ELECTRIC POWER RESEARCH INSTITUTE

2012-156 BWR Vessel & Internals Project (BWRVIP)

October 10, 2012

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: Joseph Holonich

Subject: Project No. 704 – BWRVIP Response to NRC Request for Additional Information on BWRVIP-241

Reference: Letter from John R. Jolicoeur (NRC) to David Czufin (BWRVIP Chairman), "Acceptance for Review and Request for Additional Information, for BWRVIP-241: BWR Vessel and Internals Project: Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-To-Vessel Shell Welds and Nozzle Blend Radii (TAC NO. ME6328)," dated February 15, 2012.

Enclosed are five (5) copies of the BWRVIP proprietary response to the NRC Request for Additional Information (RAI) on the BWRVIP report entitled "BWRVIP-241: BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-To-Vessel Shell Welds and Nozzle Blend Radii." The RAI was transmitted to the BWRVIP by the NRC letter referenced above.

A license renewal appendix for BWRVIP-241 is also enclosed. The BWRVIP requests that the NRC include this Appendix in their review and evaluation of BWRVIP-241. Note that this license renewal appendix also applies to BWRVIP-108NP.

Please note that the enclosed response contains proprietary information. A letter requesting that the response be withheld from public disclosure and an affidavit describing the basis for withholding this information are provided as Attachment 1. The response includes yellow shading to indicate the proprietary information. The proprietary information is also marked with the letters "TS" in the margin indicating the information is considered trade secrets in accordance with 10CFR2.390A.

Two (2) copies of a non-proprietary version of the BWRVIP response to the RAI are also enclosed. This non-proprietary response is identical to the enclosed proprietary response except that the proprietary information has been deleted.

G004
NRR

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Extra copies were sent to the PM

If you have any questions on this subject please call Randy Schmidt (PSEG Nuclear, BWRVIP Assessment Committee Technical Chairman) at 856.339.3740.

Sincerely,

Dennis Madison
Southern Nuclear
Chairman, BWR Vessel and Internals Project

ELECTRIC POWER
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Attachment 1

NEIL WILMSHURST
Vice President and
Chief Nuclear Officer

October 4, 2012

Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Request for Withholding of the following Proprietary Document:

BWRVIP Response to NRC Request for Additional Information on "BWRVIP-241: BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-To-Vessel Shell Welds and Nozzle Blend Radii"

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the information identified in the enclosed Affidavit consisting of the proprietary information owned by Electric Power Research Institute, Inc. ("EPRI") identified above (the "Response"). Proprietary and non-proprietary versions of the Correspondence and the Affidavit in support of this request are enclosed.

EPRI desires to disclose the Response in confidence to assist the NRC. The Response is not to be divulged to anyone outside of the NRC or to any of its contractors, nor shall any copies be made of the Report provided herein. EPRI welcomes any discussions and/or questions relating to the information enclosed.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (704) 704-595-2732. Questions on the content of the Report should be directed to ***Andrew McGehee*** of EPRI at (704) 502-6440.

Sincerely,

c: Sheldon Stuchell, NRC (sheldon.stuchell@nrc.gov)

1300 West W.T. Harris Boulevard, Charlotte, NC 28262-8550 USA • 704.595.2732 • Mobile 704.490.2653 • nwilmshurst@epri.com

ELECTRIC POWER
RESEARCH INSTITUTE

AFFIDAVIT

RE: Request for Withholding of the Following Proprietary Document:

BWRVIP Response to NRC Request for Additional Information on "BWRVIP-241: BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-To-Vessel Shell Welds and Nozzle Blend Radii"

I, Neil Wilmshurst, being duly sworn, depose and state as follows:

I am the Vice President and Chief Nuclear Officer at Electric Power Research Institute, Inc. whose principal office is located at 1300 W WT Harris Blvd, Charlotte North Carolina ("EPRI") and I have been specifically delegated responsibility for the above-listed Response that is sought under this Affidavit to be withheld (the "Response"). I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Response on behalf of EPRI.

EPRI requests that the Response be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information:

a. The Response is owned by EPRI and has been held in confidence by EPRI. All entities accepting copies of the Response do so subject to written agreements imposing an obligation upon the recipient to maintain the confidentiality of the Response. The Response is disclosed only to parties who agree, in writing, to preserve the confidentiality thereof.

b. EPRI considers the Response and the proprietary information contained therein (the "Proprietary Information") to constitute trade secrets of EPRI. As such, EPRI holds the Response in confidence and disclosure thereof is strictly limited to individuals and entities who have agreed, in writing, to maintain the confidentiality of the Response. EPRI made a substantial economic investment to develop the Response, and, by prohibiting public disclosure, EPRI derives an economic benefit in the form of licensing royalties and other additional fees from the confidential nature of the Response. If the Response and the Proprietary Information were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry, they would be able to use the Response for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the Response.

c. EPRI's classification of the Response and the Proprietary Information as trade secrets is justified by the Uniform Trade Secrets Act which California adopted in 1984 and a version of which has been adopted by over forty states. The California Uniform Trade Secrets Act, California Civil Code §§3426 – 3426.11, defines a "trade secret" as follows:

(1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy."

d. The Response and the Proprietary Information contained therein are not generally known or available to the public. EPRI developed the Response only after making a determination that the Proprietary Information was not available from public sources. EPRI made a substantial investment of both money and employee hours in the development of the Response. EPRI was required to devote these resources and effort to derive the Proprietary Information and the Response. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Response is highly valuable to EPRI.

e. A public disclosure of the Proprietary Information would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Proprietary Information both domestically and internationally. The Proprietary Information and Response can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

I have read the foregoing and the matters stated herein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 1300 W WT Harris Blvd being the premises and place of business of Electric Power Research Institute, Inc.

Date: 10-4-12.

Neil Wilmshurst

(State of North Carolina)
(County of Mecklenburg)

Subscribed and sworn to (or affirmed) before me on this 4th day of October, 2012 by Neil Wilmshurst, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature Deborah H. Rouse (Seal)

My Commission Expires 2nd day of April, 2016

BWRVIP Response to NRC Request for Additional Information on
"BWRVIP-241: BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-To-Vessel Shell Welds and Nozzle Blend Radii"

Non-Proprietary Version

BWRVIP Response to NRC Request for Additional Information on "BWRVIP-241: BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-To-Vessel Shell Welds and Nozzle Blend Radii"

Each item from the NRC Request for Additional Information (RAI) is repeated below verbatim followed by the response to that item.

RAI-1

The finite element stress analysis results for recirculation inlet and outlet nozzles are documented in Section 4 of this Technical Report (TR). The NRC staff compared the nozzle stress pattern and magnitude from this TR with those from the BWRVIP-108 TR, "BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the [BWR] Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii." The NRC staff found that under the heatup transient, the Pilgrim recirculation inlet nozzle (Figure 4-9), the Nine Mile Point, Unit 1, recirculation inlet nozzle (Figure 4-21), the Browns Ferry, Unit 2, recirculation outlet nozzle (Figure 4-33), and the Columbia recirculation outlet nozzle (Figure 4-45), all showed high compressive stresses at the inside surface when the steady state is approached. Page 4-3 of the TR provided explanation: "[t]his is due to the difference in thermal coefficient of expansion between the stainless steel clad and the low alloy steel nozzle material." However, the nozzle stresses reported in BWRVIP-108 for recirculation inlet and outlet nozzles showed no or insignificant compressive stresses at the inside surface under the same heatup transient. Please explain the discrepancies for inside surface stresses between these two reports, or confirm that they are caused by the BWRVIP-108 report's not considering clad in the nozzle stress analyses.

In addition, the trend of the through-thickness stress distributions at the nozzle blend radius region and the nozzle-to-shell weld region due to pressure for the Pilgrim recirculation inlet nozzle (Figure 4-8 of the TR; decreasing from the inside surface) is opposite to the trend for the Nine Mile Point, Unit 1, recirculation inlet nozzle (Figure 4-19 of the TR; increasing from the inside surface). Please provide insight or clarification to demonstrate that the opposite trend is not due to finite element modeling errors.

Response to RAI-1:

The differences for inside surface stresses between the BWRVIP-241 and BWRVIP-108 reports are due to the inclusion of cladding in the nozzle stress analyses. For the evaluation of the recirculation inlet and outlet nozzles in BWRVIP-241, the cladding is stainless steel and has a higher thermal expansion coefficient than the nozzle/vessel low alloy steel. At the steady state operating temperature of about 550°F, it does introduce a compressive stress in the cladding (or inside surface) due to the different thermal expansion coefficients. The BWRVIP-108 analysis did not consider cladding for additional conservatism and also because many of the BWR nozzle types do not have cladding.

The main reason that the stresses in the Nine Mile Point Unit 1 nozzle are different from the other nozzles is due to its unique nozzle-to-shell weld configuration. In Nine Mile Point Unit 1, the nozzle-to-shell weld is located between the pipe and vessel shell, as shown below in Figure 1 (from Figure 4-12 of BWRVIP-241). For the Pilgrim recirculation inlet nozzle, the weld is located between the nozzle forging & vessel shell as shown in Figure 2 (from Figure 4-1 of BWRVIP-241). In addition, the Nine Mile Point Unit 1 recirculation inlet nozzle is in the RPV bottom head region, as shown in Figure 3 (from Figure 4-11 of BWRVIP-241), while the Pilgrim recirculation inlet nozzle is in the RPV cylindrical body. Thus, the Nine Mile Point Unit 1 nozzle is connected to a spherical shell instead of a cylindrical shell like Pilgrim.
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Figure 1]] TS
Nine Mile Point Unit 1 Recirculation Inlet Nozzle Dimensions (shown in inches)

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Figure 2
Pilgrim Recirculation Inlet Nozzle Dimensions]] TS

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EPRI Proprietary Information

Figure 3
Location of Recirculation Inlet Nozzle, Nine Mile Point Unit 1

]] TS

RAI-2

For the Browns Ferry, Unit 2, recirculation outlet nozzle under the loss of feedwater pump transient, the maximum through-thickness hoop stress shape of the nozzle blend radius region (rapidly increasing and then rapidly decreasing) occurred sooner, at 6,710 seconds, than the minimum through-thickness hoop stress shape (rapidly increasing and then slowly increasing), at the 19,336 second (Figure 4-34(a) of the TR). This is opposite to the corresponding hoop stress distribution for the Columbia recirculation outlet nozzle under the loss of feedwater pump transient (Figure 4-47(a) of the TR), where the maximum through-thickness hoop stress shape (rapidly increasing and then rapidly decreasing) occurred later, at the 4,560 sec, than the minimum through-thickness hoop stress shape at the 904 sec. Please provide insight or clarification to demonstrate that the opposite trend is not due to finite element modeling errors.

Response to RAI-2:

The thermal stress in a component depends on four parameters:

1. Magnitude of the temperature range, ΔT
2. Rate of change in temperature, i.e. step change or slow ramp up/down
3. Direction of the temperature change, heat up or cool down
4. Thermal properties: thermal conductivity, heat capacity, boundary heat transfer coefficient and thermal expansion coefficient

Assuming the thermal properties are the same, the thermal stress is proportional to the magnitude of ΔT and rate of change in temperature. These two parameters offset each other, however. For instance, a step change of lesser ΔT could result in similar stress magnitude compared to a large ΔT at a gradual rate of change. The thermal stress responds opposite to the direction of temperature change. Decreasing temperature induces relative tensile stress and increasing temperature induces relative compressive stress. The thermal stress is also affected by thermal properties, as differences in material thermal properties dictates how fast the components respond and thus induce a non-uniform through-wall temperature distribution. In summary, thermal stress is a result of all the above mentioned factors.

In general, the through-thickness hoop stress range in the nozzle corresponds roughly to the temperature range and rate of change under the transient. High stresses are seen after a dramatic change of temperature over a short period of time. There is usually a time delay, due to heat transfer from the fluid to nozzle, and the time delay depends on the heat transfer coefficient of the fluid. These differences in maximum stresses and time of occurrence are a function of the transient definition, material properties and nozzle geometry.

For the Browns Ferry recirculation outlet nozzle, the loss of feedwater pump transient (LOFWP) begins at 3600 seconds; i.e. the X-axis in Figure 4-28 starts at 3600 seconds. The maximum through-thickness hoop stress shape of the nozzle blend radius region occurred at 3,110 seconds after the start of LOFWP transient in Figure 4-28 (or 6,710 seconds in Figure 4-34(a)), right after the nozzle experiences the highest thermal gradient (decreases 200°F over 3 minutes at time 2,380 seconds after the start of LOFWP, as shown in Figure 4-28). The lowest through-thickness hoop stress shape of the nozzle blend radius region occurred at 16,336 seconds after the start of

the LOFWP transient (or 19,936 seconds in Figure 4-34 (a)), after the nozzle achieves a steady state at the end of the transient.

For the Columbia recirculation outlet nozzle under the loss of feedwater pump transient, the maximum through-thickness hoop stress shape of the nozzle blend radius region occurred at 4,560 seconds, right after the nozzle experiences the largest temperature drop (decreases 75 °F), as well as the highest thermal gradient (decreases 75 °F over 7 minutes at a time 4,200 seconds, as shown in Figure 4-42). The lowest through-thickness hoop stress shape of the nozzle blend radius region occurred at 904 seconds, right after the nozzle experiences the smallest thermal gradient, i.e. a small drop in temperature over a relatively long time (decreases 3°F over 9 minutes at 540 seconds, as shown in Figure 4-42).

As seen in the above two cases, the maximum thermal stress occurs when there is a large temperature drop, and/or when there is a rapid temperature drop. This is understandable because, the temperature drop usually causes tensile stress at the nozzle and the existence of cladding complicates the through-wall stress profile: resulting in cladding compressive stress while keeping the nozzle outside surface in tension. Hence, the maximum temperature drop or temperature gradient corresponds to the maximum tensile stress in the nozzle. On the contrary, the minimum thermal stress occurs when there is a small temperature drop, or when there is no temperature drop, i.e. steady state. This is confirmed in both the Browns Ferry and Columbia cases.

Therefore, the differences between Browns Ferry and Columbia are due to the transient definition. This difference is not a result of finite element modeling errors. To make sure the transient definitions analyzed in BWRVIP-241 are representative, three additional recirculation nozzles, ranking right below the current Nine Mile Point, Browns Ferry and Columbia nozzles in Table 3-2 of BWRVIP-241, were chosen for the investigation. These nozzles are Oyster Creek Recirculation inlet Nozzle, Susquehanna Unit 1 Recirculation outlet nozzle and Hatch Unit 2 Recirculation outlet nozzle. It is found that the transient definitions of Oyster Creek Recirculation inlet Nozzle are identical to those of Nine Mile Point Recirculation inlet nozzle; the transient definition of the two recirculation outlet nozzles are identical to those of Browns Ferry recirculation outlet nozzle. In conclusion, the thermal transient definitions used in BWRVIP-241 are representative and bounding samples of the BWR fleet.

RAI-3

Figure 4-10 of the TR illustrated the axial and hoop stress distributions across the reactor pressure vessel (RPV) thickness along the nozzle-to-shell weld region, caused by sudden pump start of cold recirculation loop. The axial and hoop stress distributions at t = 2.136 seconds show significant compressive stresses close to inside surface of the nozzle. Please point out the time corresponding to t = 2.136 seconds in the temperature-time plot of the sudden pump start of cold recirculation loop transient shown in Figure 4-5 (b). It appears that at t = 2.136 seconds, the RPV and nozzle inside surfaces are still under the effect of cooldown from 522 °F to 130 °F and are not likely to produce compressive stresses. Please explain this observation.

Response to RAI 3:

Thermal analysis assumes that RPV and nozzle start at steady state temperature of 522°F, then cool down to 130°F at time t = 2 seconds. The time corresponding to t = 2.136 seconds in Figure 4-5(b) is shown in the figure below.

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Figure 4: Modified Figure 4-5(b): Sudden Pump Start of Cold Recirculation Loop

]] **TS**

The compressive stresses at t = 2.136 seconds is close to the inside surface and is a result of the difference in the thermal expansion coefficients between the cladding and vessel material.

RAI-4

Figure 4-21 of the TR illustrated the hoop and axial stress distribution across the RPV thickness along Path 1 (nozzle-to-shell weld) and Path 2 (nozzle blend radius) of the recirculation inlet nozzle (see Figure 4-18) of a plant under the heatup transient. Please provide the direction of the axial stress for both paths and confirm that the direction of the hoop stress is perpendicular to the paper of Figure 4-18. Provide an explanation for the sharp change of slope of the hoop stress distribution at about 0.5 inch from the inside surface in Figure 4-21(a). Please discuss whether this explanation can be applied to other nozzles showing similar stress distributions.

Response to RAI-4:

In Figure 4-18, the direction of axial stress is in the plane perpendicular to the nozzle axis and the direction of hoop stress is perpendicular to the paper. The axial/hoop directions correspond to the axial/hoop direction of the vessel shown in Figure 4-11.

The sharp change of slope of the hoop stress distribution at about 0.5 inch from the inside surface in Figure 4-21(a) is due to the difference in the thermal expansion coefficients for the cladding and vessel and due to the thermal effect. This explanation applies to other nozzles showing similar sharp change of stress close to the inside surface.

RAI-5

Regarding the probabilistic fraction mechanic (PFM) methodology, Section 5.1 of the TR states, "A modified version, VIPER-NOZ [21] was developed to include the evaluation capability for nozzles." This suggests that the supplemental PFM analyses reported in this report were performed using the VIPER-NOZ Code. Please identify the part(s) of the VIPER-NOZ Code methodology which goes beyond what were accepted in the safety evaluation (SE) for the BWRVIP-108 report and, therefore, needs NRC staff review. Please also justify the inputs (except for plant-specific geometries and thermal loading) which go beyond what were accepted in the SE for the BWRVIP-108 report.

Response to RAI-5:

The VIPER-NOZ Code is the same as that used in the BWRVIP-108 analysis and accepted in the staff's SE. The statement in BWRVIP-241 was not intended to imply that a new version of the code/methodology was used. Rather it was intended to point out that a modified version of the original VIPER code (as documented in BWRVIP-05 to evaluate RPV circumferential weld) was developed to include the evaluation capability for BWR nozzles. Therefore, BWRVIP-241 uses the same version of the VIPER-NOZ Code developed and used in BWRVIP-108.

The inputs for VIPER-NOZ (except for plant-specific geometries and thermal loading) are consistent with those accepted by the staff and documented in the SE for the BWRVIP-108 report.

RAI-6

For fatigue crack growth (FCG) analyses, it appears that the approach in the TR is more conservative than that in the BWRVIP-108 report for each of the four nozzles, as indicated in a statement of Section 5.3.1, "For conservatism, all thermal cycles, except the heatup transient are lumped as the sudden pump start on cold recirculation for FCG analysis." Similar statements can be found under Sections 5.3.2 to 5.3.4. Please confirm that the approach in the TR is more conservative than that in the BWRVIP-108 report. Please also quantify this conservatism, or confirm that the additional conservatism in the TR is insignificant.

Response to RAI-6:

For fatigue crack growth (FCG) analyses, the BWRVIP-241 takes the same approach as that documented in the BWRVIP-108 report. For each of the four nozzles, as indicated in Section 5.3.1 of the report, "For conservatism, all thermal cycles, except the heatup transient are lumped as the sudden pump start on cold recirculation for FCG analysis."

In BWRVIP-108, the thermal cycles, except for the heatup transient and cooldown transient, are lumped to the limiting transient for FCG analysis. In either case, the heatup and cooldown transient are considered as a complete cycle, from minimum to maximum back to the original state of minimum.

To assure the thermal cycles analyzed are representative, three additional recirculation nozzles, ranking right below the current Nine Mile Point, Browns Ferry and Columbia nozzles in Table 3-2, Oyster Creek Recirculation inlet Nozzle, Susquehana Unit 1 Recirculation outlet nozzle and Hatch Unit 2 Recirculation outlet nozzle, were chosen for the investigation. It is found that the thermal definitions and thermal cycles of Oyster Creek Recirculation inlet nozzle are identical to that of Nine Mile Point Recirculation inlet nozzle; the thermal definition and thermal cycles of two recirculation outlet nozzles are identical to that of Browns Ferry recirculation outlet nozzle evaluated in BWRVIP-241. Therefore, the thermal transient definitions and thermal cycles used in BWRVIP-241 are representative and bounding samples of the BWR fleet.

In conclusion, the approach used in BWRVIP-241 is the same as that used in the BWRVIP-108 report. Consequently, the conservatism is unchanged between BWRVIP-108 and BWRVIP-241.

RAI-7

Regarding the fracture mechanics (FM) models used in the analyses, Section 5.4 of the TR states that, "For the nozzle blend radius region, a nozzle blend radius corner fracture mechanics model [26] was used.... For the nozzle-to-vessel shell weld, the following crack models [27] were used...." Reference 26 is a private communication from P. M. Besuner to P. C. Riccardella. Reference 27 is a paper by C. B. Buchalet and W. H. Bamford. These references did not appear in the BWRVIP-108 report. To reduce the NRC staff's review time, please cite any NRC SE on submittals using the FM models in these two references, or provide them for NRC staff review.

Response to RAI-7:

The two references above were cited in the General Electric report for the Boiling Water Reactor Feedwater Nozzle/Sparger Program (GE NEDO-21821 [1].). This GE report addressed the cracking issue experienced in the feedwater nozzles and spargers of boiling water reactors. Figure 5 provides the reference page from NEDO-21821. As can be seen from Figure 5, the two references are items 11 and 13. The NRC's evaluation of the GE report was documented in NUREG 0619, Rev. 1 [2]. NEDO-21821 was cited in Appendix C, pages C-2 and C-7 of NUREG-0619, Rev. 1.

References:

[1] General Electric Report NEDO-21821, 'Boiling Water Reactor Feedwater Nozzle/Sparger Final Report', 78NED264, Class I, June 1978.

[2] BWR FeedWater Nozzle and Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10 (Technical Report), NUREG-0619-REV-1, Nov 1980.

REFERENCES - SECTION 9

1. *Report on the Integrity of Reactor Vessels for Light Water Power Reactors*, U.S. Atomic Energy Commission Report, Wash.-1285, Jan. 1974.

2. *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, U.S. Atomic Energy Commission Report Wash.-1400, August 1974.

3. Section 3.2.

4. J. G. Merkle, G. D. Whitman and R. H. Bryan, *An Evaluation of the HSST Program Intermediate Pressure Vessel Tests in Terms of Light-Water-Reactor Pressure Vessel Safety*, Oak Ridge National Laboratory Report, ORNL-TM-5090, November 1975.

5. *Re-evaluation of Feedwater Nozzle Cladding Defects, Millstone Unit 1*, Teledyne Materials Research Technical Report TR-2186 (a), October, 1975.

6. O.L. Bowie, *Analysis of an Infinite Plate Containing Radial Cracks Originating from the Boundary of an Internal Circular Hole*, Journal of Mathematics and Physics, Vol. 35, 1956.

7. S. Miyazono, S. Uedo, T. Kodaira, K. Shibata, T. Isozaki and N. Nakajima, *Fatigue Behavior of Nozzles of Light Water Reactor Pressure Vessel Model*, Third International Conference on Pressure Vessel Technology, ASME.

8. *Rules for Construction of Nuclear Vessels*, ASME Boiler and Pressure Vessel Code, Section III.

9. J. D. Gilman and Y. R. Rashid, *Three-Dimensional Analysis of Reactor Pressure Vessel Nozzles*, First National Conference SMIRT Volume 4, Part G, September 1971.

10. A. B. Fife, I. R. Kobsa, P. C. Riccardella, H. T. Watanabe, *Boiling Water Reactor Feedwater Nozzle/Sparger Interim Program Report* (NEDO-21480), July 1977, BWRPD, General Electric Company.

11. C. B. Buchalet, W. H. Bamford, *Stress Intensity Factor Solutions for Continuous Surface Flaws in Reactor Pressure Vessel*, ASTM-STP-590, 1975.

12. R. Labbens, A. Pellissier-Tanon, J. Heliot, *Practical Method for Calculating Stress Intensity Factors Through Weight Functions*, ASTM-STP-590, 1975.

13. Private Communication, P. M. Besuner to P. C. Riccardella, *Three-Dimensional Stress Intensity Factor Magnification Constants for Radial Feedwater Nozzle Cracks*, Failure Analysis Associates, June 29, 1976.

Figure 5: Reference Page of General Electric Report NEDO-21821 [1]

Additional Information Provided by the BWRVIP (not related to the NRC RAIs)

A License Renewal Appendix for BWRVIP-241 is provided on the following pages. The BWRVIP requests that the NRC include this Appendix in their review and evaluation of BWRVIP-241. Note that this License Renewal Appendix also applies to BWRVIP-108NP.

BWR Vessel and Internals Project

BWR Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii (BWRVIP-241), EPRI Report 1021005, October 2010.

Appendix A

BWR Nozzle Radii and Nozzle-to-Vessel Welds Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)

Appendix A
BWR Nozzle Radii and Nozzle-to-Vessel Weld Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)

The purpose of Appendix A is to demonstrate that this report provides the necessary information to comply with the technical information requirements pursuant to paragraphs 54.21 [a] and [c], and 54.22, and the NRC's findings under 54.29[a] of the license renewal rule (Reference A1). It is intended that the NRC's review and approval of Appendix A will allow utilities the option to incorporate the report and appendix by reference in a plant-specific integrated plant assessment (IPA) and time-limited aging analysis (TLAA) evaluation. If a license renewal applicant confirms that this report or BWRVIP-108NP applies to their plant's current licensing basis (CLB) and that the results of the Appendix A IPA and TLAA evaluation are in effect at their plant, then no further review by the NRC of the matters described herein is needed.

A.1 Description of the BWR Nozzles and Intended Functions

The nozzle-to-vessel weld in the BWR nozzles is the weld connecting the nozzle to the vessel and the nozzle blend radii is defined as the blend region (corner) made by the nozzle-to-vessel transition. The nozzle-to-vessel weld and blend radii of the nozzles are within the scope of the license renewal rule and are addressed in BWRVIP-108NP and BWRVIP-241. The nozzles evaluated are discussed in Section 4 of BWRVIP-108NP with nozzle dimensions given in Table 4-1. The general nozzle configuration and terminology is given in Figure 4-3 of BWRVIP-108NP. BWRVIP-241 contains a focused evaluation of recirculation inlet and outlet nozzles as shown in Figure 4-1 and 4-23, respectively. The general nozzle configurations are shown in Figures 4-1, 4-12, 4-23 and 4-36, respectively. The design, materials, operating, environmental, and other technical information is also contained in Section 4.0.

Generally, nozzles greater than 2 inches in diameter are required to ensure the capability to shut down the reactor and maintain it in a safe-shut down condition (54.4(a)(1)(ii)) and prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to 10 CFR 100 guidelines (54.4(a)(1)(iii)). The intended functions of the nozzles are to:

1. Control of the reactor water level by providing recirculation coolant
2. Cool reactor during off-normal operation mode
3. Emergency makeup and chemical purification of reactor water.

The intended functions are preserved under normal, upset, emergency, and faulted conditions. Section 4.0 describes the details of the bounding loading and load

combinations that need to be considered to determine that stress levels for the various operating conditions are consistent with the CLB.

A.2 Nozzles Subject to Aging Management Review (54.21(a)(1))

Paragraph 54.21(a)(1) of the rule provides the requirements for identifying if the nozzles are subject to aging management review. To satisfy the requirements of 54.21(a)(1), the guidance provided in the NEI industry guideline (Reference A2) was used to identify passive components and then to identify those that are long-lived. For nozzles greater than 2 inches, a screening methodology was not needed to make this determination. All of these nozzles are passive and long-lived. Therefore, the nozzles having a diameter greater than 2 inches are subject to an aging management review.

A.3 Management of Aging Effects (54.21 [a] [3])

(a) Assessment of Aging Effects and Inspection Programs

For the purpose of this Appendix, the BWR Reactor Pressure Vessel Internals Industry Report (Reference A3) and the resolution to the NRC's questions on the Industry Report are used to identify the aging mechanisms for the nozzles. Aging mechanisms are the causes of the aging effects. NUREG 1557 (Reference A4) is used to establish the correlation between the aging effects and their associated aging mechanisms. If the industry report concludes that the aging mechanism is significant, then the associated aging effect is included in this aging management review. Using this methodology and guidelines provided by BWRVIP-05 (Reference A5), it was determined that there is no material degradation mechanism. The only aging effect that requires aging management review for the nozzles is the effect of neutron fluence on material fracture toughness. For crack initiation, a fatigue analysis of the nozzles is typically addressed in the original ASME code RPV stress report. The stress and cycling ranges during the extended operating period require evaluation to demonstrate that the commitments for the current license term can be maintained. This evaluation is further discussed in Section A.4. This conclusion is consistent with the scope and intent of this report.

The causes of crack initiation and growth and a susceptibility assessment for the reactor vessel shell welds which are applicable to the nozzles are provided in Section 3.0 of BWRVIP-05 (Reference A5). The susceptibility factors of environment, materials, and stress state are discussed in Section 3.0 of BWRVIP-05. There is no service induced cracking history in the BWR RPV, as presented in Section 4.0 of BWRVIP-05.

The inspection guidelines for BWR RPV shell welds are described in Sections 3 & 4 of BWRVIP-05 which is also applicable to BWR RPV nozzles. Section 4 summarizes the inspection survey for the RPV vessels. These recommendations are based on the

results of conservative structural analyses of a representative geometry and the safety consequences of component failure. This assessment provides the framework for the basis of the inspection strategies and the guidelines for individual applicants to use in developing their plant specific inspection plans. The applicant will need to demonstrate that the structural analysis of the representative nozzle geometry and the evaluation of the safety consequences of the nozzle failure applies to their plant's CLB.

The recommended inspection procedure for nozzle-to-shell welds and nozzle blend radii is 25% of the nozzles every 10 years, as discussed in Section 1 of this report.

(b) Demonstration that the Effects of Aging are Adequately Managed

The effect of neutron fluence on material fracture toughness is the only aging effect for the nozzles that requires aging management review for license renewal. This aging effect will be managed by an inspection program incorporating the recommendations described in Section 1 and the resulting plant-specific strategy (i.e., plant-specific analysis and reinspection schedule). The inspection methods and implementation guidance addresses the:

- Location that requires inspection.
- Extent of baseline inspection for each location.
- Extent of reinspection for each location.
- Methodology for scope expansion should flaws be detected.
- Analysis methods to determine the need for corrective action and establish a reinspection schedule if flaws are detected.

Implementation of the inspection recommendations in an inspection program and the resulting plant-specific strategy will provide verification of the structural integrity of the nozzles. Therefore, there is reasonable assurance that crack initiation and growth will be adequately managed so that the intended functions of the nozzles will be maintained consistent with the CLB in the extended operating period.

A.4 Time Limited Aging Analyses (54.21 [c] [1])

The six criteria contained in the NEI industry guideline (Reference A3) were applied to identify the time limited aging analysis (TLAA) issues. That is, those calculations and analyses that:

1. Involve the nozzles,
2. Consider the effects of aging,
3. Involve time-limited assumptions defined by the current operating term,
4. Were determined to be relevant for making a safety determination,
5. Involved conclusions or provide the base for conclusions related to the capability of the nozzles to perform their intended function, and
6. Are incorporated or contained by reference in the CLB.

A fatigue analysis of the nozzle-to-vessel shell weld and nozzle blend radii is typically addressed in the original ASME Code RPV stress report. Fatigue would therefore be defined as a TLAA issue. The stress and cycling range during the extended operating period require evaluation by the applicant to demonstrate that commitments for the current license term can be maintained.

It should be noted that the actual number of thermal cycles experienced by a plant might exceed those assumed in BWRVIP-108NP. However, studies conducted by BWRVIP have concluded that stress corrosion cracking initiation and growth is a much more dominant contributor to probability of failure when compared to fatigue crack growth from thermal cycles. Therefore, the number of actual plant cycles can greatly exceed those assumed in the BWRVIP reports without having a significant effect on the results.

A.5 Exemptions (54.21 [c] [2])

Exemptions associated with the nozzles that contain TLAA analysis issues will be identified and evaluated for license renewal by individual applicants.

A.6 Technical Specification Changes or Additions (54.22)

There are no changes or additions to the technical specifications associated with the nozzles as a result of this aging management review to ensure that the effects of aging are adequately managed.

A.7 Demonstration that Activities will Continue to be Conducted in Accordance with the CLB (54.29[a])

Sections A.1, A.2, and A.3 address the requirements 54.21(a) of the rule. The nozzle components that are subject to aging management review are identified and it is demonstrated that the effects of aging are adequately managed.

Sections A.4 and A.5 address the requirements of 54.21 (c) of the rule. Plant-specific time limited aging analyses (TLAAs) and exemptions that require evaluation will be evaluated by the applicant.

Section A.6 addresses the requirements of 54.22 of the rule. There are no technical specification changes or additions necessary to manage the effects of aging for the nozzles during the period of extended operation.

Therefore, actions have been identified and have been or will be taken by utilities with BWR plants, such that there is reasonable assurance that the activities authorized by license renewal for the nozzles will be in accordance with the CLB.

A.8 References

A1 Title 10 of the: Code of Federal Regulations, Part 54, "Requirements for License Renewal of Operating Licenses for Nuclear Power Plants," (60 Federal Register 22461), May 8, 1995.

A2 Nuclear Energy Institute Report NEI 95-10 (Rev. 0), Industry Guideline for Implementing the Requirements of 10 CFR Part 54, the License Renewal Rule.

A3 NUMARC 90-03, BWR Reactor Pressure Vessel Internals License Renewal Industry Report, Revision 1, June 1992.

A4 NUREG-1557, Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal, October 1996.

A5 BWRVIP-05 Report, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," Electric Power research Institute, TR-105697, September 1995.