3.3-20
3.3-27	3.3-27
3.3-28	3.3-28
3.7-18	3.3-17
B 2.0-2	B 2.0-2
B 3.2-1	B 3.2-1
B 3.2-3	B 3.2-3
B 3.2-4	B 3.2-4
B 3.2-6	B 3.2-6
B 3.2-7	B 3.2-7
B 3.3-3	B 3.3-3
B 3.3-7	B 3.3-7
B 3.3-16	B 3.3-16
B 3.3-17	B 3.3-17
B 3.3-24	B 3.3-24
B 3.3-28	B 3.3-28
B 3.3-53	B 3.3-53
B 3.3-54	B 3.3-54
B 3.3-56	B 3.3-56
B 3.3-76	B 3.3-76
B 3.3-77	B 3.3-77

Remove	Insert
B 3.3-78	B 3.3-78
B 3.3-80	B 3.3-80
B 3.3-81	B 3.3-81
B 3.3-141	B 3.3-141
B 3.4-10	B 3.4-10
B 3.7-31	B 3.7-31
B 3.7-35	B 3.7-35
B 3.7-36	B 3.7-36

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 238 TO RENEWED FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 180 TO RENEWED FACILITY OPERATING LICENSE NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By application dated December 19, 2002, as supplemented by letters dated April 7, May 21, May 30, June 4, September 4, and September 12, 2003, Southern Nuclear Operating Company, Inc. (SNC or the licensee), requested changes to the Edwin I. Hatch Nuclear Plant, Units 1 and 2 (HNP or Hatch), Renewed Facility Operating License (RFOL) DPR-57 and RFOL NFP-5, and Technical Specifications (TSs). The proposed changes would increase the licensed power level by 1.5 percent from 2763 megawatts thermal (MWt) to 2804 MWt. The proposed changes are based on installation of the Advanced Measurement Analysis Group, Inc. (AMAG)/Westinghouse Crossflow ultrasonic flow measurement (UFM) instrumentation, resulting in improved feedwater (FW) flow measurement accuracy and an AMAG/Westinghouse CORRTEMP ultrasonic temperature measurement (UTM) system to improve the accuracy of the feedwater temperature measurement.

Specifically, the proposed changes are:

1. Paragraph 2.C(1), in RFOL DPR-57 (Unit 1) and RFOL NPF-5 (Unit 2) would be revised to authorize operation at a steady state reactor core power level not in excess of 2804 MWt (100 percent).

2. The definition of RATED THERMAL POWER (RTP) in TS 1.1 would be revised to reflect the increase from 2763 MWt to 2804 MWt.

3. The THERMAL POWER Safety Limit value in TS 2.1.1.1 would be revised to less than or equal to “24%” RTP, with the reactor steam dome pressure < 785 psig or core flow < 10 percent rate core flow.

Enclosure 3

4. The THERMAL POWER value in limiting condition for operation (LCO) 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)" Applicability would be revised to greater than or equal to 24%. The THERMAL POWER value in Required Action B.1 would be revised to less than 24%. The THERMAL POWER value in surveillance requirement (SR) 3.2.1.1 Frequency would be revised to greater than or equal to 24%.

5. The THERMAL POWER value in LCO 3.2.2, "Minimum Critical Power Ratio (MCPR)" Applicability would be revised to greater than or equal to 24%. The THERMAL POWER value in Required Action B.1 would be revised to less than 24%. The THERMAL POWER value in SR 3.2.2.1 Frequency would be revised to greater than or equal to 24%.

6. The THERMAL POWER value in TS 3.3.1.1, "RPS Instrumentation," Required Action E.1 and in SR 3.3.1.1.11 would be revised to less than 27.6%. This change is also included in the Applicable Modes or Other Specified Conditions for Functions 8 and 9 on Table 3.3.1.1-1, "Reactor Protection System Instrumentation." The THERMAL Power value in SR 3.3.1.1.2 and NOTE would be revised to greater than or equal to 24%. The Allowable Value in Table 3.3.1.1-1, "Reactor Protection System Instrumentation" for Function 2.b., "Simulated Thermal Power - High," would be revised to read $\leq$0.57W + 56.8%RTP and 115.5% RTP and the footnote (b) would be revised to read 0.57W +56.8% - 0.57ΔW RTP.

7. The THERMAL POWER value in LCO 3.3.2.2, "Feedwater and Main Turbine Trip High Water Instrumentation" Applicability would be revised to greater than or equal to 24%. The THERMAL POWER value in Required Action C.1 would be revised to less than 24%.

8. The THERMAL POWER value in LCO 3.3.4.1, "End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation" Applicability would be revised to greater than or equal to 27.6%. The THERMAL POWER value in Required Action C.2 would be revised to less than 27.6%. The THERMAL POWER value in SR 3.3.4.1.2 would be revised to greater than or equal to 27.6%.

9. The THERMAL POWER value in LCO 3.7.7, "Main Turbine Bypass System" Applicability would be revised to greater than or equal to 24%. The THERMAL POWER value in Required Action B.1 would be revised to less than 24%.

10. The associated Bases for the above TS would be revised to support the proposed change in steady state power level.

The April 7, May 21, May 30, June 4, September 4, and September 12, 2003, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 18, 2003 (68 FR 7821).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix K, “ECCS Evaluation Models,” as originally issued, required licensees to base their transient and accident analyses on an assumed power level of at least 102 percent of the licensed thermal power level. The original uncertainty assumption was mandated to account for uncertainties in determining thermal power. Specifically, the 2-percent margin was intended to address uncertainties related to heat sources in addition to instrument measurement uncertainties.

In its application, SNC requested approval to increase the HNP, Units 1 and 2 licensed thermal power level based on the installation of the AMAG/Westinghouse Crossflow UFM instrumentation for FW flow measurement. Use of the Crossflow UFM provides more accurate measurements of the FW flow due to a reduced core-thermal-power uncertainty. The Crossflow UFM system was previously reviewed and approved by the NRC staff in the NRC’s Safety Evaluation Report (SER) for ABB Combustion Engineering Nuclear Power Topical Report CENP-397-P, Rev. 1, “Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology,” dated March 20, 2000. The NRC staff concluded in its review of the generic topical report that the reduction in power measurement uncertainty does not constitute a significant change to the ECCS evaluation model as defined in 10 CFR 50.46(a)(3)(i). The Hatch specific topical report, NEDC-33085P, “Safety Analysis Report For Edwin I. Hatch Units 1 and 2 Thermal Power Optimization,” follows the format and content for Boiling Water Reactor (BWR) Thermal Power Optimization (TPO) licensing reports documented in NEDC-32938P, “Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization,” called “TLTR.”

Regulatory Background

Nuclear power plants are licensed to operate at a specified power, which at operating power levels, is indicated in the control room by neutron flux instrumentation that has been calibrated to correspond to core thermal power (CTP). CTP is determined by a calculation of the energy balance of the plant nuclear steam supply system (NSSS). The accuracy of this calculation depends primarily upon the accuracy of feedwater flow, feedwater enthalpy, and main steam enthalpy measurements. Thus, an accurate measurement of feedwater flow and temperature is necessary for calibrating nuclear instrumentation to represent CTP.

Uncertainty in the calculation of CTP can increase the probability of exceeding the power levels assumed in design-basis transient and accident analyses. In this regard, to allow for uncertainties when determining thermal power (e.g., instrument measurement uncertainties), Appendix K to 10 CFR Part 50 originally required loss-of-coolant accident (LOCA) and emergency core cooling system (ECCS) analyses to assume that the reactor had operated continuously at a power level at least 102 percent of the licensed thermal power. Initially, the 2-percent power margin uncertainty value was intended to address uncertainties related to heat sources in addition to instrument measurement uncertainties.

On June 1, 2000, the NRC issued a revision to 10 CFR Part 50, Appendix K, (65 FR 34913). The NRC staff concluded, at the time of the original ECCS rulemaking, that the 2-percent power margin requirement was based solely on considerations associated with power measurement uncertainty, as is reflected in Appendix K. The original regulation did not require licensees to demonstrate the power measurement uncertainty, but rather mandated a 2-percent margin, notwithstanding that the instruments used to calibrate neutron flux instrumentation may be more accurate than originally assumed in the ECCS rulemaking. The revised rule allows licensees to justify a smaller margin for power measurement uncertainty and to use an assumed power level less than 102 percent of the licensed power level, provided the new power level is demonstrated to account for uncertainties due to power level instrument error.

The amended rule provided licensees the opportunity to use a reduced margin if they determine that there was a sufficient benefit. Licensees may apply the margin to gain benefits from operation at higher power, or the margin could be used to relax ECCS-related technical specifications (e.g., pump flows). Another potential benefit could be in modifying fuel management strategies (e.g., possibly by altering core power peaking factors). However, the amended rule, by itself, does not allow increases in licensed power levels. Because a plant licensed RTP is a TS limit, proposals to raise the licensed RTP must be reviewed and approved under the license amendment process.

In addition to the regulatory requirement of Appendix K to 10 CFR Part 50, the NRC issued Regulatory Issue Summary 2002-03 (RIS 2002-03), "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," to provide guidance to licensees on the scope and detail of the information that should be provided to the NRC for reviewing measurement uncertainty recapture (MUR) power uprate applications to facilitate NRC staff review. This guidance was predicated on the experience gained from NRC staff reviews of MUR power uprate applications. RIS 2002-03 provided the following general guidance for staff reviews of MUR power uprate applications:

- In areas (e.g., accident/transient analyses, components, systems) for which the existing analyses of record do not bound the plant operation at the proposed uprated power level, the NRC staff will conduct a detailed review.

- In areas (e.g., accident/transient analyses, components, systems) for which the existing analyses of record do bound plant operation at the proposed uprated power level, the NRC staff will not conduct a detailed review.

- In areas that are amenable to generic disposition, the NRC staff will utilize such dispositions.

The plant-specific basis for the proposed MUR power uprate is provided in the following documents:

- GE Nuclear Energy Topical Report NEDC-33085P, "Safety Analysis Report For Edwin I. Hatch Units 1 and 2 Thermal Power Optimization,"

- GE Licensing Topical Report NEDC-32938P-A, "Licensing Topical Report, Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," dated July 2000, which was approved by the NRC staff by letter dated June 20, 2001,

- GE Licensing Topical Report NEDC-32424P-A, "Licensing Topical Report, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR1) which was approved by the NRC staff in February 1999,

- ISA-RP67.04, Part II-1994, “Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation” that was endorsed by the NRC staff in Regulatory Guide (RG) 1.105, Rev. 3, “Setpoints for Safety-Related Instrumentation,” dated December 1999,

- The licensee’s December 19, 2002, license amendment request, with enclosures,

- Combustion Engineering Nuclear Power, LLC (CENPD)-397-P-A, Rev. 01, “Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow measurement Technology,” dated May 2000, which provides information on the Crossflow UFM system design, its underlying principles of ultrasonic measurement, experimental data validating the system accuracy, and an overview of the Crossflow system installation, and

- Westinghouse LTR-NRC-02-3, “Description of Crossflow^{XT} Ultrasonic Flow Measurement System,” dated January 18, 2002, which described a modification to the Crossflow system that added four additional UFM transducers in the mounting and transducer support frame (M/TSF).

Enclosure 7, “Plant Modifications” of the licensee’s submittal dated December 19, 2002, provides information validating the feedwater venturi-based system accuracy.

The basis of the licensee’s proposed power uprate is the reduction of the measurement uncertainty of CTP by using the Crossflow UFM system to measure feedwater flow and the CORRTEMP UTM system to improve the measurement of feedwater temperature at Hatch, Units 1 and 2. The licensee referenced Combustion Engineering Topical Report CENPD-397-P-A, Rev. 01, “Improved Measurement Accuracy Using Crossflow UltraSonic Flow Measurement Technology,” that was approved by the NRC staff on March 20, 2000, for referencing in license applications for power uprates. The licensee also addressed the regulatory guidance items in RIS 2002-03.

Technical Background

The typical elements used for measuring feedwater flow are an orifice plate, a venturi meter, or a flow nozzle that generate a differential pressure proportional to the feedwater velocity in the pipe. Of the three differential pressure devices, the venturi meter is most widely used for feedwater measurement in nuclear power plants.

The major advantage of a venturi meter is that a relatively low head loss occurs as a fluid, such as, water passes through the device. However, fouling of the device is the major disadvantage of this meter or any other nozzle-based flow meter. Fouling is caused by metallic plating on the throat area of the meter, that causes the meter, over time, to indicate a higher differential pressure and, thus, a higher than actual indicated flow rate. This result leads plant operators to over-estimate plant CTP and, thereby, calibrate nuclear instrumentation high. Calibrating nuclear instrumentation high is conservative with respect to reactor safety, yet it causes the licensee to lower the plant electrical generation proportionally when the plant is operated at its thermal power rating. In addition to fouling, the transmitter and the analog-to-digital converter of the venturi meter introduce errors in the flow measurement thus necessitating removal, cleaning, and recalibration of the flow device.

Because of the desire to reduce flow instrumentation uncertainty and to operate the plant closer to the license rating, the industry assessed alternate flow measurement techniques and found the UFM to be a viable alternative. The typical UFM system consists of a set of electronic transducers that are controlled by computer electronics. The UFM system is not susceptible to fouling because the UFM system does not use a venturi for measuring feedwater flowrate.

A cross-correlation UFM-based flow measurement system was first developed by Canadian General Electric for Ontario Hydro. However, the system was not optimized for application over a wide range of flow velocities and pipe diameters. The task of optimizing the cross-correlation technique was carried out by AMAG, and CENP. This work resulted in an improved cross-correlation UFM system, Crossflow, that has been installed to measure reactor coolant flow and feedwater flow in more than 40 nuclear power plants in the United States, Canada, South America, and Europe. However, until recently licensees have not taken credit for the accuracy of the Crossflow UFM system in regulatory applications.

The NRC staff performed a technical evaluation of the licensee's proposed implementation of a Crossflow UFM system and CORRTEMP UTM system in the Hatch, Units 1 and 2 to improve the accuracy of the heat balance calculation used to calibrate the neutron flux instrumentation in support of the licensee's proposed 1.5-percent MUR power uprate. Additionally, the NRC staff reviewed the licensee's proposed TS changes associated with the proposed 1.5-percent MUR power uprate requested by the licensee.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of the proposed license amendment.

The NRC staff's review of the licensee's application is organized as follows:

- 3.1 Reactor Core and Fuel Performance
- 3.2 Reactor Coolant and Connected Systems
- 3.3 Engineered Safety Features
- 3.4 Instrumentation and Control
- 3.5 Electrical Power and Auxiliary Systems
- 3.6 Power Conversion Systems
- 3.7 Radwaste and Radiation Sources
- 3.8 Reactor Safety Performance Evaluations
- 3.9 Other Evaluations
- 3.10 RFOL and TS Changes

3.1 Reactor Core and Fuel Performance

The core thermal-hydraulic design and fuel performance characteristics are evaluated for each reload fuel cycle. The following sections address the effect of the TPO uprate on fuel design performance, thermal limits, the power/flow map, and stability.

3.1.1 Fuel Design and Operation

Fuel bundles are designed in accordance with the applicable general design criteria (GDC) of 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” to ensure that (1) the fuel bundles are not damaged during normal steady-state operation and anticipated operational occurrences (AOOs), (2) any damage to the fuel bundles will not be so severe as to prevent control rod insertion when required, (3) the number of fuel rod failures during accidents is not underestimated during accidents, and (4) the coolability of the core is always maintained. For each fuel vendor, the NRC-approved fuel design acceptance criteria and analysis methodology assure that the fuel bundles comply with the objectives of Sections 4.2 and 4.3 of the standard review plan and comply with the GDC. The fuel vendors perform thermal-mechanical, thermal-hydraulic, neutronic, and material analyses to ensure that the fuel system design can meet the fuel design limits during steady-state, AOO, and accident conditions.

The TPO uprate is planned to be implemented during the current Cycle 21 for Hatch, Unit 1, which is a mixed core of 560 fuel assemblies consisting of General Electric (GE) 13 and GE 14 fuel bundles. The TPO uprate for Hatch, Unit 2 is planned to be implemented during Cycle 18, which will have a mixed core of 560 fuel assemblies consisting of GE 13 and GE 14 fuel bundles. The Hatch reload analysis is based on the NRC-approved GE methodology described in NEDE-24011-P-A-14, “General Electric Standard Application for Reactor Fuel (GESTAR II),” dated July 2000. The NRC-approved codes and methodologies used for the licensing safety analyses are also referred to in Section 5 of the Hatch TSs. The limiting AOO and accident analyses are reanalyzed for every reload and the safety analyses are documented in Chapter 15 of the updated final safety analysis report (UFSAR). Limiting AOOs and accidents are events that could potentially affect the core operating and safety limits that ensure the safe operation of the plant.

A new mechanical fuel design is not needed to achieve the 1.5-percent power uprate, though new fuel designs may be used in the future to obtain additional operating flexibility or to maintain the fuel cycle length. The current GE 13 and GE 14 fuel types meet NRC-approved acceptance criteria, and any amendments authorizing the use of a new fuel design that does not comply with the NRC-approved fuel design criteria given in GESTAR II will require NRC review.

The slightly higher operating power and the increased steam void content will affect the core and fuel performance. The licensee may also change the power distribution in the reload design to allow more operating flexibility or to maintain the fuel cycle length. This would also affect the core and fuel performance. However, the steady-state and transient design linear heat generation rate limits for each fuel bundle ensure that the fuel plastic strain design limit and the fuel centerline melt limit will not be exceeded. The thermal-hydraulic design and the operating limits will also ensure that the probability of boiling transition will not increase at the uprated conditions.

When a new fuel type is introduced, numerous evaluations are performed as part of the reload process. These evaluations not only confirm that the approved burnup limits in the applicable core operating limits report methodology topical reports are not exceeded, but address all other impacts this new fuel type may have on operation at the TPO power level, including impacts on stability, thermal-hydraulic compatibility, radiological analyses, and hydrogen generation. As set forth in the TS, the licensee must follow acceptable methods and processes described in approved fuel vendor topical reports to perform these analyses and evaluations.

3.1.2 Thermal Limits Assessment

According to GDC 10, “Reactor Design,” of 10 CFR Part 50, Appendix A, the reactor core and the associated control and instrumentation systems are required to be designed with appropriate margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents). In the TSs, the safety limits, some LCOs, and some SRs use 25 percent of the RTP as a threshold. The 25-percent threshold for thermal monitoring is based on generic analyses of a plant with the highest average bundle power (BWR/6), operating at 50-percent power. The original plant operating licenses typically set this monitoring threshold at 25 percent of RTP. The average bundle power for the highest power density plant with the plant operating at the 100 percent of the original licensed thermal power is 4.8 MWt. At 25 percent of original rated thermal power, the average bundle power for the originally licensed plant is 1.2 MWt. For Hatch, the fuel thermal margin monitoring threshold is scaled down to 24 percent to ensure that monitoring is initiated before the average bundle power exceeds 1.2 MWt. A change in the fuel thermal monitoring threshold also requires a corresponding change to the TS reactor core safety limit for reduced pressure or low core flow.

The safety limit minimum critical power ratio (SLMCPR) protects 99.9 percent of the fuel rods from boiling transition during steady-state operation. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as a result of an AOO. The operating linear heat generation rate (LHGR) is the core operating limit that assures the fuel thermal-mechanical performance limit (i.e., the 1-percent fuel plastic strain design limit or the no-fuel-centerline-melt criterion) will not be exceeded as a result of an AOO.

The SLMCPR is calculated for every reload at the rated thermal power, using NRC-approved methodologies.

The OLMCPR is determined on a cycle-specific basis from the results of the reload transient analysis and this approach will not change. AOOs are analyzed at various points in the allowable operating domain, depending on the type of transient. The change in the MCPR is combined with the SLMCPR to establish the OLMCPR. This ensures that 99.9 percent of the rods will not reach boiling transition in the event of an anticipated transient. The licensee will calculate the OLMCPR at the uprated condition for HNP.

The steady-state and transient LHGR limits are established for every fuel design to protect against fuel centerline melt throughout the operating cycle. The licensee will determine the LHGR limits for the uprated cycle in the reload analysis for future cycles. These limits will be maintained during operation.

The maximum planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting LOCA and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, the licensee will perform LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload the licensee will confirm that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

Thus, the licensee will calculate the OLMCPR, the SLMCPR, the LHGR, and the MAPLGHR for the uprated conditions as part of the each reload analysis, using NRC-approved methodologies. The licensee will propose appropriate changes to the limits in the TSs and/or the core operating limit report. This is acceptable to the NRC staff.

3.1.3 Reactivity Characteristics

The reload core analysis will ensure that the minimum shutdown margin requirements will be met for each core design.

3.1.4 Stability

Hatch utilizes reactor stability Option-III as described in NEDO-31960-A and Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," dated November 1995. The enabled region is modified to maintain the pre-TPO absolute power and flow. The stability-based OLMCPR associated with the oscillation power range monitor setpoint assures that the critical power ratio safety limit is not violated following an instability event. There is minimal effect on stability beyond the normal cycle-to-cycle core characteristic variations that are evaluated with the reload. The TPO uprate will not significantly affect stability. Accordingly, reload stability evaluations will continue to ensure acceptable stability performance and protection for future cores operating at TPO uprate conditions. This is acceptable to the NRC staff.

3.1.5 Reactivity Control

Control Rod Drives (CRD) and CRD Hydraulic System

The CRD system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The scram and rod insertion/withdrawal functions of the CRD system depend on the operating reactor pressure and the pressure difference between the CRD system and the reactor vessel bottom head pressure. SNC determined that the CRD system is capable of performing its design functions of rapid rod insertion (scram) and rod positioning (insertion/withdrawal).

In its response to the NRC staff's request for additional information (RAI), the licensee evaluated the control rod drive mechanisms by comparing the proposed parameters to those in the design basis analysis. The licensee indicated that the reactor vessel operating and design pressure and temperature that are used in the existing design basis analysis have sufficient margins to accommodate the changes due to the proposed 1.5-percent power uprate. The licensee concluded that the existing Hatch design basis for stresses and fatigue cumulative usage factors of the CRD mechanisms remain unchanged for the proposed 1.5-percent power uprate condition.

The NRC staff has determined that the proposed power uprate will not have a significant impact on the operation of the CRD system for the following reasons:

(1) The operating dome pressure will not change, and the scram timing at steady-state power conditions will not be affected.

(2) There must be a minimum pressure differential of 250 psid between the hydraulic control unit and the vessel bottom head for normal CRD insertions and withdrawals. Since the operating dome pressure will not increase, the power uprate will have little impact on the CRD pump capacity.

Therefore, the NRC staff concludes that the CRD system will continue to perform all its safety-related functions and meet its design basis and performance requirements at the proposed uprated conditions.

3.1.6 Reactor Core and Fuel Performance - Conclusion

Based on the evaluations presented in Sections 3.1.1 through 3.1.5, the NRC staff concludes that the proposed power uprate is acceptable with respect to its impact on the reactor core and the fuel performance.

3.2 Reactor Coolant and Connected Systems

3.2.1 Nuclear System Pressure Relief/Overpressure Protection

The safety/relief valves (SRVs) provide overpressure protection for the NSSS during abnormal operational transients. The steam flow associated with the 1.5-percent power uprate can be regulated adequately by adjusting the turbine control valve (TCV) position; therefore, the operating dome pressure will not increase, and the SRV setpoints and the number of valve actuation groups will not be changed.

Tables 1-2 and 1-3 of NEDC-33085P provide the thermal-hydraulic parameters for the rated and the proposed power uprate conditions. The tables show that for a core flow of 105 percent, the steam flow rate increases by 0.8 percent for Unit 1 and by 1.5 percent for Unit 2 for the proposed power uprate conditions.

The Unit 1 and Unit 2 Cycles 21 and 18 overpressure protection analyses were performed with the NRC-approved evaluation model ODYN, described in GE topical report, NEDO-24154P-A, “Qualification of the One-dimensional Core Transient Model for BWRs,” dated February 2000. The peak calculated pressure for Unit 1, Cycle-21 is 1350 psig and the peak calculated pressure for Unit 2, Cycle-18 is 1344 psig. Both values are within the acceptance criterion of 1375 psig in NEDO-24154P-A, and are acceptable.

Since the SRVs will actuate at the current setpoints and the current American Society of Mechanical Engineers (ASME) overpressure protection analysis is based on operation at 102-percent power, the NRC staff accepts the licensee’s assessment that the SRVs will have sufficient capacity to handle the increased steam flow associated with the proposed uprate.

3.2.2 Reactor Pressure Vessel (RPV) and Internals

Fracture Toughness

Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary of light-water nuclear power reactors. It also provides adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. For the RPV, this appendix requires an evaluation of the Charpy upper-shelf energy (USE) and adjusted reference temperature (ART).

Neutron irradiation causes a decrease in the Charpy USE and an increase in the ART of the RPV beltline materials. The ART impacts the plant's pressure-temperature (P-T) limits and RPV integrity evaluations. The Boiling Water Reactor Vessel Integrity Project (BWRVIP) program in the Electric Power Research Institute (EPRI) TR-113596, "BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," (BWRVIP-74), dated September 1999 contains an integrity evaluation of BWR RPV circumferentially oriented welds and BWR RPV axially oriented welds that are based, in part, on neutron irradiation. Therefore, in order to demonstrate that neutron irradiation does not significantly impact RPV integrity resulting from the proposed power uprate, the licensee must evaluate the impact of neutron irradiation on the Charpy USE, ART, RPV circumferential welds, and RPV axial welds.

Appendix H to 10 CFR Part 50 specifies the requirements for the materials surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region. Under this program, fracture toughness test data is obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. The program may be plant-specific, where capsules are withdrawn from the licensee's reactor vessel or may be part of an integrated surveillance program (ISP) where capsules are withdrawn from vessels with similar neutron irradiation and thermal environments and the data is shared with the licensees participating in the program. The requirements for an ISP are contained in Section III.C of Appendix H to 10 CFR Part 50.

Section 3.2.1 of NEDC-33085P provides the evaluation of the impact that the power uprate will have on the fracture toughness of the Hatch reactor vessels. This section indicates that the end-of-life (EOL) fluence is calculated for the uprate condition based on the neutron fluence methodology utilized for license renewal reactor vessel integrity and P-T limit analyses. The results of these evaluations indicate that:

- The Charpy USE remains bounded by the BWR Owners Group (BWROG) equivalent margin analysis for 32 effective full power years (EFPY) (40-year life) and BWRVIP-74 for 54 EFPY (60-year life), thereby, demonstrating compliance with 10 CFR Part 50, Appendix G.

- The surface fluence increases for EOL (32 and 54 EFPY) due to the power uprate. The 32 and 54 EFPY shifts in reference temperature are increased, and consequently, necessitate a change in the ART. These values are provided in Tables 3-1a and 3-1b for Hatch, Unit 1 and Tables 3-1c and 3-1d for Hatch, Unit 2. Because the increase in

ART is negligible ($\leq$ 0.5 $^{\circ}$F), the current P-T limit Technical Specification curves remain valid.

Section 3.2.1 of NEDC-33085P indicates the reactor vessel material surveillance program consists of three capsules for each unit. Two capsules containing Charpy specimens were removed from the Hatch, Unit 1 vessel after 5.75 and 14.3 EFPY of operation; both capsules were tested, and one was reconstituted and placed back into the vessel during the Fall 1997 Refueling Outage. One capsule containing Charpy specimens was removed from Hatch, Unit 2 vessel during the Spring 1991 Refueling Outage. The remaining capsule in Hatch, Unit 1 and the remaining two capsules in Hatch, Unit 2 have been in the respective reactor vessels since plant startup. In a letter from H. L. Sumner, Jr. (SNC) to NRC dated August 9, 2002, Hatch requested NRC approval to implement the BWRVIP ISP. Hatch will comply with the withdrawal schedule specified for representative or surrogate surveillance capsules that now represent or apply to each unit in the BWRVIP ISP. Therefore, the 10 CFR Part 50, Appendix H surveillance capsule schedule for the BWRVIP ISP will govern. Implementation of the power uprate has no effect on the BWRVIP ISP withdrawal schedule.

A renewed license for Hatch was issued on June 6, 2002, that extended the operating license from 40 to 60 years. Therefore, the NRC staff evaluated the impact of 60 years of neutron radiation and power uprate on the Charpy USE, ART, RPV circumferential welds and RPV axial welds.

Charpy USE

By letter dated April 30, 1993, the BWROG submitted a topical report entitled "10 CFR Part 50 Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy in BWR-2 Through BWR-6 Vessels," to document that BWR RPVs could meet the margins of safety against fracture equivalent to those required by Appendix G of the ASME Code for Charpy USE values less than 50 ft-lb. In a letter dated December 8, 1993, the NRC staff concluded that the topical report demonstrated that the evaluated materials have the margins of safety against fracture equivalent to Appendix G to the ASME Code, in accordance with Appendix G to 10 CFR Part 50. In this report, the BWROG derived, through statistical analysis, the initial USE values for materials that originally did not have documented Charpy USE values. Using these statistically derived Charpy USE values, the BWROG predicted the end-of life (40 years of operation) Charpy USE values in accordance with RG 1.99, Rev. 2, “Radiation Embrittlement of Reactor Vessel Materials.” According to this RG, the decrease in Charpy USE is dependent upon the amount of copper in the material and the neutron fluence predicted for the material. The BWROG analysis determined that the minimum allowable Charpy USE in the transverse direction for base metal and along the weld for weld metal was 35 ft-lb.

GE performed an update to the USE equivalent margins analysis that is documented in BWRVIP-74. BWRVIP-74 provides a bounding Charpy USE for BWR plants for 54 EFPY. Specifically, the bounding analysis for Hatch-type plants (BWR/4) indicates that at 54 EFPY, the Charpy USE in the transverse direction for plates (base metal) would be at least 45 ft-lb, and the Charpy USE for the non-Linde 80 submerged arc welds would be at least 43 ft-lb. Since these values are greater than the minimum allowable Charpy USE of 35 ft-lb, these materials would have margins of safety against fracture equivalent to Appendix G to the ASME Code. Since this was a generic analysis, the licensee provided its response to RAI MAT-2, plant-specific information to demonstrate that the Hatch beltline materials meet the criteria specified in the report, consistent with the NRC staff's SER in a letter to Carl Terry, BWRVIP Chairman, dated October 18, 2001 (ADAMS Accession Number ML012920549).

The analysis in EPRI TR-113596 determined the reduction in the unirradiated Charpy USE resulting from neutron radiation using the methodology in RG 1.99, Rev. 2. Using this methodology with a correction factor of 65 percent for conversion of the longitudinal properties to transverse properties, the lowest irradiated Charpy USE at 54 EFPY for all BWR/3-6 plates is projected to be 45 ft-lb. The correction factor for specimen orientation in plates is predicated on NRC Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," July 1981. Using the RG methodology, the lowest irradiated Charpy USE at 54 EFPY for BWR non-Linde 80 submerged arc welds is projected to be 43 ft-lb. EPRI TR-113596 indicates that the percent reduction in Charpy USE for the limiting BWR/3-6 plates and BWR non-Linde 80 submerged arc welds is 23.5 percent and 39 percent, respectively. To demonstrate that the Hatch beltline materials meet the criteria specified in the report, the licensee, in response to RAI MAT-2, submitted plant-specific information that demonstrated that the percent reduction in Charpy USE for its beltline materials is less than those specified for the limiting BWR/3-6 plates and the non-Linde 80 submerged arc welds, and that the percent reduction in Charpy USE for its surveillance weld and plate are less than or equal to the values projected using the methodology in RG 1.99, Rev. 2.

In its response to RAI MAT-2, the licensee provided plant-specific information necessary to demonstrate that the Hatch beltline materials meet the criteria specified in the report. The licensee indicates that the predicted reduction in Charpy USE at 54 EFPY under power uprate conditions for the limiting plates in Hatch, Units 1 and 2 is 19 percent and 15 percent, respectively. The predicted reduction in Charpy USE at 54 EFPY for the limiting welds in Hatch, Units 1 and 2 is 33.5 percent and 24 percent, respectively. The licensee indicates that the percent reduction in Charpy USE for its surveillance weld and plate is less than the values projected using the methodology in RG 1.99, Rev. 2. The NRC staff has reviewed the information provided by the licensee, and has determined that the percent reduction in Charpy USE for the beltline materials and the surveillance materials meet the criteria specified in EPRI TR-113596. In addition, the NRC staff has also determined that the materials and surveillance data reported by the licensee are consistent with data contained in the Reactor Vessel Integrity Database (RVID). The RVID is a database maintained by the NRC staff, that contains a summary of all of the relevant materials data submitted by all licensees in their evaluations of reactor vessel integrity. Since the Hatch beltline material and surveillance weld and plate meet the specified criteria, the Hatch beltline materials will meet the margins of safety against fracture equivalent to those required by Appendix G to the ASME Code and, therefore, will meet the Charpy USE requirements of Appendix G to 10 CFR Part 50 at 54 EFPY for the power uprate conditions.

Adjusted Reference Temperature

The current P-T limits were approved for 32 and 54 EFPY in a letter dated August 29, 2000. The surface fluence increases for 32 and 54 EFPY due to the power uprate. Therefore, the ART at 32 and 54 EFPY are increased. The ART values due to the power uprate for 32 and 54 EFPY are provided in Tables 3-1a and 3-1b for Hatch, Unit 1 and Tables 3-1c and 3-1d for Hatch, Unit 2. The NRC staff concludes that the 0.5 °F increase in ART is negligible and therefore, the P-T TS curves remain valid for the proposed power uprate condition.

Circumferential RPV Weld Inspection Relief

The BWRVIP provided the technical bases supporting the elimination of RPV circumferential welds from the inservice inspection programs for BWRs in EPRI TR-113596. These technical bases are approved for the current license term, and are applicable to Hatch.

Appendix E to the NRC's "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925)," USNRC, dated July 28, 1998, documents an evaluation of the impact of license renewal from 32 EFPY to 64 EFPY on the conditional probability of vessel failure. That SER reports that the frequency of cold overpressurization events results in a total vessel failure probability of approximately 5×10^{-7}. The SER conservatively evaluates an operating period of 10 EFPY greater than what is realistically expected for a 20-year license renewal term (i.e., 48 to 54 EFPY). Therefore, this analysis provides a basis for BWRVIP-05 to be approved as a technical alternative to the current inservice inspection requirements of ASME Section XI for volumetric examination of the circumferential welds. Since this was a generic analysis, the licensee, in response to RAI MAT-3, provided plant-specific information to demonstrate that the Hatch beltline materials meet the criteria specified in the report.

In response to RAI MAT-3, the licensee compared the mean reference temperature (RT_{NDT}) for Combustion Engineering fabricated welds from the NRC staff's SER dated July 28, 1998, to the mean RT_{NDT} of the circumferential welds in Hatch, Units 1 and 2 at 54 EFPY under power uprate conditions. The mean RT_{NDT} values in the NRC staff's SER were determined for the limiting BWR RPVs that were fabricated by Combustion Engineering, Babcock and Wilcox, and Chicago Bridge and Iron. Since the Hatch RPVs were fabricated by Combustion Engineering, the results from the NRC staff's SER are applicable to Hatch. The mean RT_{NDT} of the circumferential welds in Hatch at 54 EFPY under power uprate conditions is less than the values for a Combustion Engineering vessel (using Combustion Engineering Owners Group chemistries) at 32 EFPY and 64 EFPY, which indicates that the Hatch circumferential welds will be less embrittled than the Combustion Engineering vessel in the NRC staff analysis at 32 EFPY and 64 EFPY. The NRC staff's SER indicates that the conditional failure probabilities for the Combustion Engineering vessel at 32 EFPY and 64 EFPY were 6.34×10^{-5} and 4.38×10^{-4}, respectively. Since the Hatch circumferential welds will be less embrittled than the Combustion Engineering vessel analyzed in the NRC staff's SER, the conditional failure probability for the Hatch RPVs will be less than the values specified in the NRC staff's SER for circumferential welds.

By letters dated December 2, 1998, and January 19 and February 5, 1999, the licensee requested relief from the inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of the circumferential RPV weld for Hatch, Unit 1. On March 11, 1999, the NRC approved the alternative examination program for circumferential RPV welds for Hatch, Unit 1. In response to RAI MAT-3, the licensee has demonstrated compliance with the criteria in the letter dated July 28, 1998, that transmitted the NRC staff's SER for BWRVIP-05, and has justified that they continue to meet the evaluation which granted them relief from the inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds following the power uprate and up to 54 EFPY. Hatch, Unit 2 will be requesting the same relief in the future and will submit information at the power uprate level.

Axially Oriented RPV Welds

The NRC staff's SER, dated March 7, 2000, to Carl Terry, BWRVIP Chairman, dated March 7, 2000, discusses the NRC staff's concerns related to the RPV failure frequency of the limiting axial welds and the BWRVIP's analysis of the failure frequency. The SER indicates that the RPV failure frequency attributable to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is below 5 x 10^{-6} per reactor year, given the assumptions regarding flaw density, distribution, and location described in the SER. The failure frequency is dependent upon the amount of embrittlement, which in turn is dependent upon the neutron fluence of the RPV. The NRC staff's SER evaluated 40 years of operation based on the embrittlement and neutron fluence projected for each BWR reactor vessel at that time. Because the failure frequency is tied to neutron fluence for the initial 40-year license period of BWR plants, licensees for power uprate with renewed licenses should provide plant-specific information applicable to power uprate and renewed license conditions. Since Hatch has a renewed license and will operate under power uprate conditions, the licensee, in response to RAI MAT-4, submitted information on the impact of embrittlement and neutron fluence resulting from power uprate and 54 EFPY of operation on the failure frequency of the RPVs' axial welds.

The BWRVIP identified Clinton and Pilgrim as the reactor vessels with the highest mean RT_{NDT} in the BWR fleet. The NRC staff confirmed this conclusion in its SER by comparing the information contained in the BWRVIP analysis and the information contained in the RVID for all BWR RPV axial welds. The NRC staff performed analyses of the Clinton and Pilgrim plants. The results from the NRC staff calculations are presented in Table 1. The NRC staff calculations used the basic input information for Pilgrim, with three different assumptions for the initial RT_{NDT}. The calculations of the actual Pilgrim condition used the docketed initial RT_{NDT} of –44 °C (–48 °F) and a mean RT_{NDT} of 20 °C (68 °F). A second calculation, listed as "Mod 1" in Table 1, is consistent with the BWRVIP calculations, with an initial RT_{NDT} of –18 °C (0 °F) and a mean RT_{NDT} of 47 °C (116 °F). A third calculation, with an initial RT_{NDT} of –19 °C (–2 °F) and a mean RT_{NDT} of 46 °C (114 °F), was chosen to identify the mean value of RT_{NDT} needed to provide a result that closely matches the RPV failure frequency of 5 x 10^{-6} per reactor-year.

Table 1: Comparison of Results from Staff and BWRVIP

Plant	Initial RT_{NDT} (°F)*	Mean RT_{NDT} (°F)	Vessel Failure Freq.	
			Staff	BWRVIP
Clinton	-30	91	2.73E-6	1.52E-6
Pilgrim	-48	68	2.24E-7	-------
Mod 1 **	0	116	5.51E-6	1.55E-6
Mod 2 ***	-2	114	5.02E-6	-------

* °C = 0.56 x (°F – 32)

** A variant of Pilgrim input data, with initial RT_{NDT} = 0 °F.

*** A variant of Pilgrim input data, with initial RT_{NDT} = -2 °F.

Since the BWRVIP analysis was generic, the NRC staff issued RAI MAT-4 requesting the licensee to submit plant-specific information to demonstrate that the Hatch beltline materials meet the criteria specified in the report. The licensee provided plant-specific information in response to RAI MAT-4 to demonstrate that the Hatch beltline materials meet the criteria specified in the SER. The mean RT_{NDT} of the axial welds at Hatch at 54 EFPY and power uprate conditions was less than 114 $^\circ$F for both units. This value is less than the value for Pilgrim Mod 2 in Table 1, which indicates that the Hatch axial welds at 54 EFPY and power uprate conditions will be less embrittled than the axial welds for the Pilgrim Mod 2 analysis performed by the NRC staff as set forth in its letter dated March 7, 2000. Since the Hatch axial welds will be less embrittled than the axial welds for the Pilgrim Mod 2 analysis performed by the NRC staff as set forth in its letter dated March 7, 2000, the conditional failure probability for the Hatch RPVs will be less than 5×10^{-6} per reactor-year at 54 EFPY and power uprate conditions. Therefore, the licensee has demonstrated compliance with the criteria in the NRC staff's letter dated March 7, 2000, and the NRC staff considers this issue resolved at Hatch for 54 EFPY and power uprated conditions.

RPV Monitoring Program

Section 3.1.17.1 of NUREG-1803, "Safety Evaluation Report Related to the License Renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2," contains an evaluation of the RPV Monitoring Program for 54 EFPY. As documented in this NUREG, the licensee has committed to a plant-specific RPV monitoring program or an ISP that will remove capsules at various neutron fluence levels to provide relevant data for assessment of the integrity of the RPV. The ISP was approved for 40 years of operation; but was not approved for 60 years of operation. The BWRVIP has committed to provide supplemental information to extend the ISP through 60 years of operation. Since the ISP was not approved for 60 years of operation, the renewed license was conditioned to require that prior to operation in the renewal term the licensee will notify the NRC of its decision to implement the ISP or a plant-specific program, and to provide the appropriate revisions to the UFSAR Supplement summary of the vessel surveillance material testing program. Since the power uprate has a negligible impact on embrittlement, as discussed above, no change is needed to the license condition.

The NRC staff concludes that during power uprate conditions and for 54 EFPY, the Hatch reactor vessels will comply with the fracture toughness requirements in Appendix G to 10 CFR Part 50, the circumferential welds will have sufficient fracture toughness to eliminate the inspection of circumferential welds in accordance with EPRI TR-113596, and the axial welds will have sufficient fracture toughness to satisfy the RPV failure frequency in the NRC staff's SER contained in the March 7, 2000, letter to Carl Terry. In addition, the NRC staff concludes that the license condition resulting from the Hatch license extension process is sufficient to ensure that the Hatch plant-specific RPV monitoring program or ISP will satisfy Appendix H to 10 CFR Part 50 during power uprate conditions and for 54 EFPY.

RPV Structural Evaluation

The licensee indicated that the effect of the proposed uprate for the reactor vessel components, except the FW nozzles, was evaluated in accordance with the ASME Boiler and Pressure Vessel Code, 1965 Edition with addenda to and including Summer 1966, which is the code of record for Hatch. The proposed power uprate does not change the operating reactor pressure and temperature from the current operating condition. There is no change in fuel lift and seismic loads due to the uprate. The current design basis transient analyses remain valid for the proposed power uprate. The LOCA loads were analyzed at 102-percent power level, bounding the proposed power uprate condition. Also, the recirculation design flow does not change for the power uprate. The annulus pressurization and jet reaction are not affected by the proposed power uprate. The licensee concluded that the current design basis stress and calculated stresses and fatigue usage factors (CUFs) analyses for the reactor vessel components will continue to meet the code limits and are, therefore, acceptable for the proposed power uprate. The NRC staff reviewed the licensee's code design limits and calculated stresses and CUFs and determined that the calculated stresses, and CUFs are less than the ASME Code limits and, therefore, acceptable for the proposed power uprate.

3.2.3 Reactor Internals

The licensee evaluated the internal components considering the changes in the design input parameters and loads due to the proposed 1.5-percent power uprate. The loads applicable to the internal components include reactor internal pressure difference (RIPD), LOCA, SRV, seismic, annulus pressurization, jet reaction, and fuel lift loads.

The licensee evaluated the reactor internals due to the slight increase in the FW flow and temperature, and the RIPDs. In its April 7, 2003, response to the NRC staff's RAI, the licensee indicated that for the affected limiting reactor internal components, the increase in loads is less than 1 percent and is insignificant compared to those in the design basis. There exist sufficient margins in the design basis of the reactor internals to accommodate the changes due to the 1.5-percent power uprate. As a result of its evaluation, the licensee concluded that the design basis stresses and fatigue usage factors for the reactor internal components will remain below the code allowable limits for Hatch plant operation at the proposed 1.5-percent power uprate.

The integrity of BWR reactor vessel internals is managed by BWRVIP programs. These programs consist of examination and evaluation of reactor vessel internal components that are susceptible to degradation in the BWR environment and are discussed in Section 3.1.15 of NUREG-1803. Section 3.3 of NEDC-33085P contains the licensee's evaluation of the impact of the power uprate on reactor internals including core support structures and non-core support structure components.

For evaluation of core support structures done in support of license renewal, the NRC staff identified that these structures could be susceptible to irradiation assisted stress corrosion cracking (IASCC) and void swelling. However, since the power uprate only increases the licensed power level by 1.5 percent above the current licensed thermal power level, the NRC staff determined that the power uprate will not significantly increase the susceptibility of these components to IASCC and void swelling because the increase in neutron fluence is negligible. Therefore, the BWRVIP programs, as implemented at Hatch, do not need to be revised as a result of the power uprate and the existing programs are sufficient to ensure the integrity of the reactor vessel internals for power uprate conditions because the power uprate conditions do not significantly impact the integrity of reactor internals.

On August 21, 2002, General Electric Nuclear Energy (GENE) issued Service Information Letter (SIL) No. 644, "BWR/3 steam dryer failure." SIL-644, applicable to BWR/3-style steam dryers, recommended monitoring for moisture content and other reactor parameters, and for an inspection of the cover plates at the next refueling outage for plants operating at greater than the original licensed thermal power (OLTP). The licensee implemented its extended power uprate (EPU) for Unit 1 in 1999, and 1998 for Unit 2, which represents 113.4 percent of OLTP. Hatch Unit 1 has a BWR/4 -style steam dryer, and Hatch Unit 2 has a BWR/5-style steam dryer. The licensee does not have a moisture carryover monitoring program; therefore, no established trends exist on the performance of the steam dryer. The licensee has performed visual examinations of the steam dryer, but only as part of the vessel internal inspection scope during various outages for Hatch. Following the EPU implementations, only limited inspections have been performed, and there is no trend data on existing flaw indications. As a result of SIL-644, the licensee performed concentrated inspections of the Unit 2 steam dryer during the Spring 2003 refueling outage. The inspection indicated no evidence of SIL-644 type failure mechanisms of the dryer components.

On September 5, 2003, GENE issued SIL-644, Supplement 1, “BWR steam dryer integrity.” This supplement recommends inspections, and monitoring for all BWR steam dryer types. The licensee in its letter dated September 12, 2003, committed to implement the recommendations in SIL-644, Supplement 1.

Based on the NRC staff’s review of the licensee’s previous inspections of the dryer following the implementation of the EPU and the future implementation of the recommendations in SIL-644, Supplement 1, the NRC staff has determined that operation of the Hatch steam dryers remain acceptable at the proposed power uprate conditions, pending a generic long-term solution by the industry and the NRC.

3.2.4 Flow-Induced Vibration

The licensee assessed flow induced vibration (FIV) for the proposed power uprate for limiting reactor internal components such as the shroud head and separator, steam dryers, jet pumps, jet pump sensing lines, and FW sparger. The licensee indicated that there is a slight increase in flow induced vibration for the shroud, shroud head and separator, and FW sparger because of an approximately 2-percent increase in steam and FW flow due to the power uprate. Other internal components are not affected since the maximum core flow and the maximum recirculation drive flow remain unchanged following the proposed 1.5-percent power uprate. As a result of its evaluation, the licensee concluded that vibration of safety related internal components due to flow induced vibration loads will remain within the GE acceptable stress limits of 10 ksi.

The NRC staff finds that the increases in FIV in the shroud head and separator, and FW sparger, due to the proposed power uprate, are less than 10 percent. Therefore, the NRC staff finds that the reactor internals will remain adequate and acceptable for the proposed 1.5-percent power uprate, considering the GE stress limit of 10 ksi would permit service cycles equal to 10^{11} when compared to the ASME allowable stress limit of 13.6 ksi. The NRC staff concludes that the reactor internals design remains acceptable for the FIV at the proposed power uprate condition.

3.2.5 Piping Evaluation

3.2.5.1 Reactor Coolant Piping and Components

The licensee evaluated the effects of the proposed 1.5-percent power uprate condition on the reactor coolant piping, components and their supports with regard to changes in flow rate, temperature and pressure. The licensee summarized its evaluation of reactor coolant pressure boundary (RCPB) piping inside the containment in a table in Section 3.5.1 of the amendment request. The piping systems evaluated by the licensee included the recirculation, main steam (MS) and attached piping systems (including SRV discharge line), reactor core isolation cooling (RCIC) piping, MS drain lines, RPV head vent line, FW piping (inside containment), RPV bottom head drain line, and residual heat removal (RHR), low pressure core spray, high pressure core spray, reactor water cleanup (RWCU), and standby liquid control (SLC) piping systems.

The methods used by the licensee for the piping and pipe support evaluations is described in NEDC-32938P Appendix K, "Methods and Assumptions for Piping Evaluation of TPO Uprate," which has previously been reviewed and accepted by the NRC for power uprates. For TPO uprates of $\leq$ 1.5 percent the licensee stated that there are no increases in reactor dome operating and design pressures and temperatures, MS operating and design pressures and temperatures and core flow, and a negligible increase in coolant temperature. The changes in core pressure drop and recirculation temperature have a negligible effect on the recirculation piping performance. Since these values are close to the current operating conditions, the licensee concluded that no further evaluation was necessary for the reactor coolant pressure boundary portion of all piping with the exception of the MS and FW lines and piping connected to the MS lines. There is a slight increase in the MS and FW flow rate and in the FW system operating pressure and temperature.

The licensee assessed the design basis calculation of the RCPB piping and its support components. The licensee found that the conditions under the proposed power uprate are enveloped by the current licensing basis analyses of the above identified piping systems. Therefore, the licensee concluded that the existing design basis of the RCPB piping and supports is adequate and acceptable for operation at the 1.5-percent uprate conditions, in compliance with the standards of American National Standards Institute (ANSI) B31.1, "Power Piping" 1986 Edition through 1987 Addenda, and ASME Section III, 1977 Edition through Winter 1978 Addenda, which are the codes of record for Hatch.

The NRC staff finds that the design of piping, components and their supports is adequate to maintain the structural and pressure boundary integrity of the RCPB because the current licensing basis parameters are bounding for the proposed 1.5-percent power uprate condition.

The licensee evaluated the MS piping system for conformance with the ANSI B31.1/ASME Section III Code stress criteria and for the effects of temperature, pressure, and flow on the piping stress and pipe supports. The licensee stated the current licensing basis for the MS piping system (inside containment) analyzed for pressure, temperature, and flow, envelops the TPO operating pressure, temperature, and flow. Therefore, the licensee concluded, all safety aspects of the MS piping system (inside containment) are within current licensing basis evaluations that were previously performed at 102-percent current licensing thermal power (CLTP).

The licensee evaluated the FW piping system (inside containment) for conformance with the ANSI B31.1/ASME Section III Code stress criteria, and for the effects of temperature, pressure and flow on the piping stress and pipe supports (i.e., snubbers, hangers and struts). The licensee stated the current licensing basis for the FW piping system (inside containment) evaluation for pressure, temperature, and flow, envelops the TPO operating pressure, temperature, and flow. Therefore, the licensee concluded that all safety aspects of the FW piping system (inside containment) are within the current licensing evaluations that were previously performed at 102 percent of the CLTP.

The NRC staff has reviewed the information provided by the licensee and determined that the current analyses, performed using 102 percent of the CLTP, bound the stresses of the piping at the proposed power uprate. The NRC staff considers the materials used in the various systems that comprise the RCPB piping to be adequate for the 1.5-percent power uprate since the pressure, temperature, and flow are within the original licensee's design evaluation, which was previously performed at 102 percent of the CLTP.

3.2.5.2 Balance of Plant Piping

The licensee evaluated the balance-of-plant (BOP) piping systems by comparing the original design basis conditions with those for the proposed power uprate. The BOP piping systems that are affected were determined from the uprated reactor and BOP heat balances. The piping systems evaluated include large and small bore ASME Section III Code Class 1, 2, and 3 piping and supports that are outside the RCPB. The evaluation of BOP piping was performed in a similar manner to the evaluation of RCPB piping and supports for operation at the proposed power uprate.

The licensee reviewed the piping stress analyses of record. The licensee found that there are sufficient margins between the maximum stresses and fatigue usage factors and code allowable limits in the original design basis calculation to accommodate the slight increases in temperature (<1 °F), pressure (<5 psi) and flow rate (<2 percent) for the proposed power uprate. No new postulated pipe break locations were identified in any of the system evaluated. The licensee concluded that the Hatch BOP piping and related support systems remain within code allowable of the present code of record (ASME Section III, 1977 Edition through the Winter 1978 Addenda). The NRC staff has reviewed the information provided by the licensee on design basis conditions and piping stress analyses of record and finds that the BOP systems will operate within the fatigue usage factors and code allowable limits in the original design basis and, therefore, will operate at the proposed 1.5-percent power uprate conditions without adverse effects on the piping system and its supports.

Flow Accelerated Corrosion (FAC) in Piping

The TPO power uprate will cause changes of some plant operating parameters at Hatch. These changes may affect the wear rates caused by FAC and require modification of the FAC program.

At Hatch, the following systems have a condition conducive to FAC:

- MS and associated systems
- FW piping system
- BOP piping

The MS and the associated piping systems contain components that are made from carbon steel and are susceptible to FAC. When these components are exposed to superheated steam or high quality steam, no FAC damage will occur. However, some of these pipes may contain moisture, which will create an environment conducive to FAC. The licensee has determined that after the power uprate, there will be no change in the line pressure and temperature in these pipes, but the velocity of steam will increase by about 2 percent. This will cause some increase of FAC. In the FW system, carbon steel pipes are exposed to flowing single phase water. After power uprate, there will be a slight increase of velocity and temperature causing an increase of FAC. The integrity of the BOP piping system is assured by proper design in accordance with the applicable codes and standards. However, carbon steel pipes exposed to flowing single and two phase fluids may still exhibit FAC damage. After the power uprate, the change in the fluid velocity, temperature and moisture content in the systems associated with the turbine cycle will cause an increase in FAC.

The licensee has a FAC program based on the CHECWORKS predictive code that includes wall thickness measurements. Using this program, the licensee predicted that after TPO power uprate the increase in wear rate caused by FAC will be very small. It demonstrated that increase of wear rates in the extraction steam lines to the FW heaters, which were identified as being the systems most susceptible to FAC, was less than half a percent. After implementing the TPO power uprate, the licensee will be revising its FAC program by including into the predictive code any changes in the plant operating parameters that may have some effect on FAC.

The licensee stated that operation at the TPO RTP results in some changes to parameters affecting FAC in those systems associated with the turbine cycle (i.e., condensate, FW, MS). The licensee’s inspection and evaluation for FAC in BOP systems addresses Generic Letter (GL) 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning.” The plant FAC program currently monitors the affected systems. The licensee considers the continued monitoring of the systems provides confidence in the integrity of susceptible high-energy piping systems. The licensee also stated that appropriate changes to piping inspection frequency will be implemented as a result of the TPO. This action takes into consideration adjustments to predicted material loss rates used to project the need for maintenance/replacement prior to reaching minimum wall thickness requirements. The licensee concluded that the program will provide assurance that the TPO has no adverse effect on high-energy piping systems potentially susceptible to pipe wall thinning due to FAC.

The NRC staff reviewed the licensee’s application on the systems where the TPO power uprate causes some change in FAC, and concludes that the percent change in predicted wear rate, and the change in predicted wear rates are negligible for the power uprate condition, and this will be verified by the Hatch monitoring program. The NRC staff concludes that the effect is very small and will be adequately controlled by the procedures in the licensee’s FAC program.

3.2.6 Reactor Recirculation System (RRS)

The power uprate will be accomplished by operating along extensions of the rod and core flow lines on the power/flow map. Both units are currently licensed to operate for increased core flow (ICF) operation up to a maximum core flow of 82.4 Mlb/hr. The power uprate does not necessitate an increase in the maximum allowable core flow. Therefore, the reactor recirculation flow will be maintained within the flow limits of the existing power/flow map, with 100-percent power corresponding to the proposed power uprate level. The cycle-specific reload analysis will consider the full range of the power and flow operating region.

The licensee assessed the effects of the proposed TPO uprate on the RRS and concluded the TPO uprate has a minor effect on the RRS and its components. The licensee stated that the TPO uprate does not necessitate an increase in the maximum core flow. The licensee stated that both Hatch, Units 1 and 2 are licensed for ICF operation of 105 percent of rated core flow. The licensee stated no significant reduction of the maximum flow capability occurs due to the TPO uprate because of the small increase in core pressure drop ($<$1 psi), and the effect on pump net positive suction head (NPSH) at TPO conditions is negligible. The licensee's evaluation also confirmed that no significant increase in RRS vibration occurs from the TPO operating conditions.

The NRC staff has reviewed the information provided by the licensee and concludes that the TPO uprate condition will have negligible impact on the materials used in the reactor recirculating system because there is no increase in the maximum core flow. Further, because the effect on NPSH is negligible and there is no significant increase in RRS vibration, the TPO power uprate will have no adverse effect on the RRS.

3.2.7 Reactor Core Isolation Cooling System

The RCIC system provides core cooling when the RPV is isolated from the main condenser and the RPV pressure is greater than the maximum allowable for starting a low-pressure core cooling system. The RCIC system is designed to provide rated flow over a range of reactor pressures from approximately 150 psig to the maximum pressure corresponding to the lowest opening setpoint for the SRVs. In particular, the loss-of-feedwater-flow (LOFW) transient assumes that the RCIC system will maintain water level inside the core shroud high enough to ensure that the top of the active fuel (TAF) will be covered throughout the event. The transient analysis also assumes that the low-setpoint SRVs will remove the stored and decay heat since main steam isolation valve (MSIV) closure on low water level isolates the reactor from the main condenser. The transient is a power-dependent transient and is more severe at a higher initial power since there is more stored energy and decay heat to be dissipated and the water level drops faster.

The LOFW analysis described in GE topical report NEDC-31984P, Volume 1, "Generic Evaluation of Boiling Water Reactor Power Uprate," July 1991 is applicable to Hatch because the analyses confirm that RCIC is capable of mitigating the LOFW transient. Since the proposed 1.5-percent power uprate does not increase the steady-state operating pressure or the SRV actuation setpoints, the NRC staff determined that the RCIC performance would not be affected by the proposed power uprate. Accordingly, the NRC staff concludes that since the SRVs are capable of removing stored and decay heat and that there is sufficient margin to the

TAF to maintain core cooling capability, the RCIC performance will be acceptable at the power uprate conditions.

3.2.8 Residual Heat Removal System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary-system decay heat removal after reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low-pressure coolant injection (LPCI) mode, the shutdown cooling mode, the suppression pool cooling mode, and the containment spray cooling mode.

The slightly higher decay heat has negligible effect on the operation of the RHR system in the shutdown cooling mode. The NRC staff concludes that the RHR system will perform all of its safety functions because it is bounded by the previous analyses based on 102 percent of CLTP.

3.2.9 Reactor Water Cleanup System

The function of the RWCU system is to remove solids and dissolved impurities from the reactor coolant, thereby reducing its concentration of radioactive and corrosive species. The RWCU system is a normally operating system with no safety related functions other than containment isolation. The flow through the RWCU system is not significantly affected by the reactor power and recirculation flow conditions, therefore, the increase of rated power due to the TPO power uprate will only negligibly affect system capability. There is no significant effect on operating temperature and pressure conditions in the high-pressure portion of the system. Operation at the uprated power will cause an insignificant change in the quantity of fission and corrosion products, and other soluble and insoluble impurities in the reactor water. There will be no need to make changes in water quality requirements. The NRC staff reviewed the licensee’s evaluation of the performance of the RWCU system after power uprate and finds it to be acceptable because there is no significant effect on operating temperatures and pressures in the high-pressure portion of the system. The NRC staff concludes that the safety and operational aspects of the RWCU system will continue to be satisfied at the proposed power uprate conditions because the only safety function is to isolate the RWCU, which is not affected by the TPO uprate, and there is no significant impact on RWCU system operating temperatures or pressures.

3.2.10 Reactor Coolant System and Connected Systems - Conclusion

Bases on the evaluations in Sections 3.2.1 through 3.2.9 of this SE, the NRC staff concludes that the proposed power uprate is acceptable with respect to its impact on the reactor coolant system and connected systems.

3.3 Engineered Safety Features

3.3.1 Containment Systems Performance

The licensee stated that the current containment evaluations are bounding for the 1.5-percent power uprate, because they were performed at 102 percent of the CLTP level. The licensee evaluated the containment short-term pressure and temperature response, the long-term suppression pool temperature response, containment dynamic loads, and containment isolation. The licensee stated that although the nominal operating conditions change slightly at the proposed power uprate level, the required initial conditions for containment analysis inputs remain the same.

Based on the NRC staff's review of the licensee's application and experience gained from the review of power uprate applications for similar BWR plants, the NRC staff finds that there are negligible effects on containment system performance. The NRC staff concludes that containment performance will remain acceptable after implementation of the power uprate.

Motor Operated Valves

The licensee reviewed its motor-operated-valves (MOVs) program and indicated that the existing MOVs evaluation at Hatch was performed based on 102 percent of the current power level, and using maximum expected differential pressure that bounds the proposed 1.5-percent power uprate condition. The licensee evaluated its commitments relating to GL 95-07, "Pressure Locking and Thermal Binding of Safety- Related Power-Operated Gate Valves." These commitments are associated with the pressure locking and thermal binding of safety related power operated gate valves that need to operate to carry out their intended safety function. The licensee found that the existing analysis conditions remain bounding for the 1.5-percent power uprate. The licensee also evaluated its response relating to its GL 96-06, "Assurance of Equipment Operability And Containment Integrity During Design-Basis Accident Conditions," program regarding the over-pressurization of isolated piping segments. The licensee concluded that the existing evaluation for GL 96-06 was performed based on containment design temperature and pressure, which remain bounding for the proposed power uprate. For the reasons set forth above, the NRC staff concludes that the power uprate will have no adverse effects on the safety-related valves and that the licensee's conclusions from the GL 95-07, GL 96-06, and GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" programs, remain valid.

3.3.2 Emergency Core Cooling System

The ECCS is designed to provide protection in the event of a LOCA due to a rupture of the primary-system piping. Although design basis accidents (DBAs) are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on (1) the peak cladding temperature, (2) local cladding oxidation, (3) total hydrogen generation, (4) coolable core geometry, and (5) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system pipe. Assuming a single failure of the ECCS, the LOCA analyses identify the break sizes that most severely challenge the ECCS systems and the primary containment. The MAPLHGR operating limit is based on the most limiting LOCA analysis, and the licensees perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met. Since the ECCS-LOCA analyses were performed using NRC-approved methodology and codes and appropriate plant-specific parameters, those analyses are acceptable to the NRC staff. Therefore, the NRC staff concludes that the ECCS will perform as designed and analyzed at the proposed power uprate conditions.

The ECCS for Hatch includes the high-pressure coolant injection (HPCI) system, the LPCI mode of the RHR system, the core spray (CS) system, and the automatic depressurization system (ADS).

High-Pressure Coolant Injection System

The HPCI system (with other ECCS systems as backups) is designed to maintain reactor water inventory during small- and intermediate-break LOCAs, isolation transients, and LOFW events. The HPCI system is designed to pump water into the reactor vessel over a wide range of reactor operating pressures. The HPCI system also serves as a backup to the RCIC system.

The HPCI system is required to start and operate reliably over its design operating range. During the LOFW event and isolation transients, the RCIC system maintains water level above the TAF. For MSIV closure, the SRVs open and close as required to control pressure and HPCI eventually restores water level.

The licensee evaluated the capability of the HPCI system during operation at the TPO power level to provide core cooling to the reactor to prevent excessive fuel peak cladding temperature (PCT) following small- and intermediate-break LOCAs and ensure core coverage up to the TAF in isolation transients and LOFW transients. The licensee stated that the HPCI evaluation is applicable to and is consistent with the evaluation in TLTR Section 5.6.7. The maximum reactor pressure at which the HPCI system must be capable of injecting into the vessel for the RCIC backup function was selected based on the opening set point values of SRVs. The TPO uprate does not decrease the NPSH available for the HPCI pump or increase the required NPSH.

The licensee evaluated the capability of the HPCI system to perform as designed and analyzed its performance at the TPO conditions. The licensee determined that the HPCI system can start and inject the required amount of coolant into the reactor for the range of reactor pressures associated with LOCAs and isolation transients. The ECCS-LOCA analysis is based on the current HPCI capability (see Section 3.3.3) and demonstrates that the system provides adequate core cooling. Since the ECCS-LOCA analysis demonstrates that the system provides adequate core cooling, the NRC staff concludes that the HPCI system is acceptable for TPO operation.

Core Spray System

The CS system initiates automatically in the event of a LOCA. In conjunction with other ECCS systems, the CS system provides adequate core cooling for all LOCA events. The system also provides spray cooling for long-term core cooling after a LOCA. The licensee explained that the existing CS system hardware has the capability to perform its design function at the TPO conditions and that the generic evaluation in Section 5.6.10 of the TLTR is applicable to Hatch. Since the ECCS-LOCA analysis demonstrates that the system provides adequate core cooling, the NRC staff concludes that the CS is acceptable at the proposed power uprate condition.

Low-Pressure Coolant Injection

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. In conjunction with other ECCS systems, the LPCI mode of RHR is used to provide adequate core cooling for all LOCA events. The licensee stated that the existing system has the capability to perform the design injection function of the LPCI mode for operation at the TPO conditions. Since the licensee's ECCS-LOCA analysis for TPO operation is based on the current LPCI capability (as discussed in Section 3.3.3) and demonstrates that the system provides adequate core cooling, the NRC staff finds the evaluation acceptable.

Automatic Depressurization System

The ADS uses the SRVs to reduce reactor pressure after a small-break LOCA with CS failure, allowing LPCI and the CS system to provide cooling flow to the vessel. The plant design provides for SRVs to have a minimum flow capacity. After a delay, the ADS actuates either on low water level or on high drywell pressure. The licensee stated that the ADS's ability to perform these functions is not affected by the power uprate. Since the small-break LOCA analyses assume that the ADS actuates at a bounding vessel pressure and power, the NRC staff determined that this power uprate does not affect the capability of the ADS to perform its function.

ECCS Net Positive Suction Head

The licensee stated that the most limiting case for NPSH typically occurs at the peak long-term suppression pool temperature. In addition, the licensee stated that the generic evaluation of the containment provided in Appendix G to the TLTR is applicable to HNP. Since the containment analysis was performed at 102 percent of the CLTP, the licensee concluded that there is no change in the available NPSH for systems using suppression pool water and the power uprate has no effects on the NPSH requirements.

Based on a review of the licensee's application against the current licensing basis (CLB), the NRC staff has determined that the NPSH for systems using suppression pool water remains unchanged at the 1.5-percent power uprate condition. Accordingly, the NRC staff concludes that the current NPSH analyses remain valid for the proposed power uprate conditions.

Conclusion

The LOCA analyses of record demonstrate that the HPCI system, the LPCI mode of RHR, the CS system, and the ADS have the capabilities to provide core cooling during a LOCA. These capabilities do not change for operation at the uprated conditions. Therefore, the NRC staff concludes that the ECCS will continue to meet the ECCS-LOCA analysis assumptions and design criteria at the uprated conditions.

3.3.3 Emergency Core Cooling System Performance

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The licensee stated that the ECCS performance satisfies these requirements under all LOCA conditions and the analytical models satisfy these requirements. The GE fuel was analyzed with GE's NRC-approved GESTAR II model. These analyses were performed at 102 percent of current licensed power level for the power uprate. In this analysis, the core flow of Hatch, Unit 2 and the FW temperature of Hatch, Unit 1 are used. The limiting case was the double-ended guillotine break of the recirculation suction line with a battery failure. The analyses, for each respective fuel type, yielded PCTs less than 1820 °F, peak cladding oxidation of less than 1 percent, and core-wide cladding oxidation of less than 0.2 percent. These results comply with the 10 CFR 50.46 requirements of PCT less than 2200 °F, cladding oxidation less than 17 percent, and core-wide cladding oxidation of less than 1 percent. The NRC staff accepts SNC's ECCS performance evaluation because the analytical models and codes are based on the NRC-approved methodology described in GESTAR II and the ECCS-LOCA analyses are based on bounding power and flow conditions.

3.3.4 Main Control Room (MCR) Atmosphere Control System

The licensee evaluated the current design basis analysis for a postulated accident to the MCR operators. The habitability was determined to be unchanged because the existing analysis was evaluated for accident conditions from 102 percent of CLTP. The licensee stated that the system remains capable of performing its safety function following the proposed power uprate.

Based on the NRC staff's review of the licensee's evaluation and the experience gained from the review of power uprate applications for similar BWR plants, the NRC staff finds that the lciensee's existing analysis for the MCR atmosphere control system meets applicable regulatory guidelines, and the impact on MCR operators is negligible at the proposed uprate condition. The NRC staff concludes, therefore, that the MCR atmosphere control system remains acceptable for the uprated conditions.

3.3.5 Standby Gas Treatment System (SGTS)

The SGTS minimizes the off-site and control room doses rates during venting and purging of the containment atmosphere under abnormal conditions. The system was designed to maintain the secondary containment at a slightly negative pressure during such conditions. The charcoal beds can accommodate DBA conditions at 102 percent of the CLTP. The licensee stated that the system remains capable of performing its safety function for the power uprate.

Based on the NRC staff's review of the licensee's application, the NRC staff finds that the licensee's existing analysis remain bounding because it was performed at 102 percent of the CLTP, which bounds the proposed power uprate, and concludes that the SGTS operation remains acceptable for the uprated conditions.

3.3.6 Post-LOCA Combustible Gas Control System (CGCS)

Hatch, Unit 1 controls combustible gas concentrations by use of the containment atmosphere dilution (CAD) method. The CAD method adds nitrogen to the containment to dilute the oxygen concentration below the flammability limit. At Hatch, Unit 2 combustible gas is controlled by use of hydrogen recombiners, that maintain a safe level of hydrogen inside the primary containment. A containment purge capability is provided as a backup for both units. As a result of the proposed power uprate, the post-LOCA production of hydrogen and oxygen from radiolysis is expected to increase proportionally to the increase in power level.

For Hatch, Unit 1, the addition rate of nitrogen from the CAD system is controlled and monitored in the main control room. The licensee stated that there is sufficient capacity in the CAD system to account for the increase in oxygen generation due to the power uprate. The power uprate may require the CAD system to be initiated earlier in the accident, however, the change will be insignificant. The oxygen concentration needs to be controlled within 5 volume percent (v/o) following a LOCA. The licensee stated that additional margin has been provided by designing the system to control oxygen within 4 v/o.

For Hatch, Unit 2, the initiation of the hydrogen recombiners is controlled and monitored based on the gas concentration and not by time. The power uprate will cause the recombiners to be initiated earlier during the postulated accident, however, the change will be insignificant. The hydrogen concentration needs to be controlled within 4 v/o following a LOCA. The licensee stated that additional margin has been provided by designing the system to control hydrogen within 3.5 v/o.

Based on the NRC staff's review of the licensee's application, and a review of the guidance in RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-Of-Coolant Accident," the NRC staff concludes that based on the insignificant change in time to initiate the CAD system for Hatch, Unit 1, and the hydrogen recombiners for Hatch, Unit 2, CGCS operations remains acceptable for the proposed uprated conditions.

3.3.7 Engineered Safety Features - Conclusions

Based on the evaluation in Sections 3.3.1 through 3.3.7, the NRC staff concludes that the proposed power uprate is acceptable with respect to its impact on the engineered safety features.

3.4 Instrumentation and Control

3.4.1 Crossflow System Instrumentation

The licensee's December 19, 2002, submittal requested a 1.5-percent measurement uncertainty recovery power uprate on the basis of using the Crossflow UFM system in combination with a CORRTEMP UTM system to automatically correct the installed feedwater venturi readings in the plant computer. The licensee provided supplementary information supporting the requested power uprate in letters dated May 21 and May 30, 2003.

In the licensee's initial submittal, the licensee intended to install three Crossflow UFM M/TSFs with associated UFM transducers and electronics on the "A" feedwater pipe, and a single UFM M/TSF on the "B" feedwater pipe in Hatch, Unit 1. Each M/TSF would also have an associated CORRTEMP UTM system. By letter dated May 30, 2003, the licensee described a change in this configuration in Hatch, Unit 1 to a single UFM system with an associated CORRTEMP system on each feedwater pipe. As with Hatch, Unit 1, the licensee will install one Crossflow UFM system and associated CORRTEMP system on each feedwater pipe in Hatch, Unit 2.

The Crossflow UFM system instrumentation consists of ultrasonic transducers, a signal conditioning unit, and a data processing computer (DPC). The DPC, with its software, performs digital signal processing on the demodulated ultrasonic signals and calculates the delay time for use in flow calculations.

Each UFM system consists of eight non-intrusive flow transducers mounted in a M/TSF, which is externally attached to the pipe in which the flow is to be measured. The eight transmitters produce two feedwater flow signals on the associated feedwater pipe that are then arithmetically averaged to produce a single feedwater pipe flow rate. The feedwater flow rates from the two feedwater trains are then combined to produce a total feedwater flow rate. Additionally, a CORRTEMP UTM system will be installed on each feedwater pipe to improve the accuracy of the existing resistance temperature detector (RTD)-based feedwater temperature measurements.

The purpose of installing the UFM/UTM systems is to improve the accuracy of the venturi-based feedwater flow measurement systems. This increased flow accuracy can be translated into a like improvement in the accuracy of the CTP calculation, due to the use of more accurate feedwater flow and temperature measurements in the heat balance calculation. This measurement uncertainty recapture will allow the licensee to operate the plant at a higher power level without exceeding the 10 CFR Part 50, Appendix K 102-percent power margin, thereby, recovering lost generating capacity due to feedwater venturi fouling while staying within the plant's licensed operating power level. Additionally, the UFM/UTM systems will provide an in-plant capability for periodically recalibrating the feedwater venturi flow coefficient to adjust for the adverse effects of venturi fouling. The combination of the UFM and UTM on each feedwater pipe are considered by the licensee to be an integrated feedwater flow measurement system.

Topical Report CENPD-397-P-A, Rev. 1, provided a methodology for determining measurement uncertainty of the Crossflow UFM system. This methodology used specific guidelines and equations for determining the 95/95 uncertainty values of the Crossflow UFM system input parameters. The parameters that contribute to feedwater flow measurement uncertainty are pipe inside diameter, transducer spacing, feedwater density, Crossflow time delay, pipe wall roughness, and the velocity profile correction factor (VPCF).

CENPD-397-P-A, Rev. 1, included typical uncertainties for each of the input parameters, except for the pipe wall roughness, and overall flow measurement uncertainty of the Crossflow UFM/UTM system for a typical feedwater loop (straight pipe, fully developed flow). Actual uncertainties are determined by the licensee on a plant-specific basis by using the guidelines and equations provided in the topical report. The uncertainties are affected by temperature change and, therefore, the topical report recommended improving the accuracy of the feedwater temperature instrumentation to reduce the total uncertainty of the feedwater flow measurement. The methodology specified additional correction factors to be applied to the VPCF of fully developed flow in a straight pipe when determining the VPCF for plant-specific conditions and pipe configuration.

In responding to NRC staff questions regarding the configuration of the Crossflow UFM system in the Hatch units, the licensee referenced a Westinghouse letter to the NRC dated January 18, 2002, (LTR-NRC-02-3, "Description of Crossflow^{XT} Ultrasonic Flow Measurement System"), which described a modification to the Crossflow UFM system (renamed the Crossflow^{XT} UFM system) that added additional signal conditioning capabilities and an additional set of four UFM transducers in the M/TSF, thereby, increasing the total number of UFM transducers in a M/TSF from four transducers (as described in CENPD-397-P-A) to eight transducers. NRC staff review and approval of the new Crossflow^{XT} UFM system design described in LTR-NRC-03-2 had not been requested by Westinghouse or the licensee, and the licensee had not referenced this Westinghouse letter in its initial submittal.

Although Westinghouse claimed in LTR-NRC-02-3 that the Crossflow^{XT} system will provide feedwater flow rates that have a statistic-based higher accuracy than the Crossflow system described in CENPD-397-P-A, the licensee did not credit this higher accuracy in its request for a MUR power uprate (of 1.5-percent RTP). Consequently, the NRC staff did not approve the analysis in LTR-NRC-03-2 that provided a statistical basis for future MUR power uprates in the Hatch units or other nuclear power plants.

The licensee provided uncertainty data and the resulting calculations of total CTP uncertainty. The licensee's submittal listed the contributions of individual error elements and stated that the calculations are based on accepted plant instrument uncertainty methodology. The calculation of total power measurement uncertainty using the Crossflow UFM system in combination with a CORRTEMP UTM for correcting the existing RTD-based feedwater temperature used the square root of the sum of the squares (SRSS) method to combine the individual error elements (various error elements related to Crossflow UFM system errors, pressure and moisture instrument errors, and other gains and losses). The NRC staff reviewed the calculations and found that the methodology for calculating CTP uncertainties was consistent with the SRSS method approved by the NRC staff in RG 1.105, Rev. 3, which endorsed ISA-S67.04, Part II-1994, and was, therefore, acceptable.

The licensee's calculations using a measured feedwater mass flow accuracy of 0.42 percent for the UFM/UTM system resulted in a total CTP uncertainty of -0.457 percent, +0.456-percent CTP for Unit 1, and -0.461 percent, +0.460-percent CTP for Unit 2. The NRC staff finds that the uncertainty values calculated by the licensee support the licensee's request for a 1.5-percent thermal power uprate (i.e., 2.0 percent minus 0.460 percent results in a potential margin recovery of 1.54 percent, which is greater than the licensee's requested 1.50-percent power uprate).

The Crossflow UFM/UTM system for this application was designed and manufactured under the Westinghouse quality control program, which provides for configuration control, deficiency reporting and correction, and maintenance. The current software was verified and validated under Westinghouse's Verification and Validation Program. Specific examples of quality measures included in the design, fabrication, and testing of the Crossflow system were provided in the topical report. Westinghouse's Verification and Validation program provides procedures for deficiency reporting for engineering action and notification of holders of Crossflow software.

The NRC staff's safety evaluation of CENPD-397-P, Rev. 01, included four additional criteria requirements to be addressed by a licensee requesting a power uprate. In the licensee's December 19, 2002, submittal, the licensee addressed each of the four criteria as follows:

Criterion 1: The licensee should discuss the development of maintenance and calibration procedures that will be implemented with the Crossflow UFM system installation. These procedures should include processes and contingencies for an inoperable Crossflow UFM/UTM system and the effect on thermal power measurement and plant operation.

In Enclosure 7, the licensee proposed a 72-hour allowed outage time (AOT) for the Crossflow UFM/UTM system (which includes the electronics and UFM transducers on a M/TSF and the associated CORRTEMP UTM electronics and transducer for that M/TSF) to provide sufficient time for troubleshooting, repair, calibration, and return of the system to operation. In response to a NRC staff request for clarification regarding the CORRTEMP UTM contribution to the feedwater temperature uncertainty, the licensee stated that, if a CORRTEMP UTM system became inoperable, the affected plant power would be reduced by the same amount as if a UFM system had become inoperable.

The NRC staff reviewed the licensee's uncertainty study presented in Attachment 1 to Enclosure 7. The uncertainty study described the methodology used to determine the uncertainties and weighting factors for the Crossflow UFM/UTM system instrumentation and the feedwater venturi instrumentation. The uncertainty analyses addressed the use of the existing feedwater venturis calibrated by an operable Crossflow UFM system and/or feedwater RTDs for performing the required heat balance calculations during the 72 hours immediately following a Crossflow UFM/UTM system outage. The feedwater venturi-based feedwater flow measurement uncertainty calculation included the effects of time-dependent drift, measurement and test equipment uncertainties, setting tolerance uncertainties, and the effect on process conditions such as temperature and pressure. Based on the results of this study, the licensee concluded that if the AOT of an inoperable Crossflow system was exceeded, the CTP reduction will be limited to 0.5 percent above the CLTP, or 2777 MWt, instead of 2763 MWt.

The licensee further stated that the power level at the start of a Crossflow system AOT would be maintained at 2804 MWt only if power is not significantly changed during the time interval. If the power is changed significantly during the outage, the power level would be reduced to an amount that is 0.5 percent above the CLTP value. This reduced power level value was based on the largest uncertainty in the power calculation using the feedwater venturis, which was 1.228-percent RTP for Unit 1. The NRC staff finds that requesting that the power level be reduced to an amount that is 0.5 percent above the CLTP on the basis of the accuracy of the existing feedwater flow venturis is acceptable because 10 CFR Part 50, Appendix K, does not specify the type of instrumentation that must be used to justify an increase in the RTP on the basis of measurement uncertainty recapture. Additionally, the NRC staff finds that reducing the power level to 0.5-percent above the CLTP in the event a UTM becomes inoperable is acceptable because the UTM contribution to the accuracy of the heat balance calculation is less than the UFM system contribution; consequently, the power reduction is conservative.

In Enclosure 7, the licensee stated that implementation of the power uprate license amendment will include developing the proposed procedures and documents necessary for operation, maintenance, calibration, testing, and training at the proposed power level with the new Crossflow UFM/UTM system. The licensee will revise plant maintenance and calibration procedures to incorporate Crossflow system maintenance and calibration standards prior to declaring the system operational and prior to increasing power above the current license thermal power level. In Enclosure 5 of the licensee's submittal, "Licensing Commitments," licensee commitment B.1 states that necessary maintenance and operational procedure revisions will be completed prior to implementation of the requested license amendment.

On the basis of the licensee's response to Criterion 1 in Enclosure 7 and the corresponding licensing commitments in Enclosure 5, the NRC staff finds the licensee's response to Criterion 1 to be acceptable.

<u>Criterion 2:</u> For plants that currently have the Crossflow system installed, the licensee should provide an evaluation of the operational and maintenance history of the installed UFM and

confirm that the instrumentation is representative of the Crossflow UFM and is bounded by the requirements set forth in Topical Report CENPD-397-P-A, Rev. 01.

In Enclosure 7, the licensee stated, "This criterion is not applicable to Plant Hatch, Units 1 and 2. Plant Hatch currently uses flow venturis for the feedwater flow measurement contribution to the CTP computation. The installation and operation of the Crossflow system is in anticipation of approval of the proposed amendment. Installation of the systems will be completed prior to implementation of the requested license amendment."

In Enclosure 5 of the licensee's submittal, licensee commitment A.1 stated that implementation of the Crossflow system will be completed prior to implementation of the requested license amendment and prior to raising the rated thermal power above 2763 MWt, the CLTP. Additionally, licensee commitment A.2 stated that validation of the assumed Crossflow system measurement uncertainty will be performed prior to implementation of the requested license amendment and prior to raising the power level above 2763 MWt.

On the basis of the licensee's response to Criterion 2 in Enclosure 7 and the corresponding licensing commitments in Enclosure 5, the NRC staff finds the licensee's response to Criterion 2 to be acceptable.

<u>Criterion 3:</u> The licensee should confirm that the methodology used to calculate the uncertainty of the Crossflow UFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternate methodology is used, the application should be justified and applied to both the venturi and the Crossflow UFM for comparison.

The licensee performed the heat balance uncertainty analyses of the Crossflow UFM/UTM system and the feedwater venturi system using the instrument uncertainty methodology described in ISA-RP67.04, Part II-1994. The feedwater temperature uncertainty was based on implementation of a CORRTEMP UTM feedwater temperature measurement system as an integrated part of the Crossflow UFM system for both the existing feedwater system and the proposed Crossflow system. Using the methodology described in ISA-RP67.04, Part II-1994, the licensee first established a baseline condition using heat balance measurement and process parameters in the plants. The licensee then varied each measurement and process parameter independently by a nominal error (e.g., ±1 °F feedwater temperature) to determine the affect (weighting factor) of that parameter on the CTP. The process parameter uncertainties were then multiplied by these weighting factors to obtain weighted measurement and process parameter contributions to the CTP calculation (e.g., ±0.1309-percent RTP/°F feedwater temperature in Hatch, Unit 1). These weighted contributions were then combined using a SRSS process to determine the random uncertainty of the heat balance calculation for both the Crossflow system and the feedwater venturi system.

Bias values for feedwater flow venturi fouling (for the feedwater flow venturi uncertainty analysis only) and RWCU flow (for both analyses) were then combined with the SRSS random uncertainties to determine the total CTP calculation uncertainty. Since application of the feedwater flow venturi fouling bias would result in less conservatism with respect to the total CTP calculation uncertainty, and application of the RWCU bias would result in more conservatism, the licensee applied the RWCU bias value to determine the total CTP calculation uncertainty.

The licensee provided tables listing the measurement and process uncertainties, the corresponding weighting factors, the contribution to the CTP calculation of the weighted parameter results of the uncertainty calculations, the total bias error, and the total CTP uncertainty. The NRC staff verified the process by which the total CTP uncertainties had been calculated, found the process to be consistent with the SRSS statistical methodology for combining uncertainties described in ISA-RP67.04, Part II-1994, that was endorsed by the NRC staff in RG 1.105, Rev. 3; and, therefore, found the process to be acceptable for both the feedwater flow venturi CTP calculation uncertainties and the Crossflow system CTP calculation uncertainties for Hatch, Units 1 and 2.

Criterion 4: The licensee of a plant at which the installed Crossflow UFM was not calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation) should provide additional justification. This justification should show either that the meter installation is independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibration and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated Crossflow UFM, the licensee should confirm that the plant-specific installation follows the guidelines in the Crossflow UFM system topical report.

In Enclosure 7, the licensee stated that both Unit 1 and Unit 2 calibrations of the UFMs will be performed with a mockup laboratory installation that models the actual plant configuration. A further clarification from the licensee revealed that the laboratory hydraulic modeling was created to calibrate the UFM system in Hatch, Unit 1. The licensee found that the Hatch, Unit 2 piping configuration at the location of the M/TSF installations and test data confirmed that the feedwater flow was fully developed at the location of the UFM system. Therefore, laboratory calibration of the Hatch, Unit 2 UFM system was not necessary.

For Hatch, Unit 1, the feedwater piping has a nonstandard limited run of straight pipe for calibrating the UFM system in situ; therefore, it was necessary for the licensee to perform a laboratory calibration of the Unit 1 UFM system. The laboratory calibrations of the Hatch, Unit 1 UFM systems were performed at the National Research Center of Canada in Ottawa using the weigh tank method and standards traceable to a National Institute of Standards and Technology reference. The test procedure consisted of two configuration tests.

The first test was a baseline test, in which UFM systems were placed along the model piping downstream of a simple 90 degree elbow. This represented a baseline test configuration, in which the correction factor 15 pipe diameters downstream of the pipe elbow is only a function of the Reynolds number under the field conditions, as described in CENPD-397-P-A.

The second test configuration differed from the baseline piping configuration in that the 90 degree piping elbow was replaced with a 90 degree elbow followed immediately by a 45 degree elbow, which is the piping configuration in Hatch, Unit 1. The difference between the plant configuration and the baseline configuration represented the Hatch, Unit 1 VPCF. The licensee concluded that the flow profile and meter factors would be representative of the plant-specific installation, and that no additional justification would be required to meet Criterion 4.

The methodology for determining the VPCF in Hatch, Unit 1 was consistent with the VPCF methodology approved by the NRC staff in CENPD-397-P-A, Rev. 01. The in situ calibration methodology to be used in Hatch, Unit 2 was also approved by the NRC staff by its approval of CENPD-327-P. The NRC staff, therefore, concludes that the methodologies used by the licensee for calibrating the Crossflow UFM systems in the Hatch plants are acceptable.

In Enclosure 5, licensee commitment A.2 states that the licensee will validate the assumed Crossflow UFM system measurement uncertainty prior to implementing the requested license amendment and prior to raising the power level above 2763 MWt (the CLTP).

On the basis of the licensee's confirmation that the Crossflow UFM system will be equivalent to known calibration and plant configurations for Hatch, Units 1 and 2, and the licensee's validation of the Crossflow UFM system performance prior to implementing the license amendment, the NRC staff finds the licensee's response to Criterion 4 to be acceptable.

In addition to the four criteria stated above, RIS 2002-03 indicated that the licensee should provide information that specifically addressed the aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric (heat balance). These aspects included maintaining calibration, controlling the software and hardware configuration, performing corrective actions, reporting deficiencies to the manufacturer, and receiving and addressing manufacturer deficiency reports.

In Enclosure 7, the licensee stated that calibration and maintenance will be performed using site procedures developed from the Crossflow system technical manual and plant operating and maintenance manuals. The licensee further stated that all work would be performed in accordance with site work control procedures. Additionally, verification of acceptable Crossflow system operation would be provided by local onboard system diagnostics. Furthermore, calibration of other instrumentation that contributes to the power calorimetric calculation would be performed periodically. Measurement and test equipment, setting tolerances, calibration frequencies, and instrumentation accuracy were evaluated by the licensee and accounted for in the licensee's uncertainty evaluation described above. In Enclosure 5, licensee commitment B.1 stated that necessary maintenance and operational procedure revisions will be completed prior to implementation of the requested license amendment.

On the basis of the information presented in Enclosure 7, which is consistent with the RIS 2002-03 criteria, and licensee commitment B.1, the NRC staff finds acceptable the licensee's planned maintenance and calibration activities to be performed with the Crossflow UFM/UTM system.

In Enclosure 7, the licensee stated that the software and hardware configuration for the Crossflow system and all other instrumentation that affect the CTP calorimetric will be controlled through the plant modification process that will provide for a licensee evaluation of any changes necessary to conform with the licensee's uncertainty study for the Crossflow system. The licensee further stated that changes to hardware and software are evaluated through the licensee's 10 CFR 50.59 process. The NRC staff finds the process by which the licensee will control the hardware and software changes via the 10 CFR 50.59 process to be acceptable.

In Enclosure 7, the licensee stated that corrective actions involving maintenance will be performed by instrumentation and control (I&C) maintenance personnel, who will be qualified in accordance with the Hatch I&C Training Program, and who will be formally trained on the

Crossflow system. In Enclosure 5, licensee commitment B.1 stated that necessary maintenance and operational procedure revisions will be completed prior to implementation of the requested license amendment.

On the basis of the information presented in Enclosure 7, which is consistent with the RIS 2002-03 criteria, and licensee commitment B.1, the NRC staff finds acceptable the licensee's planned maintenance training activities to be performed on the Crossflow UFM/UTM system.

In Enclosure 7, the licensee stated that although the Crossflow system is non-safety related for this application, corrective actions will be performed by qualified maintenance personnel in accordance with the licensee's Condition Reporting system. Additionally, reliability of the Crossflow system will be monitored by licensee personnel in accordance with the provisions of the licensee's Condition Reporting system. The licensee further stated that the Crossflow system purchase order includes the provision that the Crossflow system vendor (Westinghouse) shall inform the licensee of any deficiencies in accordance with maintenance agreement reporting provisions. Manufacturer deficiency reports are addressed by provisions in the licensee's Condition Reporting system. These activities are consistent with the requirements of 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program," which requires, in part, that the licensee shall identify the structures, systems and components that shall be covered by the quality assurance program; and activities affecting quality shall be controlled under suitably controlled conditions. The NRC staff, therefore, finds the licensee's control over the Crossflow system corrective actions to be acceptable.

The NRC staff finds that the licensee sufficiently resolved the plant-specific concerns about Crossflow UFM/UTM maintenance and calibration, hydraulic configuration, processes and contingencies for an inoperable Crossflow UFM. Additionally, the licensee used an approved methodology to calculate the plant-specific Crossflow UFM/UTM system measurement uncertainty, the venturi-based feedwater flow measurement uncertainty, and the power measurement uncertainty. The NRC staff concludes, therefore, that the licensee has addressed in an acceptable manner the criterion in RIS 2002-03, and the criteria set forth in the NRC safety evaluation approving Topical Report CENPD-397-P-A, Rev. 1, regarding implementation of the Crossflow UFM system in Hatch, Units 1 and 2.

3.4.2 Instrumentation and Controls - Conclusion

The licensee's justifications for implementing a Crossflow UFM/UTM system in the Hatch, Units 1 and 2 plants to reduce the heat balance measurement uncertainty and thereby allow an increase in reactor power by 1.5-percent RTP are in accordance with Appendix K to 10 CFR Part 50, the guidance provided in RIS-2002-03, topical reports that have been approved by the NRC staff, and the licensee's uncertainty analyses. Based on its evaluation, the NRC staff finds the proposed change acceptable in regard to the impact of the proposed power uprate on instrumentation and controls.

3.5 Electrical Power and Auxiliary Systems

3.5.1 Alternating Current (ac)/Off-site Power

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covers the information, analyses, and documents for the offsite power system and the stability studies for the electrical transmission grid. The focus of the review relates to the basic acceptance criterion that the loss of the nuclear unit, the largest operating unit on the grid, or the loss of the most critical transmission line will not result in the loss of offsite power to the plant. Branch Technical Position Instrumentation and Control System Branch (BTP ICSB) 11, "Stability of Offsite Power Systems," outlines an acceptable approach to addressing the issue of stability of offsite power systems. Acceptance criteria are based on GDC 17, "Electric Power Systems," of Appendix A to 10 CFR Part 50. Specific review criteria are contained in standard review plan (SRP) Sections 8.1 and 8.2, Appendix A to 8.2 and BTPs Power System Branch-1 and ICSB-11.

As described by the licensee's amendment request dated December 19, 2002, the main generator on each unit is re-rated at 1050 million volt amps (MVA) with the EPU effort. The main generator provides power through the isolated phase bus at 24 kV to both the main transformer and the unit auxiliary transformers. The preferred ac power source provides offsite ac power to the auxiliary power distribution system for the startup, operation, or shutdown of the station. The preferred ac power also provides a source of offsite ac power to all emergency loads necessary for the safe shutdown of the reactor. The electrical distribution system has been previously evaluated to conform to GDC 17.

Grid Stability

An EPU was implemented for Hatch, Unit 1 in 1999 and for Hatch, Unit 2 in 1998 and involved re-rating the main generator and support systems as well as performing a detailed grid stability analysis. The stability analysis performed for the EPU identified only one contingency for which the critical clearing time (CCT) was approaching the actual breaker failure clearing time (BFCT). The particular contingency in question was for a three-phase fault on the low side of the 500-230 kV auto-transformer with a failure of breaker 540 to operate. The CCT for this contingency was found to be 10 cycles while the actual BFCT was 9.5 cycles. This represented the bounding contingency to be evaluated for the TPO. The results of the study indicated that for the TPO, the bounding contingency remains acceptable. Evaluations were performed for the generator control and protection, and the generator protective relays. These evaluations concluded that TPO does not necessitate any changes to the relays. An analysis was also performed to identify the maximum expected VARs out of the generators at the TPO conditions. At the TPO conditions, the maximum expected VARs out are within the generator reactive capability curves indicating the generators will remain stable at the TPO conditions thereby maintaining grid stability. A calculation was performed to verify that the generator operating range is still a minimum of 95 percent to a maximum of 100 percent of the 24 kV rated generator voltage for both Hatch, Units 1 and 2. A steady state voltage analysis was performed for the TPO conditions for a peak load (2003). The analysis used the two worst case scenarios for summer peak conditions and the 2002-2003 worst case valley loading conditions that were identified in the 2002 Hatch steady state UFSAR study. The analysis concluded that the proposed TPO meets the minimum 230 kV bus voltage criteria of 101.3-104.9 percent. Therefore, the TPO will have an insignificant impact on grid stability.

The NRC staff reviewed the licensee's submittal and concludes that the impact of the power uprate on the grid stability is insignificant. Therefore, the plant continues to be acceptable with respect to grid stability with this power uprate.

Main Generator

The TPO of 1.5 percent will increase the maximum Unit 1 generator to 1025 MVA and the Unit 2 generator output to 1032 MVA. The EPU effort re-rated the generators to 1050 MVA. The output for generators is bounded by the previous EPU re-rate. At the TPO conditions, the maximum expected VARs out are within the generator reactive capability curves of each unit indicating the generator of each unit will remain stable at the TPO conditions

The NRC staff reviewed the main generator capability curve of each unit and concludes that the generators will continue to operate at the anticipated power uprate and, therefore, the design is acceptable.

Main Power Transformer

The ratings of the main generator step-up transformers are 1008 MVA for Unit 1 and 998 MVA for Unit 2. At maximum generator MWe output and normal expected MVARs being absorbed, the expected loading on the main generator step-up transformers is 917 MVA for Unit 1 and 955 MVA for Unit 2. Since these loads are less than the transformer ratings, the transformers are capable of handling the TPO loadings.

The NRC staff reviewed the licensee's submittal and concludes that the anticipated power uprate of 1.5 percent is below the maximum main transformer design rating of each unit and, therefore, operating the main power transformers at the uprated power condition is acceptable.

Isophase Bus

The EPU effort also re-rated the isophase busses to 1050 MVA. The re-rated current through the isophase busses at 1050 MVA is 26,588 amps. The maximum expected current through an isophase bus is 26,133 amps. This is bounded by the re-rated current value. Normal expected generator output is much less than the 1032 MVA (956 MVA for Unit 1 and 996 MVA for Unit 2) resulting in substantially lower current through the isophase busses.

The NRC staff reviewed the licensee's submittal and concludes that the impact of power uprate of 1.5 percent is below the design rating of the isophase bus of each unit and, therefore, operating the isophase bus at the uprated power condition is acceptable.

Startup Transformer (SUT)

The SUTs are rated as follows:

SUT 1C and 2C 28.0 MVA (FOA @ 65 °C)
SUT 1D and 2D 33.6 MVA (FOA @ 65 °C)

At the current license power level, the maximum load on SUT 1C or 2C is approximately 22 MVA. The maximum increase in loading due to the TPO on SUT 1C or 2C is about 24 kilowatt (kW) from the recirculation motor generator (MG) set motor drives. The resulting MVA is much less than the transformer rating of 28 MVA. There is no increase in emergency loads due to TPO. The maximum emergency load represented by the 4.16 kV busses is approximately 9.8 MVA. Therefore, SUT 1 C and 2C are also adequately rated to carry the maximum emergency loads. At the current license power level, the maximum load on SUT 1D or 2D is 19.1 MVA. The maximum expected total load due to the TPO on SUT 1D or 2D is 19.2 MVA. With this load increase, the load on SUT ID or 2D remains much less than the rating of 33.6 MVA.

The NRC staff reviewed the licensee's submittal and concludes that the startup transformers loading resulting from the 1.5-percent power uprate is below their maximum design rating and, therefore, operating the startup transformers at the uprated power condition is acceptable.

Unit Auxiliary Transformer (UAT)

The UATs are rated as follows:

UAT IA and 2A 33.6 MVA (FOA @ 65 °C)
UAT lB and 2B 28.0 MVA (FOA @ 65 °C)

At the current license power level, the maximum load on UAT lA or 2A is 19.1 MVA. The maximum expected total load due to the TPO on UAT lA or 2A is 19.2 MVA. With this load increase, the load on UAT IA or 2A remains much less than the rating of 33.6 MVA. At the current license power level, the maximum load on UAT lB or 2B is approximately 22 MVA. The maximum total load due to the TPO on UAT lB or 2B is about 22.1 MVA from the recirculation MG set motor drives. The resulting MVA is much less than the transformer rating of 28 MVA. Since the resulting MVA for all UATs is much less than the transformer ratings, the UATs are capable of supporting the additional load imposed by the TPO. The NRC staff reviewed the licensee's submittal and concludes that the unit auxiliary transformer loading resulting from the 1.5-percent power uprate is below its maximum design rating and, therefore, operating the unit auxiliary transformers at the uprated power condition is acceptable.

The NRC staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the offsite power system and concludes that the offsite power system will continue to meet the requirements of GDC 17 following implementation of the proposed power uprate. The NRC staff further concludes that the impact of the proposed power uprate on grid stability is insignificant. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the offsite power system.

3.5.2 AC/Onsite Power

The ac onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to the safety-related equipment. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the ac onsite power system. Acceptance criteria are based on GDC 17 as it relates to the capability of the ac onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

The emergency diesel generators (EDGs) are each rated at 3200 kW. The ac onsite power system for units 1 and 2 consists of five EDGs 1A, 1B, 1C, 2A, and 2C. EDGs 1A and 1C supply Unit 1 emergency buses 1E and 1G respectively. EDGs 2A and 2C supply Unit 2 emergency buses 2E and 2G respectively. EDG 1B is a shared unit and can supply either Unit 1 or Unit 2 on complete loss of offsite power. The load distribution on emergency buses during a LOCA is about 3100 kW on each of the buses 1E, 1F, 2E and 2F, and 3136 kW on Bus 1G. The EDGs automatically supply ac power to the Class 1E buses in order to provide motive and control power to equipment required for a safe shutdown of the plant. Station loads under accident conditions are based on equipment nameplate data. The auxiliary loads may experience a small increase in flow and pressure due to power uprate. Because, these changes are small, the motor demand for each of these loads remain bounded by the existing design. Therefore, under accident conditions, the ac onsite electrical power system is adequately sized.

The NRC staff has reviewed the licensee’s submittal for the effect of the proposed power uprate on the ac onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system’s functional design. The NRC staff further concludes that the ac onsite power system will continue to meet the requirements of GDC 17 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the onsite ac power system.

3.5.3 Direct Current (dc) Power

The dc power systems include those dc power sources and their distribution systems and auxiliary supporting systems provided to supply motive or control power to safety-related equipment. The NRC staff’s review covers the information, analyses, and referenced documents for the dc onsite power system. Acceptance criteria are based on GDC 17 and 10 CFR 50.63 as they relate to the capability of the dc onsite electrical power to facilitate the functioning of structures, systems, and components important to safety. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2

The licensee reviewed the dc loading requirements and did not identify any reactor power-dependent loads. Operation at the uprated power level does not increase any loads or revise control logic.

The NRC staff has reviewed the licensee’s submittal for the effect of the proposed power uprate on the dc onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system’s functional design. The NRC staff further concludes that the dc onsite power system will continue to meet the requirements of GDC 17 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the dc onsite power system.

3.5.4 Fuel Pool

The fuel pool cooling and cleanup system (FPCCS) removes heat from the spent fuel assemblies stored in the spent fuel pool (SFP) in order to maintain the pool temperature at, or below, its design temperature during normal plant operations. In addition, the FPCCS reduces activity, maintains water clarity, and maintains the cooling function during and after a seismic event.

The licensee stated that the fuel pool heat load increases slightly as a result of the power uprate. However, the new heat load is within the design basis heat load for the FPCCS, and it will not result in a delay in removing the RHR system from service (i.e., the duration of supplemental cooling will not be increased). The licensee determined that the SFP cooling is adequate by calculating the heat load generated by a full-core discharge plus remaining space filled with spent fuel discharged at regular intervals.

Regarding other fuel pool design considerations, the crud activity and corrosion products in the SFP can increase slightly; however, the licensee determined that this increase is insignificant and the water quality will be maintained by the FPCCS. In addition, the licensee determined that the normal radiation levels around the SFP may increase slightly; however, the licensee determined that the increase will not significantly increase the operational doses to personnel or equipment. Also, there is no effect on the design of the spent fuel racks because the original SFP design temperature is not exceeded.

Based on the NRC staff's review of the licensee's evaluation against the CLB and experience gained from the review of power uprate applications for similar BWR plants, and in view of the foregoing, the NRC staff finds that the FPCCS, in combination with the RHR system, can maintain the SFP temperature at, or below, design limits for all offload conditions at the proposed 1.5-percent proposed power uprate level.

3.5.5 Service Water Systems

The licensee stated that the safety-related plant service water (PSW) system serves as the heat sink for all systems cooled by either the safety-related and non-safety related systems during all planned operations in all operating states, and to provide makeup to the plant circulating water system. Each of the two divisions of PSW supply cooling water to one redundant train of safety-related equipment. After passing through isolation valves, the two safety-related headers combine into one header that supplies the non-safety-related equipment.

The licensee evaluation of the PSW system following the implementation of the power uprate indicates that the increase in heat loads for the components affected are within the existing design heat loads, except for the reactor building closed cooling water (RBCCW) heat exchangers. The licensee stated that the evaluation indicates that the PSW is capable of handling the additional demand from the RBCCW heat exchanger resulting from the power uprate without any significant effect on the PSW outlet temperature. The evaluation indicates an increase of $<0.1\ ^{o}F$ in PSW discharge temperature, therefore, the system is adequate for the power uprate conditions.

The licensee stated that the main condenser, circulating water, and normal heat sink systems are designed to remove the heat rejected to the condenser and maintain a low condenser pressure. The 1.5-percent power uprate increases the heat rejected to the condenser and may reduce the difference between the operating pressure and minimum condenser vacuum; however, the licensee's evaluation of the design duty over the actual yearly range of circulating water inlet temperatures determined that the condenser, circulating water system, and the heat sink are adequate for the power uprate. The licensee stated that the plant will remain within the state thermal discharge limits during operation at power uprate conditions.

The licensee stated that the heat loads on the RBCCW system do not increase significantly, except for the increase in the fuel pool heat load (1.5 percent) and a slight increase in the heat loads from the reactor recirculation pump and MG set coolers. The evaluation indicates that there will be a small increase in the total heat load for the RBCCW heat exchanger; however, it will be within the design capacity of the system. After implementation of the power uprate, there will be no significant effect on the outlet water temperature of the heat exchanger. The evaluation concluded that all the components served by the RBCCW heat exchanger will be adequately cooled without any effect on the RBCCW design conditions after the power uprate.

The chilled water systems for the reactor/radwaste building, primary containment, turbine building, and control building provide the cooling water to the non-safety-related coolers located in the respective buildings. The licensee's evaluation for the power dependent heating, ventilation, and air conditioning (HVAC) systems indicate that there is no increase in the heat loads for any of these buildings. The licensee concluded that the design of the building chilled water systems is not affected by the power uprate.

The ultimate heat sink (UHS) is provided by the Altamaha River. The licensee has determined that as a result of operation at the new power level, the post-LOCA UHS water temperature increases slightly, primarily due to higher reactor decay heat. This results in a higher UHS evaporation rate. The licensee stated that the existing UHS system provides a sufficient quantity of water at a temperature $\leq$ 95 ^{o}F (design temperature) following a design basis LOCA. As stated above, the plant will remain within the state thermal discharge limits during operation at power uprate conditions.

Based on the NRC staff's review of the licensee's evaluations and the experience gained from the review of power uprate applications for similar BWR plants, the NRC staff has determined that operation at the proposed 1.5-percent proposed power uprate level does not change the design aspects and operation of the PSW systems, and that operation of the plant's water system will remain bounded by current analyses. Therefore, the NRC staff finds that the impact of plant operations at the proposed power uprate level of these systems acceptable.

3.5.6 Standby Liquid Control System

The SLC system provides the alternate means of attaining and maintaining cold shutdown conditions, assuming no control rod movement, as required by GDC 26, "Reactivity Control System Redundancy and Capability." The shutdown capability of the SLC system and the necessary boron solution are evaluated each reload cycle. Since the SRV setpoints are not changed for the proposed power uprate, the uprate will have no effect on the rated injection flow. The licensee determined that the capability of the SLC system to provide its backup shutdown function is unchanged and the system will continue to meet the requirements of 10 CFR 50.62. Because the uprate will not change the operating parameters of the SLC system, the NRC staff determined that the SLC continues to meet applicable requirements.

3.5.7 Power Dependent Heating, Ventilation and Air Conditioning

The function of the HVAC systems is to prevent extreme thermal environmental conditions from impacting personnel and equipment by ensuring that design temperatures are not exceeded. The licensee stated that the proposed power uprate will cause the recirculation pump and MG sets to contribute a slight increase in the heat load to the HVAC system. The FW piping temperature is expected to increase by approximately 2 °F.

The licensee stated that the increase in surface temperature of the pipe will be minimal because the FW piping is insulated. The increased temperature will still be within the temperature range specified by the piping insulation design specifications with the result that there is no effect on the specified heat losses from the piping. The increased heat load, due to the increased horsepower specifications of the recirculation pump and MG sets, will still be within the design requirements of the drywell coolers. Other areas are unaffected by the power uprate because the process temperatures and electrical heat loads are not impacted.

Based on the NRC staff's review of the licensee's evaluation and the experience gained from the review of power uprate applications for similar BWR plants, and in view of the above, the NRC staff has determined that operation of the HVAC systems remains acceptable at the proposed power uprate conditions.

3.5.8 Fire Protection

The licensee stated that fire detection and suppression systems are not expected to be impacted by plant operation at the proposed 1.5-percent power uprate power level since there are no physical plant configuration or combustible load changes resulting from the proposed power uprate operations. In addition, the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and the operator actions necessary to mitigate the consequences of a fire are not affected by the proposed power uprate conditions.

Based on the NRC staff's review of the licensee's evaluation and the experience gained from the review of power uprate applications and fire protection programs for similar BWR plants, the NRC staff finds that the safe shutdown systems and procedures used to mitigate the consequences of a fire will continue to meet 10 CFR 50.48 and 10 CFR Part 50, Appendix R, and will not be affected by plant operation at the proposed 1.5-percent power uprate level.

3.5.9 Electrical Power and Auxiliary Systems - Conclusions

Based on the evaluation in Sections 3.5.1 through 3.5.8, the NRC staff concludes that the proposed power uprate is acceptable with respect to its impact on the Hatch electrical power and auxiliary systems.

3.6 Power Conversion Systems

3.6.1 Turbine-Generator

The turbine-generator is designed with a maximum flow-passing and generator capability at rated conditions to ensure that the design rated output is achieved. The NRC approved the EPU of 8 percent for Hatch, Units 1 and 2 on October 22, 1998. The licensee, in evaluating for the EPU, performed steam specification calculations to determine the turbine steam path conditions for the current licensed thermal power. The thermodynamic models evaluated the turbine and generator stationary and rotating components for increased loadings, pressure drops, thrusts, stresses, overspeed capability, and other design considerations to ensure that design limits were not exceeded and that operation remained acceptable at the EPU conditions. These evaluations also included valves, control systems, and other support systems. These evaluations were performed for 102 percent of the rated conditions. The excess capacity, or flow margin, ensured that the turbine generator can meet rated conditions for continuous operating capability with allowances for variations in flow coefficients from expected values, manufacturing tolerances, and other variables that may affect the flow-passing capability of the unit. The increased throttle steam flows is approximately 101.3 percent (Hatch, Unit 1), and 100.8 percent (Hatch, Unit 2) of current rated flow. The small increase in flow, temperature, and pressure are bounded by the flow margins that were included in the EPU evaluations and, therefore, the turbines will support operation at the power uprate conditions. The turbine generator values and performance will be monitored during power ascension testing and at the new power uprate conditions during normal operation. The rotor missile analysis for the EPU were performed for a 10-percent power uprate, therefore, this power uprate of 1.5 percent is bounded by that previous analysis. The licensee stated that the overspeed calculations performed for the EPU compared the entrapped steam energy contained within the turbine and associated piping, after the stop valves trip and the sensitivity of the rotor train for the capability of overspeeding. The licensee further indicated that although the entrapped energy increased slightly for this power uprate, no change in the overspeed trip settings is necessary.

The NRC staff reviewed the licensee’s evaluation and concludes that the turbine will operate within the current capacity limits for the electrical output at the proposed power uprate level. Therefore, the existing rotor missile and turbine overspeed analyses remain acceptable for operation at the proposed power uprate conditions.

3.6.2 Condenser and Steam Jet Ejectors

The licensee evaluated the impact of the power uprate on condenser performance based on current circulating water system flow. The licensee’s evaluation determined that the design margin in the condenser heat removal capability can accommodate the additional heat rejected for operation at the proposed power uprate. Additionally, the licensee determined that air leakage into the condenser does not increase as a result of the proposed 1.5-percent power uprate, and the small increase in hydrogen and oxygen flows from the reactor do not effect the steam jet air ejector performance because the design was based on operation at flows significantly greater than those needed at the proposed uprated power.

The NRC staff reviewed the licensee’s evaluation and concludes that the operation of the condenser and steam jet ejectors at the proposed uprated power level is bounded by the current design capabilities of the condenser and steam jet ejectors. Therefore, the existing condenser and steam jet ejectors remain adequate.

3.6.3 Turbine Bypass System (TBS)

The TBS is designed to control steam pressure when reactor steam generation exceeds turbine specifications during unit startup, sudden load reductions, and cooldowns. The TBS allows excess steam flow from the reactor to the condenser without going through the turbine. The system is designed so that sudden load reductions within the capacity of the TBS can occur without a reactor scram. The TBS was originally designed for a steam flow capacity of a nominal 25 percent, for Hatch, Unit 1, and 23.5 percent, for Hatch, Unit 2. Following the EPU, the steam flow capacity for Hatch, Unit 1 was 21.4 percent, and Hatch, Unit 2 was 20.6 percent.

The steam bypass capacity at the proposed power uprate is a nominal 21.2 percent (Hatch, Unit 1), and 20.2 percent (Hatch, Unit 2) of the 100-percent power uprate RTP steam flow rate. The transient analyses that credit the turbine bypass system availability use the absolute flow capacity. The licensee stated in section 9.1 of NEDC-33085P that the transient is based on the generic evaluations of Appendix E of the TLTR and that the evaluations and conclusions of Appendix E are applicable to Hatch. Appendix E demonstrated that the effect of the power uprate is small. Additionally, the licensee will confirm the results of the limiting analysis as part of the normal reload analysis.

Based on the NRC staff's assessment of the licensee's evaluation against the criteria and analyses in the TLTR, which demonstrates that the transient analyses which credit the TBS indicate that the effect of the power uprate is small and experience gained from the review of power uprate applications for similar BWR plants, the NRC staff concludes that operation of the steam bypass pressure control system remains acceptable at the uprated conditions.

3.6.4 Feedwater and Condensate Systems

The FW and condensate systems are not safety-related; however, their performance may have an effect on plant availability and the capability to operate reliably at the uprated power conditions. The licensee reviewed the HNP FW heaters, heater drains, condensate demineralizers (CDs), and FW and condensate pumps to demonstrate that the components are capable of performing in the proper design range and, therefore, provide the slightly higher flow rate for the proposed power uprate conditions at the desired temperature and pressure. Additionally, the licensee determined that the FW control valves are capable of maintaining water level control at the proposed power uprate level. The licensee evaluated the operation of the FW and condensate systems and found that sufficient design margin exists in both systems for operation during normal and transient conditions at the proposed power level.

The NRC staff reviewed the licensee's evaluation against the CLB and has determined that the FW and condensate system design bounds the operating conditions at the proposed power uprate level. Accordingly, the NRC staff concludes that operation of the FW and condensate systems remains acceptable at the proposed power uprate level.

3.6.5 Condensate Demineralizers

The licensee evaluated the impact of the power uprate on the CDs and determined that no measurable effect results from the power uprate. The licensee determined that the CDs will experience no pressure increase at the power uprate conditions. The licensee stated that the CD resin run lengths are limited by their total integrated flow processing capacity (total gallons through the vessel) such that in-service lifetimes should be inversely proportional to condensate flow. Although the run time may slightly decrease, this is not a significant radwaste burden.

The NRC staff reviewed the licensee's submittal and, based on its knowledge of CD design and operation, has determined that the reduction in resin run time is reasonable. Accordingly, the NRC staff concludes that operation of the CDs remain acceptable at the proposed power uprate level.

3.6.6 Power Conversion Systems - Conclusion

Based on the NRC staff's review, as discussed in Sections 3.6.1 through 3.6.5 above, the NRC staff finds that the power conversion systems can accommodate plant operations at the proposed 1.5-percent power uprate level. Therefore, the NRC staff finds that operation of the power conversion systems at the proposed power uprate level remains acceptable.

3.7 Radwaste and Radiation Sources

3.7.1 Liquid and Solid Waste Management

The liquid radwaste system collects, monitors, processes, stores, and returns processed radioactive waste to the plant for reuse, discharge, or shipment. In its application, the licensee stated that the backwash of the CD pre-filters, cleaning of CD polisher resins, and replacement of CD resin are the largest source of liquid and solid waste. The 2-percent increased flow rate due to the power uprate results in a reduction in time between backwashes and cleanings, however this does not affect plant safety and is not a significant radwaste burden. The licensee also stated that the RWCU filter demineralizer may require more frequent replacements, due to slightly higher levels of activation and fission products at the increased power level. The floor drain collector and waste collector subsystems will not experience a significant increase in volume due to operation at the proposed 1.5-percent power uprate level.

The licensee concluded that the activated corrosion products in the liquid wastes are expected to increase proportionally to the power uprate, and the total volume of processed waste is not expected to increase significantly as a result of the proposed power uprate. The licensee also concluded that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, will continue to be met based on their assessment of the plant operating effluent reports and the slight increases expected from the 1.5-percent power uprate.

Based on the NRC staff's review of the licensee's evaluation against the CLB, the NRC staff concludes that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, applicable to the liquid radwaste management system, will continue to be satisfied at the proposed 1.5-percent increase in power level since there will only be a slight increase in the volume processed by the system, and a slight increase in the activation and fission products in the liquid wastes.

3.7.2 Gaseous Waste Management

The gaseous waste systems collect, control, process, and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. The gaseous waste management systems include the offgas system and various building ventilation systems that function to reduce radioactive gaseous releases from the plant.

The licensee stated that the amount of fission products released into the coolant is dependent on the number and nature of fuel rod defects and is not dependent on reactor power. Therefore, the licensee concluded that the activity of airborne effluents released through building vents is not expected to increase significantly due to the proposed 1.5-percent power uprate. The licensee administratively controls radioactive releases which, therefore, are not a function of reactor core power. The impact of the 1.5-percent power uprate on the reactor fuel is addressed in Section 3.1 of this SE.

The licensee evaluated the impact of the 1.5-percent power uprate on the offgas system, including the effects of hydrogen water chemistry and noble metal injection. The licensee stated that the increases in H_2 and $O_{2,}$ due to the power uprate, remain well within the capacity of the system and the system radiological release rate is administratively controlled. Therefore, the licensee concluded that gaseous effluents are expected to remain well within the release limits following implementation of the 1.5-percent power uprate.

Based on a review of the licensee's evaluation against the CLB, the NRC staff concludes that the gaseous radwaste management system will continue to operate acceptably at the proposed uprated conditions, since the activity of the airborne effluents is not expected to increase significantly and releases are administratively controlled.

3.7.3 Radwaste and Radiation Sources - Conclusion

Based on the evaluation in Sections 3.7.1 and 3.7.2 of this SE, the NRC staff concludes that the proposed power uprate is acceptable with respect to its impact on radwaste and radiation sources.

3.8 Reactor Safety Performance Evaluation

3.8.1 Reactor Transients

AOOs are abnormal transients that are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, GDC 10, 15, and 20. GDC 10 requires that the reactor core and associated control and instrumentation systems be designed with sufficient margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation and during AOOs. GDC 15 requires that sufficient margin be included to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operating conditions and AOOs. GDC 20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure the specified fuel design limits are not exceeded during normal operating conditions and AOOs.

The SRP provides further guidelines: (1) pressure in the reactor coolant and MS system should be maintained below 110 percent of the design values according to the ASME Code, Section III, Article NB-7000, "Overpressure Protection;" (2) fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during normal operating conditions and AOOs; (3) an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and (4) an incident of moderate frequency, in combination with any single active-component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel cladding perforations are acceptable.

Chapter 15 of the Hatch plant UFSAR contains the design basis analyses of the effects of an AOO resulting from changes in the system parameters such as (1) a decrease in core coolant temperature, (2) a increase in reactor pressure, (3) a decrease in reactor coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, and (6) a decrease in reactor coolant inventory. The facility's responses to the most limiting transients are analyzed each reload cycle and corresponding changes in the MCPR are added to the SLMCPR to establish the OLMCPR. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

The licensee will perform the reload analysis at the uprated conditions using an NRC-approved methodology. The licensee determined that the thermal limits to ensure the fuel cladding integrity will be maintained for operation at the uprated conditions during AOOs and accidents. Since the licensee is using an NRC-approved methodology, with appropriate cycle-specific inputs, the NRC staff concludes that deferring the limiting transient analyses to the reload analysis for the cycle implementing the power uprate is acceptable with regard to the impact of AOOs at the proposed power uprate level.

3.8.2 Design Basis Accidents

The potential radiological consequences of DBAs are proportional to the quantity of radioactive material released to the environment. This release is a product of the radioactive material released from the core or from the reactor coolant system (RCS), the transport of the released material to the effluent release point, and the transport (e.g., atmospheric dispersion) in the environment. The transport in the environment is not considered further as it is not affected by the power uprate. In general, the inventory of fission products in the core and the quantity of radioactive material in the RCS is directly proportional to the operating power level. An increase in the RTP, as proposed here, can be expected to increase the inventory of radioactive material available for release. The transport of the released material is dependent on plant process parameters, such as process stream flows, temperatures, and pressures. An increase in the RTP and any associated plant modifications could affect the assumptions made in previous consequence analyses.

The NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of design basis accidents, performed by SNC in support of its proposed license amendment. Information regarding these analyses was provided in Enclosure 8 to the application, NEDC-33085P. Section 9.2, "Design Basis Accidents," of this enclosure addressed the impact of the proposed power uprate on previously analyzed radiological consequences of DBAs. By letter dated August 8, 1997, SNC proposed license amendments that would increase the maximum licensed thermal power by 8 percent, from the then current 2558 MWt to the current 2763 MWt. In support of that application, the licensee re-evaluated the radiological consequences of affected DBAs at an uprated reactor power level of 2818 MWt (102 percent of the requested increase in rated thermal power to 2763 MWt). That request was approved for Hatch, Units 1 and 2 as Amendments 214 and 155 respectively. These analyses assumed a measurement uncertainty of 2 percent. The measurement uncertainty for the Crossflow instrumentation is about ±0.5 percent. The sum of the proposed power increase of 1.5 percent and the Crossflow measurement uncertainty is about 2 percent. As such, analyses originally performed using a power level equivalent to 102 percent of RTP continue to be bounding and re-analysis is not necessary. The remaining UFSAR accident analyses are either not dependent on the RTP or are adequately bounded by the other events. As such, the NRC staff determined that there can be no increase in the postulated radiological consequences of DBAs previously analyzed.

3.8.3 Special Events

3.8.3.1 Anticipated Transient Without Scram (ATWS)

ATWS is an AOO with failure of the reactor protection system to initiate a reactor scram to terminate the event. The requirements for ATWS are specified in 10 CFR 50.62 and requires BWR facilities to have the following mitigating features for an ATWS event:

(1) a SLC system with the capability of injecting a borated water solution with reactivity control equivalent to the control obtained by injecting 86 gpm of a 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside-diameter reactor vessel,

(2) an alternate rod injection (ARI) system that is designed to perform its function in a reliable manner and that is independent all the way from sensor output to the final actuation device, and

(3) equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

Hatch meets the ATWS mitigation requirements defined in 10 CFR 50.62. The SLC systems are capable of boron injection equivalent to 86 gpm, and the licensee has installed an ARI system and automatic RPT logic.

BWR facilities are also analyzed against certain ATWS acceptance criteria to demonstrate their ability to withstand an ATWS event. These criteria include criteria for maintaining fuel integrity (the core and fuel must maintain a coolable geometry), primary system integrity (the peak reactor vessel pressure must remain below 1500 pig), and containment integrity (the containment temperature and pressure must not exceed the design limit).

Section 5.3.5 and Appendix L of the TLTR, present a generic evaluation of the sensitivity of BWRs to an ATWS event after a TPO uprate. The topical report provides an ATWS acceptance criterion margin to determine whether a plant-specific evaluation is needed at the TPO power level. The TLTR states that if the suppression pool temperature margin criterion of 2 °F is not met, a plant-specific ATWS containment analysis is required. The ATWS analysis performed at 100 percent of CLTP demonstrated a margin of 74 °F and 133 °F for Hatch, Units 1 and 2 respectively. The maximum calculated suppression pool temperature calculated for the TPO is 217 °F (MSIV closure event). The lower temperature limit (i.e., the point at which ECCS NPSH no longer can be assured) is 219.2 °F and 220.2 °F for Hatch, Units 1 and 2 respectively. Since there is sufficient margin to meet the suppression pool temperature criterion, no plant-specific ATWS analysis for suppression pool temperature was performed for TPO conditions. However, because Hatch does not have sufficient margin to the ASME Service Limit C peak vessel bottom pressure limit of 1500 psig at the current licensed thermal power level, a plant-specific ATWS analysis was performed for the TPO uprate using the NRC staff-approved methodology given in ELTR1. The limiting case was a pressure-regulator-fail-open event at the beginning of the cycle conditions. The calculated peak vessel bottom head pressure is 1497 psig for the TPO uprate.

Based on the margin criteria and justification provided in the TLTR, the analyses performed by GE on the sensitivity of BWRs to an ATWS, and the available margin for peak ATWS parameters, the NRC staff accepts the licensee evaluation. Accordingly, the NRC staff concludes that Hatch meets the ATWS rule requirements specified in 10 CFR 50.62.

3.8.3.2 Station Blackout (SBO)

Station blackout refers to the complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. Station blackout involves the loss of offsite power concurrent with turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources." The NRC staff's review focuses on the impact of the of the proposed power uprate on the plant's ability to cope with and recovery from an SBO event for SBO are based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and Appendix B to SRP 8.2.

The licensee evaluated the plant for conformance with the SBO rule. This evaluation was conducted according to the assumptions of RG 1.155 and NUMARC 87-00. The licensee evaluated the following:

* The adequacy of the condensate/reactor coolant inventory
* The capacity of the Class 1E batteries
* The SBO compressed nitrogen requirements
* The ability to maintain containment integrity
* The effect of loss of ventilation on rooms that contain equipment essential for plant response to an SBO event.

The NRC staff's review indicates that the 4-hour coping duration has not changed and the plant currently has margins of 23,000 gallons to the available condensate storage inventory volume and margins of 87 ^{o}F and 146 ^{o}F (for Hatch, Units 1 and 2 respectively) to the containment peak temperature limit. Class 1E battery capacity and the compressed air system are unaffected by power uprate, and power uprate will not increase demand on these systems for SBO scenarios. Therefore, the capacity of these systems will remain adequate.

The NRC staff has reviewed the licensee's submittal on the effect of the proposed power uprate on the plant's ability to cope with and recover from an SBO event for the period of time established on the plant's licensing basis. The plants have adequate condensate inventory for decay heat removal during an SBO of 4-hour duration. The NRC staff concludes that the licensee has adequately evaluated the effects of the proposed power uprate on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following the implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to an SBO.

3.8.4 Reactor Safety Performance Evaluations - Conclusions

Based on the evaluations in Sections 3.8.1 through 3.8.3 of this SE, the NRC staff concludes that the proposed power uprate is acceptable with respect to its impact on reactor safety performance.

3.9 Other Evaluations

3.9.1 High-Energy Line Break Analysis (HELB)

The licensee stated that since the operating temperatures and pressures change only slightly at the uprated conditions, there is no change in the HELB mass and energy releases. As indicated by the licensee in its amendment request, the nominal vessel dome pressure and other portions of the reactor coolant pressure boundary remain at current operating temperatures or lower. The existing HELB analyses were performed assuming 102 percent of the CLTP, which bounds the proposed 1.5-percent power uprate condition. Therefore, the licensee concluded that the existing HELB analysis, break locations, pipe whip and jet impingement analyses remain unchanged. The existing pipe whip restraints, jet impingement shields, and their supporting structures are also adequate for the proposed 1.5-percent power uprate condition. Based on its review, the NRC staff concurs with the licensee's conclusion, because the existing HELB analyses bound the proposed uprate condition.

3.9.2 Environmental Qualification (EQ)

Electrical equipment specified in 10 CFR 50.49 is qualified based on normal and accident environmental conditions in the containment and portions of the auxiliary and turbine buildings. In its application, the licensee addressed power uprate issues related to EQ of electrical equipment. The licensee stated that it performed the current main steam line break, DBA-LOCA, and containment analyses at 102 percent of CLTP. Therefore, the equipment qualification envelope inside containment continues to be applicable for operation at the uprated power conditions. The licensee also stated that the normal radiation profiles, both inside and outside containment, and the accident temperature, pressure, and humidity environments outside containment were based on the CLTP. The licensee performed an analysis and determined that all of the equipment, both inside and outside of the containment, is qualified and the installed equipment remains within its qualification envelope at the uprated conditions.

The staff has reviewed the licensee's evaluation of the effects of the proposed power uprate on the EQ of the electrical equipment and has determined that the electrical equipment remains within its qualification envelope at the proposed uprated power conditions. Accordingly, the staff concludes that the equipment continues to meet the relevant requirements of 10 CFR 50.49. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to EQ of electrical equipment.

3.9.3 Operator Training and Human Factors

3.9.3.1 Operator Actions

The licensee stated in the December 19, 2002, application that "For TPO uprate conditions, operator response to transient, accident, and special events is not affected. Operator actions for maintaining safe shutdown, core cooling, containment cooling, etc., do not change for the TPO uprate." The NRC staff concludes that the licensee has adequately addressed the question of operator actions sensitive to the power uprate by describing the lack of affect on operator response and actions.

3.9.3.2 Operating Procedures

The licensee stated that operating procedures and guides will be changed to reflect the actions necessary during UFM outages, to reflect the accuracy of the core thermal power and FW flow measurements. The operating procedures will be modified to reflect the approved allowed outage times for the UFMs, detailing the steps that must be taken to reduce power level. The licensee stated that the emergency operating procedures (EOP) action thresholds are plant unique and will be addressed using standard procedure updating processes. The licensee further indicated that the TPO uprate will have a negligible or no effect on the operator action thresholds and to the EOP's in general. The licensee also stated that they will change all necessary procedures prior to implementation of the power uprate. The NRC staff finds that the necessary procedures will be changed or updated prior to implementation and, therefore, will appropriately address plant operation at the new power level.

3.9.3.3 Control Room Controls, Displays and Alarms

The licensee has identified the all necessary hardware and software to provide the UFM input to the plant computer and automatically correct the FW flow venturi readings to the UFM readings. The modification will rescale the average power range monitors to provide 0-100-percent power output for 0-2804 MWt input. This modification will also rescale the flow-referenced high power trip function for two-loop and single-loop operation. This modification will also rescale the plant computer to associate 100-percent RTP with 2804 MWt. These modifications will be completed prior to implementation of the power uprate. The licensee will provided operator training on the plant changes and operational aspects of the plant due to the power uprate prior to implementation. The NRC staff finds that the licensee has adequately identified the changes that will occur to controls, displays, and alarms as a result of the proposed power uprate and appropriately described how these changes will be implemented. The NRC staff concludes that the control room controls, displays and alarms will be appropriately modified to support operations at the power uprate conditions.

3.9.3.4 Control Room Plant-Referenced Simulator

The licensee stated that it will provide the changes to the plant simulator to provide the UFM readings on the plant computer and average power range monitors (APRMs), to reflect the correct percent RTP value to megawatt thermal correlations and show the corrections to the FW flow venturi readings. These modifications, including the modifications to the simulator, will be completed prior to implementation of the power uprate. The licensee will provide operator training on the plant changes and operational aspects of the plant due to the power uprate prior to implementation. Simulator changes and validation for the TPO uprate will be performed in accordance with ANSI/American Nuclear Society (ANS) 3.5-1985, "Nuclear Power Plant Simulators for Use in Operator Training." The NRC staff finds that the licensee has adequately identified the changes that will occur to controls, displays, and alarms as a result of the proposed power uprate and appropriately described how these changes will be implemented. The NRC staff concludes that the control room plant-referenced simulator will be appropriately modified to support operations at the power uprate conditions.

3.9.3.5 Operator Training Program

The licensee stated that no additional training (apart from normal training for plant changes) is required to operate the plant in the TPO uprate condition. In Enclosure 7 of the licensee's submittal dated December 19, 2002, the licensee stated that they will provide operator training on the plant changes and operational aspects of the plant prior to implementation. Minor changes to the power/flow map, flow-referenced setpoint, and the like, will be communicated through normal operator training. The licensee will make changes to operator training procedures and guides to reflect actions needed during UFM outages, to reflect accuracy of the core thermal power, and FW flow measurements. The operator training on the plant changes and operational aspects of the plant due to the power uprate will be completed prior to implementation. The NRC staff finds that the licensee has adequately addressed the changes to the operator training program to ensure operators have the necessary skills and knowledge to operate the plant at the increased power level. The licensee has also identified that operators will be trained prior to the implementation of the power uprate. The NRC staff concludes that the operator training program will be appropriately modified to address changes associated with operations at the power uprate conditions.

3.9.3.6 Operator Training and Human Factors - Conclusion

The NRC staff has reviewed the licensee's planned actions related to human factors area and concludes that the licensee has adequately considered the impact of the proposed 1.5-percent power uprate on changes to operator actions, procedures, plant hardware and software, and associated training programs to ensure that operator's performance is not adversely affected by the proposed modifications. The NRC staff finds that the proposed power uprate is acceptable with respect to its impact on human factors considerations.

3.10 Renewed Facility Operating License and Technical Specifications Changes

The licensee proposed to revise the RFOL and TSs as follows to reflect the increase in licensed power level from 2763 MWt to 2804 Mwt:

(1) Change to RFOL DPR-57 (Unit 1) and NPF-5 (Unit 2)

Paragraph 2.C(1), "Maximum Power Level," would be revised to authorize operation at a steady state reactor core power level not in excess of 2804 Mwt (100%).

(2) Change to TS 1.1, "Definitions"

The definition of RATED THERMAL POWER would be revised to reflect the increase from 2763 MWt to 2804 Mwt.

(3) Change to TS 2.1.1.1, "Reactor Core SLs"

The THERMAL POWER Safety Limit value would be reduced from less than or equal to "25%" RTP to less than or equal to "24%" RTP, with the reactor steam dome pressure < 785 psig or core flow < 10% rate core flow. The objective of this revision is to maintain the original generic basis of 4.8 MWt per fuel bundle.

(4) Change to TS 3.2.1, “Average Planar Linear Heat Generation Rate (APLHGR)”

The THERMAL POWER value in the LCO 3.2.1 Applicability would be reduced from greater than or equal to 25% to greater than or equal to 24%. The THERMAL POWER value in Required Action B.1 would be reduced from less than 25% to less than 24%. The THERMAL POWER value in SR 3.2.1.1 Frequency would be reduced from greater than or equal to 25% to greater than or equal to 24%. The objective of these changes is to reflect the change to Safety Limit 2.1.1.1.

(5) Change to TS 3.2.2, “Minimum Critical Power Ratio (MCPR)”

The THERMAL POWER value in LCO 3.2.2 Applicability would be reduced from greater than or equal to 25% to greater than or equal to 24%. The THERMAL POWER value in Required Action B.1 would be reduced from less than 25% to less than 24%. The THERMAL POWER value in SR 3.2.2.1 Frequency would be reduced from greater than or equal to 25% to greater than or equal to 24%. The objective of these changes is to reflect the change to Safety Limit 2.1.1.1.

(6) Change to TS 3.3.1.1, “Reactor Protection System (RPS) Instrumentation”

The THERMAL POWER value in Required Action E.1 and in SR 3.3.1.1.11 would be reduced from less than 28% to less than 27.6%. This change is also reflected in the Applicable Modes or Other Specified Conditions for Functions 8 and 9 on Table 3.3.1.1-1, “Reactor Protection System Instrumentation.” The objective of these changes is to reduce the reactor power when the turbine stop valve (TSV) closure scram, TCV fast closure scram, and EOC-RPT are bypassed. Reactor power is sufficiently low enough that a scram or trip is not required to mitigate a turbine-generator trip.

The THERMAL Power value in SR 3.3.1.1.2 and NOTE is reduced from greater than or equal to 25% to greater than or equal to 24%. The objective of this change is to reflect the change to Safety Limit 2.1.1.1.

The Allowable Value in Table 3.3.1.1-1 for Function 2.b., “Simulated Thermal Power - High,” would be revised to read $\leq$0.57W + 56.8% RTP and 115.5% RTP and the footnote (b) would be revised to read 0.57W + 56.8% - 0.57ΔW RTP. The flow-referenced APRM Allowable Values for both two-loop and single-loop operation are unchanged in units of absolute core thermal power versus recirculation drive flow. Since the values are expressed in percent of RTP, they decrease in proportion to the power uprate.

(7) Change to TS 3.3.2.2, “Feedwater and Main Turbine Trip High Water Instrumentation”

The THERMAL POWER value in LCO 3.3.2.2 Applicability would be reduced from greater than or equal to 25% to greater than or equal to 24%. The THERMAL POWER value in Required Action C.1 would be reduced from less than 25% to less than 24%. The objective of these changes is to reflect the change to Safety Limit 2.1.1.1.

(8) Change to TS 3.3.4.1, “End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

The THERMAL POWER value in LCO 3.3.4.1 Applicability would be reduced from greater than or equal to 28% to greater than or equal to 27.6%. The THERMAL POWER value in Required Action C.2 would be reduced from less than 28% to less than 27.6%. The THERMAL POWER value in SR 3.3.4.1.2 would be reduced from greater than or equal to 28% to greater than or equal to 27.6%. The objective of these change is to reduce the TSV closure scram, TCV fast closure scram, and EOC-RPT bypass values, in percent RTP, by the ratio of the power increase, since the scrams and trips are based on the absolute thermal power and not the percent RTP.

(9) Change to TS 3.7.7, “Main Turbine Bypass System”

The THERMAL POWER value in LCO 3.7.7 Applicability would be reduced from greater than or equal to 25% to greater than or equal to 24%. The THERMAL POWER value in Required Action B.1 would be reduced from less than 25% to less than 24%. The objective of these changes is to reflect the change to Safety Limit 2.1.1.1.

The RFOL and TS changes reflect the proposed increase in licensed power level based on installation of the Crossflow UFM system for FW flow and temperature measurements. Based on the evaluations discussed in Section 3.1 through 3.9 of this SE, the NRC staff concludes that the above-described changes to the RFOL and TS, which maintain the absolute thermal power limits in the TS, are within the licensing basis of the Hatch, Units 1 and 2 plants and are acceptable.

Additionally, enclosure 3 of the December 19, 2002, submittal included proposes changes to the TS bases changes that support the propose change in steady state power level. The proposed TS bases changes are consistent with the changes to the license and TSs. The NRC staff has no objections to the TS bases changes.

5.0 REGULATORY COMMITMENTS

To support the proposed MUR power uprate, the licensee made the following licensing commitments (as stated):

A. Plant Modifications

1. Implementation of the Crossflow system will be completed prior to implementation of the requested license amendment and prior to raising the rated thermal power (RTP) above 2763 MWt.

2. Validation of the assumed Crossflow system measurement uncertainty will be performed prior to implementation of the requested license amendment and prior to raising the power level above 2763 MWt.

B. Administrative Changes

1. Necessary maintenance and operational procedure revisions will be completed prior to implementation of the requested license amendment.

2. Operational procedure revisions will include the Crossflow system out-of-service administrative technical requirements.

C. Startup Testing

1. Core power from the average power range monitors (APRMs) will be rescaled to the uprated power level prior to exceeding the current licensed power level. Any necessary adjustments of the APRM alarm and trip settings will be made.

2. Demonstration of an acceptable fuel thermal margin will be performed prior to and during power ascension at each steady-state heat balance power level (95% and 100% of the current licensed power level and 100% of the uprated power level).

   Fuel thermal margin will be projected to the uprated RTP point after the measurement at 95% and 100% of current licensed power level are taken to show the estimated margin. The demonstration of core and fuel conditions will be performed using current Plant Hatch methods.

3. In preparation for operation at the uprated power level, routine measurements of reactor and system pressures, flows, and selected major rotating equipment vibration will be taken near 95% and 100% of the current licensed power level and at 100% of the uprated power level.

4. The operational aspect of the power uprate will be demonstrated by performing turbine pressure regulator controller and feedwater controller testing during power ascension testing. Reactor pressure control system testing, consistent with the guidelines of NEDC-222085P, “Safety Analysis Report for Edwin I. Hatch Units 1 and 2 THERMAL POWER OPTIMIZATION,” dated December 2002, will be performed during power ascension testing.

   During these tests, a water level change of $\pm$ 3 inches and pressure setpoint changes of $\pm$ 3 psi will be used. If necessary, the controllers and actuator elements will be adjusted.

   a. The performance of the feedwater level control system will be recorded at 95% and 100% of the current licensed power level and confirmed at the uprated power level during power ascension.

b. The turbine pressure controller setpoint will be readjusted at 95% current licensed power level and held constant. Adjusting the pressure setpoint prior to recording the baseline power ascension data establishes a consistent basis for measuring the performance of the reactor and the turbine control valves.

D. Training

1. Minor changes (e.g. power/flow map and flow-reference setpoint changes) will be communicated through normal operator training.

2. Simulator changes and validation for the power uprate will be performed in accordance with ANSI/ANS 3.5-1985.

The NRC staff considered the above commitments as part of its evaluation in Section 3.0 above and finds the commitments appropriate for the proposed MUR power uprate. The NRC staff has conditioned the implementation of the proposed MUR power uprate on completion of the above commitments.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (68 FR 7821). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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