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SOUTHERN COMPANY
Energy to Serve Your World℠

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50-366

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
Measurement Uncertainty Recapture Power Uprate (MURPU)
Request for Additional Information (RAI) Response

Ladies and Gentlemen:

By letter dated December 19, 2002, Southern Nuclear Operating Company (SNC) submitted to the NRC a Technical Specifications amendment request for the Edwin I. Hatch Nuclear Plant Units 1 and 2. The proposed amendment increases the authorized maximum power level for both units from the current limit of 2763 MWt to 2804 MWt. The NRC/NRR Hatch Project Manager, via electronic communication, forwarded to SNC several Requests for Additional Information containing NRC Staff review requests related to SNC's December 19, 2002 submittal.

Enclosure 1 provides documentation of the NRC's requests followed by SNC's responses and contains proprietary information as defined in 10CFR2.790. This information is provided in confidence and should be withheld from public disclosure. The proprietary marking (double-underlined font with brackets) indicates the specific lines of information that are considered proprietary.

Enclosure 2 provides a non-proprietary version of the applicable NRC RAIs and SNC responses. Please note that a portion of NRC RAI RSB-4 itself contains proprietary information as indicated in Enclosure 1. As presented in Enclosure 1 and 2, NRC RAI RSB-4 has been rewritten to ensure that the proprietary information is not inadvertently released.

The GE proprietary information affidavits (Enclosures 3 and 4) identify that the designated information has been handled and classified as proprietary to GE. SNC hereby requests that the designated information be withheld from public disclosure in accordance with the provisions of 10CFR2.790 and 9.17.

This letter contains no NRC commitments.

A001

Please let me know if you have any questions or comments regarding this submittal.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

HLS/twl/whc

Enclosures: Enclosure 1 – MURPU RAI Responses with Proprietary Information
Enclosure 2 – MURPU RAI Non-Proprietary Responses
Enclosure 3 – General Electric Affidavit for Proprietary Information
Enclosure 4 – General Electric Affidavit for Proprietary Information

cc: Southern Nuclear Operating Company
Mr. J. D. Woodard, Executive Vice President
Mr. P. H. Wells, General Manager – Plant Hatch
Document Services RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. S. D. Bloom, NRR Project Manager – Hatch
Mr. N. P. Garrett, Acting Senior Resident Inspector – Hatch

Enclosure 3

General Electric Affidavit for Proprietary Information

Edwin I. Hatch Nuclear Plant
Measurement Uncertainty Recapture Power Uprate (MURPU)
Request for Additional Information (RAI) Response

General Electric Company

AFFIDAVIT

I, David J. Robare, state as follows:

(1) I am Technical Projects Manager, Technical Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in The information sought to be withheld is contained in Attachment 2 to GE letter GE-HATCH-TPO-083, Michael Dick (GE) to Timothy Long (SNC) , Task T1302: Hatch TPO Project, *Response to NRC's Request forAdditional Information (RAI) RSB 1, 2, 4 thru 8,* dated March 12, 2003. The proprietary information is identified by a double underline inside square brackets.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission,</u> 975F2d871 (DC Cir. 1992), and <u>Public Citizen Health Research Group v. FDA,</u> 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

8) The information identified in paragraph (2), above, is classified as proprietary because it contains responses containing or based on detailed results of analytical models, methods and processes, including computer codes for BWRs.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 12th day of March, 2003.

David J. Robare
General Electric Company

Enclosure 4

General Electric Affidavit for Proprietary Information

Edwin I. Hatch Nuclear Plant
Measurement Uncertainty Recapture Power Uprate (MURPU)
Request for Additional Information (RAI) Response

General Electric Company

AFFIDAVIT

I, George B. Stramback, state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Attachment 2 to letter GE-HATCH-TPO-084, *Task 1302: Hatch TPO, Response to NRC's Request For Additional Information (RAI) MCB 1 thru 4 and MAT 1 thru 4, Proprietary and Non-Proprietary Versions*, dated March 26, 2003. The proprietary information in Attachment 2 (*GE-HATCH -TPO-084, GE Responses to NRC RAIs MCB 4,* Proprietary), is identified as red, double underlined font within brackets.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains further details regarding the GE proprietary report *NEDC-33085P, Safety Analysis Report for Edwin I. Hatch Units 1 and 2 Thermal Power Optimization,* Class III (GE Proprietary Information), dated December 2002, which contains detailed results of analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR").

The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 26th day of March 2003.

George B. Stramback

George B. Stramback
General Electric Company

Enclosure 2

MURPU RAI Non-Proprietary Responses

Edwin I. Hatch Nuclear Plant
Measurement Uncertainty Recapture Power Uprate (MURPU)
Request for Additional Information (RAI) Response

NRC RAI ESB-1
Provide details about the grid stability analysis including assumptions and results for the power uprate condition.

SNC RAI ESB-1 Response
An Extended Power Uprate (EPU) was implemented on Hatch Unit 1 in 1999 and on Hatch Unit 2 in 1998. The EPU represented a power level increase of 8% from a previous uprate of 5% 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 autotransformer with a failure of breaker 540 to operate. The CCT for this contingency, during a valley load condition, was found to be 10 cycles while the actual BFCT was 9.5 cycles. This represented the bounding contingency to be evaluated for the measurement uncertainty recapture power uprate (MURPU).

The electrical power levels that were evaluated for EPU were 930 MWe and 940 MWe for Units 1 and 2 respectively. The MURPU represents only a 5 MWe increase for Unit 1 (935 MWe) and a 10 MWe increase for Unit 2 (950 MWe) over what was analyzed for the EPU. Two cases were run for the MURPU – one with EPU levels of 930 MWe and 940 MWe and one with the proposed power levels of 935 MWe and 950 MWe for Unit 1 and Unit 2 respectively. The system load level used a light load case (50% of peak). The CCT for the breaker failure contingency on the low side of the autotransformer was found to decrease by only 0.25 cycles due to the MURPU conditions. The difference between the EPU CCT and the MURPU at valley load conditions is expected to be about the same as those at light system load conditions. The small difference in CCT of 0.25 cycles indicates this contingency is acceptable for the MURPU conditions. Since the bounding contingency remains acceptable, then those not analyzed also remain acceptable for the MURPU conditions.

Evaluations were performed for the generator control and protection, and the generator protective relays. These evaluations concluded that MURPU does not require any changes to the relays including the underactive ampere limiter (URAL) settings. An analysis was also performed to identify the maximum expected VARs out of the generators at the MURPU conditions. At the MURPU conditions, the maximum expected VARs out are within the generator reactive capability curves indicating the generators will remain stable at the MURPU conditions thereby maintaining grid stability.

A calculation was performed to verify that the generator operating range is still a minimum of 95% to a maximum of 100% of the 24 kV rated generator voltage for both Unit 1 and Unit 2.

SNC RAI ESB-1 Response (continued)

A steady state voltage analysis was performed for the MURPU conditions for 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 FSAR Study. The analysis concluded that the proposed MURPU meets the minimum 230 kV bus voltage criteria of 101.3% - 104.9%.

Therefore it was concluded that the MURPU will have an insignificant impact on grid stability.

NRC RAI ESB-2

Table 6-1 "TPO Plant Characteristics" provides the uprated ratings. Provide in detail the existing ratings and the effect of the power uprate on the following equipment:

i. *Main generator*
ii. *Isophase bus*
iii. *Main power transformer*
iv. *Startup transformer*
v. *Unit auxiliary transformer*

SNC RAI ESB-2 Response

i. Main Generator

The MURPU of 1.5% will increase the maximum Unit 1 generator output 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.

ii. Isophase Bus

The EPU effort also re-rated the isophase busses to 1050 MVA. Re-rated current through the isophase busses at 1050 MVA is 26,588 amps (1050 MVA @ 22.8 KV). The maximum expected current through an isophase bus is 26,133 amps (1032 MVA @ 22.8 kV). 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.

iii. 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

iii. Main Power Transformer (continued)
transformer ratings the transformers are capable of handing the MURPU requirements.

iv. Startup Transformers
The Startup transformers (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 MURPU on SUT 1C or 2C is about 24 kW from the recirculation MG set motor drives. The resulting MVA is much less than the transformer rating of 28 MVA.

The maximum emergency load represented by the 4.16 kV busses is approximately 9.8 MVA. Therefore, SUT 1C 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 increase due to the MURPU on SUT 1D or 2D is 43 kVA. With this load increase, the load on SUT 1D or 2D remains much less than the rating of 33.6 MVA.

The maximum emergency load represented by the 4.16 kV busses for SUT 1D or 2D was previously evaluated during EPU for a 3% overload condition. The analysis performed for EPU envelopes the load increase for the MURPU in that the overload condition remains below the 3% overload condition previously evaluated.

v. Unit Auxiliary Transformers
The Unit Auxiliary Transformers (UATs) are rated as follows:

UAT 1A and 2A	33.6 MVA (FOA @ 65°C)
UAT 1B and 2B	28.0 MVA (FOA @ 65°C)

At the current license power level, the maximum load on Unit Auxiliary Transformer (UAT) 1A or 2A is 19.1 MVA. The maximum expected increase due to the MURPU on UAT 1A or 2A is 43 kVA. With this load increase, the load on UAT 1A or 2A remains much less than the rating of 33.6 MVA.

At the current license power level, the maximum load on UAT 1B or 2B is approximately 22 MVA. The maximum increase in loading due to the MURPU on UAT 1B or 2B is about 24 kW from the

v. **Unit Auxiliary Transformers (continued)**
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 MURPU.

NRC RAI RSB-1

Section 1.2.1, TPO Analysis Basis - Clarify which unit's parameters were used for the analysis at 102% of CLTP and explain in detail why the analysis is bounding for both units. What are the differences between the two units? The feedwater temperature for Unit 1 at TPO conditions is 392 degrees F (Table I-2) compared with 427 degrees F (Table I-3 for Unit 2). List all the differences between the units which impact the analysis.

SNC RAI RSB-1 Response

The significant differences between Hatch Unit 1 and Unit 2 are the rated core flow, rated steam flow, rated feedwater flow and the final feedwater temperature. The difference in rated core flow for the units is due to different core orificing. The difference in final feedwater temperature, steam flow, and feedwater flow are due to differences in the feedwater heating capacities of the units that allows Hatch Unit 2 to achieve a higher final feedwater temperature (i.e., Unit 2 has an additional stage of feedwater heating).

The TPO evaluations that are based on previously bounded analyses performed at 102% of the CLTP where "bounding" parametric data was assumed are discussed below.

ECCS-LOCA Analysis.

In this analysis, the core flow of Hatch Unit 2 and the feedwater temperature of Hatch Unit 1 were used. The result of the bounding values is that the break flow from a postulated recirculation line break is higher due to higher annulus subcooling. The use of these "bounding" parameters results in conservative, e.g., higher, PCT results.

Containment Analysis

For the short-term containment system response analysis, Hatch Unit 1 parameters were used for calculating the blowdown flow. This was considered conservative, since the use of Hatch Unit 1 parameters results in a higher break flow and a resulting higher peak containment pressure.

For the long-term containment system analysis, Hatch Unit 1 Vessel Steam flow was used to minimize vessel inventory loss, and Hatch Unit 2 feedwater temperature and feedwater flow were used to maximize energy input into the containment. This results in the calculation of conservatively higher suppression pool temperatures.

Loss of Feedwater Flow Analysis
Hatch Unit 2 parameters were used as a bounding analysis for the Plant Hatch Extended Power Uprate evaluation. The higher feedwater and steam flows for Hatch Unit 2 result in a more rapid reactor vessel inventory loss than for Hatch Unit 1. The use of Hatch Unit 2 parameters results in a greater challenge to the loss of feedwater flow acceptance criterion of maintaining water level above the Top of Active Fuel upon a loss of Feedwater flow, and is therefore conservative.

NRC RAI RSB-2
Section 1.3.2 Reactor Performance Improvement Features - Confirm that the analyses performed for reactor performance improvement features bounds 101.5 power level.

SNC RAI RSB-2 Response
The performance improvement features and Equipment out of service option listed in section 1.3.2 of NEDC-33085P were considered in the Plant Hatch TPO evaluations and remain acceptable for the proposed 1.5% power uprate.

NRC RAI RSB-3
Section 2.1 Fuel Design and operation - Describe the current operating Cycle for both units, specify GE fuel type which are in the core now and planned for TPO

SNC RAI RSB-3 Response
For the TPO uprate, there are no changes in fuel type required. The Hatch units are also in the process of transitioning from 18 to 24 month fuel cycles. It is this change, rather than the TPO uprate, that has required the introduction of GE14 (10x10) fuel in both units. Over the next 3 cycles on each unit, the older GE13 (9x9) fuel will gradually be replaced by GE14 fuel. No other fuel type changes are under consideration at this time.

The GE14 fuel design is being introduced into the Hatch units under the General Electric Standard Application for Reactor Fuel (GESTAR-II) Amendment 22 licensing process. Revision 1 of NEDC-32868P, "GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)," demonstrates in detail that GE14 fuel is in compliance with the licensing acceptance criteria for fuel as described in Amendment 22 of GESTAR-II, except for a few plant-specific initial GE14 evaluations which have been separately addressed by Southern Nuclear.

A brief description of each core loading is provided below.

Hatch-1 Cycle-21
Hatch-1 Cycle 21 started operation in April 2002 in its first 24-month fuel cycle. The TPO uprate is planned for June 2003 during mid-cycle. The Cycle 21 core is a conventional core loading utilizing 224 fresh (40%) GE14 fuel bundles at 3.98 weight percent enrichment designed to achieve 15,780 MWd/st incremental cycle exposure. From the previous cycle, 336 GE13 bundles remain. All fuel assemblies loaded in Cycle 21 have barrier cladding and debris filter lower tie plates.

Hatch-2 Cycle-18
Hatch-2 Cycle 18 started operation in March 2003 in its first 24-month fuel cycle. The TPO uprate is planned for June 2003 in the early part of the cycle. The Cycle 18 core is a conventional core loading utilizing 232 fresh (41.4%) GE14 fuel bundles at 3.98 weight percent enrichment designed to achieve 16,430 MWd/st incremental cycle exposure. From the previous cycle, 328 GE13 bundles remain. All fuel assemblies loaded in Cycle 18 have barrier cladding and debris filter lower tie plates.

NRC RAI RSB-4
Section 2.2, Thermal Limits Assessment

It is stated that "For the Plant Hatch TPO uprate, [ ]." (Suggested non-proprietary rewrite: It is stated that the historical 25% RTP value was rescaled.) If this is true, additional clarifications are required.

SNC RAI RSB-4 Response
For Plant Hatch TPO, [ ]. However, additional justification is not required since the historical 25 percent value was reduced.

The TPO Licensing Topical Report (TLTR), NEDC-32938P, Sections 5.8 and F4.2.11, provide the justification for the reduction in the historical value of 25 percent RTP. The TLTR requires [

] Since the 25 percent value was reduced, no additional justification was provided.

To clarify the TLTR criteria, [

]

[ ], the fuel thermal margin monitoring threshold is scaled down, if necessary, to ensure that the monitoring is initiated [

]. For Plant Hatch, the historical value of 25% was reduced to 24%.

__NRC RAI RSB-5__
Section 3.1, Nuclear system pressure relief/overpressure protection - Identify the GE approved methodology and refer to the analyses given in the current reload analyses.

__SNC RAI RSB-5 Response__
The approved methodology for the overpressure analysis is documented in GESTAR II. The applicable references are:

- Qualification of the One-Dimensional Core Transient Model for BWR's, NEDO-24154, Vol. 1 and 2: October 1978.
- Qualification of the One-Dimensional Core Transient Model for BWR's, NEDE-24154-P, Vol. 3: October 1978.

The current over-pressurization analyses results are reflected in the following reload analyses for Hatch Unit 1 and 2:

Supplemental Reload Licensing Report for EDWIN I. HATCH NUCLEAR POWER PLANT UNIT 1 Reload 20 Cycle 21, 0000-0002-7058-SRLR, March 2002.

- Overpressure analysis results (MSIV Closure/Flux Scram) = 1350 psig.

Supplemental Reload Licensing Report for EDWIN I. HATCH NUCLEAR POWER PLANT UNIT 2 Reload 17 Cycle 18, 0000-0007-0430-SRLR, March 2003.

- Overpressure analysis results (MSIV Closure/Flux Scram) = 1344 psig.

NRC RAI RSB-6

Section 3.6, Reactor Recirculation System - What is the licensed maximum core flow for Hatch plant? Discuss the pump the cavitation interlock aspects.

SNC RSI RSB-6 Response

The licensed maximum core flow for Plant Hatch is 82.4 Mlb/hr, as stated in the Plant Hatch TPO Safety Analysis Report (TSAR), NEDC-33085P, Table 1-2.

[

]

NRC RAI RSB-7

Section 4.3, Emergency Core Cooling System Performance - What is the limiting break and the calculated PCT?

SNC RAI RSB-7 Response

The current limiting break is the DBA recirculation suction line with a dedicated diesel battery failure. The Licensing Basis PCT is 1820 °F for both GE13 and GE14.

NRC RAI RSB-8

Section 9.3.1, ATWS - The previous ATWS analysis performed at CLTP demonstrated a margin of 74 degrees F for Unit 1 and 133 degrees F for Unit 2. We have seen much lower margins for Mark-I plants like Hatch. Why these margins are so high?

SNC RAI RSB-8 Response

Plant Hatch has a higher margin because the suppression pool temperature limit is higher relative to other Mark I plants addressed by recent TPOs.

A review of the previous Safety Analysis Reports associated with TPOs and Extended Power Uprates (EPU) demonstrates that the basis for the suppression pool limit is different for certain plants. Typically, for those plants with a higher margin, the suppression pool temperature limit is based on the containment design temperature limit (e.g., Plant Hatch and Duane Arnold). Typically, for those plants with a lower margin, the suppression pool temperature limit is lower due to bounding post-accident conditions.

Although not necessarily applicable to the ATWS events, the lower temperature limits of the bounding post-accident conditions (less than the containment design temp) are imposed by the individual utilities. The maximum ATWS suppression pool temperature calculated for the TPO is 217° F (ATWS-MSIV closure transient). To make a direct comparison of the lower temperature limit, the point at which ECCS NPSH no longer can be assured would be the lower temperature limit for Hatch. The lower temperature limit is 219.2°F and 220.2°F for Units 1 and 2 respectively. Therefore, for the Hatch TPO, the ECCS NPSH is adequate based on the long-term temperature/pressure response of the suppression pool following an ATWS event with the corresponding licensing basis overpressure.

NRC RAI MTB-1

In the submittal the licensee states that after TPO small changes of velocity, temperature, and moisture content occurring in the systems carrying hot fluid will not be significantly affected by the Flow Accelerated Corrosion. In order to support your statement provide a few examples of the change of wear rates caused by the power increase in the most FAC susceptible systems.

SNC RAI MTB-1 Response

The SNC FAC Program uses EPRI's CHECWORKS™ to aid in determining component wear rates and inspection priority. A CHECWORKS™ wear rate analysis was performed to compare wear rates between the current FAC model based on current licensed thermal power (CLTP) and the MURPU.

The extraction steam to the 10th stage feedwater heater and the extraction steam to the 12th stage feedwater heater have been identified as piping systems most susceptible to FAC. The CHECWORKS™ model for the Unit 2 10th stage extraction line consists of 16 piping segments with a total of 151 components. Each component in each segment for the current model was compared to a model using MURPU conditions. The greatest average wear rate value increase as a result of the MURPU for a component was for the upstream section of a 36" tee and resulted in an increase of approximately 0.021 mils/year (~ 0.328% increase). The component inspection period was reduced by approximately 99 weeks over an original inspection period of approximately 2699 weeks. The next component with the greatest wear rate increase was the branch of a 36" tee and resulted in an increased wear rate of approximately 0.017 mils/year (~0.157% increase), and a reduced inspection period of approximately 13 weeks over an original inspection period of approximately 627 weeks.

The CHECWORKS™ model for the Unit 2, 12th stage extraction line consists of 16 piping segments with a total of 152 components. The greatest average wear rate increase as a result of TPO was for a 36" elbow which resulted in an approximate increase of 0.034 mils/year (~0.234% increase), and a reduced inspection period of approximately 7 weeks over an original inspection period of approximately 134 weeks. The next component with the greatest wear rate was also for a 36" elbow and resulted in an increased wear rate of approximately 0.026 mils/year (~0.143%), and a reduced inspection period of approximately 18 weeks over an original inspection period of approximately 820 weeks.

It should be noted that the inspection period is defined as time to a calculated $T_{critical}$. For the Hatch CHECWORKS™ models, the calculated $T_{critical}$ value is the T_{min} value based on Hoop Stress and is calculated by the software's own algorithms. Recently, the industry has taken the approach that bending loads should be considered when determining T_{min} for a given component. At Hatch, additional conservatism is included by not allowing T_{min} to drop below 0.3 $T_{nominal}$. Therefore, $T_{critical}$ is determined by the larger of T_{hoop}, $T_{bending}$, or 0.3 $T_{nominal}$. As components are physically measured during outages, wear rates and time to $T_{critical}$ are updated and the program model is revised.

The Hatch FAC model will be formally updated for MURPU conditions and CHECWORKS™ will continue to be used to aid in ranking components for inspection to determine the need for maintenance or replacement prior to reaching minimum wall thickness requirements. Adherence to the Hatch FAC Program provides assurance that the MUPUR has no adverse effect on high energy piping systems potentially susceptible to pipe wall thinning due to FAC.

NRC RAI MCB-1

In reference to Section 2.5, you indicated that the generic discussion in TPO Licensing Topical Report (TLTR) Section 5.6.3 and J.2.3.3 applies to Plant Hatch. As such, the Hatch Control Rod Drive Mechanisms (CRDMs) are not affected by the TPO uprate and no further evaluation of CRDMs is required. However, the TPO licensing topical report (TLTR) is still under review and has not been approved by the staff. Provide a summary discussing your determination of how the TLTR generic evaluation for CRDMs is applicable to Plant Hatch. Confirm that the existing design Basis stress and fatigue analysis of the CRDMs remains unchanged for the proposed 1.5 percent power uprate.

SNC RAI MCB-1 Response

The components of the CRDM, which form part of the primary pressure boundary, have been designed in accordance with the applicable ASME B&PV

Code. The CRDM structural and functional integrity is acceptable for a bottom head pressure of at least 1,250 psig. The CRD mechanism also has been evaluated for higher postulated abnormal operating pressures and conditions that subsequently apply the maximum CRD pump discharge pressure to the CRD mechanism internal components.

The CRD mechanism has been evaluated for the proposed 1.5% power uprate operating conditions and found to be acceptable. The CRDM qualification is based on the temperature and internal reactor differential pressure changes caused by 1.5% power uprate operating conditions relative to the CRDM structural design margins. Therefore, the existing design basis analysis for stress and fatigue cumulative usage of the CRDMs remains unchanged for the proposed 1.5% power uprate for Hatch.

NRC RAI MCB-2
In reference to Section 3.2.2 of Attachment 2 to the amendment request, you indicate that the effect of TPO was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of ASME Boiler and Pressure Vessel Code. For the components under consideration, the 1965 Edition of the Code with addenda to and including Winter 1966, which is the construction code of record, was used as the governing code. You also indicate that if a component underwent a design modification, the governing code for that component was the code used in the stress analysis of the modified component. Provide a summary of the components that were modified and the code editions/code cases (if applicable) other than the code of record that were used for the power uprate evaluation.

SNC RAI MCB-2 Response
No design modifications to components under consideration were performed since the extended power uprate as stated in Section 3.2.2 of the Hatch TSAR. Also, no design modifications to components under consideration were required as a result of the TPO.

NRC RAI MCB-3
In reference to Section 3.3.2, you indicated that consistent with TLTR, only Normal and Upset conditions were evaluated because the TPO loads for Faulted and Emergency conditions are bounded by the previous analysis performed at 102 percent of CLTP. The evaluation considered the effect of TPO on pressure, temperature, weight, seismic and flow loads, as applicable, and was performed consistent with the design bases for the responses. You also stated that the TPO loads were either bounded by the design basis values or the changes were insignificant, and therefore, the reactor internal components, remains qualified for the TPO uprate. Provide a summary of the quantitative evaluation to

demonstrate that there exist sufficient safety margins to accommodate the changes due to the proposed power uprate on the reactor internals.

<u>SNC RAI MCB-3 Response</u>
The loads in most of the internals due to TPO were smaller than those of the design basis. Those components were deemed qualified with no further detailed evaluation. The loads in a few components showed a very marginal increase (less than 1%) compared to those in the design basis. These changes were judged to be insignificant and acceptable based on the existing margins in the design basis. These components were also considered to be qualified with no further detailed evaluation.

<u>NRC RAI MCB-4</u>
In reference to Section 3.4, you stated that the safety-related Main Steam (MS) and Feedwater (FW) piping have minor increased flow rate and flow velocities resulting from the TPO uprate. You also indicated that the MS and FW piping vibration is expected to increase by about 3.5 percent, and that operating experience shows no evidence of vibration problems in MS and FW lines for CLTP operating conditions. Therefore, you concluded that the MS and FW lines vibrations remain within acceptable limits during TPO. Provide a technical basis or quantitative evaluation for your conclusion. Also, confirm that the existing design basis analysis of the current MS and FW lines vibration has sufficient safety margin to accommodate the vibration increase due to the proposed 1.5 percent uprate.

<u>SNC RAI MCB-4 Response</u>
A vibration monitoring program on main steam line and feedwater piping was performed during EPU implementation at Plant Hatch. [

] Thus, there is still sufficient margin with respect to the allowable limits.

<u>NRC RAI MCB-5</u>
In reference to Section 3.5.1, you stated that the effect of the TPO uprate with no nominal vessel dome pressure increase is negligible for the reactor coolant pressure boundary portion of all piping except for portions of the FW lines, MS lines, and piping connected to the FW and MS line. The MS and FW lines were evaluated for compliance with ANSI B31.1/ASME Section III Code stress criteria for the proposed power uprate condition. Clarify how both ANSI B31.1 "Power

Piping" Code and ASME Section III Code were used in the evaluation of MS and FW lines piping and supports. Provide the Code of record and the Code Edition that were used for the evaluation of MS and FW lines at the proposed power uprate condition.

<u>SNC RAI MCB-5 Response</u>

The FW and MS lines were previously evaluated at 102% of the current licensed power level which represents the current licensing basis for the systems. The piping systems (piping and supports) were evaluated for compliance with the applicable code stress criteria. That is, the non safety-related portions of the systems were evaluated for compliance with ANSI B31.1 Power Piping Code whereas the safety-related portions of the systems were evaluated for compliance with ASME Section III.

For the TPO, the uprate conditions for the main steam and feedwater piping systems were evaluated against the current licensing basis parameters. This evaluation concluded that the current licensing basis parameters bound the TPO uprate conditions. Since the current evaluations for pressure, temperatures and flow envelopes the TPO operating conditions, further code evaluations were not necessary for the FW and MS lines.

For the TPO uprate, no new Codes (including Addendums) were introduced during the evaluation process. The present Codes of record for Hatch Unit 1 are ANSI B31.7, 1969 through 1970 and ANSI B31.1, 1967. For Unit 2, they are ASME III, 1971 and ANSI B31.1, 1967 through 1971. Later editions of these Codes (ANSI B31.1, 1986 Edition and B31b-1987 Addenda, and ASME Section III, 1977 Edition with Addenda through Winter 1978) were used for the current licensing basis evaluations.

<u>NRC RAI MAT-1</u>

Section 3.2.1 of NEDC-33085P indicates the end-of-life (EOL) fluence is calculated for the TPO uprate conditions and from the fluence for current conditions to evaluate the vessel against the requirements of 10 CFR, Appendix G. The licensee is requested to identify the methodology utilized in calculating the neutron fluence. Identify whether the methodology has been previously reviewed and approved by the NRC and include all references.

<u>SNC RAI MAT-1 Response</u>

Consistent with the discussion in Section 5.5.1.5 in the TPO Licensing Topical Report (TLTR), NEDC-32938P, the operating (or currently licensed) fluence values (which are based upon flux wire measurements and not upon Reg. Guide 1.190 methods) are increased approximately proportional to the power increase.

The TPO LTR is currently under NRC review. However, the use of the current fluence valves based on measured flux wires from surveillance capsules has been used in previous TPO applications and approved by the NRC.

__NRC RAI MAT-2__

General Electric performed an equivalent margins analysis, which is documented in EPRI TR-113596, "BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," BWRVIP-74, dated September 1999. EPRI TR-113596 indicates that the percent reduction in Charpy upper shelf energy (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 Plant Hatch beltline materials meet the criteria specified in the report, the licensee should demonstrate that the percent reduction in Charpy USE for the beltline materials at 60 years of operation and TPO is less than those specified for the limiting BWR/3-6 plates and the non-Linde 80 submerged arc welds, and 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 Regulatory Guide 1.99, Revision 2.

__SNC RAI MAT-2 Response__

Please see Tables MAT2-1 through MAT2-4, which demonstrate compliance with the guidelines set forth in BWRVIP-74 for Equivalent Margin Analysis for 60 years (54 EFPY) for Hatch Units 1 and 2.

__NRC RAI MAT-3__

Plant Hatch will use an approved technical alternative in lieu of ultrasonic testing of RPV circumferential shell welds. The technical alternative is discussed in the staff's final SER of the BWR Vessel and Internals Project BWRVIP-05 Report, which is contained in a letter, dated July28, 1998 to Carl Terry, BWRVIP Chairman. In that letter, the staff concludes that, since the failure frequency for circumferential welds in BWR plants is significantly below the criteria specified in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," and the core damage frequency (CDF) of any BWR plant, and since continued inspection would result in a negligible decrease in an already acceptably low value, elimination of the ISI for RPV circumferential welds is justified. The staff's letter indicates that BWR applicants may request relief from the inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds by demonstrating (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the evaluation, and (2) they have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the report.

Section A.4.5 of Report BWRVIP-74 indicates that the staff's SER conservatively evaluated BWR RPVs to 64 effective full-power years (EFPY), which is 10 EFPY greater than what is realistically expected for the end of the license renewal period. Since this was a generic analysis, the applicant must provide plant-specific information to demonstrate that the Plant Hatch beltline materials meet the criteria specified in the report for 60 years of operation and TPO.

SNC RAI MAT-3 Response

Please see Table MAT3-1, which demonstrates the insignificant change from current licensed basis to TPO for the limiting circumferential welds.

NRC RAI MAT-4

In its letter dated July 28, 1998, to Carl Terry, BWRVIP Chairman, the staff identified a concern about the failure frequency of axially oriented welds in BWR RPVs. In its response to this concern, the BWRVIP provided evaluations of axial weld failure frequency in letters dated December 15, 1998 and November 12, 1999. The staff's evaluation of these analyses is contained in a letter dated March 7, 2000, to Carl Terry. The SER that is enclosed in that letter indicates that the RPV failure frequency as a result of the failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is below 5×10^{-6} per reactor year, given the assumptions regarding flaw density, distribution, and location described in the SER. Since the results apply only for the initial 40-year license period of BWR plants, Hatch must provide plant-specific information applicable to 60 years of operation and TPO.

SNC RAI MAT–4 Response

Please see Table MAT4-1, which demonstrates the insignificant change from current licensed basis to TPO for the limiting axial welds.

Table MAT2-1
Equivalent Margin Analysis for Plate Material for Plant Hatch Unit 1 for 54 EFPY

Plate Equivalent Margin Analysis

PLANT APPLICABILITY VERIFICATION FORM
FOR Hatch Unit 1 - BWR 4/MK I

BWR/3-6 PLATE

Surveillance Plate USE:

%Cu = 0 12

1st Capsule Fluence = 2 4x10^{17} n/cm^2

2nd Capsule Fluence = 4 6x10^{17} n/cm^2

Unirradiated to 1st Capsule Measured % Decrease = 4 (Charpy Curves)

Unirradiated to 2nd Capsule Measured % Decrease = -5 (Charpy Curves)

1st Capsule Rev 2 Predicted % Decrease = 9 (Rev 2, Figure 2)

2nd Capsule Rev 2 Predicted % Decrease = 10 (Rev 2, Figure 2)

Limiting Beltline Plate (Heat C4337-1) USE

%Cu = 0 17

54 EFPY 1/4 T Fluence = 2.5x10^{18} n/cm^2

Rev 2 Predicted % Decrease = 19(Rev 2, Figure 2)

Adjusted % Decrease = N/A (Rev 2, Position 2.2)

19% $\leq$ 23.5%, so vessel plates are bounded by equivalent margin analysis

Table MAT2-2
Equivalent Margin Analysis for Weld Material for Plant Hatch Unit 1 for 54 EFPY

Weld Equivalent Margin Analysis

PLANT APPLICABILITY VERIFICATION FORM
FOR Hatch Unit 1 - BWR 4/MK I

BWR/2-6 WELD

Surveillance Weld USE:

%Cu = 0 30

1st Capsule Fluence = 2.4×10^{17} n/cm^2

2nd Capsule Fluence = $4\ 6 \times 10^{17}$ n/cm^2

Unirradiated to 1st Capsule Measured % Decrease = Unknown (Charpy Curves)

Unirradiated to 2nd Capsule Measured % Decrease = -16 (Charpy Curves)

1st Capsule Rev 2 Predicted % Decrease = 19 (Rev 2, Figure 2)

2nd Capsule Rev 2 Predicted % Decrease = 22 (Rev 2, Figure 2)

Limiting Beltline Weld (Heat 1P2815) USE:

%Cu = 0 316

54 EFPY 1/4 T Fluence = $2\ 5 \times 10^{18}$ n/cm^2

Rev 2 Predicted % Decrease = 33.5(Rev 2, Figure 2)

Adjusted % Decrease = N/A (Rev 2, Position 2.2)

33.5% $\leq$ 39%, so vessel welds are bounded by equivalent margin analysis

Table MAT2-3
Equivalent Margin Analysis for Plate Material for Plant Hatch Unit 2 for 54 EFPY

Plate Equivalent Margin Analysis

PLANT APPLICABILITY VERIFICATION FORM
FOR Hatch Unit 2 - BWR 4/MK I

BWR/3-6 PLATE

Surveillance Plate USE:

%Cu = 0 08

1st Capsule Fluence = 2.3×10^{17} n/cm^2

Unirradiated to 1st Capsule Measured % Decrease = 0 (Charpy Curves)

1st Capsule Rev 2 Predicted % Decrease = 7 (Rev 2, Figure 2)

Limiting Beltline Plate (Heat C8579-2) USE:

%Cu = 0 11

54 EFPY 1/4 T Fluence = 2.8×10^{18} n/cm^2

Rev 2 Predicted % Decrease = 15 (Rev 2, Figure 2)

Adjusted % Decrease = N/A (Rev 2, Position 2.2)

15% $\leq$ 23.5%, so vessel plates are bounded by equivalent margin analysis

Table MAT2-4
Equivalent Margin Analysis for Weld Material for Plant Hatch Unit 2 for 54 EFPY

Weld Equivalent Margin Analysis

PLANT APPLICABILITY VERIFICATION FORM
FOR Hatch Unit 2 - BWR 4/MK I

BWR/2-6 WELD

Surveillance Weld USE:

%Cu = 0.13

1st Capsule Fluence = 2.3×10^{17} n/cm^2

Unirradiated to 1st Capsule Measured % Decrease = -1 (Charpy Curves)

1st Capsule Rev 2 Predicted % Decrease = 11 (Rev 2, Figure 2)

Limiting Beltline Weld (Heat 10137) USE:

%Cu = 0 216

54 EFPY 1/4 T Fluence = 1.7×10^{18} n/cm^2

Rev 2 Predicted % Decrease = 24 (Rev 2, Figure 2)

Adjusted % Decrease = N/A (Rev 2, Position 2.2)

24% ≤ 39%, so vessel welds are bounded by equivalent margin analysis

Table MAT3-1
Circumferential Weld Data per BWRVIP-05 for CLTP and TPO

Group	CE (VIP) 32 EFPY	CE (CEOG) 32 EFPY	CE (VIP) 64 EFPY	CE (CEOG) 64 EFPY	Hatch 1 54 EFPY (CLTP)	Hatch 1 54 EFPY (TPO)	Hatch 2 54 EFPY (CLTP)	Hatch 2 54 EFPY (TPO)
Cu%	0.13	0.183	0.13	0.183	0.197	0.197	0.047	0.047
Ni%	0.71	0.704	0.71	0.704	0.06	0.060	0.049	0.049
CF	151.7	172.2	151.7	172.2	91.0	91.0	31.0	31.0
Fluence ($10^{19}n/cm^2$)	0.20	0.20	0.40	0.40	0.24	0.24	0.24	0.25
ΔRT_{NDT} (°F)	86 4	98.1	113.2	128.5	55	56	19	19
$RT_{NDT(U)}$ (°F)	0	0	0	0	-10	-10	-50	-50
Mean RT_{NDT} (°F)	86 4	98.1	113.2	128.5	45	46	-31	-31
P(F/E) NRC	2 81E-5	6 34 E-5	1.99 E-4	4.38 E-4	---	---	---	---
P(F/E) BWRVIP	No failure	---	---	---	---	---	---	---

Note: All values rounded for consistency. Mean RT_{NDT} calculated using peak surface fluence, and does not include a margin term.

Table MAT4-1
Axial Weld Data per BWRVIP-05 for CLTP and TPO

Group	MOD 2	Hatch 1 54 EFPY (CLTP)	Hatch 1 54 EFPY (TPO)	Hatch 2 54 EFPY (CLTP)	Hatch 2 54 EFPY (TPO)
Cu%		0.316	0.316	0.216	0.216
Ni%		0.724	0.724	0.043	0 043
CF		219.0	219.0	98.0	98.0
Fluence ($10^{19}n/cm^2$)		0.35	0.35	0.24	0.25
ΔRT_{NDT} (°F)		155	156	61	61
$RT_{NDT(U)}$ (°F)	-2	-50	-50	-50	-50
Mean RT_{NDT} (°F)	114	105	106	11	11
P(F/E) NRC	5.02 E-6	---	---	---	---
P(F/E) BWRVIP	No failure	---	---	---	---

Note: All values rounded for consistency. Mean RT_{NDT} calculated using peak surface fluence, and does not include a margin term.