



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

June 9, 2014

Mr. Timothy S. Rausch  
Senior Vice President and Chief Nuclear Officer  
PPL Susquehanna, LLC  
769 Salem Boulevard  
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - RELIEF  
REQUESTS FOR THE FOURTH 10-YEAR INSERVICE INSPECTION  
INTERVAL (TAC NOS. MF2705 THROUGH MF2714)

Dear Mr. Rausch:

By letter dated August 30, 2013, as supplemented by letters dated January 31, 2014, and April 28, 2014, PPL Susquehanna, LLC (the licensee) submitted Relief Requests 4RR-02, 4RR-05, 4RR-06, 4RR-07, and 4RR-08 for the fourth 10-year inservice inspection (ISI) interval for the Susquehanna Steam Electric Station (SSES), Units 1 and 2. Relief Requests 4RR-02, 4RR-05, 4RR-06, 4RR-07, and 4RR-08 request the use of alternatives to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI (ASME Code) at SSES.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the licensee requested to use alternatives in 4RR-02, 4RR-05, 4RR-06, and 4RR-08 on the basis that the alternatives provide an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee requested to use the proposed alternative in 4RR-07, on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff determined that the proposed alternatives described in requests 4RR-02, 4RR-06, and 4RR-08 provide an acceptable level of quality and safety. Accordingly, the NRC staff concludes, as stated in the enclosed safety evaluation (SE), that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i). The NRC staff determined that the proposed alternatives described in requests 4RR-05 and 4RR-07 provide reasonable assurance of structural integrity or leak tightness of the subject components and complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes, as stated in the enclosed SE, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the Nuclear Regulatory Commission staff authorizes the proposed alternatives in requests 4RR-02, 4RR-05, 4RR-06, 4RR-07, and 4RR-08 for the fourth ISI interval at SSES, which began on June 1, 2014, and is currently scheduled to end on May 31, 2024.

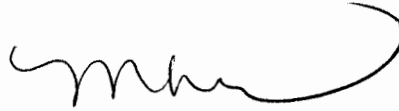
All other requirements of 10 CFR 50.55a and ASME Code, Section XI, for which relief was not specifically requested and approved, remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

T. Rausch

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If you have any questions, please contact the SSES Project Manager, Mr. Jeffrey A. Whited, at [jeffrey.whited@nrc.gov](mailto:jeffrey.whited@nrc.gov) or 301-415-4090.

Sincerely,

A handwritten signature in black ink, appearing to read 'Meena K. Khanna', with a large, sweeping flourish at the end.

Meena K. Khanna, Chief  
Plant Licensing Branch I-2  
Division of Operating Reactor Licensing  
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosure:  
Safety Evaluation

cc w/encl: Distribution via ListServ



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING RELIEF REQUESTS 4RR-02, 4RR-05, 4RR-06, 4RR-07, AND 4RR-08

ASSOCIATED WITH THE FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL

PPL SUSQUEHANNA, LLC

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 50-388

1.0 INTRODUCTION

By letter dated August 30, 2013,<sup>1</sup> as supplemented by letters dated January 31, 2014,<sup>2</sup> and April 28, 2014,<sup>3</sup> PPL Susquehanna, LLC (the licensee) submitted Relief Requests 4RR-02, 4RR-05, 4RR-06, 4RR-07, and 4RR-08 for the fourth 10-year inservice inspection (ISI) program for the Susquehanna Steam Electric Station (SSES), Units 1 and 2. Relief Requests 4RR-02, 4RR-05, 4RR-06, 4RR-07, and 4RR-08 request the use of alternatives to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI (ASME Code) at SSES.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the licensee requested to use alternatives in 4RR-02, 4RR-05, 4RR-06, and 4RR-08 on the basis that the alternatives provide an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee requested to use the proposed alternative in 4RR-07, on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

The fourth 10-year ISI interval at SSES, Units 1 and 2, began on June 1, 2014, and is currently scheduled to end on May 31, 2024. The applicable ASME Code edition and addenda for the fourth 10-year ISI Interval at SSES, Units 1 and 2, is the 2007 Edition through the 2008 Addenda.

The regulations in 10 CFR 50.55a(a)(3) state, in part, that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the NRC, if the licensee demonstrates (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.

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<sup>1</sup> Agencywide Documents Access and Management System (ADAMS) Accession No. ML13247A167.

<sup>2</sup> ADAMS Accession No. ML14031A081.

<sup>3</sup> ADAMS Accession No. ML14118A443.

The regulations in 10 CFR 50.55a(g)(4) state, in part, that:

Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME B&PV Code (or ASME OM [Operation and Maintenance] Code for snubber examination and testing) that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and that are incorporated by reference in paragraph (b) of this section [10 CFR 50.55a], to the extent practical within the limitations of design, geometry and materials of construction of the components. . .

The regulations in 10 CFR 50.55a(g)(4)(ii) state, in part, that:

Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section [10 CFR 50.55a] 12 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in NRC Regulatory Guide [RG] 1.147, Revision 16, when using Section XI; or Regulatory Guide 1.192 when using the OM Code, that are incorporated by reference in paragraph (b) of this section), subject to the conditions listed in paragraph (b) of this section . . .

Based on the above, and subject to the following technical evaluation, the Nuclear Regulatory Commission (NRC) staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternatives requested by the licensee.

### 3.0 TECHNICAL EVALUATION

All of the requested alternatives outlined below are requested for the duration of the fourth 10-year ISI Interval which began on June 1, 2014, and is currently scheduled to end on May 31, 2024. The applicable ASME Code edition and addenda for SSES, Units 1 and 2, during the fourth 10-year ISI interval is the 2007 Edition through the 2008 Addenda.

#### 3.1 Licensee's Alternative Request 4RR-02

The licensee's submittal stated that 4RR-02 was being provided as an administrative placeholder because this relief request was submitted in the second 10-year ISI Interval as 2RR-22, and was subsequently approved by the NRC staff in a safety evaluation (SE) dated February 28, 2001,<sup>4</sup> until the end of the initial license for both units, which includes the fourth 10-year ISI interval.

However, because the initial license period ends at midnight on July 17, 2022, for SSES, Unit 1, and midnight on March 24, 2024, for SSES, Unit 2, while the fourth 10-year ISI interval ends May 31, 2024, the staff determined that a part of the fourth 10-year ISI interval is not covered by the 2RR-22. Therefore, the licensee requested that the NRC staff review and disposition Relief Request 4RR-02 for the fourth 10-year ISI Interval. By letter dated January 31, 2014, the

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<sup>4</sup> ADAMS Accession No. ML010330383.

licensee submitted a revised Relief Request 4RR-02, which corrected the requested duration of the relief to cover the entire fourth 10-year ISI interval.

### 3.1.1 Licensee's Request

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposed an alternative that would eliminate the requirement to inspect the reactor pressure vessel (RPV) circumferential welds except for the areas of intersection with the axial welds, consistent with the guidance provided in Generic Letter (GL) 98-05, "Boiling Water Reactor [BWR] Licensees Use of the BWRVIP-05 [Boiling Water Reactor Vessels and Internals Project] Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998,<sup>5</sup> and the NRC staff's SE for the BWRVIP-05 report issued on July 28, 1998.<sup>6</sup>

The licensee requested approval to implement the alternative RPV examination in lieu of the ISI requirements for circumferential welds in the ASME Code, Section XI, 2007 Edition through the 2008 Addenda, Table IWB-2500-1, Examination Category B-A, Item Number B1.11 volumetric examination of RPV circumferential welds. The components for which the alternative is requested are Weld IDs: AA, AB, AC, AD, and AE. The components are all ASME Code Class 1.

#### Basis for the Alternative

The licensee's basis for applying the alternative related to circumferential welds is a demonstration that the SSES, Units 1 and 2, RPV circumferential welds meet the two conditions from the NRC staff's SE of BWRVIP-05, as communicated in GL 98-05.

The first condition of GL 98-05 is that at the end of the license renewal period, the circumferential welds will satisfy the limiting conditional failure probability for circumferential welds from the NRC staff's Final Safety Evaluation Report (FSER) for BWRVIP-05.

The licensee provided a table intended to illustrate that the conditional failure probability of the limiting SSES, Units 1 and 2, RPV circumferential welds are bounded by the values in Table 2.6-4 for the Limiting Plant-Specific Analyses (32 effective full-power years (EFPY)) of the NRC's evaluation of BWRVIP-05. The licensee stated that the chemistry factor, shift in  $RT_{NDT}$  [Reference Temperature for Nil Ductility Transition] due to irradiation ( $\Delta RT_{NDT}$ ), unirradiated  $RT_{NDT}$  ( $RT_{NDT(U)}$ ), and mean  $RT_{NDT}$  are determined in accordance with the guidelines of RG 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.<sup>7</sup> Table 1 of Relief Request 4RR-02 provides input parameters listed above for SSES, Units 1 and 2, and the NRC limiting plant-specific analysis parameters at 54 EFPY from Table 2.6-4 of the NRC's SE of BWRVIP-05. The results of the licensee's evaluation showed that the bounding mean  $RT_{NDT}$  at 54 EFPY for SSES, Units 1 and 2, is less than the NRC limiting  $RT_{NDT}$  at 32 EFPY. The licensee therefore concluded that the circumferential welds in the SSES, Units 1 and 2, RPVs at 54 EFPY, which are conservatively enveloped for the fourth 10-year ISI Interval, would continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC staff's FSER of BWRVIP-05. See Table 1 in Section 3.1.2 of this SE for the specific values of the parameters provided by the licensee.

<sup>5</sup> ADAMS Legacy Accession No. 9811030134.

<sup>6</sup> ADAMS Legacy Accession No. 9808040037.

<sup>7</sup> ADAMS Accession No. ML003740284.

The second criterion from GL 98-05 is that licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC staff's FSER for BWRVIP-05.

In its basis for compliance with the second criterion, the licensee stated that it has procedures in place which monitor and control reactor temperature and water inventory during all aspects of cold shutdown, which would minimize the likelihood of a low temperature over-pressurization (LTOP) event from occurring. The licensee further stated that these procedures are reinforced through operator training. Procedural controls described by the licensee for prevention of cold overpressure events can be found on page 3 of Attachment 2 to the licensee's letter dated January 31, 2014.

### 3.1.2 NRC Staff Evaluation

The licensee requested the alternative to eliminate the examination of the RPV circumferential welds required by the ASME Code, Section XI, Table IWB-2500 for the fourth 10-year ISI interval. The licensee's basis for relief is meeting the two conditions from the NRC staff's FSER of BWRVIP-05, as communicated in GL 98-05.

The NRC staff previously authorized the same alternative for the remainder of the original license period in an SE dated February 28, 2001. In accordance with the requirements from the NRC staff's FSER of BWRVIP-05, for plants to be granted relief from inspection of circumferential welds, the NRC staff concluded that the conditional failure probability would have to be much less than the limiting conditional probability of failure for RPVs fabricated by Chicago Bridge & Iron Company (CB&I) for 32 EFPY, which was determined to be  $2 \times 10^{-7}$  per reactor year. On this basis, the staff concluded relief from inspection of all circumferential welds was acceptable for SSES, Units 1 and 2, through the end of the current license.

With respect to the first condition of GL 98-05, which requires that at the expiration of the license, the circumferential welds must satisfy the limiting conditional failure frequency for circumferential welds from the NRC staff's FSER for BWRVIP-05, this can be satisfied by demonstrating the mean  $RT_{NDT}$  for the limiting circumferential weld for a plant is less than the value listed for the limiting RPV for a particular fabricator, as given in Table 2.6-4 of the NRC staff's FSER of BWRVIP-05. Although Relief Request 4RR-02 did not identify the manufacturer for the SSES, Units 1 and 2, RPVs, the staff verified via the Reactor Vessel Integrity Database that the manufacturer is CB&I for both RPVs. This is also consistent with Section 4.2.5 of the SSES, Units 1 and 2, License Renewal Application (LRA).

The NRC staff compared the inputs for calculating the mean  $RT_{NDT}$  for the limiting circumferential welds for SSES, Units 1 and 2, to the information for the same welds in the LRA. The copper, nickel,  $RT_{NDT(U)}$ , and 54 EFPY RPV inner diameter neutron fluence values are all consistent with the values reported in the LRA. In Section 4.2.1 of NUREG-1931, "Safety Evaluation Report (SER) Related to the License Renewal of SSES, Units 1 and 2,"<sup>8</sup> the staff found that the neutron fluence values given in the LRA for the SSES, Units 1 and 2, RPVs, were calculated in accordance with an NRC-approved methodology consistent with RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,"<sup>9</sup> and are therefore acceptable.

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<sup>8</sup> ADAMS Accession No. ML093170792.

<sup>9</sup> ADAMS Accession No. ML010890301.

The NRC staff also verified that since the submittal of the LRA for SSES, Units 1 and 2, no changes to the input parameters have resulted from the testing of representative materials in the BWR Integrated Surveillance Program (ISP) for the limiting SSES, Units 1 and 2, circumferential welds. BWRVIP-86, Revision 1-A: "BWR Vessel and Internals Project Updated BWR Integrated Surveillance Program (ISP) Implementation Plan,"<sup>10</sup> indicates that the ISP representative materials for the limiting circumferential welds in the SSES, Units 1 and 2, RPVs are not from the same material heat number as the SSES welds. Section 5.6 of BWRVIP-86, Revision 1-A states that if the heat of the material does not specifically match the limiting heat of the beltline material for that vessel, the chemistry factor for the limiting beltline material will be determined by the tables in RG 1.99. This position was approved in the NRC staff's SE dated February 1, 2002,<sup>11</sup> of the original topical report describing the ISP, BWRVIP-86-A: "BWR Vessel and Internals Project Updated BWR Integrated Surveillance Program (ISP) Implementation Plan".<sup>12</sup> Therefore, results from the ISP are not used to predict the shift in  $RT_{NDT}$  due to irradiation of the SSES, Units 1 and 2, beltline materials, and do not affect the predicted mean  $RT_{NDT}$  of the limiting circumferential welds.

The NRC staff performed a confirmatory calculation of the mean  $RT_{NDT}$  value for the SSES, Unit 2, limiting circumferential weld using the input values from the licensee's submittal. To determine the mean  $RT_{NDT}$ , the NRC staff added the  $\Delta RT_{NDT}$  calculated using the methodology of RG 1.99, Rev. 2 to  $RT_{NDT(u)}$ . For the SSES, Unit 2, limiting circumferential weld, the staff calculated a mean  $RT_{NDT}$  of 17 °F at 32 EFPY, which is less than the mean  $RT_{NDT}$  value of 41.9 °F at 32 EFPY calculated by the licensee. The NRC staff also calculated a higher  $\Delta RT_{NDT}$  value of 37 °F compared to 30.9 °F calculated by the licensee. The NRC staff determined that the  $\Delta RT_{NDT}$  value used by the licensee was calculated using the fluence at the one-quarter of the RPV thickness location (1/4T location), which results in a lower fluence due to attenuation through the RPV thickness, when compared to the RPV inner diameter fluence used in the NRC staff's calculation. However, the licensee also applied a margin term in determining the Mean  $RT_{NDT}$ . Also, a supplemental SE to BWRVIP-05 was issued on March 7, 2000,<sup>13</sup> which clarifies that "mean  $RT_{NDT}$ " does not include a margin term, and revises Table 2.6-4 of the initial SE to correct the chemistry factor for the limiting CB&I RPV circumferential weld.

Therefore, the licensee's mean  $RT_{NDT}$  value is conservative. The inputs and results of the NRC staff's confirmatory calculation are shown in the third column of Table 1.

The NRC staff also calculated the mean  $RT_{NDT}$  for the limiting SSES, Unit 1, circumferential weld at 32 EFPY to confirm that the mean  $RT_{NDT}$  value for the SSES, Unit 2, weld is limiting. For the SSES Unit 1 weld, the NRC staff used copper, nickel,  $RT_{NDT(u)}$ , and 54 EFPY neutron fluence values from the SSES LRA. Based on this calculation, the NRC staff confirmed the SSES, Unit 2, circumferential weld has a higher mean  $RT_{NDT}$  than the most limiting SSES, Unit 1, circumferential weld. Therefore, the licensee's analysis is bounding for both SSES of the units since it used the higher mean  $RT_{NDT}$  of the two units.

The NRC staff notes that the licensee used the 54 EFPY neutron fluence values, corresponding to 60 calendar years of operation or the end of the period of extended operation (PEO), to calculate the mean  $RT_{NDT}$  for the limiting weld, but compared this value to the limiting CB&I RPV

<sup>10</sup> ADAMS Accession No. ML131760082.

<sup>11</sup> ADAMS Accession No. ML020380691.

<sup>12</sup> ADAMS Accession No. ML023190487.

<sup>13</sup> ADAMS Accession No. ML003690281.

mean  $RT_{NDT}$  for 32 EFPY. However, if the licensee had compared the mean  $RT_{NDT}$  for SSES, Units 1 and 2, to the 64 EFPY value from Table 2.6-5 of the NRC staff's FSER of BWRVIP-05, the SSES, Units 1 and 2 mean  $RT_{NDT}$  would also be bounded because the 64 EFPY CB&I mean  $RT_{NDT}$  is higher than the corresponding 32 EFPY value. The licensee's use of the 54 EFPY fluence value is conservative because the requested relief is only through the fourth 10-year ISI interval, which will occur nearly 20 years prior to the end of the PEO for SSES, Units 1 and 2.

Based on the above, the NRC staff finds that the licensee has demonstrated that the mean  $RT_{NDT}$  values for SSES, Units 1 and 2, will remain bounded by the generic mean  $RT_{NDT}$  value for an RPV fabricated by CB&I through the end of the fourth 10-year ISI interval. Therefore, the conditional failure probability of the SSES, Units 1 and 2, RPVs with no circumferential weld examinations will remain bounded through the end of the fourth 10-year ISI interval by the limiting conditional failure probability from the NRC staff's final FSER of BWRVIP-05.

The NRC staff reviewed the licensee's description of the operator training and operational procedures that prevent cold overpressure events and determined that the procedures and training that are in place at SSES should result in a very low probability of cold overpressurization events. The NRC staff also notes the description of procedures and training for prevention of cold overpressurization events in the current relief request is essentially identical to that contained in the previous relief request 2RR-02, which was approved in the staff's February 28, 2001, SE.

**Table 1 – Evaluation of Limiting 32 EFPY Adjusted Reference Temperature Value for SSES RPV Circumferential Welds**

Parameter Description	SSES Units 1 and 2 Comparative Parameters at 32 EFPY for the Bounding Circumferential Weld Wire Heat/Lot 62463/E 204A27A* - Licensee	SSES Units 1 and 2 Comparative Parameters at 32 EFPY for the Bounding Circumferential Weld Wire Heat/Lot 62463/E 204A27A - Staff Calculation	NRC Limiting Plant Specific Analyses Parameters at 32 EFPY SER Table 2.6-4
Copper (Cu), Weight %	0.06	0.06	0.10
Nickel (Ni), Weight %	0.89	0.89	0.99
Chemistry Factor (CF)	82	82	134.9**
End-of-Life (EOL) Inner Diameter Fluence, $\times 10^{19}$ n/cm <sup>2</sup>	0.118	0.118	0.51
End-of-Life (EOL) 1/4T Fluence, $\times 10^{19}$ n/cm <sup>2</sup>	0.082	n/a	n/a
Fluence Factor	0.38 (1/4T)	0.4508 (ID)	0.81 (ID)
$\Delta RT_{NDT}$ , °F	30.9	37	109.5
$RT_{NDT(U)}$ , °F	-20	-20	-65
Mean $RT_{NDT}$ , °F	41.9	17	44.5

\* The footnote to Relief Request 4RR-02 Table 1 stated that this data is for SSES, Unit 2 and envelopes SSES, Unit 1

\*\* The licensee's table gave this CF as 109.5 °F. This is based on the staff's July 28, 1998, SE of BWRVIP-05. This value was incorrect, and was subsequently corrected to 134.9 °F via the staff's March 7, 2000, supplemental SE of BWRVIP-05 (ML003690281). The mean  $RT_{NDT}$  is unchanged from the initial SE of BWRVIP-05.



The NRC staff finds the information submitted by the licensee related to the RPV circumferential welds supports the determination that the conditional probability of failure at the end of fourth 10-year ISI interval is bounded by the limiting conditional probability of failure for a CB&I-fabricated RPV. This finding is based on the projected mean  $RT_{NDT}$  of the limiting circumferential weld materials for SSES, Units 1 and 2, which is less than the mean  $RT_{NDT}$  value associated with the limiting conditional failure probability for a CB&I RPV from the NRC staff's FSER of BWRVIP-05.

Additionally, the licensee will continue to implement operator training and procedures to limit the frequency of cold overpressure events to the amount specified in the NRC staff's FSER of BWRVIP-05. Therefore, the licensee has met the two plant-specific conditions required by the NRC staff's FSER of BWRVIP-05 to obtain relief from inspection of the circumferential RPV welds.

Based on the above evaluation, and pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff finds that the licensee's proposed alternative in 4RR-02 from the requirements of the ASME Code, Section XI Table IWB-2500-1 Examination Category B-A, Item B1.11 pertaining to RPV circumferential shell welds, provides an acceptable level of quality and safety. Therefore, the use of the proposed alternative in 4RR-02 is authorized for the fourth 10-year ISI interval at SSES, Units 1 and 2.

### 3.2 Licensee's Alternative Request 4RR-05

#### 3.2.1 Licensee's Request

The licensee requested the use of Code Case N-795, "Alternative Requirements for BWR Class 1 System Leakage Test Pressure Following Repair/Replacement Activities," which is intended to provide alternative test pressure for Class 1 pressure tests following repair/replacement activities which occur subsequent to the periodic Class 1 pressure test required by Table IWB-2500-1, Category B-P and prior to the next refueling outage on those components that cannot be isolated. This does not include repair/replacement activities on the reactor vessel, and components that can be isolated will be pressure tested at a pressure that is required by IWB-5221(a).

The licensee has defined the required test pressure according to IWA-5211(a) for components within the Reactor Coolant Pressure Boundary as a minimum of 1035 psig. The proposed alternative is to perform the system leakage test and VT-2 examination in accordance with Code Case N-795 at 932 psig (90 percent of the required pressure) with a minimum hold time of 1 hour for uninsulated components and an 8-hour hold time for insulated components during maintenance, forced outages, or outages other than refueling outages.

The basis for this request is that the performance of the Category B-P pressure test for each refueling outage, places SSES, Units 1 and 2, in a position of significantly reduced margin, approaching the fracture toughness limits defined in the technical specification (TS) pressure temperature (P-T) curves. To violate these curves would place the vessel in an LTOP condition. With strict operational control procedures, specific component alignment, and operations staff training regarding LTOP, the licensee indicated that it may be considered acceptable to be at this reduced margin condition for the purpose of verifying the leakage status/integrity of the primary system in order to meet the ASME Code Section XI, Category B-P requirements prior to startup from a refueling outage. However, the licensee indicated that performing this evolution more frequently, as would be required to fulfill the pressure tests in this relief request, would increase the overall risks to the plant.

### 3.2.2 NRC Staff Evaluation

The licensee requested the use of an alternative under 10 CFR 50.55a(a)(3)(i), which is a proposed alternative that provides an acceptable level of quality and safety. However, the NRC staff evaluated this request under Section 50.55a(a)(3)(ii), which authorizes a proposed alternative when complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The NRC staff considers structural integrity to be provided through the combination of design requirements, controls placed on welding and fabrication, nondestructive examination, and the pressure test. The NRC staff considers the acceptable level of safety and quality to be achieved when all of the requirements for these are met. The NRC staff acknowledges that there are instances where conducting a pressure test at normal operating pressure could present a hardship. These instances may include the restart of a BWR from a short outage when high decay heat load remains in the nuclear fuel. The NRC staff also acknowledges that to obtain pressures equivalent to those at 100 percent power level during shutdown conditions to require abnormal system line-ups and reduced margin to LTOP. The NRC staff also acknowledges that when normal operating pressure equivalent to 100 percent power is reached during a normal plant startup, radiation levels may be high and result in higher dose rates to the plant workers while conducting a pressure test visual examination for leaks.

ASME Code Case N-795 provides an alternative test pressure for some Class 1 pressure tests following repair/replacement activities at BWR plants. The licensee argues that performance of this primary system pressure test at a BWR places the unit in a position of significantly reduced margin, approaching the fracture toughness limits defined in the TS P-T curves. In addition, reactor pressure corresponding to 100 percent rated power cannot be obtained, at a large majority of BWR units, during normal startup operations at low power levels. The pressure control system does not allow the setpoint to approach the 100 percent pressure value and the core reload analysis does not cover the elevated pressure at low power levels conducive to personnel entry into the drywell. The alternative was developed because some BWR licensees believe that the Class 1 pressure tests performed at pressures corresponding to 100 percent reactor power require abnormal plant conditions and alignments that increase risk. Specifically, to obtain the test pressures corresponding to 100 percent rated power, and still allow access for the examination, a large majority of BWRs must perform a pressure test which requires the primary system to be isolated (including shutdown cooling). During this test, the vessel is filled essentially water solid while at a greatly reduced margin to cold overpressure conditions. BWR owners believe that an alternative test performed at slightly reduced pressures and normal plant conditions would still allow for an adequate leak examination and would reduce the time required to perform this test.

For an existing through-wall defect, the leak rate would be proportional to the square root of the differential pressure driving the leak. The lower pressure of the code case would provide more than 90 percent of the flow that would result from the pressure corresponding to 100 percent power through a postulated through-wall defect in the pressure boundary. To account for the reduced pressure, the code case proposes to increase hold times to allow for more leakage from the pressure boundary. It should be noted that neither the Class 1 pressure tests to satisfy the periodic test required under IWB-2500-1, Category B-P, nor the pressure tests required following repair or replacement activities on the reactor vessel are addressed by the code case. In addition, the Class 1 pressure test required at the end of each refueling outage will still be performed at a pressure corresponding to 100 percent reactor power. Code Case N-795 specifies a 15 minute hold time for non-insulated components and a 6 hour hold time for insulated

components. However, the NRC staff believes that a 1 hour hold time for non-insulated components and 8 hours for insulated components is justified, since 15 minutes may not allow sufficient time for an adequate examination. The licensee has agreed to the longer hold times required by the conditions set by the NRC.

Based on the above evaluation, and pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff finds that the licensee's proposed alternative in 4RR-05 to use Code Case N-795, provides reasonable assurance of structural integrity or leak tightness of the subject components and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the use of the proposed alternative in 4RR-05 is authorized for the fourth 10-year ISI interval at SSES, Units 1 and 2.

### 3.3 Licensee's Alternative Requests 4RR-06

#### 3.3.1 Licensee's Request

The affected components are ASME Code Class 1, 2, and 3, snubber attachments.

The 2007 Edition through the 2008 Addenda of ASME Section XI contains Figure IWF-1300-1(f), which depicts the examination boundaries for snubbers. The boundaries indicate that the attachment of the snubber to the pressure boundary and building structure is required to be examined in accordance with IWF-2000.

Table IWF-2500-1 requires a VT-3 visual examination of Class 1 (FI.10), Class 2 (FI.20), Class 3 (FI.30) piping supports, and Class 1, 2, and 3, (FI.40) component supports. The percentages for each Class are also identified: Class 1 (25 percent (%)), Class 2 (15%), and Class 3 (10%). The total percentage sample shall be comprised of supports from each system (such as Main Steam, Feedwater, or RHR), where the individual sample sizes are proportional to the total number of non-exempt supports of each type and function within each system.

In Attachment 3 of the alternative request submitted by letter dated August 30, 2013, the licensee stated, in part, that:

#### Reason for Request

Snubbers were removed from ASME Section XI in the 2006 Addenda. Figure IWF-1300-1(f) was added to show the examination boundaries for snubbers which excluded the snubber including the pivot and clevis pins (see Figure 1 below).

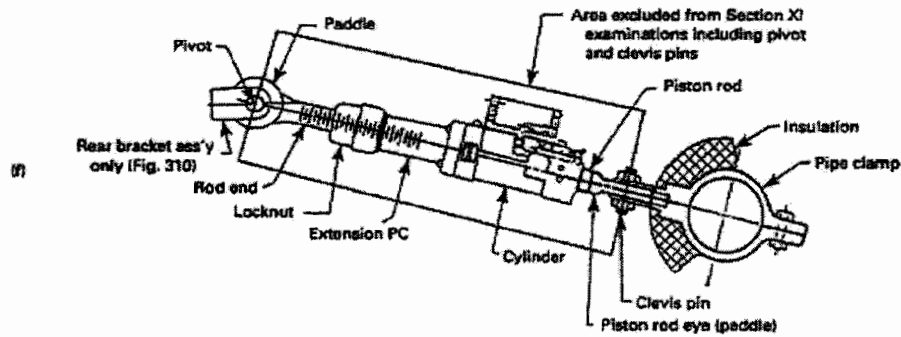


Figure 1

The attachments for the snubber to the pressure boundary (via pipe clamps etc.) and to the building structure are still included as part of the ASME Section XI examination boundary. This means that both the Snubber Program and the ISI Program requires tracking and scheduling two different examination boundaries for one component.

In order to eliminate the duplication of effort by tracking two different examination boundaries for one component, SSES requests incorporating ... both examination boundaries as shown in the Figure 2 below into the Snubber Program. In addition, incorporating both examination boundaries into one program provides a better understanding of the condition of the snubber and its associated attachment to the pressure boundary or building structure. A 100% visual examination of all safety related snubbers will be performed on an examination frequency determined by the [ASME] O&M Code 2004 Edition through the 2006 Addenda and Code Case OMN-13 (Note that Code Case OMN-13 has been found acceptable in RG 1.192). The examination method used for the snubber and their attachments will be the VT-3 visual examination in accordance with ASME Section XI, IWA-2213.

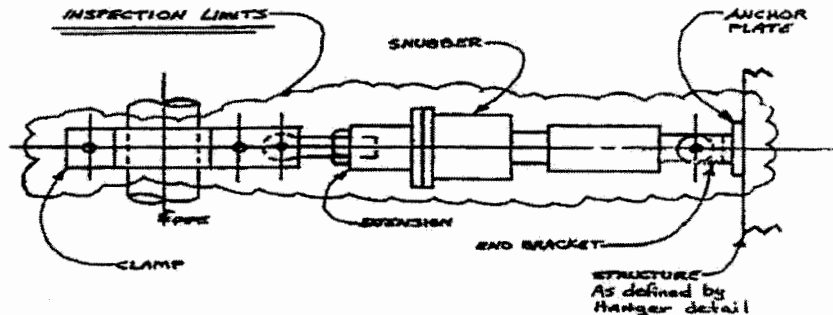


Figure 2

Proposed Alternative and Basis for Use

[The ASME] O&M Code Case OMN-13 requires 100% safety related snubbers to be examined and evaluated at least once every 10 years. This exceeds the requirements of ASME Section XI, IWF-2500-1 tables which only requires 25% of Class 1, 15% of Class 2, and 10% of Class 3 required over a 10-year interval.

Performing both examination boundaries in the Snubber Program using VT-3 qualified personnel to perform the examinations provides a better understanding of the snubber and attachments. This will meet both the [ASME OM Code] visual examination and ASME Section XI examination requirements. This reduces the number of required examinations and personnel required to accomplish both requirements with one examination. Performing the examination on all snubber attachments in accordance with the [ASME OM] Code frequency exceeds the required percentage requirements of ASME Section XI.

In response to a request for additional information (RAI), which was submitted by letters dated January 31, 2014, and April 28, 2014, the licensee stated, in part, that:

1. Examinations of snubbers and their associated supports will be administrated and scheduled per the PPL Snubber Program under the requirements of the ASME OM Code 2004 Edition through the 2006 Addenda. Examination of the snubber attachments is required per ASME Section XI Code 2007 Edition with the 2008 Addenda and the PPL ISI Program. Visual examination of the associated attachments will be performed at the same time as the required visual examination of the snubber. This is being done as a dose and time saving effort. For both examinations, a VT-3 qualified inspector will be performing the examination. Examination of 100 percent of the snubber attachments exceeds Code requirements as defined in the ASME Section XI Code.
2. Snubbers and associated attachments will be administered under the PPL Snubber Program which is committed to the ASME OM Code 2004 Edition through the 2006 Addenda.

Supports (without snubbers) will be administered under the PPL ISI Program which is committed to the ASME Section XI Code 2007 Edition through the 2008 Addenda. While under two different programs, visual inspections for both programs will be conducted by VT-3 qualified individuals.

3. For those snubber attachments that are covered by insulation, the insulation will be removed prior to the VT-3 Visual Examination.
4. The following items are listed in IWF-2500:
  - (a) Mechanical connections to pressure retaining components and building structures
  - (b) Weld connections to building structure
  - (c) Weld and mechanical connections at intermediate joints in multiconnected integral and nonintegral supports

- (d) Clearances of guides and stops, alignment of supports, and assembly of support items
- (e) Hot or cold settings of spring supports and constant load supports
- (f) Accessible sliding surfaces

PPL will examine the above items within the boundary identified in Figure 2 of the Request for Alternative 4RR-06 as they pertain to the snubber attachment configuration. For the snubber itself, ISTD-4200 will be followed as applicable.

- 5. The requirement found in IWF-2430 to examine the supports immediately adjacent to a snubber that is found exceeding the acceptance standards, and that requires corrective measures, will be examined regardless of whether the adjacent support includes a snubber.
- 6. PPL implemented the use of Code Case OMN-13 during its third ten-year inspection interval after satisfactorily meeting the requirements of ISTD-4251 and ISTD-4252. The last visual examinations were performed in 2012 (Unit 2) and 2013 (Unit 1). In both cases the numbers of snubber failures as part of the visual examination program were below the required threshold for reduction of snubber visual examination frequency.

The "interval" for the PPL Snubber Program is the same as the PPL ISI Program. Alignment of the two programs is not necessary. The beginning of the Fourth Ten-Year Inspection Interval for both the PPL Snubber and PPL ISI Programs is June 1, 2014.

[While using Code Case OMN-13], should the number of unacceptable snubbers exceed the limits as prescribed in [table] ISTD-4252-1, the frequency of visual examinations [will be changed from ten-year to two-year (refueling outage)]. Only snubbers and their associated attachments will be under the requirements of this reduced visual inspection frequency.

- 7. Visual examination will be carried out in accordance with ASME Section XI, IWA-2213(a) thru IWA-2213(g). Visual examinations will be carried out by VT-3 qualified examiners.
- 8. Under the PPL Snubber Program, should a snubber or its associated attachment fail its visual examination, then the snubber will be removed for functional testing. The visual failure and the reason for the failure will also be entered into the SSES Corrective Action Program (CAP) system. Should the snubber fail its functional test, it is then considered a visual failure and the requirements of ASME OM Code ISTD-4240 then apply.

A failure of a support (without a snubber) will follow the requirements of IWF-2430.

- 9. New snubbers added will be examined in accordance with the ASME OM Code. New supports (without snubbers) will be examined in accordance with ASME Section XI, IWF-2410(c).

### 3.3.2 NRC Staff Evaluation

The NRC staff reviewed three issues of interest: (1) the snubber (pin-to-pin) inservice examination requirements; (2) the snubber attachment visual examination; and (3) the proposed alternative.

The SSES Units 1 and 2 snubber program for the fourth 10-year ISI interval is based on ASME OM Code 2004 Edition through 2006 Addenda, whereas the visual inspection of attachments for snubber's attachments and non-snubber supports (pressure boundary attachment to the building attachment), for the fourth 10-year ISI interval is based on ASME Code Section XI 2007 Edition through the 2008 Addenda.

The 2007 Edition through the 2008 Addenda of ASME Code Section XI contains Figure IWF-1300-1(f), which depicts the examination boundaries for a snubber (as shown in Figure 1 above). The boundaries indicate that the attachment of the snubber to the pressure boundary and building structure is required to be examined in accordance with IWF-2000, whereas the excluded snubber (pin-to-pin) is to be examined per ASME OM Code.

The licensee states that in order to eliminate the duplication of effort by tracking two different examination boundaries for one component, SSES, Units 1 and 2, requests an alternative to establish one examination boundary as shown in Figure 2 above.

The NRC staff reviewed all the information provided in the licensee's submittal and its response to the RAIs, and found the proposed examination boundaries of snubber and attachments, as shown in Figure 2 in lieu of Figure 1 boundaries, to be acceptable based on the following:

1. Incorporating both examination boundaries (snubbers and their attachments) into one program provides a better understanding of the condition of snubber and its associated attachments, without sacrificing any quality and safety.
2. SSES, Units 1 and 2, will be using VT-3 qualified personnel for both snubbers and its associated attachments. Performing both visual examinations of snubber and its attachments under the proposed boundaries in the snubber program using VT-3 qualified personnel to perform examinations provides a better understanding of the snubber and attachments.
3. The 2004 Edition through the 2005 Addenda of the ASME Code, did not specify any boundaries between snubber and its attachments. In the 2006 Addenda of the ASME Code, new boundaries between the snubber and its attachments were introduced via Figure IWF-1300-1(f), because the snubber (pin-to-pin) examination and testing requirements were moved into the ASME OM Code. Visual examination of snubber attachments by VT-3 qualified personnel meets the inspection requirement of ASME Code Section XI, 2007 Edition through the 2008 Addenda.
4. ASME OM Code Case OMN-13 requires 100 percent of the safety-related snubbers to be examined and evaluated at least once every 10 years. Now, based on the proposed boundaries of Figure 2, SSES, Units 1 and 2, will also perform 100 percent visual examination of snubber attachments along with the snubbers. This exceeds the requirements of the ASME Code Section XI, IWF-2500-1 tables (for attachments) which



only require 25% of Class 1, 15% of Class 2, and 10% of Class 3 required over a 10-year interval.

5. Visual examination of the snubber and associated attachments will be performed at the same time to save the time and dose.

Based on the above evaluation, and pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff finds that the licensee's proposed alternative in 4RR-06 of using boundaries as specified in Figure 2, which includes the snubber and its attachments, meets or exceeds the inspection requirements of IWF-2000 and therefore provides an acceptable level of quality and safety. Therefore, the use of the proposed alternative in 4RR-06 is authorized for the fourth 10-year ISI interval at SSES, Units 1 and 2. The snubber attachment inspections will be tracked by the Snubber Program in lieu of the ISI Program.

### 3.4 Licensee's Alternative Requests 4RR-07

#### 3.4.1 Licensee's Request

In Attachment 2 of the RAI response submitted by letter dated April 28, 2014, the licensee stated, in part, that:

[ASME Code, Section XI, IWB-2500,] [t]able IWB-2500-1 [Code Category B-P, Item Number B15.10] requires [that all Class 1 pressure retaining components be subject to a system leakage test with a Visual] a VT-2 [ ] examination each refueling outage. Subparagraph IWB-5221(a) requires that the system leakage test be performed at a pressure not less than the pressure corresponding to 100 percent rated reactor power.

The Reactor Pressure Vessel Head Flange Leak Detection Line is separated from the reactor pressure boundary by one passive membrane, a silver plated O-ring located on the vessel flange. A second O-ring is located on the opposite side of the tap in the vessel flange [ ]. This line is required during plant operation in order to indicate failure of the inner flange seal O-ring. Failure of the inner O-ring is the only condition under which this line is pressurized.

The configuration of this system precludes manual testing while the vessel head is removed because the odd configuration of the vessel tap [ ], combined with the small size of the tap and the high test pressure requirement (1035 psig minimum), prevents the tap in the flange from being temporarily plugged. The opening in the flange is only 3/16 of an inch in diameter and is smooth walled making a high pressure temporary seal very difficult. Failure of this seal could possibly cause ejection of the device used for plugging into the vessel.

A pneumatic test performed with the head installed is precluded due to the configuration of the top head. The top head of the vessel contains two grooves that hold the O-rings. The O-rings are held in place by a series of retainer clips. The retainer clips are contained in a recessed cavity in the top head (See Figure 4RR-07.1). If a pressure test was performed from the leak-off line side with the head on, the inner O-ring would be pressurized in a direction opposite to what it would see in normal operation. This test pressure would result in a net inward



force on the O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The O-ring material is a thin silver plating and could very likely be damaged by this deformation into the recessed areas on the top head.

In addition to the problems associated with the O-ring design that preclude this testing it is also questionable whether a pneumatic test is appropriate for this line. Although the line will initially contain steam if the inner O-ring leaks, the system actually detects leakage rate by measuring the level of condensate in a collection chamber. This would make the system medium water at the level switch. Finally, the use of a pneumatic test performed at a minimum of 1035 psig would represent an unnecessary risk in safety for the inspectors and test engineers in the unlikely event of a test failure, due to the large amount of stored energy contained in air pressurized to 1035 psig.

System leakage testing of this line is precluded because the line will only be pressurized in the event of a failure of the inner O-ring. It is extremely impractical to purposely fail the inner O-ring in order to perform a test.

A VT-2 visual examination will be performed on the accessible portions of the line after the refueling cavity has been filled to its normal refueling water level for at least 4 hours. For sections of the line that are inaccessible for direct VT-2 visual examination, examination will include the surrounding area underneath the piping for evidence of leakage, as permitted by IWA-5241(b). The static head developed due to the water above the vessel flange during flood-up will allow for the detection of any gross indications in the line. This examination will be performed with the frequency specified by Table IWB-2500-1 for a System Leakage Test (once each refueling outage).

### 3.4.2 NRC Staff Evaluation

The licensee requested relief from ASME Code, Section XI, Table IWB-2500-1, which requires a visual examination to be performed during a system leakage test. Since the licensee is proposing to perform the required visual examination at a lower pressure, the NRC staff finds that the request is actually from the requirements of IWB-5221, which sets the requirement that the leakage test shall be conducted at the pressure corresponding to 100 percent rated reactor power.

In order to perform the required test, the licensee could pressurize between the reactor vessel head O-rings, but this could possibly damage the inner O-ring. If the inner O-ring was damaged, the licensee would need to replace the O-ring set. The time and radiation exposure to remove and reinstall the RPV head to replace the O-rings would be a significant burden on the licensee. The licensee could install a plug; however, this would subject workers to high doses at the reactor vessel flange area. In addition, due to the configuration of the vessel tap, combined with the high test pressure requirements, there is the possibility that the device used for plugging the vessel could become lost in the reactor vessel, which could lead to fuel damage or other damage within the reactor coolant system.

The licensee has proposed performing a VT-2 visual examination of the accessible areas each refueling outage on the piping subjected to the static pressure head when the reactor cavity is filled. The VT-2, with a 4-hour wait time after the refueling cavity is filled to the normal refueling

water level, will allow the licensee to detect any significant through-wall defects in the reactor vessel leak-off line. The frequency of the inspections will ensure that upon startup from a refueling outage that the reactor coolant leak-off line does not have significant through-wall defects. The inner O-ring seals the reactor vessel and head and, along with the leak off piping, will provide pressure integrity. Although the reactor vessel leak-off line will initially contain steam if the inner O-ring leaks, the system detects leakage rate by measuring the level of condensate in a collection chamber. This will alert the licensee to the onset of O-ring leakage. The licensee is required to monitor both unidentified leakage (<5 gpm) and total leakage (<25 gpm) from the reactor coolant system pressure boundary. Should the inner O-ring and leak off piping no longer be capable of withstanding the pressure, and the leakage rates do not meet the TS requirements, shutdown of the reactor would be required. There is reasonable assurance that any problems in the subject piping would be detected through these measures. The NRC staff concludes that requiring compliance with the IWB-5221 system pressure test requirements results in a hardship or unnecessary difficulty without a compensating increase in the level of quality and safety.

Based on the above evaluation, and pursuant to 10 CFR 50.55a(a)(3)(ii) the NRC staff finds that the licensee's proposed alternative in 4RR-07 to perform a VT-2 visual examination on the accessible portions of the Flange Seal Leak Detection Line Pressure Retaining Components after the refueling cavity has been filled to its normal refueling water level for at least 4 hours, provides reasonable assurance of structural integrity or leak tightness of the subject components and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the use of the proposed alternative in 4RR-07 is authorized for the fourth 10-year ISI interval at SSES, Units 1 and 2.

### 3.5 Licensee's Alternative Request 4RR-08

#### 3.5.1 Licensee's Request

IWC-2500, Table IWC-2500-1, Code Category C-H, Item Number C7.10 requires that all Class 2 pressure retaining components be subject to a system leakage test with a Visual, VT-2 examination each inspection period. The system leakage test is to be performed at the system pressure obtained while the system, or portion of the system, is in service performing its normal operating function. The licensee requested relief for the control rod drive (CRD) accumulators and associated piping.

In Attachment 5 of the alternative request submitted by letter dated August 30, 2013, the licensee stated, in part, that:

As required by the SSES Technical Specifications, the CRD Accumulator Pressure must be greater than or equal to 940 psig. Once a week, the accumulator pressure is verified for each accumulator in accordance with SSES Technical Specifications. Additionally, the accumulator pressure is continuously monitored by system instrumentation. Since the accumulators are isolated from the source of makeup nitrogen, continuous monitoring of the CRD Accumulators serves as a pressure decay type test. Should accumulator pressure fall below approximately 980 psig, an alarm is received in the control room. The pressure for the accumulator is recorded and the accumulator is recharged and checked for leaks in accordance with SSES procedures. Should a leak be detected, corrective actions are taken to repair the leak in accordance with SSES procedures.

The licensee is requesting relief from the VT-2 visual examination requirements in Table IWC-2500-1 on the basis that continuous monitoring of the accumulator pressure and a TS required walkdown of each accumulator exceed the ASME Code Section XI requirement for a VT-2 visual examination.

### 3.5.2 NRC Staff Evaluation

Since monitoring the nitrogen side of the accumulators is continuous, any leakage from the accumulator would be detected by normal system instrumentation. An additional VT-2 visual examination performed once per inspection period would not provide an increase in safety, system reliability or structural integrity. The NRC staff finds that continuous pressure decay monitoring and a weekly TS required walkdown for the nitrogen side of the CRD accumulators is an acceptable alternative to the requirements of Table IWC-2500-1.

Based on the above evaluation, and pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff finds that the licensee's proposed alternative in 4RR-08 to perform continuous pressure decay monitoring on each accumulator provides an acceptable level of quality and safety. Therefore, the use of the proposed alternative in 4RR-08 is authorized for the fourth 10-year ISI interval at SSES, Units 1 and 2.

## 4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternatives described in requests 4RR-02, 4RR-06, and 4RR-08 provide an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i). As set forth above, the NRC staff determines that the proposed alternatives described in requests 4RR-05 and 4RR-07 provide reasonable assurance of structural integrity or leak tightness of the subject components and complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the proposed alternatives in requests 4RR-02, 4RR-05, 4RR-06, 4RR-07, and 4RR-08 for the fourth ISI interval at SSES, Units 1 and 2, which began on June 1, 2014, and is currently scheduled to end on May 31, 2024.

All other requirements of 10 CFR 50.55a and ASME Code, Section XI, for which relief was not specifically requested and approved, remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

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Date: June 9, 2014

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- 2 -

If you have any questions, please contact the SSES Project Manager, Mr. Jeffrey A. Whited, at [jeffrey.whited@nrc.gov](mailto:jeffrey.whited@nrc.gov) or 301-415-4090.

Sincerely,

/RA/

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Plant Licensing Branch I-2  
Division of Operating Reactor Licensing  
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

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