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May 17, 2011

SOUTHERN COMPANY

Docket Nos.: 50-366

NL-11-0242

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

Edwin I. Hatch Nuclear Plant - Unit 2
Proposed Alternative in Accordance With 10 CFR 50.55a(a)(3)(i)
HNP-ISI-ALT-11

Ladies and Gentlemen:

Pursuant to 10 CFR 50.55a(a)(3)(i), Southern Nuclear Operating Company (SNC) hereby requests approval of an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, 2001 Edition through 2003 Addenda for Edwin I. Hatch Nuclear Plant (HNP) Unit 2. Currently, examination of reactor pressure vessel (RPV) circumferential shell welds is required in Table IWB-2500-1, category B-A, Item B1.11.

SNC requests NRC approval of Alternative HNP-ISI-ALT-11, Version 1 to continue the elimination of the RPV circumferential shell weld examinations as previously allowed by relief request RR-38 and the NRC's safety evaluation (SE) dated January 28, 2005, from July 31, 2007 through the period of extended operation (PEO), which expires on June 13, 2038.

This alternative is requested because compliance with the specified requirements would result in hardship without a commensurate increase in the level of quality and safety. According to 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph 50.55a(a)(3) may be used, when authorized by the NRC, if an applicant demonstrates that the proposed alternatives would provide an acceptable level of quality and safety.

By letter dated March 29, 2004, as supplemented by letter dated September 13, 2004, SNC submitted a proposed alternative (RR-38) to the requirements of the 1989 Edition of Section XI of the ASME Code for HNP Units 1 and 2. This alternative proposed the elimination of the circumferential shell weld examinations based on BWRVIP-05. During the review of RR-38, the NRC staff and SNC agreed that RR-38 should be amended to include the PEO (i.e., through its current 60-year renewed license).

The NRC's response was communicated in a January 28, 2005 safety evaluation (ADAMS Accession Number ML050130317). The NRC concluded that SNC's proposed alternative provided an acceptable level of quality and safety and it was authorized pursuant to 10 CFR 50.55a (a)(3)(i). However, the NRC staff determined that the neutron fluence values used in the alternative were based on a neutron fluence code (RAMA) that had not been reviewed and approved by the NRC staff. The NRC concluded that the fluence values were conservative through July 31, 2007. Therefore, the NRC staff limited the duration of RR-38 until July 31, 2007 pending SNC's use of an NRC approved fluence calculation methodology.

The NRC has addressed the use of RAMA fluence modeling methodology as defined in a May 13, 2005 letter (TAC NO. MB9765) from W.H. Bateman (NRC) to Bill Eaton (BWRVIP Chairman). The letter granted conditional approval for the use of RAMA as indicated in Enclosure 4.

After approval of RAMA, Alternative HNP-ISI-ALT-8 (which is similar to this proposed Unit 2 alternative) was submitted to the NRC for HNP-Unit 1. This alternative was approved by NRC letter dated December 6, 2007. (ADAMS Accession No. ML073130188). Approval for HNP-Unit 2 was not requested at that time because the HNP Unit 2 fluence evaluation had not been completed. A fluence evaluation for HNP Unit 2 using the currently approved RAMA code has been completed. The report indicated that there is a relatively small increase in projected fluence at the end of 60 years (50.1 Effective Full Power Years (EFPY)). Based upon the completion of the HNP Unit 2 fluence evaluation, SNC is requesting approval to continue the elimination of the HNP Unit 2 RPV circumferential shell weld examinations until the end of the period of extended operation; June 13, 2038.

Additional supporting information is provided in the Enclosure 1.

SNC requests approval of this proposed alternative by May 1, 2012 to support the Spring 2013 examinations required during the 3rd period of the 4th interval.

This letter contains no NRC commitments. If you have any questions, please contact N. J. Stringfellow at 205-992-7037.

Respectfully submitted,

Mark J Ajluni

M. J. Ajluni
Nuclear Licensing Director

MJA/PAH/lac

Enclosures: 1. Proposed Alternative in Accordance With 10 CFR 50.55a(a)(3)(i), HNP-ISI-ALT-11
2. Southern Nuclear Relief Request RR-38
3. NRC Evaluation of RR-38 (TAC Nos. MC2381 and MC2382) dated January 28, 2005
4. NRC Safety Evaluation of Proprietary EPRI Reports (TAC No. MB9765) dated May 13, 2005

References:

1. NL-04-1764 dated 09/13/1974, "Third Ten-Year Inservice Inspection Program, Submittal of Revised relief request RR-38."
2. NRC Letter of 01/28/2005, "Evaluation of Relief Request (RR) Number 38 (TAC Nos. MC2381 and MC2382).
3. NRC Letter of May 13, 2005, [Safety Evaluation of proprietary EPRI Reports, "BWR Vessel and Internals Project, RAMA Fluence Methodology Benchmark manual (BWRVIP-114," "RAMA Fluence Methodology Benchmark Manual-Evaluation of Regulatory Guide 1.1990 Benchmark Problems (BWRVIP-115)," "RAMA Fluence Methodology-Susquehanna Unit 2Surveillance Capsule Fluence Evaluation For cycles 1-5 (BWRVIP-117), " AND "RAMA Fluence Methodology Procedures Manual (BWRVIP-121)," and "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)" (TAC No. MB9765).
4. BWR-VIP-05, "Structural Integrity Calculation Package HTCH-@!Q-302R1, "Hatch Revised material Data Review and ART Calculations."
5. Transware Enterprises Document SNC-FLU-002-R-001, Revision 0, "Edwin I. Hatch Unit 2 reactor Pressure Vessel Fluence Evaluation at the End of Cycle 18 and 50.1 EFPY" dated April 23, 2007.
6. Final SER of the BWR Vessel Internals Project BWRVIP-05 Report, July 28, 1998. (TAC No. MA93925).
7. Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05, dated March 7, 2000. (Adams Accession No. ML031430372).

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. D. R. Madison, Vice President – Hatch
Ms. P. M. Marino, Vice President – Engineering
RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Mr. V. M. McRee, Regional Administrator
Mr. P. G. Boyle, NRR Project Manager– Hatch
Mr. E. D. Morris, Senior Resident Inspector – Hatch

Edwin I. Hatch Nuclear Plant-Unit 2
Proposed Alternative in Accordance With 10 CFR 50.55a(a)(3)(i)
HNP-ISI-ALT-11

Enclosure 1

Proposed Alternative in Accordance With 10 CFR 50.55a(a)(3)(i)
HNP-ISI-ALT-11

Edwin I. Hatch Nuclear Plant - Unit 2
Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i)
HNP-ISI-ALT-11

1. Requested Date For Approval and Basis

Approval is requested by May 1, 2012 to support the Spring 2013 examinations required during the 3rd Period of the 4th Interval.

2. ASME Code Component(s) Affected

Class 1 Category B-A Pressure retaining circumferential welds in the Reactor Pressure Vessel (RPV).

3. Applicable Code Edition and Addenda

ASME Section XI, 2001 Edition through the 2003 Addenda.

4. Applicable Code Requirements

Examination of RPV circumferential shell welds as required in Table IWB-2500-I, Category B-A, Item B1.11.

5. Reason for Request

Southern Nuclear Operating Company (SNC) proposes to continue the elimination of the RPV circumferential shell weld examinations as previously allowed by Relief Request RR-38 (Enclosure 2) and the NRC's safety evaluation (SE) dated January 28, 2005 (Enclosure 3) from July 31, 2007 through the period of extended operation (PEO). The PEO is the period of the HNP Unit 2 renewed license, which expires June 13, 2038.

6. Background

Per the NRC SE dated July 28, 1998 and Generic Letter 98-05, it was indicated that BWR licensees could request relief from the inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential reactor pressure welds (ASME Section XI Code, Table IWB-2500-1, Examination Category B-A, Item 1.11, Circumferential Shell Welds) by demonstrating:

1. At the expiration of their license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in this evaluation.

2. Licensees have implemented operator training and established procedures that limit the frequency of cold over pressure events to the amount specified in this report.

Based on these two requirements, SNC:

- Was previously granted approval for permanent deferral (during the initial 40-years of operation) of the HNP Unit 2 *augmented examination* requirements for the circumferential welds pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5).

- Justified relief from the volumetric examination of the circumferential RPV welds during the PEO.

Subsequently, by letter dated March 29, 2004, as supplemented by letter dated September 13, 2004, SNC submitted a proposed alternative (RR-38) to the requirements of 1989 Edition of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2. This alternative proposed the elimination of the circumferential shell weld examinations based on BWRVIP-05. During the review of RR-38, the NRC staff and SNC agreed that RR-38 should be amended to include the PEO (i.e., through its current 60-year renewed license).

The NRC's response was communicated in a January 28, 2005 safety evaluation (ADAMS Accession Number ML050130317). The NRC concluded that SNC's proposed alternative provided an acceptable level of quality and safety and it was authorized pursuant to 10 CFR 50.55a (a)(3)(i). However, the NRC staff determined that the neutron fluence values used in the alternative were based on a neutron fluence code that had not been reviewed and approved by the NRC staff. The NRC concluded that the fluence values were conservative through July 31, 2007. Therefore, the NRC staff limited the duration of RR-38 to July 31, 2007 pending SNC's use of an NRC approved fluence calculation methodology.

7. Proposed Alternative and Basis for Use

Proposed Alternative

SNC proposes an alternative to continue the elimination of the RPV circumferential shell weld examinations for HNP Unit 2 as previously allowed by Relief Request RR-38 and the NRC's safety evaluation for a period of time from July 31, 2007 through the PEO.

Basis for Use

The NRC has now addressed the approval and use of RAMA fluence modeling methodology as defined in a May 13, 2005 letter (TAC NO. MB9765) from W.H. Bateman (NRC) to Bill Eaton (BWRVIP Chairman). The letter granted conditional approval for the use of RAMA as indicated in Enclosure 4 of the May 13, 2005 letter. The NRC staff's conditions and SNC's response to the conditions are given below.

1. **Condition** - For plants that are similar in core, shroud and downcomer-vessel geometry to that of the Susquehanna and Hope Creek plants, i.e.,

BWR-IV plants, the RAMA methodology can be applied without a bias for the calculation of vessel neutron fluence.

Response – HNP - Unit 2 is a BWR-IV plant that is similar in core, shroud and downcomer-vessel geometry to that of the Susquehanna and Hope Creek plants (see Enclosure 4, Section 4.1). Therefore, the RAMA methodology can be applied without a bias for the calculation of vessel neutron fluence.

2. **Condition** - For plants (or plant groups) with a different geometry than that of the Susquehanna or Hope Creek plants, a plant-specific application for RPV neutron fluence is required to establish the value of a bias.

   Response – Not applicable to HNP - Unit 2.

3. **Condition** - Relevant benchmarking will be required for shroud and reactor internals applications.

   Response: Not applicable for this alternative.

A fluence model using the currently approved RAMA code was constructed and results reported in TransWare Enterprises Inc. report SNC-FLU-002-R-001 revision 0 dated 4/23/07. The report indicated that there is a relatively small increase in projected fluence at the end of 60 years (50.1 Effective Full Power Years (EFPY)).

Based on this data and updated Adjusted Reference Temperature (ART) calculations performed by Structural Integrity Associates (SIA), the HNP Unit 2 circumferential welds and axial continue to satisfy the limiting conditional failure probability at the end of the PEO. (Note: HNP Unit 2 is currently in its 31st year of commercial operation).

Circumferential Welds

The projected fluence value calculated by RAMA for the limiting circumferential weld at the end of the period of extended operation (PEO) at HNP Unit 2 is higher than that previously reported for HNP Unit 2 in Enclosure 2 of Relief Request 38 (RR-38). RR-38 used a GE developed fluence methodology, which was approved on a limited time basis (through July 2007). As shown in the table below, the fluence at the end of the PEO was previously stated in RR-38 as 0.244×10^{19} n/cm^2 while the RAMA calculated fluence used for this alternative at the end of the PEO is 0.324×10^{19} n/cm^2. The end of the PEO for HNP Unit 2 in RR-38 was based on an assumed 90% capacity factor, or 54 EFPY (effective full power years of operation). For the RAMA calculations, a more detailed evaluation of the capacity factor was performed, which used actual power history thru cycle 18 and resulted in the end of the PEO being defined at 50.1 EFPY.

For comparison purposes, updated values for the limiting HNP Unit 2 circumferential shell weld at the end of the PEO and values specified for

Combustion Engineering (CE) fabricated RVs in Table 2.6-5 of the July 28, 1998, NRC SER are shown below in Table 1. Also included for comparison are the values used as the basis for RR-38.

As demonstrated in Table 1, the projected fluence using the RAMA methodology increased. However, the HNP Unit 2 RT_{NDT} is still bounded by the mean RT_{NDT} calculated using either CE(VIP) or CE(CEOG) chemistry data as provided in the 1998 NRC SER for BWRVIP-05.

Table 1

	Limiting 64 EFPY CE-VIP Case Study Table 2.6-5 of NRC SE for BWRVIP-05	Limiting 64 EFPY CEOG Case Study Table 2.6-5 of NRC SE for BWRVIP-05	HNP Unit 2 Limiting Circ. Weld at License Expiration (54 EFPY) from RR-38	HNP Unit 2 Limiting Circ. Weld at License Expiration (50.1 EFPY) from SIA ART Calculation
Cu%	0.13	0.183	0.047	0.047
Ni%	0.71	0.704	0.049	0.049
Chemistry Factor (CF)	151.7	172.2	31.0	31.0
Fluence (10^{19} n/cm^2)	0.40	0.40	0.244	0.324
Delta RT_{NDT}	113.2	128.5	19.2	18.9
Initial RT_{NDT}	0	0	-50.0	-50.0
Mean RT_{NDT}	113.2	128.5	-30.8	-31.1
P(F/E) NRC	1.99×10^{-4}	4.38×10^{-4}	-------	-------
P(F/E) BWRVIP	-------	-------	-------	-------

Axial Welds

The projected fluence value calculated by RAMA for the limiting axial weld at the end of PEO at HNP Unit 2 higher than that previously reported for HNP Unit 2 in Enclosure 2 of RR-38. RR-38 used a GE developed fluence methodology, which was approved on a limited time basis (through July 2007). As shown in Table 2, the fluence at the end of the PEO was previously stated in RR-38 as 0.244×10^{19} n/cm^2 while the RAMA calculated fluence used for this alternative at the end of the PEO is 0.299×10^{19} n/cm^2. The end of the PEO for HNP Unit 2 in RR-38 was based on an assumed 90% capacity factor, or 54 EFPY (effective full power years of operation). For the RAMA calculations, a more detailed evaluation of the capacity factor was performed, which used actual power history thru cycle 18 and resulted in the end of the PEO being defined at 50.1 EFPY.

Table 2 has been provided to compare the axial weld information provided in RR-38 versus the RAMA information. The NRC issued a revised SER on BWRVIP-05 on March 7, 2000 and it indicated that the limiting axial welds should be compared with data in Table 3 of that document (Mod 2 in Table below). As demonstrated in Table 2, the $RT_{NDT(U)}$ remained unchanged and the Mean RT_{NDT} decreased; therefore, the values continue to be bounded.

In addition, axial welds and intersecting portions of circumferential welds will be examined to the extent practical, dependent upon interference by another component or restrictions due to the geometrical configuration. For those cases where the reduction in coverage is greater than 10%, relief will be requested pursuant to 10 CFR 50.55a requirements.

Table 2

	Mod 2	HNP Unit 2 Limiting Axial Weld at License Expiration (54 EFPY) from RR-38	HNP Unit 2 Limiting Axial Weld at License Expiration (50.1 EFPY) from SIA ART Calculation
Cu%	-------	0.216	0.216
Ni%	-------	0.043	0.043
Chemistry Factor (CF)	-------	98	98
Fluence (10^{19} n/cm^2)	-------	0.244	0.299
Delta RT_{NDT}	116	60.6	56.3
Initial RT_{NDT}	-2.0	-50.0	-50.0
Mean RT_{NDT}	114	10.6	6.3
P(F/E) NRC	5.02×10^{-6}	-------	-------
P(F/E) BWRVIP	-------	-------	-------

Conclusion

The HNP Unit 2 circumferential and axial welds continue to satisfy the limiting conditional failure probability at the end of the PEO and SNC has previously demonstrated that operator training and established procedures limit the frequency of cold over pressure events. Therefore, this alternative will continue to provide an acceptable level of quality and safety and approval is requested per 10 CFR 50.55a(a)(3)(i).

8. Precedents

Southern Nuclear Relief Request RR-38 (Enclosure 2)

A similar request was approved for HNP Unit 1 on December 6, 2007. (ADAMS Accession Number ML073130188) (Note: HNP Unit 2 relief was not requested at the same time as HNP Unit 1 due to the Fluence Evaluation not being completed for HNP Unit 2.)

9. References

BWR-VIP-05, Structural Integrity Calculation Package HTCH-21Q-302R1, "Hatch Revised Material Data Review and ART Calculations"

Transware Enterprises Document SNC-FLU-002-R-001 Revision 0, "Edwin I. Hatch Unit 2 Reactor Pressure Vessel Fluence Evaluation at the End of Cycle 18 and 50.1 EFPY", April 23, 2007

Final Safety Evaluation of the BWR Vessel Internals Project BWRVIP-05 Report, July 28, 1998. (TAC No. MA93925)

Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05, March 7, 2000 (ADAMS Accession No: ML031430372)

10. Status

Awaiting NRC Approval

Edwin I. Hatch Nuclear Plant-Unit 2
Proposed Alternative in Accordance With 10 CFR 50.55a(a)(3)(i)
HNP-ISI-ALT-11

Enclosure 2

Southern Nuclear Relief Request RR-38

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SOUTHERN COMPANY
Energy to Serve Your World℠

September 13, 2004

Docket Nos.: 50-321
50-366

NL-04-1764

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

Hatch Nuclear Plant
<u>Third 10-Year Interval Inservice Inspection Program</u>
<u>Submittal of Revised Relief Request RR-38</u>

Ladies and Gentlemen:

By letter dated March 29, 2004 Southern Nuclear Operating Company (SNC) submitted RR-38 to allow the deletion of the Section XI required RPV circumferential shell weld examinations (during the remainder of the 40 year initial license) based on NRC approved BWRVIP-05 (BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations). During the review of relief request RR-38, the NRC and SNC agreed that the relief request should be amended to include the period of extended operation (PEO). Accordingly, the attached revised relief request extends the requested duration to include the PEO.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

HLS/il/daj

Attachment: Revised Relief Request RR-38

cc: <u>Southern Nuclear Operating Company</u>
Mr. J. T. Gasser Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Hatch
RTYPE: CHA02.004

<u>U. S. Nuclear Regulatory Commission</u>
Dr. W. D. Travers, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

SOUTHERN NUCLEAR OPERATING COMPANY
EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2
THIRD 10-YEAR INTERVAL
REQUEST FOR RELIEF NO. RR-38

I. System/Component for Which Relief is Requested: This Relief Request applies to the Reactor Pressure Vessel (RPV) circumferential shell weld examinations for Hatch Units 1 and 2.

II. Code Requirements: The following 1989 Edition of ASME Section XI Code requirements apply to this request.

- IWB-2500 requires components to be examined as specified in Table IWB-2500-1.
- Table IWB-2500-1, Category B-A, Item No. B1.11 requires that all circumferential welds be essentially 100% examined.

III. Code Requirement from Which Relief is Requested: Southern Nuclear Operating Company (SNC) proposes to permanently exclude the examination of RPV circumferential shell welds as required in Table IWB-2500-1, Category B-A, Item No. B1.11 [This request is applicable for the current 40-year license and the Period of Extended Operation (PEO)].

IV. Background Information: By letter dated September 28, 1995 the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted BWRVIP-05 (BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations) to the NRC. BWRVIP-05 initially proposed to reduce the inspection coverage of the BWR RPV shell welds from essentially 100% of all RPV shell welds to 50% of the longitudinal welds and 0% of the circumferential welds. By letter dated October 29, 1996 the BWRVIP modified the recommendation in BWRVIP-05 to examine essentially 100% of the longitudinal welds and 0% of the circumferential welds (except for that portion of a circumferential weld intersecting with the longitudinal weld being examined).

The NRC issued their final safety evaluation (SE) for BWRVIP-05 by letter dated July 28, 1998. The SE stated that, "BWR licensees may request relief from the inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential reactor pressure welds (ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item 1.11, Circumferential Shell Welds) by demonstrating: (1) at the expiration of their license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in this evaluation, and (2) they have implemented operator training and established procedures that limit the frequency of cold over pressure events to the amount specified in this report." The SE indicated that the NRC staff concluded that a near-term safety concern did not exist; however, the NRC staff identified a need to evaluate the high conditional failure probabilities for axial welds. In a request for additional information, the NRC requested the BWRVIP to provide a more realistic potential for axial weld failures due to cold over-pressure events and to provide the failure frequency of axial welds based on NRC recommendations.

On November 10, 1998 the NRC issued Generic Letter 98-05 (Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds) to provide guidance for licensees to request relief from the augmented examination requirements for circumferential RPV shell welds. [By letter dated December 2, 1998 SNC requested approval to permanently exclude the examination of the Hatch Unit 1 RPV circumferential shell welds, based on this guidance, and by letter dated March 11, 1999 the NRC issued an SE for Hatch Unit 1 granting this request pursuant to 10 CFR 50.55a(a)(3)(i).]

By letters dated December 15, 1998 and November 12, 1999 the BWRVIP supplied additional information regarding axial weld failure probabilities. By letter dated March 7, 2000 the NRC issued a supplement to the July 28, 1998 SE concluding that, "the RPV failure frequency due to the failure of the limiting axial welds in the BWR fleet are below 5 x 10^{-6} per reactor-year, consistent with RG 1.154, given the assumptions described in the attached SE." Therefore, the issue with axial welds was resolved.

By letter dated January 31, 2001, in response to a request for additional information (RAI) for the Hatch Units 1 and 2 License Renewal Application (LRA), SNC supplied Hatch Units 1 and 2 RPV weld conditional failure probabilities and information regarding cold over-pressure events to the NRC. The NRC concluded in Section 4.6.2 of the October 5, 2001 Safety Evaluation Report that SNC has justified relief from the inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds during the PEO. The information supplied to the NRC in response to the RAIs is provided in Enclosure 1. Because of issues associated with the conditional failure probability of axial welds during the PEO, conditional failure probabilities for axial welds were also provided to the NRC and are included in this relief request (for information purposes) as Enclosure 2.

V. <u>Technical Basis</u>: Per the NRC SE dated July 28, 1998 and Generic Letter 98-05, BWR licensees may request relief from the inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential reactor pressure welds (ASME Section XI Code, Table IWB-2500-1, Examination Category B-A, Item 1.11, Circumferential Shell Welds) by demonstrating:

1. At the expiration of their license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in this evaluation.

2. Licensees have implemented operator training and established procedures that limit the frequency of cold over pressure events to the amount specified in this report.

Based on these two requirements, the NRC has previously:

- Granted approval for permanent deferral (during the initial 40-years of operation) of the Hatch Unit 1 augmented examination requirements for the circumferential welds pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5).

- Indicated that SNC has justified relief from the volumetric examination of the circumferential RPV welds during the PEO.

Hatch Units 1 and 2 are bounded by the NRC analysis for circumferential weld limiting conditional failure probabilities during and at the end of the PEO, as shown in Enclosure 1. Therefore, at the expiration of the initial 40-year license period, the Hatch Units 1 and 2 circumferential welds also will satisfy the limiting conditional failure probability for circumferential welds. (Note: Hatch Unit 1 is currently in its 29th year of commercial operation and Hatch Unit 2 is currently in its 25th year of commercial operation).

SNC has previously demonstrated that operator training and established procedures limit the frequency of cold over pressure events. This information was supplied to the NRC in the December 2, 1998 Hatch Unit 1 submittal (for the permanent deferral of the augmented examination requirements), which was subsequently approved by the NRC in the March 11, 1999 SE. This information was later referenced by SNC in the January 31, 2001 response to License Renewal RAIs, where, it was also noted that the operator training and procedures for Hatch Units 1 and 2 are the same. Extracts of this information are shown in Enclosure 3.

VI. Alternative Examinations: Axial welds and intersecting portions of circumferential welds will be examined to the extent practical, dependent upon interference by another component or restrictions due to the geometrical configuration. For those cases where the reduction in coverage is greater than 10%, relief will be requested pursuant to 10 CFR 50.55a requirements.

VII. Justification for Approval: At the expiration of the PEO (60 years) and therefore the initial 40-year license period as well (which corresponds to the start of the PEO), the Hatch Units 1 and 2 circumferential welds will satisfy the limiting conditional failure probability for circumferential welds. Procedures and training used to limit cold over-pressure events are the same for both Hatch units (approved for Hatch Unit 1 by NRC letter dated March 11, 1999). The NRC has previously concluded that elimination of the Hatch Units 1 and 2 circumferential weld examinations during the PEO is justified and the NRC has previously granted approval for the permanent deferral of the augmented circumferential weld examination requirements for Hatch Unit 1. Therefore, approval should be granted to eliminate the examination of the Hatch Units 1 and 2 RPV circumferential shell welds pursuant to 10 CFR 50.55a(a)(3)(i).

VIII. Implementation Schedule: Required for the Hatch 2 RPV weld examinations during the 18th Refueling Outage (currently scheduled to begin in February 2005).

IX. Relief Request Status: Awaiting NRC approval.

ENCLOSURE 1

EVALUATION OF LIMITING CONDITIONAL FAILURE PROBABILITIES FOR HATCH CIRCUMFERENTIAL WELDS DURING THE PERIOD OF EXTENDED OPERATION

By letter dated January 31, 2001, in response to a request for additional information (RAI) for the Hatch License Renewal Application (LRA), SNC supplied Hatch RPV weld conditional failure probabilities to the NRC. RAI 4.6-1 addressed the circumferential welds, and as shown below, the Hatch RPV conditional failure probability for circumferential welds is bounded by the NRC analysis.

> "The Hatch limiting circumferential weld properties from Tables 3-1 and 3-2 of the LRA Appendix E are compared to the information in Table 2.6-4 and Table 2.6-5 from the staff SER on BWRVIP-05."
>
> "The NRC staff used materials and fluence data in Tables 2.6-4 and 2.6-5 to evaluate failure probability of BWR circumferential welds at 32 and 64 EFPY. The NRC used Mean RT_{NDT} for the comparison. Mean RT_{NDT} is defined as: RT_{NDT} + ΔRT_{NDT}. The Mean RT_{NDT} used by the NRC have been compared to the Hatch values derived using Appendix E of the LRA. The Hatch 1 and Hatch 2 values at 54 EFPY are bounded by the 32 EFPY analysis by the NRC by at least 40 ^{0}F, and almost 75 ^{0}F at 64 EFPY. Although a conditional failure probability has not been calculated, the fact that the Hatch 54 EFPY value is bounded by the 32 and 64 EFPY value the staff used leads to the conclusion that Hatch RPV conditional failure probability is bounded by the NRC analysis."

See the table below for the comparison of values.

Group	CE(VIP) 32 EFPY	CE(CEOG) 32EFPY	CE(VIP) 64 EFPY	CE(CEOG) 64 EFPY	Hatch 1 54 EFPY	Hatch 2 54 EFPY
Cu%	0.13	0.183	0.13	0.183	0.197	0.047
Ni%	0.71	0.704	0.71	0.704	0.060	0.049
CF	151.7	172.2	151.7	172.2	91.0	31.0
Fluence (10^{19} n/cm^2)	0.20	0.20	0.40	0.40	0.236	0.244
ΔRT_{NDT} (^{0}F)	86.4	98.1	113.2	128.5	55.5	19.2
$RT_{NDT(U)}$ (^{0}F)	0	0	0	0	-10	-50
Mean RT_{NDT} (^{0}F)	86.4	98.1	113.2	128.5	45.5	-30.8
P(F/E) NRC	2.81E-5	6.34E-5	1.99E-4	4.38E-4	---	---
P(F/E) BWRVIP	No Failure	---	---	---	---	---

References:

1. Hatch License Renewal Application, Appendix E, Tables 3-1 and 3-2.
2. Final SER of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925), dated July 28, 1998.
3. GE-NE-A00-05389-08, July 1995 Power Uprate Evaluation Task Report for Edwin I. Hatch Plant Units 1 and 2, 110% Power Uprate Revised Impact on Vessel Fracture Toughness.
4. GE-NE-A13-00402-9, March 1998 Extended Power Uprate Evaluation Task Report for Edwin I. Hatch Plant Units 1 and 2 Revised Impact on Vessel Fracture Toughness.
5. BWRVIP-74 – BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines, TR-113596.
6. Structural Integrity Associates Letter, SIR-00-160, Rev. 0, December 18, 2000.

ENCLOSURE 2

EVALUATION OF LIMITING CONDITIONAL FAILURE PROBABILITIES FOR HATCH AXIAL WELDS DURING THE PERIOD OF EXTENDED OPERATION

In a response to RAI 4.6-1, SNC supplied Hatch RPV axial weld conditional failure probabilities to the NRC. As shown below, the Hatch RPV conditional failure probability for axial welds is bounded by the NRC analysis. RAI-4.6-2 states (in part):

"The SER in the May 7, 2000 letter supercedes the analysis in the July 28, 1998 letter. Therefore, the applicant should revise its analysis to compare the mean RT_{NDT} for the Plant Hatch axial welds to the mean RT_{NDT} for Pilgrim Mod 2."

In response, SNC stated:

"The Hatch limiting axial weld properties from Table 3-1 and 3-2 of Appendix E are compared to the information in Table 2.6-4 and Table 2.6-5 from the staff SER on BWRVIP-05. The NRC noted that it issued a revised SER on BWRVIP-05 on March 7, 2000 and that the limiting axial welds should be compared with data in Table 3 of that document (Mod 2 in Table below). Mean RT_{NDT} is defined as: Mean $RT_{NDT} = RT_{NDT} + \Delta RT_{NDT}$. The Mean RT_{NDT} used by the NRC have been compared to the Hatch values derived using Appendix E of the LRA. A comparison of the Mean RT_{NDT} values from the NRC report with the Hatch data shows that the NRC analysis bounds the Hatch welds. Although a conditional failure probability has not been calculated, the fact that the Hatch 54 EFPY value is less than the 64 EFPY value the staff used leads to the conclusion that Hatch is bounded by the NRC analysis."

Group	Mod 2	Hatch 1 54 EFPY	Hatch 2 54 EFPY
Cu%		0.316	0.216
Ni%		0.724	0.043
CF		219	98.0
Fluence (10^{19} n/cm^2)		0.347	0.244
ΔRT_{NDT} (°F)		155.1	60.6
$RT_{NDT(U)}$ (°F)	-2	-50	-50
Mean RT_{NDT} (°F)	114	105.1	10.6
P(F/E) NRC	5.02E-6	---	---
P(F/E) BWRVIP		---	---

References: See circumferential weld references.

ENCLOSURE 3

EVALUATION OF OPERATOR TRAINING AND ESTABLISHED PROCEDURES

Plant Hatch has procedures in place which monitor and control reactor pressure, temperature, and water inventory during all aspects of cold shutdown and refueling operations which minimizes the likelihood of a Low Temperature Over-Pressurization (LTOP) event from happening. In addition to procedural controls, periodic Licensed Operator Training further reduces the possibility of occurrence of LTOP events. Initial Licensed Operator Training and Simulator Training of plant heatup and cooldown events includes performance of surveillance tests and monitoring which ensure pressure-temperature curve compliance. In addition, periodic operator training reinforces management's expectations for strict procedural compliance.

Finally, Southern Nuclear operating personnel continuously review industry operating experiences to ensure that Plant Hatch procedures consider the impact of actual events, including LTOP events. Appropriate changes to procedures and training are then implemented to preclude similar situations from occurring at Plant Hatch.

Based on the above, the probability of an LTOP event at Plant Hatch is considered to be less than or equal to that used in the NRC evaluation.

Edwin I. Hatch Nuclear Plant-Unit 2
Proposed Alternative in Accordance With 10 CFR 50.55a(a)(3)(i)
HNP-ISI-ALT-11

Enclosure 3

NRC Evaluation of RR-38
(TAC Nos. MC2381 and MC2382) dated January 28, 2005

January 28, 2005

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 - EVALUATION OF RELIEF REQUEST (RR) NUMBER 38 (TAC NOS. MC2381 AND MC2382)

Dear Mr. Sumner:

By letter dated March 29, 2004, as supplemented by letter dated September 13, 2004, Southern Nuclear Operating Company, Inc. (SNC or the licensee), submitted proposed alternatives to the requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(ii) for the Edwin I. Hatch Nuclear Plant (Hatch), Units 1 and 2. SNC proposed two additional relief requests, RR-39 and RR-40, in the March 29, 2004, letter. The Nuclear Regulatory Commission (NRC) staff reviewed RR-39 and RR-40, and provided the results of that review in a letter to you dated January 7, 2005.

The following paragraphs summarize the NRC staff's findings regarding RR-38:

In RR-38, the licensee proposed an alternative inspection program for the circumferential welds in the reactor vessels (RVs) of Hatch, Units 1 and 2. The alternative program applies a probabilistic fracture toughness analysis to justify eliminating volumetric examinations of the RV circumferential welds, as required in Table IWB-2500-1 of Section XI of the ASME Code, and the augmented volumetric inspections for these welds, as required in 10 CFR 50.55a(g)(6)(ii)(A)(2). The NRC staff completed their review and determined that the licensee's proposed alternative provides an acceptable level of quality and safety. Therefore, the licensee's alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

The March 29, 2004, letter requested that RR-38 be approved for the remainder of the 40-year initial license. During review of the RR, the NRC staff and SNC agreed that the RR should be amended to include the period of extended operation (i.e., through its current 60-year renewed license). Upon further review, the NRC staff determined that the neutron fluence values used in the RR were based on a calculational code (i.e., RAMA) that had not been reviewed and approved by the NRC staff. The licensee had initially anticipated that the NRC staff's review and acceptance of the RAMA code for calculating reactor pressure vessel neutron fluence would be complete by the time this RR was needed. The NRC staff is currently reviewing, but has not yet approved the use of the RAMA code for neutron fluence calculations. As a result, the NRC staff reviewed the licensee's current, NRC approved neutron fluence analyses and concluded that the fluence values calculated using this methodology are conservative through July 31, 2007. Therefore, the NRC staff limits the duration of RR-38 until July 31, 2007.

Please note that if you plan to seek approval for RR-38 beyond July 31, 2007, you must submit a revised RR based on an analysis that uses an NRC staff-approved neutron fluence calculation methodology (e.g., RAMA, if approved for use by the NRC staff at that time). This issue was discussed with SNC staff in a teleconference on September 20, 2004.

The NRC staff's Safety Evaluation is enclosed. If you have any questions, please contact Christopher Gratton at 301-415-1055.

Sincerely,

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosure: As stated

cc w/encl: See next page

Please note that if you plan to seek approval for RR-38 beyond July 31, 2007, you must submit a revised RR based on an analysis that uses an NRC staff-approved neutron fluence calculation methodology (e.g., RAMA, if approved for use by the NRC staff at that time). This issue was discussed with SNC staff in a teleconference on September 20, 2004.

The NRC staff's Safety Evaluation is enclosed. If you have any questions, please contact Christopher Gratton at 301-415-1055.

Sincerely,

/RA/

John Nakoski, Chief, Section 1
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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosure: As stated

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RELIEF REQUEST (RR) NO. RR-38

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

In Letter No. NL-04-0478 dated March 29, 2004, as supplemented by Serial Letter No. NL-04-1764 dated September 13, 2004, Southern Nuclear Operating Company, Inc. (SNC, the licensee), proposed an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI and applicable Addenda (henceforth Section XI), regarding the volumetric examination requirements for reactor vessel (RV) circumferential shell welds at the Edwin I. Hatch Nuclear Plant (Hatch), Units 1 and 2. The licensee also requested relief from the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g)(6)(ii)(A)(2), as they pertain to performing augmented volumetric inspections of the RV circumferential welds at Hatch, Units 1 and 2.

Nuclear Regulatory Commission (NRC, the Commission) staff approval of the RR would authorize the use of a proposed alternative to these volumetric examinations requirements in accordance with the alternative probabilistic fracture mechanics methods discussed in Electric Power Research Institute Topical Report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," and with the NRC's guidelines for proposing these alternative programs, as established in NRC Generic Letter (GL) 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds."

2.0 REGULATORY EVALUATION

2.1 Inservice Inspection Requirements

Inservice inspection (ISI) of the ASME for Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of

Enclosure

paragraph (g) may be used, when authorized by the NRC, if: (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection [ISI] of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable Code of record for the second 10-year ISI for Hatch, Units 1 and 2 is the 1989 Edition of the ASME Code, Section XI.

2.2 Augmented Inservice Inspections Requirements for RV Shell Welds

10 CFR 50.55a(g)(6)(ii)(A)(2) requires licensees to augment their reactor vessel examinations by implementing, as part of the ISI interval in effect on September 8, 1992, the examination requirements for RV shell welds specified in Item B1.10, Section XI, Table IWB-2500-1, Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel." Section XI Item B1.10 includes the volumetric examination requirements for both circumferential RV shell welds, as specified in Section XI Item B1.11, and longitudinal RV shell welds, as specified in Section XI Item B1.12. 10 CFR 50.55a(g)(6)(ii)(A)(2) defines "essentially 100% examination" as covering 90 percent or more of the examination volume of each weld.

2.3 Additional Regulatory Guidance

2.3.1 BWRVIP-05 Report

By letter dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, May 16, June 4, June 13, and December 18, 1997, and January 13, 1998, the Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP), a technical committee of the BWR Owners Group (BWROG), submitted proprietary report BWRVIP-05. The BWRVIP-05 report evaluates the current inspection requirements for RV shell welds in BWRs, formulates recommendations for alternative inspection requirements, and provides a technical basis for these recommended requirements. As modified, the BWRVIP-05 proposed to reduce the scope of inspection of BWR RV welds from essentially 100 percent of all RV shell welds to examination of 100 percent of the axial (i.e., longitudinal) welds and essentially zero percent of the circumferential RV shell welds, except for the intersections of the axial and circumferential welds. In addition, the report includes proposals to provide alternatives to ASME Code requirements for successive and additional examinations of circumferential welds, provided in paragraph IWB-2420 and IWB-2430 respectively, of Section XI of the ASME Code.

On July 28, 1998, the NRC staff issued a Safety Evaluation Report (SER) on BWRVIP-05. This evaluation concluded that the failure frequency of RV circumferential welds in BWRs was sufficiently low to justify elimination of ISI of these welds. In addition, the evaluation concluded that the BWRVIP proposals on successive and additional examinations of circumferential welds were acceptable. The evaluation indicated that examination of the circumferential welds will be performed if axial weld examinations reveal an active degradation mechanism. The NRC staff supplemented this SER in an SER to the BWRVIP dated March 7, 2000. In this SER, the NRC staff updated the interim probabilistic failure frequencies for RV axial shell welds and revised the Table 2.6-4 to correct a typographical error in the 32 effective full power years (EFPY) Mean RT_{NDT} value cited for the limiting Chicago Bridge and Iron (CB&I) case study for circumferential welds. The correction changed the 32 EFPY Mean RT_{NDT} value for the CB&I case study from 109.5 •F to 134.9 •F.

In the BWRVIP-05 report, the BWRVIP committee concluded that the conditional probabilities of failure for BWR RV circumferential welds are orders of magnitude lower than that of the axial welds. As a part of its review of the report, the NRC conducted an independent probabilistic fracture mechanics assessment of the results presented in the BWRVIP-05 report. The NRC staff's assessment conservatively calculated the conditional probability of failure values for RV axial and circumferential welds during the initial (current) 40-year license period and at conditions approximating an 80-year vessel lifetime for a BWR nuclear plant. The failure frequency is calculated as the product of the frequency for the critical (limiting) transient event and the conditional probability of failure for the weld.

The NRC staff determined the conditional probability of failure for axial and circumferential welds in BWR vessels fabricated by CB&I, Combustion Engineering (CE), and Babcock and Wilcox (B&W). The analysis identified a cold overpressure event that occurred in a foreign reactor as the limiting event for BWR RVs, with the pressure and temperature from this event used in the probabilistic fracture mechanics calculations. The NRC staff estimated that the probability for the occurrence of the limiting overpressurization transient was 1×10^{-3} per reactor year. For each of the vessel fabricators, Table 2.6-4 of the NRC staff's SER of March 7, 2000, identifies the conditional failure probabilities for the plant-specific conditions with the highest projected reference temperature (for that fabricator) through the expiration of the initial 40-year license period. Table 2.6-5 of NRC staff's SER of July 28, 1998, identifies the conditional failure probabilities for the plant-specific conditions with the highest projected reference temperature (for that fabricator) through the expiration of an 80-year license period, which constitutes the licensing basis if two 20-year extended periods of operation have been granted for a BWR-designed nuclear power plant.

2.3.2 Generic Letter 98-05

On November 10, 1998, the NRC issued GL 98-05 that states that BWR licensees may request permanent (i.e., for the remaining term of operation under the existing, initial license) relief from the ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds (ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, "Circumferential Shell Welds") by demonstrating conformance with the following safety criteria:

(1) At the expiration of the operating license, the licensees will have demonstrated that limiting probability of failure for their limiting RV circumferential welds will continue to satisfy (i.e., be less than) the limiting conditional failure probability for circumferential weld assessed in the applicable BWRVIP-05 limiting case study.

(2) Licensees have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the NRC staff's July 28, 1998, SER.

In GL 98-05, the NRC staff stated that licensees applying the BWRVIP-05 criteria would need to continue performing the volumetric inspections of all axial RV shell welds that are required by the ASME Code, Section XI, Table IWB-2500-1, Inspection Category B-A, Item B1.12, and the augmented volumetric inspections of the RV axial shell welds that are required under 10 CFR 50.55a(g)(6)(ii)(A)(2). For plants that are currently licensed to operate in accordance with their initial 40-year operating licenses, the limiting case studies are provided in Table 2.6-4 of the revised SER on BWRVIP-05 dated March 7, 2000. For plants that have been granted operating licenses to operate for an extended period of operation, the limiting case studies are provided in Table 2.6-5 of the NRC staff's SER of July 28, 1998.

3.0 TECHNICAL EVALUATION

3.1 Code Requirement for Which Relief is Requested

The licensee requested relief from the following requirements in the 1989 Edition of Section XI of the ASME Code (Section XI):

- Subarticle IWB-2500, Table IWB 2500-1, Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," No. B1.11, "Circumferential Shell Welds."

3.2 Licensee's Proposed Alternative to the ASME Code

Using the guidelines of GL 98-05 and Topical Report BWRVIP-05 and the NRC staff's determination in its July 28, 1998, SER on BWRVIP-05, the licensee proposed to use a probabilistic fracture mechanics evaluation for the circumferential shell welds in the Hatch, Units 1 and 2 RVs as the basis for eliminating the required volumetric examinations and augmented volumetric examinations for the welds through the expiration of the extended periods of operation for Hatch, Units 1 and 2.

The licensee proposed the following alternative in lieu of performing the required volumetric examinations of the RV circumferential shell welds:

> Axial welds and intersecting portions of circumferential welds will be examined to the extent practical, dependent upon interference by another component or restrictions due to the geometrical configuration. For those cases where the reduction in coverage is greater than 10%, relief will be requested pursuant to 10 CFR 50.55a requirements.

In SNC Letter No. NL-04-1764, dated September 13, 2004, the licensee clarified that the alternative inspection program in RR-38 is requested through the expiration of the periods of extended operation for Hatch, Units 1 and 2.

3.3 Licensee's Bases for Alternative

The licensee based RR-38 on the NRC's RR provisions of GL 98-05 and the guidelines of BWRVIP-05. The licensee cited the following acceptance criteria as the bases for evaluating the acceptability of RR-38.

> Per the NRC SE dated July 28, 1998 and Generic Letter 98-05, BWR licensees may request relief from the inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential reactor pressure welds (ASME Section XI Code, Table IWB-2500-1, Examination Category B-A, Item 1.11, Circumferential Shell Welds) by demonstrating:
>
> 1. At the expiration of their license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in this evaluation . . . .[GL 98-05 Safety Criterion 1].
>
> 2. Licensees have implemented operator training and established procedures that limit the frequency of cold over pressure events to the amount specified in this report . . . .[GL 98-05 Safety Criterion 2].

3.3.1 License Basis for Conforming with GL 98-05 Safety Criterion 1 - Criterion for Conditional Probabilities of Failure

In letter dated March 29, 2004, the licensee provided its 54 EFPY Mean RT_{NDT} calculations for the limiting circumferential welds in the Hatch, Units 1 and 2 RVs (Refer to Enclosure 1 of RR-38) in order to support its basis for meeting GL 98-05 Safety Criterion 1 and to demonstrate that the 54 EFPY Mean RT_{NDT} values for Hatch, Units 1 and 2 are bounded by the Mean 64 EFPY RT_{NDT} value for the limiting CE-VIP case study.

3.3.2 License Basis for Conforming with GL 98-05 Safety Criterion 2 - Criterion on Mitigating the Probability of Cold Overpressurization Events

The licensee provided the following technical basis for meeting GL 98-05 Safety Criterion 2:

> SNC has previously demonstrated that operator training and established procedures limit the frequency of cold over pressure events. This information was supplied to the NRC in the December 2, 1998 Hatch Unit 1 submittal (for the permanent deferral of the augmented examination requirements), which was subsequently approved by the NRC in the March 11, 1999 SE. This information was later referenced by SNC in the January 31, 2001 response to License Renewal RAIs, where, it was also noted that the operator training and procedures for Hatch Units 1 and 2 are the same. Extracts of this information are shown in Enclosure 3 [of SNC Letter NL-04-478, dated March 29, 2004].

4.0 NRC STAFF EVALUATION

As discussed in Section 2.3.2 of this SE, GL 98-05 provides two criteria that BWR licensees requesting relief from ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential RV welds (ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, Circumferential Shell Welds) must satisfy. These criteria are intended to demonstrate that the conditions at the applicant's plant are bounded by those in the safety evaluation. The licensee will still need to perform the required inspections of "essentially 100 percent" of all axial welds.

4.1 Neutron Fluence Calculation for RR-38

For any given RV circumferential or axial weld material, the conditional probability of failure increases with the material's neutron fluence value and mean RT_{NDT} value, as projected to the expiration of the operating license for the facility. GL 98-05 stipulates that, at the expiration of the operating license, the mean RT_{NDT} estimates for circumferential welds should satisfy the limiting conditional failure probability for the weld materials, as stated in the NRC staff's SER of July 28, 1998. The neutron fluence values for the RV circumferential welds at the inside surface of the RV are critical inputs to the mean RT_{NDT} estimate calculations.

SNC's current method for calculating neutron fluence does not conform with the NRC staff's recommended methodology in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure VEssel Neutron Fluence." SNC had committed to have a conforming neturon fluence calculational methodology (i.e., the RAMA code) for Hatch, Units 1 and 2 approved by December 15, 2004. However, the approval of the RAMA code has been delayed. In a letter dated July 13, 2004 (SNC Letter No. NL-04-1123), SNC requested to revise its commitment date for having a neutron fluence calculational methodology that is compliant with RG 1.190, from December 15, 2004, to July 31, 2007. SNC stated that the previous commitment date was selected arbitrarily (based on the expectation that RAMA would be approved by this date), and that there was sufficient conservatism in the pressure-temperature (P-T) curves calculated using the current fluence methodology to allow the plant to operate safely until July 31, 2007. The NRC approved the commitment change request in a letter to SNC dated September 9, 2004. The NRC staff is currently reviewing the RAMA code and anticipates that the review will be complete before July 31, 2007.

NRC staff experience has shown that neutron fluence values that are calculated using methods that do not conform with RG 1.190 are typically within +/- 40 percent of the values that would be obtained using the recommended methodology of RG 1.190. As stated previously, SNC's current fluence methodology does not conform with RG 1.190. SNC stated in a Letter No. NL-04-1152, also dated July 13, 2004, that based on the current fluence methodology, the estimated fluence values on August 1, 2007, for Hatch Units 1 and 2, will be 24.2 and 22.1 EPFY, respectively. These values are 44.8 percent and 40.9 percent of the 54 EFPY neutron fluence values estimated by the current methodology for the Hatch, Units 1 and 2, respectively. As a result, neutron fluence estimates for August 1, 2007, are more conservative than the values estimated for the fluence at 54 EFPY, even considering the 40 percent adjustment for the nonconforming methodology.

Therefore, the NRC staff finds the licensee's neutron fluence estimates are acceptable to warrant approval of fluence values used in the 54 EFPY Mean RT_{NDT} analyses. However, the NRC staff is limiting the acceptance of RR-38 to the period through July 31, 2007. SNC also stated in NL-04-1152 that the P-T limit curves will be evaluated and revised, if necessary at that time.

4.2 Circumferential Weld Conditional Failure Probability

The NRC staff's SER for the BWRVIP-05 report evaluated the conditional failure probabilities for axial and circumferential shell welds in the limiting BWR RV designs manufactured by different vendors, including RVs manufactured using by CE, CB&I, and B&W. The SER also reported the Mean RT_{NDT} calculations and values that were derived from the conditional failure probabilities for the limiting case studies. For a plant granted a renewed operating license, the evaluation criteria for the limiting conditional failure probabilities and Mean RT_{NDT} values are those listed for the limiting case studies specified in Table 2.6-5 of the staff's SER of July 28, 1998. The associated limiting case studies, conditional failure probabilities, and Mean RT_{NDT} values listed in Table 2.6-5 of the NRC staff's SER are limited to plants that have accumulated no more than 64 EFPY of power operation.

The renewed operating licenses for Hatch, Units 1 and 2 were approved and issued by the NRC on January 15, 2002. In the renewed operating licenses, the staff granted power operation through August 6, 2034, for Hatch, Unit 1 and June 13, 2038, for Hatch, Unit 2, which represent operations through 54 EFPY of power operation. The period of applicability in Table 2.6-5 of the SER on BWRVIP-05 is bounding for operations of the Hatch, Units 1 and 2 reactors to the expiration of the extended operating licenses and is representative of the evaluation for RR-38. Since the Hatch, Units 1 and 2 RVs were fabricated by CE, the CE-VIP limiting case study in Table 2.6-5 provides the applicable case-study conditional probability of failure value and Mean RT_{NDT} value criterion for the evaluation of RR-38.

In the license renewal application for the Hatch, Units 1 and 2 reactors, SNC identified the calculation of the Mean RT_{ndt} values for the Hatch, Units 1 and 2 RV circumferential welds as a time-limiting aging analysis (TLAA) for the application. In the staff's evaluation in Section 4.6 of NUREG-1803, "Safety Evaluation Report Related to the License Renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2 (December 2001)", the NRC staff concluded that SNC had performed a valid TLAA analysis to justify re-submittal of the alternative inspection proposal for the Hatch, Units 1 and 2 RV circumferential welds to the expiration of the extend periods of operation for the reactor units. SNC's submittal of RR-38 on March 29, 2004, as amended in SNC Letter No. NL-04-1764, dated September 13, 2004, was performed to justify elimination of the volumetric examinations and augmented volumetric examinations for the RV circumferential welds through the expiration of the extended periods of operation for Hatch, Units 1 and 2.

The NRC staff performed an independent calculation of the Mean RT_{NDT} values for the limiting Hatch, Units 1 and 2 RV circumferential welds through 54 EFPY. Table 4.1-1 on page 8 of this SE provides a summary of the Mean RT_{NDT} values calculated by the staff for the Hatch, Units 1 and 2 RV through 54 EFPY and a comparison of the staff's Mean RT_{NDT} values to both the corresponding Mean RT_{NDT} values calculated by SNC and the Mean RT_{NDT} value criterion for the limiting CE-VIP case study at 64 EFPY.

The results in Table 4.1-1 demonstrate that the Mean RT_{NDT} values calculated by the licensee for the Hatch, Units 1 and 2 RV circumferential welds are less than that for the limiting CE-VIP case study and are in agreement with those calculated by the NRC staff. Based on this analysis, the NRC staff concludes that SNC has provided a valid basis for concluding that the conditional probability of failure values for the Hatch, Units 1 and 2 RV circumferential welds are sufficiently low to justify elimination of the volumetric examinations that are required for these welds through 54 EFPY. However, for the reasons stated in Sections 4.1 and 5.0 of this SE, the NRC staff is limiting its authorization for elimination of the required examinations until July 31, 2007.

Table 4.1-1 Comparison of NRC and SNC 54 EFPY Mean RT_{NDT} Calculations to the 64 EFPY Mean Calculations for the Limiting CE-VIP Case Study on BWRVIP-05

	Limiting 64 EFPY CE-VIP Case Study	NRC 54 EFPY Mean RT_{NDT} Calculations for Hatch, Unit 1 (Note 1)	SNC 54 EFPY Mean RT_{NDT} Calculations for Hatch, Unit 1 (Note 1)	NRC 54 EFPY Mean RT_{NDT} Calculations for Hatch, Unit 2 (Note 1)	SNC 54 EFPY Mean RT_{NDT} Calculations for Hatch, Unit 2 (Note 1)
Alloy % Cu	0.13	0.197	0.197	0.047	0.047
Alloy % Ni	0.71	0.060	0.060	0.049	0.049
$RT_{NDT(U)}$ (°F)	0	-10.0	-10.0	-50.0	-50.0
Fluence (10^{19} n/cm^2)	0.400	0.236	0.236	0.244	0.244
Chemistry Factor	151.7	91.4	91.0	30.7	31.0
ΔRT_{NDT} (°F)	113.2	55.7	55.5	19.0	19.2
Mean RT_{NDT} (°F)	113.2	45.7	45.5	-31.0	-30.8
NRC Established Probability of Failure [P(F/E)]Criterion for Case / Result for Plant Specific Calculation	1.99E-4 (no failure: Note 2)	Mean RT_{NDT} is Lower than Case Study Mean RT_{NDT} : Criterion is met. (Note 2)	Mean RT_{NDT} is Lower than Case Study Mean RT_{NDT} : Criterion is met. (Note 2)	Mean RT_{NDT} is Lower than Case Study Mean RT_{NDT} : Criterion is met. (Note 2)	Mean RT_{NDT} is Lower than Case Study Mean RT_{NDT} : Criterion is met. (Note 2)

Notes:

1. For the Hatch, Units 1 and 2 RVs, the limiting circumferential weld materials determined by the staff were equivalent to those determined by SNC. For Hatch-1, the limiting RV circumferential weld is 1-313A, which was fabricated from weld heat No. 90099. For Hatch-2, the limiting RV circumferential weld is 301-871, which was fabricated from weld heat No. 4P6052.

2. If the plant-specific Mean RT_{NDT} is less than the Mean RT_{NDT} associated with Limiting Case Study, the staff concludes that probability of failure for the plant-specific circumferential weld under review will be less that for the limiting circumferential weld in the Limiting Case Study. BWR plants that meet this criterion may conclude that the probability of failure for the limiting circumferential RV welds is sufficient to justify elimination of the volumetric examinations required by Section XI of the ASME Code (Examination Category B-A, Item B.1.11) and augmented volumetric examinations for the circumferential welds required by 10 CFR 50.55a(g)(6)(ii)(A)(2).

4.3 Minimizing the Possibility of Low Temperature Overpressurization

The licensee stated that its bases for meeting Acceptance Criterion 2 of GL 98-05 and for demonstrating that the licensee has implemented acceptable procedures and controls for mitigating a low-temperature-overpressurization event are given in the "Consideration of Low Temperature - Over Pressurization Events" section of the licensee's enclosure to SNC Serial Letter HL 5710, dated December 2, 1998, and are applicable to the evaluation of RR-38. In this letter, the licensee stated that the following operational controls, procedural controls, and staff training practices are in place to minimize the possibility of a low temperature overpressurization event.

4.3.1 Operational and System Design Considerations

From an operational basis, the reactor feedwater system (RFS), high pressure coolant injection system (HPCI), reactor core isolation cooling system (RCIC), and standby liquid control system (SLC) are the high pressure systems that provide coolant or have the potential to provide coolant at high pressure into the RV. The HPCI and RCIC pumps are steam driven and cannot function during cold shutdown. The RFS pumps automatically trip during transient and postulated loss-of-coolant accidents (LOCA) conditions, and are manually tripped during routine reactor shutdowns. Since RFS pumps are steam driven, they cannot be operated during cold shutdown condition. Although not addressed in the licensee's submittal, the NRC staff noted that SLC is solely a manual injection system; there are no automatic starts associated with SLC at Hatch, Units 1 and 2. Operator initiation of the SLC occurs only in accordance with applicable Hatch, Units 1 and 2 emergency operating procedures and would not occur during normal utility planned shutdowns of the reactors or during transient operating conditions. SLC might be manually initiated during a postulated design basis LOCA; however, if manually initiated, the SLC injection rate of approximately 40 gallons per minute (gpm) would allow operators sufficient time to control RV water level and pressure during the postulated event.

The core spray system (CS), low-pressure coolant injection/residual heat removal system (LPCI/RHR), control rod drive system (CRD), and reactor water cleanup system (RWCU) can also inject coolant into the reactor. CS and LPCI/RHR are low pressure emergency core cooling systems (ECCS), whose pumps' create a shutoff head of 375 pounds per square inch differential (psid) for CS and 223 psid for LPCI/RHR. Should either of these systems be started (i.e., inject as designed) during cold shutdown, the resulting reactor pressure and temperature would be below the P-T limits. CRD and RWCU use a feed-and-bleed process to control RV level and pressure during normal cold shutdown conditions.

4.3.2 Procedural Considerations

Plant-specific normal operating and transient operating procedures have been established to provide guidance to the operators regarding compliance with the Technical Specification P-T limits. These procedures direct operators to respond to any unexpected or unexplained rise in reactor water level, which could result from spurious actuation of an injection system. The procedural actions include preventing condensate pump injection, securing ECCS system injection, tripping CRD pumps, terminating all other injection sources, and lowering RV level via the RWCU system. In addition, plant-specific emergency operating procedures have been established to ensure that proper operating actions are followed during postulated design-basis LOCAs. The emergency procedures include control of reactor water level, reactor pressure, and reactor temperature during these postulated events and instructions on how and when SLC should be manually initiated.

4.3.3 Operator Training

The licensee emphasized that training and testing of control room operators is an integral part of ensuring the abilities of the operators to implement these procedures. On the basis of the P-T limits of the operating systems, operator training, and established plant-specific procedures, the licensee determined that a nondesign-basis cold overpressure transient is unlikely to occur.

4.3.4 Staff Determination on SNC's Basis for Meeting Acceptance Criterion 2 of GL 98-05

The staff concluded that, based on the licensee's information provided about the systems that inject at high pressures, operator training, and plant-specific procedures at Hatch, Units 1 and 2, the possibility of a low temperature overpressurization event will be minimized, and thus, the licensee has provided a sufficient basis to support the NRC staff's approval of the alternative examination request for circumferential shell welds in the Hatch, Units 1 and 2 RVs.

5.0 <u>CONCLUSION FOR RR-38</u>

The NRC staff has completed its review of the licensee's submittal and determined that the licensee conforms to the applicable safety evaluation criteria in NRC GL 98-05 and in the BWRVIP-05 report. The NRC staff has also determined that the licensee has acceptably demonstrated that the conditional probability of failure values for the Hatch, Units 1 and 2 RV circumferential welds are sufficiently low to justify the elimination of the augmented volumetric examinations required by 10 CFR 50.55a(g)(6)(ii)(A)(2), and the volumetric examinations required by the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11.

Based on this analysis, the NRC staff concludes that the licensee's proposed alternative will provide an acceptable level of quality and safety in lieu of performing the required volumetric examinations. Therefore, the licensee's alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

SNC's letter dated March 29, 2004, requested that RR-38 be approved for the remainder of the 40-year initial license. During review of the RR, the NRC staff and SNC agreed that the RR should be amended to include the period of extended operation. Upon further review, the NRC staff determined that the neutron fluence values used in the RR were based on a calculational code (RAMA) that had not been review and approved by the NRC staff. The licensee had initially anticipated that the NRC staff's review and acceptance of the RAMA code for calculating reactor pressure vessel neutron fluence would be complete by the time this RR was needed. The NRC staff is currently reviewing, but has not yet approved the use of the RAMA code for neutron fluence calculations. As a result, the NRC staff reviewed the licensee's current, NRC-approved neutron fluence analyses and concluded that the fluence values calculated using this methodology are conservative through July 31, 2007. Therefore, the NRC staff limits the duration of RR-38 until July 31, 2007.

Additional requirements of the ASME Code, Section XI for which relief has not been specifically requested and approved by the NRC staff remain applicable, including third party reviews by the Authorized Nuclear Inservice Inspector.

Principal Contributor: J. Medoff, DE
L. Lois, DSSA

Date: January 28, 2005

Edwin I. Hatch Nuclear Plant-Unit 2
Proposed Alternative in Accordance With 10 CFR 50.55a(a)(3)(i)
HNP-ISI-ALT-11

Enclosure 4

NRC Safety Evaluation of Proprietary EPRI Reports
(TAC No. MB9765) dated May 13, 2005

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

May 13, 2005

Bill Eaton, BWRVIP Chairman
Entergy Operations, Inc.
Echelon One
1340 Echelon Parkway
Jackson, MS 39213-8202

SUBJECT: SAFETY EVALUATION OF PROPRIETARY EPRI REPORTS, "BWR VESSEL AND INTERNALS PROJECT, RAMA FLUENCE METHODOLOGY MANUAL (BWRVIP-114)," "RAMA FLUENCE METHODOLOGY BENCHMARK MANUAL-EVALUATION OF REGULATORY GUIDE 1.190 BENCHMARK PROBLEMS (BWRVIP-115)," "RAMA FLUENCE METHODOLOGY-SUSQUEHANNA UNIT 2 SURVEILLANCE CAPSULE FLUENCE EVALUATION FOR CYCLES 1-5 (BWRVIP-117)," AND "RAMA FLUENCE METHODOLOGY PROCEDURES MANUAL (BWRVIP-121)," AND "HOPE CREEK FLUX WIRE DOSIMETER ACTIVATION EVALUATION FOR CYCLE 1 (TWE-PSE-001-R-001)" (TAC NO. MB9765)

Dear Mr. Eaton:

By letters dated June 11, 2003, June 26, 2003, August 5, 2003, October 29, 2003, and March 24, 2004, respectively, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the following Electric Power Research Institute (EPRI) proprietary reports for staff review and approval, "BWR Vessel and Internals Project, RAMA Fluence Methodology Manual (BWRVIP-114)," "RAMA Fluence Methodology Benchmark Manual-Evaluation of Regulatory Guide 1.190 Benchmark Problems (BWRVIP-115)," "RAMA Fluence Methodology-Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5 (BWRVIP-117)," "RAMA Fluence Methodology Procedures Manual (BWRVIP-121)," and "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)."

The reports listed above provide and support a methodology which is a new approach to neutron transport that has been developed by the BWRVIP for determining neutron fluence to the reactor pressure vessel (RPV) and internal components of BWR plants. The Radiation Analysis Modeling Application (RAMA) code will be applied in the reactor beltline region defined by the top and bottom planes of the active fuel and the inner wall of the biological shield. The methodology employs the RAMA computer code for evaluating the neutron flux from the core through the downcomer, vessel internals, and through the RPV wall.

The staff has completed its review of the proposed methodology and finds that the methodology performs as described; however, the BWRVIP did not quantify the bias and uncertainty required for the qualification of the methodology, as stated in RG 1.190, "Radiation Embrittlement of Reactor Vessel Materials." Therefore, the staff's approval is conditional based on the following criteria: (1) for plants that are similar in core, shroud and downcomer-vessel geometry to that of the Susquehanna and Hope Creek plants, the RAMA methodology can be applied without a bias for the calculation of vessel neutron fluence, (2) for plants (or plant groups) with a different geometry than that of the Susquehanna or Hope Creek plants, a plant-specific application for RPV neutron fluence is required to establish the value of a bias, and (3) relevant benchmarking will be required for shroud and reactor internals applications.

The staff evaluation of the proposed RAMA methodology is attached. Please contact Meena Khanna of my staff at 301-415-2150 if you have any further questions regarding this subject.

Sincerely,

Matthew A. Mitchell for

William H. Bateman, Chief
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: BWRVIP Service List

U.S. NUCLEAR REGULATORY COMMISSION OFFICE OF NUCLEAR REACTOR REGULATION SAFETY EVALUATION OF BWR VESSEL AND INTERNALS PROJECT, SAFETY EVALUATION OF PROPRIETARY EPRI REPORTS, "BWR VESSEL AND INTERNALS PROJECT, RAMA FLUENCE METHODOLOGY MANUAL (BWRVIP-114)," "RAMA FLUENCE METHODOLOGY BENCHMARK MANUAL-EVALUATION OF REGULATORY GUIDE 1.190 BENCHMARK PROBLEMS (BWRVIP-115)," "RAMA FLUENCE METHODOLOGY-SUSQUEHANNA UNIT 2 SURVEILLANCE CAPSULE FLUENCE EVALUATION FOR CYCLES 1-5 (BWRVIP-117)," "RAMA FLUENCE METHODOLOGY PROCEDURES MANUAL (BWRVIP-121)," AND "HOPE CREEK FLUX WIRE DOSIMETER ACTIVATION EVALUATION FOR CYCLE 1 (TWE-PSE-001-R-001)"

1.0 INTRODUCTION

1.1 Background

By letters dated June 11, 2003, June 26, 2003, August 5, 2003, October 29, 2003, and March 23, 2004, respectively, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the following Electric Power Research Institute (EPRI) proprietary reports for staff review and approval, "BWR Vessel and Internals Project, RAMA Fluence Methodology Manual (BWRVIP-114)," "RAMA Fluence Methodology Benchmark Manual-Evaluation of Regulatory Guide 1.190 Benchmark Problems (BWRVIP-115)," "RAMA Fluence Methodology-Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5 (BWRVIP-117)," "RAMA Fluence Methodology Procedures Manual (BWRVIP-121)," and "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)." These reports were supplemented by letter dated September 20, 2004, in response to the staff's request for additional information (RAI) dated April 20, 2004.

The BWRVIP-114 report describes the theory of the neutron transport calculation methodology and the uncertainty analysis. The BWRVIP-115 report documents benchmarking of the neutron fluence calculation methodology against two reactor pressure vessel (RPV) simulator measurements, a PWR surveillance capsule measurement and a calculational benchmark. The BWRVIP-117 and TWE-PSE-001-R-001 reports present plant-specific surveillance capsule neutron fluence benchmark comparisons for the Susquehanna and Hope Creek plants, respectively. The BWRVIP-121 report provides the standard procedures for carrying out neutron fluence calculations using this methodology.

The proposed methodology is essentially a new approach that has been developed by the BWRVIP for determining the fast ($E \geq 1.0$ MeV) neutron fluence accumulated by the RPV and internal components of BWR plants. The methodology employs the RAMA computer code for evaluating the neutron flux from the core through the downcomer, vessel internals and through the RPV wall. An important feature of the methodology is that the neutron transport calculation is 3-dimensional, rather than a synthesis of two 2-dimensional calculations that is used in the finite differences method on which presently approved methodologies are based. An additional feature of this approach is that the computer modeling of the physical geometry is represented without approximation. The RAMA code will be applied in the reactor beltline region defined by the top and bottom planes of the active fuel and the inside surface of the biological shield. The methodology employs the most recent BUGLE-96 nuclear transport and reaction-specific

ENCLOSURE

measured activity cross section data. The BWRVIP calculation and uncertainty methodology is summarized in Section 2. The technical evaluation is presented in Section 3, and the limitations and conclusions are provided in Section 4.

1.2 Purpose

The staff reviewed the reports discussed above to determine whether the BWRVIP's proposed methodology will provide an acceptable method for determining the fast (E ≥ 1.0 MeV) neutron fluence accumulated by the RPV and internal components of BWR plants.

1.3 Regulatory Evaluation

The basis for this review is Regulatory Guide (RG) 1.190, "Radiation Embrittlement of Reactor Vessel Materials." RG 1.190 is based on General Design Criterion (GDC) 14, 30 and 31, and describes the attributes of neutron transport methodologies which are acceptable to the staff. The basic feature of an acceptable methodology is that the code is benchmarked by acquiring and evaluating a statistically significant database of measurement-to-calculation ratios and the resulting bias and uncertainty are within certain limits.

2.0 SUMMARY OF THE EPRI BWRVIP VESSEL NEUTRON FLUENCE METHODOLOGY

2.1 RPV Neutron Fluence Calculation Methodology

The BWRVIP neutron fluence calculational methodology employs the RAMA code to evaluate the neutron flux through the core, vessel internals, and vessel geometry. The code uses the BUGLE-96 cross-section library to calculate the neutron transport and to determine the reaction-specific measured activities. The RAMA code employs a combinatorial geometry method which allows an exact representation of geometrically complex components. This is accomplished by building the desired internal component using various primitive geometry elements (Ref. 8).

The neutron transport calculation is based on the following: (1) the three-dimensional transport equation is integrated by attenuating the neutron fluence along discrete rays according to the macroscopic cross-section and optical path in the intersected region, (2) a set of parallel rays are chosen in both a radial and axial plane and the neutron fluence is determined on this grid, (3) to account for the various possible directions of particle transport, rays are defined on a discrete set of angular quadratures, and (4) anisotropic scattering is treated using a Legendre expansion of the neutron scattering cross-section.

The neutron source is determined based on the core power density and the region-wise power distribution. The RAMA source accounts for the exposure dependence of the core neutron source and allows for a detailed pin power description of the source distribution. Typically, reflective boundary conditions are applied on the planes that define the angular sector of the geometry being calculated (typically, a core octant or quadrant), and vacuum boundary conditions are applied at the outer radial boundary (e.g., the outside wall of the RPV) and on upper and lower axial boundaries.

In order to facilitate comparisons of measurements to calculated values (as instructed by RG 1.190), RAMA calculates the corresponding quantities for the measured reaction rates. RAMA determines the time-dependent neutron flux and tracks the target and reaction product nuclides.

The RAMA methodology includes a detailed neutron fluence uncertainty analysis. The parameters making a significant contribution to the neutron fluence calculation uncertainty are identified and RAMA is used to determine numerical sensitivity coefficients for these parameters. The uncertainty contribution from these parameters is determined by combining the numerical sensitivities with the estimates of the input parameter uncertainties. When making comparisons to benchmark measurements, the calculation-to-measurement (C/M) differences are combined using a covariance matrix to determine the uncertainty contribution from the measurements. The overall calculation uncertainty and bias are determined based on the C/M differences and the calculation input parameter uncertainties.

2.2 Calculation of the RPV Benchmarks

In validating the RAMA methodology, comparisons of RAMA predictions were performed for the following four benchmarks: (1) the Oak Ridge National Laboratory (ORNL) Pool Critical Assembly (PCA) benchmark experiment (Ref. 9), (2) the VENUS-3 engineering benchmark experiment (Ref. 10), (3) the H. B. Robinson-2 (HBR-2) RPV benchmark measurement (Ref. 11), and (4) the Brookhaven National Laboratory (BNL) RPV calculation benchmark of NUREG-6115 (Ref. 12). The PCA and VENUS-3 experiments are well-documented RPV mock-ups, including high accuracy dosimetry measurements. The PCA core includes twenty-five material test reactor (MTR) curved-plate type fuel assemblies and the simulator geometry includes a thermal shield, RPV, and void box outside the RPV. The PCA dosimetry measurements were made at positions in front and behind the thermal shield, at locations in front and behind the RPV, and at RPV internals locations. The PCA dosimetry measurements include the Np-237 (n, f), U-238 (n, f), In-115 (n, n'), Ni-58 (n, p) Co-58 and Al-27 (n, α) Na-24 reactions. The RAMA model is 3-dimensional and includes a radial quadrant of the PCA geometry, the full height of the core and the regions above and below the core. Detailed comparisons presented for both the thermal shield (or core shroud) and RPV locations indicate good agreement with the dosimetry measurements.

The VENUS-3 core consists of twelve 15x15 pressurized water reactor (PWR) fuel assemblies and the simulator geometry includes the baffle, core barrel, neutron pad and RPV simulator. The VENUS-3 dosimetry measurements include the Ni-58 (n, p) Co-58, In-115 (n, n'), and Al-27 (n, α) Na-24 reactions. The RAMA model is 3-dimensional and includes a radial quadrant of the simulator geometry, the full height of the core, and the regions above and below the core. Detailed comparisons are presented for the core, baffle, and core barrel and indicate good agreement with the measurements.

The HBR-2 benchmark experiment provides a well-documented set of dosimetry measurements for a full-height operating PWR, including core barrel, thermal shield and RPV. The HBR-2 dosimetry measurements include Np-237 (n, f), U-238 (n, f), Ni-58 (n, p) Co-58, Fe-54 (n, p) Mn-54, Ti-46 (n, p) Sc-46 and Cu-63 (n, α) Co-60. The measurements were made at an in-vessel capsule and at a cavity location. The HBR-2 RAMA model is 3-dimensional and provides a detailed representation of an octant of the problem geometry for a centrally-located axial region of the core. The model extends from the center of the core out to the outer surface of the biological shield. Detailed comparisons are presented for both the in-vessel surveillance capsule and the cavity measurements, and indicate good agreement with the measured data.

BNL NUREG-6115 provides the detailed specification and corresponding numerical solutions for a BWR RPV neutron fluence benchmark problem. The benchmark problem provides a reference calculation for a configuration that is typical of an operating BWR which includes the downcomer and RPV neutron fluences and the dosimeter response at an in-vessel surveillance capsule. The surveillance capsule dosimetry includes the Np-237 (n, f), U-238 (n, f), Ni-58 (n, p) Co-58, Fe-54 (n, p) Mn-54, Ti-46 (n, p) Sc-46, and Cu-63 (n, α) Co-60 reaction rates. The RAMA model is 3-dimensional and provides a detailed representation of an octant of the problem geometry over an axial region that includes the core as well as the regions above and below the core. The model extends from the center of the core out to the outer surface of the biological shield. Detailed comparisons are presented for both the RPV neutron fluences and the dosimetry reaction rates. The surveillance capsule comparisons indicate good agreement for all reaction rates. The downcomer and RPV neutron fluence comparisons indicate that RAMA is conservative relative to the reference solution.

2.3 Calculation of the Susquehanna Neutron Fluence Measurements

As part of the RAMA plant-specific qualification, RAMA transport calculations have been performed for the Susquehanna Unit 2 surveillance capsule that was removed at the end of Cycle 5. In order to validate the fast ($E \geq 1.0$ MeV) neutron fluence evaluations of the Susquehanna RPV, comparisons of the calculated and measured neutron fluence have been made to determine the neutron fluence calculational uncertainty and to identify any systematic bias in the neutron fluence predictions. The Cycle 5 surveillance capsule was located in the downcomer, radially at a position close to the innerwall of the RPV, and azimuthally 30° from the core flats. The surveillance capsule included three each of the following dosimeter wires: copper, nickel, and iron. The measured activities included the Cu-63 (n, α) Co-60, Ni-58 (n, p) Co-58, and Fe-54 (n, p) Mn-54 dosimetry reactions. The measurements were of high quality and were reported to have uncertainties on the order of a few percent.

The RAMA calculational model was based on detailed plant data provided by the Pennsylvania Power and Light (PPL) Company. The geometry data were taken from plant drawings and used to model the surveillance capsule and various core, core shroud, jet pump/riser and RPV components. RAMA provided a geometry model of high accuracy in which both the Cartesian geometry of the core boundary and the cylindrical geometry of the jet pump/riser components were represented without approximation. The RAMA model included a one-eighth (45°) azimuthal sector and the radial geometry from the center of the core out to the inner wall of the biological shield.

The core neutron source was based on the Susquehanna Cycles 1-5 operating history. Three-dimensional power, void and exposure distributions were constructed from the plant operating history files. The pin-wise gradient and exposure dependence of the neutron source for the fuel assemblies on the core periphery were included. Each cycle was described by a representative set of operating state-points. The neutron fluence accumulated by the capsule dosimeters was determined by an appropriate weighting of the RAMA state-point calculations. An extensive set of sensitivity calculations was also performed to ensure the stability and convergence of the numerical solution.

RAMA calculations of the dosimeter activities were performed and compared with the measurements (dps/g). The average C/M overall measurement was found to be very close to unity indicating that there is no significant bias in the RAMA neutron fluence predictions. The standard deviation of all C/M values was less than 20% as recommended in RG 1.190 (Section 1.4.3). In order to provide an independent assessment of the accuracy of the RAMA neutron fluence prediction, a detailed analytic uncertainty analysis was also performed. The important input parameter uncertainties were identified and an estimate of the uncertainty in each parameter was determined. The uncertainty in each parameter was propagated through the RAMA calculation using numerical sensitivity calculations. The resultant analytic estimate of the RAMA neutron fluence calculation uncertainty, corresponding to the observed C/M standard deviation, was also shown to be less than 20%.

2.4 Calculation of the Hope Creek Neutron Fluence Measurements

RAMA transport calculations were performed for the surveillance capsule removed from the Hope Creek RPV at the end of the first cycle. In order to validate the fast ($E \geq 1.0$ MeV) neutron fluence evaluations of the RPV, comparisons of the calculated and measured neutron fluence have been made to determine the neutron fluence calculational uncertainty and to identify any systematic bias in the neutron fluence predictions. The first cycle surveillance capsule was located in the downcomer, radially at a position close to the innerwall of the RPV, and azimuthally at 33° from the core flats. It is noted that two additional capsules are located at 121° and 299°. The surveillance capsule included three copper and three iron flux wires. The measured activities included the Cu-63 (n, α) Co-60 and Fe-54 (n, p) Mn-54 dosimetry reactions. The measurements were reported to have uncertainties on the order of a few percent. The copper activity was corrected for the presence of Co-59 impurity of about 0.25 parts per million (ppm).

The RAMA calculational model was based on detailed plant data. The geometry data were taken from plant drawings and used to model the surveillance capsule, the core, core shroud, jet pump/riser, and RPV components. RAMA provided a geometry model of high accuracy in which both the Cartesian geometry of the core boundary and the cylindrical geometry of the jet pump/riser components were represented without approximation. The RAMA model included a one-eighth (45°) azimuthal sector and the radial geometry from the center of the core to the biological shield.

The core neutron source was based on the first cycle's operating history. Three-dimensional power, void, and exposure distributions were constructed from the plant operating history files. The pin-wise gradient and exposure dependence of the neutron source for the fuel assemblies on the core periphery were included. The neutron fluence accumulated by the capsule dosimeters was determined by an appropriate weighting of the RAMA state-point calculations. An extensive set of sensitivity calculations was also performed to ensure the stability and convergence of the numerical solution.

RAMA calculations of the dosimeter activities were performed and compared with the measurements (dps/gm). The average C/M overall measurement was found to be very close to unity indicating that there is no significant bias in the RAMA neutron fluence predictions. The standard deviation of all C/M values was less than 20% as recommended in RG 1.190 (Section 1.4.3). In order to provide an independent assessment of the accuracy of the RAMA neutron fluence prediction, a detailed analytical uncertainty analysis was also performed. The important input parameter uncertainties were identified and an estimate of the uncertainty in each parameter was determined. The uncertainty in each parameter was propagated through the RAMA calculation using numerical sensitivity calculations. The resultant analytical estimate of the RAMA neutron fluence calculation uncertainty, corresponding to the observed C/M standard deviation, was also shown to be less than 20%.

3.0 TECHNICAL EVALUATION

The staff's review of the BWRVIP neutron fluence methodology focused on the details of the application of the neutron fluence calculation methodology and the qualification of the methodology provided by the benchmark comparisons and the plant-specific C/M database.

3.1 RPV Neutron Fluence Calculation Methodology

In the RAMA transport calculation, the neutron flux is determined by summing the contributions from a set of particle ray tracings through the problem geometry. The accuracy of this technique depends on the specific problem geometry, as well as the number and distribution of the rays used to track the neutrons through the geometry. In addition, the components that are associated with the problem geometry are represented with a discrete set of spatial regions (i.e., a spatial mesh). Because the neutron flux is averaged over these regions, a mesh-related uncertainty is introduced into the calculation. Since both of these numerical uncertainties are sensitive to the problem geometry, they require an evaluation that accounts for the geometry.

By letter dated April 20, 2004, the staff requested that the BWRVIP address the specific tests and criteria used to assure the adequacy of the number of rays and volumes used in the RAMA neutron fluence calculations for plant-specific applications. By letter dated September 29, 2004, the BWRVIP indicated that in plant-specific model applications of the RAMA fluence methodology, numerical sensitivity calculations will be performed to assure the adequacy of the number of particle tracking rays and the number of volumes used to represent component geometry in the RAMA neutron fluence evaluations. The staff found this approach acceptable.

The RAMA geometry model represents the individual components and regions of the problem geometry using a library of pre-calculated geometry elements. The modeling of the reflector region surrounding the core is particularly complicated in that it involves geometry elements that have both planar and cylindrical side boundaries. However, RAMA provides an exact representation of the true geometry (i.e., preserves the exact location, orientation and shape of all surfaces defining the physical geometry). For example, in the case of these reflector regions, the BWRVIP indicated in its letter dated September 29, 2004, that the geometry model allows for complex geometries, including the transition between the rectangular core and the cylindrical core shroud, to be precisely represented.

The RAMA code has the necessary mechanisms for geometrical representation, neutron scattering and neutron transport approximations. Therefore, the staff finds the RAMA code acceptable, based on its structural features.

3.2 Calculation of the RPV Benchmarks

The RPV benchmark calculations are performed to evaluate the accuracy of RAMA and to identify any systematic bias in the proposed licensing methodology. In order for the benchmark comparisons to reflect the difference between the benchmark and the proposed methodology, the methods used in the benchmark calculations must be the same as the proposed licensing methods. By letter dated April 20, 2004, the staff requested that the BWRVIP identify the differences between the methods used in performing the RAMA benchmark analyses in the BWRVIP-115 report and the methods that will be used in performing the calculations of the RPV and core shroud neutron fluence. By letter dated September 29, 2004, the BWRVIP indicated that the methods used in performing the RAMA benchmark analyses are the same as the methods that will be used in performing BWR RPV and core shroud neutron fluence calculations. The staff found this acceptable in that there would be no inconsistencies in the methods used.

The BWRVIP-115, BWRVIP-117, and TWE-PSE-001-R-001 reports present the RAMA analysis of a set of simulator calculations and operating reactor benchmarks which provide the basis of the Susquehanna and Hope Creek applications of the RAMA neutron fluence methodology. However, it is expected that as additional surveillance capsules are removed, new benchmark C/M data will become available. RG 1.190 requires that as new measurements become available, they shall be incorporated into the C/M database and the neutron fluence calculational bias and uncertainty estimates shall be updated as necessary.

By letter dated April 20, 2004, the staff requested that the BWRVIP address how it will ensure that new measurements are incorporated in the C/M database and that the neutron fluence bias and uncertainty will be updated in a timely manner. In its response by letter dated September 29, 2004, the BWRVIP stated that comparisons to measured surveillance capsule and benchmark dosimetry are maintained in a database that is updated as additional plant capsule evaluations are performed using the RAMA methodology. In addition, the BWRVIP stated that currently, TransWare Enterprises, Inc. (a primary contractor to the BWRVIP) maintains a surveillance capsule and benchmark dosimetry measurement database. The BWRVIP further stated that it would consider options of establishing a mechanism to collect and evaluate new C/M data. Based on the above, the staff found the BWRVIP's response acceptable.

The staff's review of this section established that the RAMA methodology is applied to the benchmarks in the same manner (approximations, cross-sections, etc.) as applied in plant-specific applications, therefore, the staff is in agreement that if a bias exists in the proposed code, it should appear in the benchmarks.

3.3 Results of the Susquehanna Dosimetry Measurements

The Susquehanna, Unit 2 surveillance capsule contained three of each of the following dosimeter wires; copper, iron and nickel. The RAMA calculated ratios and the corresponding measured specific activity (dps/g) C/M ratios are close to unity and display very good agreement. The individual ratios are well within the 20% limit specified in RG 1.190. In addition, the standard deviation is just a few percent.

In accordance with the guidance in RG 1.190, the BWRVIP-117 report includes an analytical neutron fluence uncertainty analysis. This analysis is important since it provides an independent estimate of the plant-specific Susquehanna RAMA neutron fluence calculational uncertainty. The uncertainty analysis requires that estimates of the major components of the uncertainty be determined and the uncertainties be propagated through the RAMA neutron fluence calculation. The uncertainty propagation is performed using numerical component sensitivity as calculated by RAMA. The important uncertainty components have been identified and include the following: (1) capsule and flux wire locations, (2) RPV inner radius, (3) core void fraction, (4) peripheral bundle power, and (5) iron cross-sections. In order to make an accurate determination of the RAMA uncertainty, reliable estimates of the component uncertainties are required.

By letter dated April 20, 2004, the staff requested that the BWRVIP discuss the basis for the parameter uncertainty for the components/locations listed above. In its letter dated September 29, 2004, the BWRVIP indicated that the uncertainty estimates for these components/locations is based on the following: (1) as-built measurements, (2) design drawing tolerances,
(3) experience estimates of ±5% variation in computed void fraction, (4) reported accuracy of core simulation analysis, and (5) experience estimates of ±5% in the cross section, respectively. In addition, the staff noted that Table 5-3 of the BWRVIP-117 report provided the values of the calculated bias and total uncertainty. The BWRVIP also displayed the calculation of the total uncertainty and bias from the C/M and the analytic uncertainty with weighting factors inversely proportional to the analytic and C/M variances in the BWRVIP-117 report. The staff finds the BWRVIP's response to the staff's request for additional information and the values of the bias and uncertainty, as provided in the BWRVIP-117 report, acceptable because the values are well within the limits set forth in RG 1.190.

3.4 Results of the Hope Creek Dosimetry Measurements

The Hope Creek surveillance capsule contained three copper dosimeter wires and three iron dosimeter wires. The surveillance capsule was irradiated during the first cycle for 377.9 effective full power days. The RAMA code calculated the specific dosimeter activity to the corresponding measured specific activity (dps/g). The C/M ratios are close to unity and displayed very good agreement. The individual dosimeter ratios are well within the 20% limit, as specified in RG 1.190, and the standard deviation is just a few percent. However, it was noted that unlike the Susquehanna case, the Hope Creek calculation does not include an analytical uncertainty and bias calculation.

4.0 CONCLUSION

4.1 BWR RPV Neutron Fluence

Based on the staff's review of the BWRVIP-114, -115, -117, and -121 reports, the TWE-PSE-001-R-001 report, and the supporting documentation, the staff concludes that the BWRVIP methodology, as described in these reports, provides an acceptable best-estimate plant-specific prediction of the fast (E ≥1.0 MeV) neutron fluence for BWR RPVs. This acceptance is limited to the axial region defined by the core active fuel height. The best-estimate RPV neutron fluence prediction is determined using the RAMA transport code, detailed plant-specific geometry, core operating history, and the BUGLE-96 nuclear data library with a minimum of a P_3 Legendre polynomial approximation in the iron inelastic scattering.

With respect to the calculation of BWR RPV neutron fluence, the staff concludes that based on the plant-specific benchmark data presently available, no calculational bias is required for the application of the methodology to plants of similar geometrical design to Susquehanna and Hope Creek, i.e., BWR-IV plants. However, in order to provide continued confidence in the proposed neutron fluence methodology for the BWR RPVs, the acceptance of this methodology is subject to the following conditions for plants which do not have geometries similar to the cited BWR-IV's:

- To apply the RAMA methodology to plant groups which have geometries that are different than the cited BWR-IV's, at least one plant-specific capsule dosimetry analysis must be provided to quantify the potential presence of a bias and assure that the uncertainty is within the RG 1.190 limits

  and

- Justification is necessary for a specific application based on geometrical similarity to an analyzed core, core shroud, and RPV geometry. That is, a licensee who wishes to apply the RAMA methodology for the calculation of RPV neutron fluence must reference, or provide, an analysis of at least one surveillance capsule from a RPV with a similar geometry.

4.2 Reactor Internals

EPRI's stated objective for this submittal included neutron fluence calculations for reactor internals. Neutron fluence values for reactor internal components are used to either quantify irradiation assisted stress corrosion cracking (IASCC) susceptibility, or to quantify helium formation which could affect the weldability of reactor internals components. IASCC depends on fast (E ≥ 1.0 MeV) neutron fluence, while helium formation is a function of thermal, epithermal, and fast neutron fluence. The calc ulational accuracy requirements for reactor internals are not the same as those for the RPV, and are not covered by the guidance in RG 1.190. In addition, the submittal does not include any benchmarking for reactor internals' neutron fluence calculations. Therefore, the staff will review qualification of RAMA for reactor internals applications on a case-by-case basis, based on consideration of C/M values and the associated accuracy requirements.

Licensees who wish to use the RAMA methodology for the calculation of neutron fluence at reactor internals locations must reference, or provide, an analysis which adequately benchmarks the use of the RAMA methodology for uncertainty and calculational bias based on the consideration of: (1) the location at which the neutron fluence is being calculated, (2) the geometry of the reactor, and, (3) the accuracy required for the application. In addition, if a licensee qualifies RAMA for calculating, for example, helium generation at one location (e.g., the core shroud), this qualifies RAMA for the same reactor and purpose at other reactor internals locations (e.g., at the location of the jet pumps).

4.3 Assembling a Statistically Significant Database

EPRI stated that efforts are underway to assemble a database which will enable the staff to remove any limitations placed on the use of the RAMA methodology. For such an effort to be successful, the staff expects that the neutron fluence uncertainty analysis and determination of the calculational bias for the relevant fleet of plants will be updated, as additional measurements are taken and as additional data become available. The results of the updated analysis, including the C/M ratios, should be submitted to the staff for review and approval.

5.0 REFERENCES

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