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SUSQUEHANNA STEAM ELECTRIC STATION
REQUEST FOR ADDITIONAL INFORMATION (RAI) FOR THE REVIEW OF THE SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (LRA) SECTIONS 4.2.3, 4.2.4, 4.2.5, 4.2.7, and 4.7.3
PLA-6283

Docket Nos. 50-387 and 50-388

References:

1) *PLA-6110, Mr. B. T. McKinney (PPL) to Document Control Desk (USNRC), "Application for Renewed Operating License Numbers NPF-14 and NPF-22," dated September 13, 2006.*

2) *Letter from Ms. E. H. Gettys, (USNRC) to Mr. B. T. McKinney (PPL), "Requests for Additional Information for the Review of the Susquehanna Steam Electric Station, Units 1 and 2, License Renewal Application," dated September 5, 2007.*

In accordance with the requirements of 10 CFR 50, 51, and 54, PPL requested the renewal of the operating licenses for the Susquehanna Steam Electric Station (SSES) Units 1 and 2 in Reference 1.

Reference 2 is the original request for additional information related to LRA Sections 4.2.3, 4.2.4, 4.2.5, 4.2.7, and 4.7.3. The RAI's 4.7.3-1 and 4.7.3-2 were modified as a result of teleconferences between NRC and PPL on 8/23, 9/12, 9/26, and 10/2. The enclosure to this letter provides the PPL response to each of the NRC RAI'S.

There are two new regulatory commitments contained herein as a result of these responses. These commitments are related to RAI response 4.7.3-1 regarding core plate hold down bolts and RAI 4.7.3-2 regarding the need to address the NRC's pending Safety Evaluation Report on BWRVIP-76.

If you have any questions, please contact Mr. Duane L Filchner at (610) 774-7819.

A120
NRR

I declare, under penalty of perjury, that the foregoing is true and correct.

Executed on: 10/18/2007

for BT McKinney

B. T. McKinney

Enclosure: PPL Responses to Request for Additional Information
(Sections 4.2.3, 4.2.4, 4.2.5, 4.2.7, and 4.7.3)

Copy: NRC Region I
Ms. E. H. Gettys, NRC Project Manager, License Renewal, Safety
Mr. R. V. Guzman, NRC Sr. Project Manager
Mr. R. Janati, DEP/BRP
Mr. F. W. Jaxheimer, NRC Sr. Resident Inspector
Mr. A. L. Stuyvenberg, NRC Project Manager, License Renewal, Environmental

Enclosure to PLA-6283
PPL Responses to
Request for Additional Information
(Sections 4.2.3, 4.2.4, 4.2.5, 4.2.7, and 4.7.3)

NRC RAI 4.2.3-1:

License renewal application (LRA) Section 4.2.3 states, "It may be noted that ART [adjusted reference temperature] values are well below the 200 F [as] suggested in Section 3 of Regulatory Guide 1.99 and are, thus, acceptable for the period of extended operation." Section 3 of Regulatory Guide 1.99, Revision 2 is for new plants and is not applicable to Susquehanna Units. Unlike the pressurized-thermal-shock (PTS) screening criteria used for evaluating the pressurized water reactor RPV material reference temperatures at the end of license fluence, there is no criteria for evaluating the RPV ARTs. The significance of ARTs is considered in the pressure and temperature (P-T) limit evaluation. Please revise LRA Section 4.2.3 and the associated updated final safety analysis renewal review (UFSAR) Supplement summary description so that Section 3 of Regulatory Guide 1.99 is not referenced.

PPL Response:

The SSES LRA Sections 4.2.3 and A.1.3.1.3 are revised to remove the reference to Section 3 of Regulatory Guide 1.99.

The third and final paragraph of LRA Section 4.2.3 is revised as follows:

> Remove sentence "It may be noted that ART values are well below the 200°F suggested in Section 3 of Regulatory Guide 1.99 and are, thus, acceptable for the period of extended operation."
>
> In place of the removed sentence, add the following:
>
> The ART values projected to 54 EFPY are used to develop Pressure-Temperature (P-T) limit curves, as discussed in LRA Section 4.2.4. There are no limits or specific acceptance criteria for the projected ART values.

The second and final paragraph of LRA Section A.1.3.1.3 is revised as follows:

> Remove sentence "The 60-year projected ART values are well below the temperature limit provided in Section 3 of Regulatory Guide 1.99 and are, thus, acceptable for the period of extended operation."
>
> In place of the removed sentence, add the following:
>
> The ART values projected to 54 EFPY are used to develop Pressure-Temperature (P-T) limit curves. There are no limits or specific acceptance criteria for the projected ART values.

NRC RAI 4.2.4-1:

LRA Section 4.2.4 states that calculations were performed to develop P-T limits for both units for the extended period of operation, using the 54 effective full-power year (EFPY) fluence values discussed in LRA Section 4.2.1. However, since the applicant did not include the revised P-T limits valid for 54 EFPY in the LRA for the staff review, it is inappropriate to state in LRA Section 4.2.4 that: "The 54 EFPY P-T curves for Units 1 and 2 demonstrate that there is sufficient operating margin for hydrostatic tests, heatup, cooldown, and core critical operation to the end of the period of extended operation." Please revise LRA Section 4.2.4 and its associated UFSAR Supplement summary description by taking this statement out. In addition, it is suggested to mention in the UFSAR Supplement summary description that Susquehanna Steam Electric Station (SSES) will submit the 54 EFPY P-T limits for NRC review and approval at the appropriate time to comply with Title 10 of the *Code of Federal Regulations*, Part 50, Appendix G.

PPL Response:

The SSES LRA Sections 4.2.4 and A.1.3.1.4 are revised to remove the statement regarding the acceptability of the 54 EFPY P-T curves.

The second paragraph of LRA Section 4.2.4 is revised as follows:

> Remove sentence "The 54 EFPY P-T curves for Units 1 and 2 demonstrate that there is sufficient operating margin for hydrostatic tests, heatup, cooldown, and core critical operation to the end of the period of extended operation."

The second paragraph of LRA Section A.1.3.1.4 is revised as follows:

> Remove sentence "The P-T curves for Units 1 and 2 at 54 EFPY demonstrate that there is sufficient operating margin for hydrostatic tests, heatup, cooldown, and core critical operation to the end of the period of extended operation."
>
> Add sentence "PPL will submit future P-T curve updates to the NRC as necessary to comply with 10 CFR 50 Appendix G."

NRC RAI 4.2.5-1:

In the July 28, 1998, safety evaluation report on Boiling Water Reactor Vessel and Internals Project (BWRVIP)-05, the NRC staff concluded that examination of the RPV circumferential shell welds would need to be performed if the corresponding volumetric examinations of the RPV axial shell welds revealed the presence of an age-related degradation mechanism. Confirm whether or not previous volumetric examinations of the RPV axial shell welds have shown any indication of cracking or other age-related degradation mechanisms in the welds. Please also confirm whether there are any flaw evaluations performed to-date on RPV flaws as a result of previous volumetric examinations of the RPVs, and, if flaw evaluations exist, why they are not considered as time-limited aging analysis (TLAAs).

PPL Response:

Inservice Inspection (ISI) volumetric examinations of the RPV axial shell welds for SSES Units 1 and 2 have shown no indications of cracking or other age-related degradation mechanisms in the welds. Therefore, no flaw evaluations have been required.

NRC RAI 4.2.7-1:

LRA Section 4.2.7 states that the SSES RPVs are bounded by the generic analysis that is discussed in NEDO-10029, "An Analytical Study on Brittle Fracture of GE-BWR Vessels Subject to the Design Basis Accident," because the reference temperature nil ductility (RT_{NDT}) shifts at 54 EFPY (35.9 °F for Unit 1 and 38.4 °F for Unit 2) are less than the RT_{NDT} shift of 50 °F used in the generic analysis. Please confirm whether the current licensing basis relies on the NEDO-10029 conclusions in addressing the reflood thermal shock issue and whether NEDO-10029 has been reviewed and approved by the NRC. If NEDO-10029 has been approved by the staff and utilized within the current SSES licensing basis, summarize the technical basis for determining the adequacy of this TLAA based solely on RT_{NDT} shift (as opposed to the ART of the limiting material). If NEDO-10029 was not approved, please provide the report for staff review to determine the acceptance of the methodology and results for use in the extended period of operation. Further, please demonstrate that the driving force based on the plant-specific design basis accident and SSES RPV geometry is bounded by the generic analysis.

PPL Response:

As documented in the SSES FSAR Section 3.13.1, the CLB for SSES Units 1 and 2 credits the analysis documented in NEDO-10029 to address the concern for brittle fracture of the reactor vessel due to reflood following a postulated loss of coolant accident. The analysis contained in NEDO-10029, by its reference in the FSAR, is considered to have been accepted by the NRC for the current license term. The following discussion provides the technical basis for the adequacy of the TLAA for the license renewal term.

The thermal shock analysis documented in NEDO-10029 [1] assumed a design basis loss of coolant accident (LOCA) followed by a low pressure coolant injection accounting for the full effects of neutron embrittlement at the end of 40 years. The analysis showed that the total maximum vessel irradiation (E > 1 MeV) at the mid-core inside of the vessel would be 2.4×10^{17} n/cm^2, which was considered to be below the threshold level of any nil-ductility temperature shift for the vessel material. As a result, it was concluded that the irradiation effects on all locations of the reactor vessels could be ignored. However, this analysis only bounded 40 years of operation.

The original analysis in NEDO-10029 has since been superseded by an analysis for BWR-6 vessels [2]. The more recent analysis is applicable to the SSES BWR-4 vessels because it evaluates the bounding LOCA event, a main steam line break, for a vessel design very similar to that of the SSES vessels. Because the vessel diameter has an insignificant effect on thermal stresses for thin-walled pressure vessels, the more recent analysis was considered to be applicable to all BWR-6 vessel diameters, including the 251" vessels. The SSES vessels are 251" diameter vessels. In addition, a vessel wall thickness of 6" was evaluated in the more recent BWR-6 analysis. The SSES vessels have a wall thickness of 6.1875". A critical parameter in the analysis is the material temperature at a depth of one-quarter of the wall thickness from the inside diameter, referred to as 1/4T. Because the SSES vessels have a slightly greater wall thickness, the temperature change (cooldown) due to the reflood event at the 1/4T depth would lag that of the BWR-6 vessel, because it would take longer for the cooldown to travel deeper into the vessel wall. This makes the BWR-6 analysis conservative for the SSES vessels, because a lower temperature is the worst case. Therefore, the more recent BWR-6 analysis is considered to be applicable to the SSES BWR-4 reactor vessels, as well as conservative for the SSES vessels.

The more recent BWR-6 analysis assumes end-of-life material toughness, which in turn depends on end-of-life adjusted reference temperature (ART). As stated above, the critical location for the fracture mechanics analysis is at 1/4T. For this event, the peak stress intensity at 1/4T occurs at approximately 300 seconds after the LOCA. The analysis shows that at 300 seconds into the thermal shock event, the temperature of the vessel wall at 1.5 inches deep (1/4T on the 6" thick BWR-6 vessel) is approximately 400°F. The maximum stress intensity factor, K, at 300 seconds is approximately 100 ksi$\sqrt{\text{in}}$. The acceptability of this K on a plant-specific basis for SSES can be determined by considering a revised allowable fracture toughness applicable to the SSES vessels for 54 EFPY. ART values for the SSES vessels for 54 EFPY of operation are described in Section 4.2.3 of the SSES LRA and tabulated in Table 4.2-7 for Unit 1 and Table 4.2-8 for Unit 2. The maximum calculated reactor vessel beltline material ART is 72.4°F for the Unit 1 Lower Intermediate Shell #2. Using the relationship shown in Figure G-2210-

1 of ASME Code, Section XI, Nonmandatory Appendix G [3], it is observed that the allowable material fracture toughness resides on the upper shelf of 200 ksi$\sqrt{in}$ for a ($T - RT_{NDT}$) value of (400 – 72.4) = 327.6°F. Because the maximum applied stress intensity factor, K, of 100 ksi$\sqrt{in}$ is less than the available fracture toughness of 200 ksi$\sqrt{in}$ after 54 EFPY, brittle fracture of the SSES reactor vessels due to vessel reflood following a design basis LOCA is not possible during the period of extended operation.

References:

1. NEDO-10029, "An Analytical Study on Brittle Fracture of GE-BWR Vessels Subject to the Design Basis Accident," June 1969, GE Proprietary.
2. Ranganath, S., "Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident," Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, August 1979, Paper G1/5.
3. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Nonmandatory Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 1998 Edition including 2000 Addenda.

NRC RAI 4.2.7-2:

All recent LRAs for plants with BWRs (e.g., Monticello, Brunswick, and Browns Ferry units) addressed the issue of reflood thermal shock analysis of RPV core shroud. Please address this TLAA, or explain why this topic is not a TLAA for SSES, Units 1 and 2.

PPL Response:

The CLB documentation identified in LRA Section 4.1.1 does not include a reflood thermal shock analysis for the RPV core shrouds for SSES Units 1 and 2. The review for TLAA identified no references or any other indication that a reflood thermal shock analysis for the core shrouds exists for SSES. Because there is no evidence of an analysis in the CLB documentation and no analysis was found, this topic is not a TLAA for SSES Units 1 and 2.

There is no regulatory requirement for this analysis. It is PPL's understanding that during early BWR licensing, the ACRS had raised a concern about reflood thermal shock. Typically, the question and answer were documented in the FSARs of those plants asked to address the issue, and the analyses would then be considered part the licensing basis. There is no record to indicate that PPL was asked to address thermal shock on the core shroud. As a result, there is no analysis for it in the SSES CLB.

NRC RAI 4.7.3-1:

LRA Section 4.7.3 states that, based on BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," the core plate hold-down bolts will have at least 81-percent preload remaining at 54 EFPY and, based on the GE extended power uprate (EPU) analyses of the core plate hold-down bolts, the preload at the end of 60 years would be adequate to prevent lateral motion of the core plate for the period of extended operation. This conclusion is not supported by any SSES plant-specific evaluation. Please provide the following additional information:

1. Demonstrate the applicability of the BWRVIP-25 loss of preload analysis to the SSES Units. Identify the temperature of the bolts during the normal operation and the projected bolt neutron fluence at the end of the period of extended operation for the SSES Units. Provide a plant-specific evaluation demonstrating that the loss of preload due to stress relaxation for the SSES RPV core plate hold-down bolts is bounded by the value of 19 percent from Appendix B of BWRVIP-25.

2. Perform a plant-specific core plate hold-down bolt analysis using the BWRVIP-25 Appendix A methodology, demonstrating that the axial and bending stresses for the mean and highest loaded hold-down bolts will not exceed the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III allowable stresses for Pm (primary membrane) and Pm + Pb (primary membrane plus bending) as a result of a plant-specific reduction in the bolt preload at the end of the extended period of operation. State clearly the assumptions on which the plant-specific analysis was based.

3. Provide sufficient information regarding the "GE EPU analyses" on the core plate hold-down bolts so that the staff can determine whether the SSES hold-down bolts are adequate to prevent lateral motion of the core plate for the period of extended operation.

PPL Response:

In a telecon with the NRC on 9/12/07, it was agreed that PPL would address this RAI with a revision to LRA Section 4.7.3. The revised discussion will not reference the GE EPU analyses. Therefore, item (3) in the RAI is not applicable and does not need to be addressed.

PPL revises LRA Section 4.7.3 as follows:

> The NRC safety evaluation report that references BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," (Reference 4.8.14) for license renewal

identifies loss of preload on the core plate rim hold-down bolts as one of the TLAA that must be addressed by applicants seeking license renewal.

PPL will address the loss of preload on the core plate rim hold-down bolts by taking one of the following two actions:

1. PPL will perform a SSES plant-specific evaluation consistent with BWRVIP-25 to demonstrate that the core plate rim hold-down bolts will be capable of preventing lateral displacement of the core plate for the period of extended operation. The evaluation will determine the maximum expected reduction in the bolt preload at the end of the period of extended operation, considering all applicable parameters (i.e., operating temperature, operating loads, and irradiation effects) and demonstrate the acceptability of the final preload at the end of the period of extended operation. Using the methodology of BWRVIP-25 Appendix A, the evaluation will also determine the primary membrane and bending stresses for the limiting bolt(s) to demonstrate that ASME Code allowables are not exceeded as a result of the reduction in bolt preload at the end of the period of extended operation. The evaluation will also provide either a) justification for not inspecting the core plate hold-down bolts, or b) an inspection strategy to ensure an adequate number of bolts are intact to prevent lateral displacement of the core plate. The evaluation will be submitted to the NRC for review no less than two years prior to the period of extended operation.

2. PPL will install core plate wedges to structurally replace the lateral load resistance provided by the hold-down bolts. With wedges installed, any loss of preload on the core plate rim hold-down bolts during the period of extended operation will have no effect on the lateral stability of the core plate. The wedges will be installed prior to entering the period of extended operation.

If the evaluation described as Action 1 above is unable to demonstrate acceptable bolt preload or bolt stress values at the end of the period of extended operation, appropriate corrective action will be taken prior to entering the period of extended operation. The installation of core plate wedges described as Action 2 above is considered an appropriate and acceptable corrective action.

Disposition: 10 CFR 54.21(c)(1)(ii) - Based on the commitment to perform the evaluation described as Action 1 above, the TLAA associated with core plate rim hold-down bolt loss of preload will be projected to the end of the period of extended operation. Otherwise, corrective action will be taken that will supplant the TLAA.

Similarly, PPL proposes to revise LRA Section A.1.3.6.3 as follows (same as above, excluding the Disposition statement):

The NRC safety evaluation report that references BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," (Reference 4.8.14) for license renewal identifies loss of preload on the core plate rim hold-down bolts as one of the TLAA that must be addressed by applicants seeking license renewal.

PPL will address the loss of preload on the core plate rim hold-down bolts by one of the following two actions:

1. PPL will perform a SSES plant-specific evaluation consistent with BWRVIP-25 to demonstrate that the core plate rim hold-down bolts will be capable of preventing lateral displacement of the core plate for the period of extended operation. The evaluation will determine the maximum expected reduction in the bolt preload at the end of the period of extended operation, considering all applicable parameters (i.e., operating temperature, operating loads, and irradiation effects) and demonstrate the acceptability of the final preload at the end of the period of extended operation. Using the methodology of BWRVIP-25 Appendix A, the evaluation will also determine the primary membrane and bending stresses for the limiting bolt(s) to demonstrate that ASME Code allowables are not exceeded as a result of the reduction in bolt preload at the end of the period of extended operation. The evaluation will also provide either a) justification for not inspecting the core plate hold-down bolts, or b) an inspection strategy to ensure an adequate number of bolts are intact to prevent lateral displacement of the core plate. The evaluation will be submitted to the NRC for review no less than two years prior to the period of extended operation.

2. PPL will install core plate wedges to structurally replace the lateral load resistance provided by the hold-down bolts. With wedges installed, any loss of preload on the core plate rim hold-down bolts during the period of extended operation will have no effect on the lateral stability of the core plate. The wedges will be installed prior to entering the period of extended operation.

If the evaluation described as Action 1 above is unable to demonstrate acceptable bolt preload or bolt stress values at the end of the period of extended operation, appropriate corrective action will be taken prior to entering the period of extended operation. The installation of core plate wedges described as Action 2 above is considered an appropriate and acceptable corrective action.

As a result of the above, a new commitment will be added on Table A-1. It was identified during the preparation of the response to RAI 4.7.3-1 that the plant-specific response to BWRVIP-25 Applicant Action Item 5 on page C-7 in Appendix C of the LRA contained a commitment that was not included on Table A-1. That commitment is now included with the commitment described in A.1.3.6.3. The commitment to be added to Table A-1 is as follows:

Table A-1
SSES License Renewal Commitments

Item Number	Commitment	FSAR Supplement Location (LRA App.A)	Enhancement or Implementation Schedule
55) Core Plate Hold down Bolts	PPL will either (1) obtain NRC approval of a SSES plant specific evaluation consistent with BWRVIP-25 to demonstrate that the core plate rim hold-down bolts will be capable of preventing lateral displacement of the core plate for the period of extended operation. The plant specific evaluation will address the inspection strategy for the hold-down bolts, or (2) install core plate wedges to structurally replace lateral load resistance provided by the bolts.	A.1.3.6.3	Prior to the period of extended operation.

NRC RAI 4.7.3-2:

The original RAI was revised by the NRC following discussion with PPL. The revised RAI statement is documented in an e-mail, dated 9/28/07, from Evelyn Gettys, NRC, to Duane Filchner, PPL.

Revised RAI

LRA Appendix C discussed the applicant's response to BWRVIP report application action items. The BWRVIP reports addressing the TLAA regarding irradiation assisted stress corrosion cracking (IASCC) in austenitic stainless steel RPV internals are BWRVIP-25, BWRVIP-26-A, "BWR Top Guide Inspection and Flaw Evaluation Guidelines," BWRVIP-76, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," and BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines." Although managing TLAA using aging management programs or other measures is stated in Appendix C in your response to BWRVIP report application action items, IASCC in austenitic stainless steel RPV internals should be discussed in the TLAA section to build the connection between LRA Section 4.0 and Appendix C. Further, although the NRC review of BWRVIP-76 (LR) for compliance with the license renewal rule has not been completed, you need to make a commitment to follow all BWRVIP-76 (LR) requirements and limitations and to address the conditions imposed by the staff in the pending NRC staff's safety evaluation on this report.

PPL Response:

This RAI is addressed by amending the LRA with new subsections in LRA Section 4.7 and LRA Appendix A, and a new commitment will be added to Table A-1, as follows:

New subsection in LRA Section 4.7:

4.7.4 Irradiation Assisted Stress Corrosion Cracking (IASCC)

Austenitic stainless steel reactor internal components exposed to a neutron fluence of greater than 5 x 10^{20} n/cm^2 (E > 1 MeV) are susceptible to irradiation assisted stress corrosion cracking (IASCC) in the BWR environment. As discussed in LRA Section 4.2.1, analyses were performed to determine neutron fluence for extended power uprate (EPU) conditions and for extended operation out to 54 effective full power years (EFPY). These projected fluence values were used to identify the components that would exceed the threshold fluence for IASCC.

The following reactor internal components have been identified as being susceptible to IASCC for the period of extended operation for SSES Units 1 and 2:

- Top Guide
- Core Shroud
- In-Core Flux Monitoring Dry Tubes
- Core Plate

The components identified as being susceptible to IASCC will require aging management to identify and address potential degradation (crack initiation and growth) prior to any loss of intended function.

All identified components have been evaluated for IASCC by the BWRVIP, as described in the inspection and evaluation guideline reports for each component: BWRVIP-26-A for the Top Guide; BWRVIP-76 for the Core Shroud; BWRVIP-47-A for the In-Core Flux Monitoring Dry Tubes; and BWRVIP-25 for the Core Plate. The inspection and evaluation guidelines of the identified BWRVIP reports will be implemented under the BWR Vessel Internals Program for SSES. As stated in LRA Appendix B, B.2.9, the program, with enhancement, will be consistent with NUREG-1801, XI.M9, "BWR Vessel Internals." In addition to the actions implemented under the BWR Vessel Internals Program, specific requirements imposed for license renewal have been specified for certain components. These additional requirements and actions have been identified in the responses to the BWRVIP Applicant Action Items in LRA Appendix C.

It is noted that BWRVIP-76 is currently under review by the NRC for compliance to the license renewal rule. Any future conditions, requirements, or limitations imposed by the NRC's safety evaluation for license renewal for BWRVIP-76 will be addressed by PPL.

Disposition: 10 CFR 54.21(c)(iii) - The aging effects due to IASCC of reactor internal components will be adequately managed for the period of extended operation.

New subsection in LRA Appendix A:

A.1.3.6.4 Irradiation Assisted Stress Corrosion Cracking (IASCC)

Austenitic stainless steel reactor internal components exposed to a neutron fluence of greater than 5×10^{20} n/cm^2 (E > 1 MeV) are susceptible to irradiation assisted stress corrosion cracking (IASCC) in the BWR environment. Analyses were performed to determine neutron fluence for extended power uprate (EPU) conditions and for extended operation out to 54 effective full power years (EFPY). The projected fluence values are used to identify the components that exceed the threshold fluence for IASCC.

The following reactor internal components have been identified as being susceptible to IASCC for the period of extended operation for SSES Units 1 and 2:

- Top Guide
- Core Shroud
- In-Core Flux Monitoring Dry Tubes
- Core Plate

The components identified as being susceptible to IASCC require aging management to identify and address potential degradation (crack initiation and growth) prior to any loss of intended function.

All identified components have been evaluated for IASCC by the BWRVIP, as described in the inspection and evaluation guideline reports for each component: BWRVIP-26-A for the Top Guide; BWRVIP-76 for the Core Shroud; BWRVIP-47-A for the In-Core Flux Monitoring Dry Tubes; and BWRVIP-25 for the Core Plate. The inspection and evaluation guidelines of the identified BWRVIP reports are implemented by the BWR Vessel Internals Program for SSES.

New Commitment for Table A-1:

Table A-1 SSES License Renewal Commitments			
Item Number	**Commitment**	**FSAR Supplement Location (LRA App.A)**	**Enhancement or Implementation Schedule**
56) BWRVIP-76	PPL will address any future conditions, requirements, or limitations imposed by the NRC's safety evaluation for license renewal for BWRVIP-76.	A.1.3.6.4	Prior to the period of extended operation.