ELECTRIC POWER RESEARCH INSTITUTE

2014-049 BWR Vessel & Internals Project (BWRVIP)

April 10, 2014

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-18, Revision 2

Reference: Letter from Joseph J. Holonich (NRC) to Dennis Madison (BWRVIP Chairman), Request for Additional Information on BWRVIP [Boiling Water Reactor (BWR) Vessel Internals Project]-18 Revision 2, "BWR Core Spray Internals and Flaw Evaluation Guidelines" (TAC NO. ME8809)," dated September 27, 2013.

Enclosed are five (5) copies of the BWRVIP proprietary response to the NRC Request for Additional Information (RAI) on the BWRVIP report entitled "BWRVIP-18, Revision 2: BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Evaluation Guidelines." The RAI was transmitted to the BWRVIP by the NRC letter referenced above.

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.

If you have any questions on this subject please call Ron DiSabatino (Exelon, BWRVIP Assessment Committee Technical Chairman) at 717.456.3685.

Sincerely,

Andrew McGehee, EPRI, BWRVIP Program Manager
Dennis Madison, Southern Nuclear, BWRVIP Chairman

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NRR

Additional copies were sent to the PM

ELECTRIC POWER RESEARCH INSTITUTE

KENNETH CANAVAN
Director, Plant Technology

April 10, 2014

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 Information Included in:

Response to NRC Request for Additional Information (RAI) on BWRVIP-18, Revision 2: BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines

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 report identified in the enclosed Affidavit consisting of the proprietary information owned by Electric Power Research Institute, Inc. ("EPRI") identified in the attached report. Proprietary and non-proprietary versions of the Response and the Affidavit in support of this request are enclosed.

EPRI desires to disclose the Proprietary Information in confidence to assist the NRC review of the enclosed submittal to the NRC. The Proprietary Information 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 Proprietary Information 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) 595-2731. Questions on the content of the Report should be directed to Andy McGehee of EPRI at (704) 502-6440.

Sincerely,

CHARLOTTE OFFICE
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Customer Service 800.313.3774 • www.epri.com

ELECTRIC POWER RESEARCH INSTITUTE

AFFIDAVIT

RE: Request for Withholding of the Following Proprietary Information Included In:

Response to NRC Request for Additional Information (RAI) on BWRVIP-18, Revision 2: BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines

I, Kenneth Canavan, being duly sworn, depose and state as follows:

I am the Director, Plant Technology at Electric Power Research Institute, Inc. whose principal office is located at 3420 Hillview Avenue, Palo Alto, California. ("EPRI") and I have been specifically delegated responsibility for the above-listed report that contains EPRI Proprietary Information that is sought under this Affidavit to be withheld "Proprietary Information". I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Proprietary Information on behalf of EPRI.

EPRI Information is identified in yellow shading with double square brackets. [[This sentence is an example.]] Tables containing EPRI proprietary information are identified with double square brackets before and after the object. 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.

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

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information (see e.g., 10 C.F.R. § 2.390(a)(4)):

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

b. EPRI considers the Proprietary Information contained therein to constitute trade secrets of EPRI. As such, EPRI holds the Information in confidence and disclosure thereof is strictly limited to individuals and entities who have agreed, in writing, to maintain the confidentiality of the Information.

c. The information sought to be withheld is considered to be proprietary for the following reasons. EPRI made a substantial economic investment to develop the Proprietary Information 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 Proprietary Information. If 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 Proprietary Information for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the Proprietary Information.

d. EPRI's classification of 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:

"'Trade secret' means information, including a formula, pattern, compilation, program device, method, technique, or process, that:

(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."

e. The Proprietary Information contained therein are not generally known or available to the public. EPRI developed the Information 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 Proprietary Information. EPRI was required to devote these resources and effort to derive the Proprietary Information. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Proprietary Information is highly valuable to EPRI.

f. 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 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: 04/10/2014

Kenneth Canavan

Kenneth Canavan

(State of North Carolina)
(County of Mecklenburg)

Subscribed and sworn to (or affirmed) before me on this 10th day of April, 2014, by Kenneth Canavan, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature Deborah H. Rose (Seal)

My Commission Expires 2nd day of April, 2016

Response to NRC Request for Additional Information (RAI) on
BWRVIP-18, Revision 2: BWR Vessel and Internals Project, BWR Core Spray Internals
Inspection and Flaw Evaluation Guidelines

Non-Proprietary Version

ENCLOSURE

BWRVIP Response to NRC Requests for Additional Information on BWRVIP-18, Revision 2

Each Request for Additional Information (RAI) received from the NRC is repeated verbatim below followed by the BWRVIP Response.

RAI 1

Section 5.1.2 of TR BWRVIP-18, Revision 2 (the TR) indicated that the limit load evaluation methodology in the TR is the one described in Appendix C of Section XI of the *American Society of Mechanical Engineers Boiler and Pressure Vessel Code* (ASME Code). The NRC staff confirmed that the TR's assertion is true up to the 2001 Edition of the ASME Code. The 2004 and later edition of the ASME Code made two major changes to the Appendix C methodology:

(1) The definition of flow stress σ_f was revised from $3S_m$ (S_m is the allowable design stress intensity as determined from Table I-1.0 of ASME Code, Section III, Appendix I) to $(S_y + S_u)/2$, using ASME Code specified yield and ultimate strength of the material, or $\sigma_f = (\sigma_y + \sigma_u)/2$ if the measured yield and ultimate strength of the material are available.

(2) The equations connecting the applied stresses and the failure bending stress for the flux welds and non-flux welds (i.e., Equations 5-5 and 5-6 of the TR) were revised to reflect different safety factors for membrane and bending stresses.

Please revise the limit load methodology to be consistent with the later editions of the ASME Code because, for instance, your proposed old Appendix C approach is non-conservative for Level C loading when compared to the current Appendix C approach.

Further, the TR proposed the limit load methodology described in BWRVIP-76 (Reference 8) as an alternative. Please confirm that the correct reference number for BWRVIP-76 is 12, not 8.

<u>BWRVIP Response to RAI 1</u>

The BWRVIP has previously reviewed the effect of changes in non-mandatory Appendix C, Section XI, ASME Code on the structural integrity assessment of BWR internal components and has concluded that a change to the BWRVIP guidance is not warranted. There are two primary reasons for this conclusion.

First, the ASME makes periodic revisions to the Code. The revisions are either included as Code Cases or as outright revisions to the Code that are published periodically. The general practice is to allow the user to specify the ASME Section XI Code edition and addenda to be used for a specific plant's interval for inservice inspection. The user is allowed the option of using either their current code of record or a later Code. This approach has been accepted by the NRC staff unless the revisions in the later Code versions are specifically not approved by the NRC. Thus, the use of the earlier Code editions (1989 to 2001 editions) in BWRVIP-18, Revision 2 is consistent with accepted practice.

Secondly, the impact of the revised code rules is very small for flaw evaluation of core spray piping. The flaw evaluation methodology in BWRVIP-18-A (and the subsequent revisions) was

based on Appendix C in the 1989 edition of the ASME Code. The newer Code rules referred to in this RAI are based on the revised Appendix C of Section XI first published in the 2002 Addenda and later formally issued in the 2004 Edition of the ASME Code (referred herein as 'new Code rules'). The major changes in the new Code rules were the new definition of the flow strength and different safety factor for membrane and bending stresses for different operating conditions (Level A, B, C and D). The BWRVIP conducted an evaluation of the difference between the new Code rules and the 1989-2001 Code rules (referred herein as 'old Code rules'). The objective of the study was to determine if there was a compelling reason to revise the current guidelines to incorporate the 2004 Appendix C rules for evaluation of flaws in BWR internal piping. The evaluation concluded that there were no significant differences in allowable flaw size for core spray piping determined using the new and old code rules. The primary reason is that, while the new Code rules would indicate that allowable flaw size is somewhat smaller for Level C conditions (and only Level C conditions), the core spray evaluations are bounded by Level B or Level D conditions for which the Code changes make little difference. The key points in the evaluation are summarized here.

Table 1 shows a comparison of the key aspects of the old and new Code rules. Table 2 shows the specific structural factors in the new Code rules. The effects of the flow stress differences and the different structural factors are summarized below.

The flow stress is defined as (Sy + Su) / 2 rather than 3 Sm where Sy and Su are the Code values of the yield and ultimate strength respectively. For Type 304 or 316 stainless steel at 550°F, the new flow stress is (18.8 + 63.5) / 2 = 41.15 ksi rather than 3 Sm= 3 x 16.9 = 50.7 ksi used in the previous rules. This would imply that the 2004 Code rules are more conservative for austenitic piping by a factor of 1.23 (50.7/41.15).

However, the structural factors were defined for Level A-D conditions (Table 1) and separate factors were defined for both membrane and bending stresses. Taking the Level B factors as an example, the new rules require a factor of 2.4 for membrane and 2.0 for bending instead of the 2.77 used in the old rules. This introduces a somewhat less conservative element in the calculation with respect to the structural factors under the new rules. If one averages the new structural factors for bending and membrane stress (2.4 + 2) / 2 = 2.2 and compares it to the old factor of 2.77, it is seen that the new rules would be less conservative by the factor of 1.26. Thus, when the conservatism due to the flow stress (1.23) is included with the reduced safety factors (1.26), one can conclude (at least for Level B conditions) that the 2004 Code rules and the 2001 Code rules essentially lead to the same overall structural factors.

This conclusion holds true for all conditions except Level C. For level C conditions, the combination of the new safety factors and the revised flow stress do, in fact, lead to a situation where the old Code rules are non-conservative by 15 or 20 percent. However, Level C is not limiting for the core spray piping for the following reasons:

- As shown in Table 3 which lists the load combinations from BWRVIP-18, Rev. 2, there are no load combinations that involve Level C loading for the core spray piping. Table 3 shows the typical load combinations used for the evaluation of

BWR internal core spray piping. The load combinations of interest are for Level B or Level D conditions.

- The membrane stress (for which the difference between the new and old Code rules are most significant) in the core spray piping is small since the differential pressure between the OD and ID is not significant.

- Generally, seismic loading is the major load source for the core spray piping, Seismic loads are considered only for Level B or Level D load combinations. Because of the higher structural factor for level B, it is the limiting condition in most cases.

Based on the fact that the old rules and the new code rules lead to the same overall structural factors for Level B conditions which are governing, it was concluded that no changes in the structural evaluation methodology are technically warranted.

The BWRVIP considered making the changes to the evaluation methodology solely to be in strict compliance with the latest version of the Code. However, since it was not necessary from a technical perspective and since it would require a significant cost in time and personnel resources by the industry to revise flaw evaluations and flaw handbooks with no corresponding increase in safety or change in the final outcome of the evaluation, the proposal was rejected.

In response to the last item in RAI #1, the BWRVIP agrees that the correct reference number for BWRVIP-76 is 12, not 8. The typo will be corrected in the –A version of the report.

Table 1. Comparison of the features of the Old and New Code Rules

Old Code Rules (1989 to 2001 Codes)	New Code Rules (Post 2004 Codes)
The flow stress is assumed to be equal to 3Sm where Sm is the ASME Code Design Stress Intensity. For Type 304 or 316 stainless steel at 550°F, the flow stress is 3 Sm= 3 x 16.9 = 50.7 ksi.	The flow stress is defined as (Sy + Su) / 2 rather than 3 Sm where Sy and Su are the Code values of the yield and ultimate strength respectively. For Type 304 or 316 stainless steel at 550°F, the new flow stress is (18.8 + 63.5) / 2 = 41.15 ksi.
The structural factors are based on two classifications – 2.77 for Service Levels A and B (normal and upset conditions, respectively) including design conditions and 1.39 for Service Levels C and D (emergency and faulted conditions, respectively). This is consistent with the IWB-3600 of Section XI.	The structural factors were defined for Level A-D conditions (Table 2) and separate factors were defined for both membrane and bending stresses.
Different Z factors for SMAW and SAW welds.	Z factor is the same for all flux welds and is set equal to the value for SAW welds
The structural factors are specified for (Pm+Pb).	Separate factors for both membrane and bending stresses as shown in Table 2.

Table 2. Structural Factors for Circumferential Cracks the New Code

Service Level	Old Code Rules	New Code Rules	
	Structural Factor on Pm+Pb	Membrane Stress Pm SF_m	Bending Stress Pb SF_b
A	2.77	2.7	2.3
B	2.77	2.4	2.0
C	1.39	1.8	1.6
D	1.39	1.3	1.4

Table 3 Typical Load Combinations for Internal Core Spray Piping and Spargers

[[

Content Deleted
EPRI Proprietary Information

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RAI 2

Section 5.1.4 of the TR presents leak rate calculation methods, including a method documented in Electric Power Research Institute (EPRI) NP-3596-SR, "PICEP: Pipe Crack Evaluation Program (Revision 1)." This TR section also proposes 10 steps to predict leak rates from inaccessible welds. The NRC staff has the following questions:

(1) Step 3 is related to computing the leak rate from each similar accessible weld that is judged to have a through-wall flaw. Please confirm that this judgment means that a through-wall flaw is assumed if (a) UT results of the weld indicated that after crack growth, the flaw depth will be 75 percent of the wall thickness, and (b) VT inspection conducted identified flaw indications. Please supplement the judgment if there are cases other than (a) and (b) mentioned above.

(2) Step 4 mentioned the total number of similar accessible leakage welds for a plant. Please confirm that for each plant the total number is based on all inservice inspection records of that plant since the first day of its operation.

(3) Step 9 contains a typo "•" in the first line and should be corrected if you plan to revise the TR.

<u>BWRVIP Response to RAI 2</u>

1. Flaws detected in core spray piping by either VT or UT are considered to be through-wall. The depth information acquired in a UT inspection is not considered to be essential information. In developing the response to this RAI, it became clear that the through-wall assumption for flaws detected by UT is not described as clearly as it could be in BWRVIP-18, Revision 2. Section 5.1.1 (Flaw Characterization) will be revised as follows:

   All indications detected visually or with UT must be considered to be through-wall for the purposes of structural and leakage evaluations.

2. The NRC is correct: the number of accessible leakage welds is based on results of all inspections since the first day of plant operation.

3. The typo does not appear in the EPRI published version of the report. It may be an artifact of the version of Adobe software used to read the file. For reference, Step 9 should read:

   "When the number of inaccessible leakage welds determined from Step 6 is ≤ 1 the cumulative ratio for the inaccessible leakage weld determined from Step 7 will be 1.0, and the estimated leak rate for the inaccessible weld will be equal to the highest calculated leak rate for the accessible leakage welds."

RAI 3

Table 5-1 of the TR contains three typos: "Table 3-5" was mentioned in each of the three boiling water reactor (BWR) designs in Table 5-1. However, Table 3-5 does not exist in the TR. It should be Table 3-4 instead.

BWRVIP Response to RAI 3

The NRC is correct that the table reference is incorrect. The correct reference is to Table 3-1. This will be corrected in the –A version of the report.

RAI 4

Section 5.2 of the TR provides general guidance for evaluating the structural integrity of core spray piping brackets. It states, "A limit load approach similar to that outlined in [13] could be used to evaluate the adequacy of the cracked cross-section." Please provide the following information related to the brackets and their structural integrity: (1) material, (2) operating experience about cracking, (3) crack growth rate, and (4) the limit load methodology for the bracket side of a solid circular cross section, noting that, even if the limit load methodology of ASME Code, Section XI, Appendix C (i.e., Reference 13) is followed by all licensees, inconsistent limit load methodologies for brackets may be developed.

BWRVIP Response to RAI 4

1. Material
   The core spray piping is supported by brackets that are welded either to the RPV wall or to the core shroud. The piping brackets are made of stainless steel and welded with stainless or Alloy 182 weld metal.

2. Operating experience about cracking
   As described in Section 3.4.1 of BWRVIP-251, cracking in the piping brackets themselves is nonexistent: "Regarding piping brackets .... there are no significant indications reported aside from a limited number of tack weld indications, none of which were determined to detrimentally affect component function."

3. Crack Growth Rate
   Since no cracking has been observed it is not possible to determine the in-situ crack growth rates. However, for the purpose of flaw evaluation, the NRC-approved crack growth rate of [[ ]] in/hour would be used at each crack tip. TS

4. Limit Load Methodology for the Bracket Side
   The BWRVIP has not developed detailed flaw evaluation guidance for all components and sub-components. Doing so would be an immense effort given the variety of cracking that could be envisioned. For core spray pipe brackets, only general guidance is provided (e.g. limit load evaluation methodology can be used) as is done for many other components in the BWRVIP Program. While the details of each analysis might be slightly different, the analyses would be based on the sound technical principles of the Code.

RAI 5

Section 5.3.1 of the TR provides general guidance for evaluating the structural integrity of core spray sparger brackets due to bracket-side heat affected zone (HAZ) cracking. It states, "A limit load analysis can be used to evaluate the adequacy of the cracked cross-section." Please provide the following information related to the sparger brackets and their structural integrity: (1) material, (2) operating experience about cracking, (3) crack growth rate, and (4) the limit load methodology for the bracket side of a solid rectangular cross section, noting that even if the limit load methodology of ASME Code, Section XI, Appendix C is followed by all licensees, inconsistent limit load methodologies for brackets may be developed. In addition, as an alternative to flaw evaluation on cracked brackets, the TR proposed a functionality analysis discounting the cracked brackets. For this approach, please provide information regarding (1) existence of such an analysis, generic or plant-specific and (2) detailed guidance for performing such a functionality analysis including consideration of the additional flow induced vibration loads and loose part generation.

BWRVIP Response to RAI 5

1. Material
   The core spray sparger is supported by brackets that are welded to the shroud. The piping brackets are made of stainless steel and welded with stainless weld metal. The shroud is also stainless steel.

2. Operating experience about cracking
   Inspection data for BWR/2-5 welded sparger brackets indicate a number of IGSCC indications. BWRVIP-251 (Section 3.4.1) states that of the 410 brackets inspected, only 19 have shown some degree of cracking. All of the cracking reported is in the shroud HAZ material and not in the sparger bracket itself. These cracks have occurred in both 304SS and L-Grade core shrouds.

3. Crack Growth Rate
   Available inspection data on sparger bracket cracking does not permit a determination of a component-specific crack growth rate. However, the crack growth rate associated with the bracket welds is not expected to be different than the typical IGSCC growth rates in BWR shroud welds or other components. For the purpose of crack growth evaluation (lengthening), the NRC-accepted value of [[ ]] in/hour is used for determining the length growth at each crack tip. TS

4. Limit Load Methodology for the Bracket Side
   The BWRVIP has not developed detailed flaw evaluation guidance for all components and sub-components. Doing so would be an immense effort given the variety of cracking that could be envisioned. For core spray sparger brackets, only general guidance is provided as is done for many other components in the BWRVIP Program. The general guidance for sparger brackets is found in Section 5.1.2 of BWRVIP-18 where a limit load method consistent with Section XI is recommended. While the details of any two

evaluations developed by utilities may differ slightly, they will be solidly based on the technical foundation of the Code. See also the response to RAI #1.

Detailed guidance has not been developed for the functionality evaluation referenced in Section 5.3.1. Since it is not currently anticipated that such an evaluation will be required by utilities and since additional guidance is not available by reference to Section XI, the functionality evaluation will be deleted from BWRVIP-18, Rev 2 as an option for evaluating sparger brackets.

RAI 6

Section 5.3.2 of the TR provides general guidance for evaluating the structural integrity of core spray sparger brackets due to shroud-side HAZ cracking. It states that, "the generic analyses demonstrate that only a small fraction of the bracket-to-shroud weld is required to maintain the function of the bracket." Provide details of this generic analysis to support the quoted statement.

<u>BWRVIP Response to RAI 6</u>

The referenced analysis was based on a plant specific evaluation for a BWR/3 plant where a surface crack [[ ]] inches long was found on the shroud side of the weld. The stresses on the bracket for the Level D condition were determined to be Pm = [[ ]] ksi and Pm+Pb= [[ ]]. The allowable through-wall crack length (a_{allow}) can be calculated as follows: TS TS

$$\sigma_f\,(L - a_{allow}) = \sigma * L * SF$$

σ_f = flow stress = 3 Sm = 50.7 ksi

L = total length of the fillet weld = 8 inches

SF = structural factor = 1.39

σ = applied stress in the bracket = [[ ]] TS

The allowable flaw size is [[ ]] inches. Since the weld length was 8 inches, the required uncracked ligament is [[ ]] inches. Thus only a small fraction (~15%) of the bracket-to-shroud weld is required to maintain the structural integrity of the bracket. Note also that the analysis conservatively assumes through-wall cracking. TS TS

RAI 7

Appendix A of the TR indicated that for an assumed nominal flow, the flow-induced stress was calculated as 70 psi. Please describe the nature of this "flow induced stress" if it is not the hoop stress caused by the bounding internal pressure. Appendix A also suggested that the internal pressure for the leakage calculation was 79 psi, presumably at the assumed nominal flow and bounding pressure of 150 psi. Please explain how the 79 psi was derived from a "process diagram drawing for the plant."

BWRVIP Response to RAI 7

The 70 psi value represents the flow induced vibration stress. It was based on an assumed value of the acceleration. Since the analysis is provided merely as an example, its exact value is unimportant.

The process diagram is a schematic diagram showing major portions of the system including valves, pumps, orifices and strainers. The diagram indicates nodes at various points of interest. For each node, the process diagram specifies in tabular form, the different modes of operation, flow rates, pressures and temperature. The 79 psi value was taken from the process diagram at the point where the leakage calculation is performed. Again, since this is merely an example, the exact value is unimportant.

RAI 8

Appendix A of the TR presents thermal loads in a Table at the top of Page A-3. Please

(1) confirm whether the temperatures (for reactor pressure vessel (RPV), shroud, and shroud support legs), RPV pressure, displacements, and core spray pipe temperature in the table for each transient represent peak values during that transient.

(2) confirm that the only calculated values in the table are displacements.

(3) describe the nature of "the vessel thermal cycle drawing" and its relationship to the plant's design-basis transients.

BWRVIP Response to RAI 8

The values in the table on Page A-3 were based on a sample calculation for a specific plant and were intended only as an example.

1. The temperatures and pressures listed in the table represent typical values based on the plant design transients. They are not meant to be construed as peak values that bound the fleet. As stated in the response to RAI 7 above, since the analysis is presented as an example, the exact value used is unimportant.
2. The only calculated values are displacements.
3. The vessel thermal cycle drawing describes the design basis transients for an individual plant. These transients are used for the ASME Code stress analysis of the vessel components.

RAI 9

Appendix A of the TR presents sample calculated stresses for the core spray piping in a Table at the bottom of Page A-3. Please

(1) state the references for the loads (i.e., DW(P), CSIN(P), LOCAD(S), SSEIZ(P), SSEIY(P), and SSEDZ(S)) of the load combination in Note 1 of Page A-3 for the sample plant.

(2) confirm whether the loads are in the current licensing basis of the sample plant. Identify the loads which are not in the current licensing basis of the sample plant and have not been reviewed by the NRC.

(3) repeat the discussion requested in RAI 9 (2) for other load combinations mentioned in Section 4 of the TR.

BWRVIP Response to RAI 9

The calculated stresses shown in the table are the results of the example calculation presented in Appendix A. It must be understood that the example was presented merely to demonstrate the method of performing a calculation and that the results were not used to develop the inspection guideline listed elsewhere in BWRVIP-18, Rev. 2. As such, the pedigree and the precise values of the input parameters to the calculation are unimportant.

1. The values of the parameters DW(P), etc. were taken from the design basis for the plant used in the example calculation (a BWR/4).
2. It is not known whether the licensing basis for the plant has changed since the calculation was performed (circa 1995).
3. It is not known if the loads shown in Section 4 of the report are currently in the licensing basis for the sample plant. As described in the preamble to Section 4 of BWRVIP-18, Rev. 2, a plant must utilize the loads and load combination in its current FSAR when performing flaw evaluations. If specific loads and/or load combinations are not contained in the FSAR, then the load definitions and combinations listed in Section 4 shall be used. This methodology has been used in all of the BWRVIP Inspection Guidelines and has been universally accepted in previous Safety Evaluations.

RAI 10

Appendix A of the TR discussed leak rate calculation and assessment. The NRC staff has two questions:

(1) It states, "Leakage from other sources such as from the slip-fit joint between the sleeve and the core spray nozzle, any access hole cover repair, shroud, etc., should be included to obtain the total leakage...." The TR should consider providing guidance to estimate leakage from other sources if it is not readily available from the plant's operating record.

(2) Confirm that each plant has an existing "standard" loss-of-coolant accident (LOCA) analysis that would show the allowable reduction of the core spray flow that would still satisfy the peak cladding temperature limits. Otherwise, the TR should consider providing guidance for developing this LOCA analysis.

BWRVIP Response to RAI 10

1. The note is a reminder that plants are required to consider leakage from all sources when performing the integrated leakage evaluation that is required for compliance with the plant safety analysis. The leakage from locations such as slip-joints, drain holes, etc. are typically found in the plant safety analysis (e.g. LOCA analysis). Since methods for calculating leakage through cracks is not available in plant documentation, the BWRVIP provides such methods in each of the BWRVIP Inspection Guidelines. With this additional guidance plants have a complete methodology for assessing integrated leakage. No additional guidance is necessary.

2. Each plant has a LOCA-ECCS analysis that accounts for all leakage paths and demonstrates acceptable PCT limits. The analysis is governed by 10 CFR 50.46(b) and 10 CFR 50 Appendix K. Additional BWRVIP guidance is not judged to be necessary due to other regulatory governance.

RAI 11

Appendix A of the TR presents sample calculated stresses for the sparger in a Table at the top of Page A-5. Please

(1) explain why the sparger geometry will make the thermal bending stress become zero at the Tee-Box location. Does the zero thermal bending stress occur at the Tee-Box location only? Or, at all locations of the sparger?

(2) confirm whether the sparger loads used in this example are in the current licensing basis of the sample plant. If the sparger loads are not in the current licensing basis of the sample plant and have not been reviewed by the NRC, provide additional information regarding the sparger loads so that, if accepted by the NRC, other plants can follow the same approach to define sparger loads for their plants.

BWRVIP Response to RAI 11

1. As with other example calculations in the Appendix, the exact values used in the sample analyses are unimportant. The thermal bending stress in the region of the tee-box was likely considered to be zero because the tee box is not restrained by any support. The sparger is free to expand in the circumferential direction and is only constrained by the supports in the radial and axial direction. The thermal bending stress at other locations on the sparger may not be zero.

2. See response to RAI 9.

Inspection Requirements:

The BWRVIP-251, "Technical Bases for Revision of the BWRVIP-18 Core Spray Inspection Program," report dated May 2012, was used by the BWRVIP as technical bases for revising the inspection criteria for the core spray and sparger systems. The BWRVIP-251 report was submitted to the NRC staff for information only, and it is a critical report that provided justification for the reduction in inspection requirements for the core spray and sparger system. The NRC staff reviewed the BWRVIP-251 report and developed the following RAI questions on the technical justification for the revised inspection requirements for the core spray and sparger systems.

RAI 12

Identify which of the welds listed in the BWRVIP-251 report- Tables 3-2, 3-3, 3-4, 3-5, 3-8, and 3-9 and, sparger bracket welds are most susceptible to intergranular stress corrosion cracking due to: (a) high stress and (b) effect of cold work. Since cold work cannot be quantified, the NRC staff requests that BWRVIP, in its evaluation, should take into account the possibility that the welds were previously cold worked during the original fabrication. How many of the welds in a given category can have through wall leak without compromising the: (a) safety consequence specifically, safe shutdown mode; and, (b) functionality of the system.

BWRVIP Response to RAI 12

None of the welds in the core spray system are subject to significant applied stresses. The majority of the stress that causes crack initiation and growth is related to residual stress which is similar in magnitude for all welds in the system. The BWRVIP assumes that the residual stress at all of the major piping and sparger welds will be high enough to initiate a crack; consequently, all welds in the system are considered in the inspection program.

As described by the staff, cold work is difficult to quantify and varies from plant to plant and weld to weld. It is somewhat dependent on the methods of fabrication that were in use at the time the plant was constructed. The BWRVIP agrees that some degree of cold work could be present at all welds and, as described above, all welds are considered in the inspection program.

While it is difficult to predict which welds will ultimately be most susceptible to cracking, the inspection data described in BWRVIP-251 shows that the majority of the core spray pipe weld cracking incidents reported are associated with weld locations P3, P5, P8a, and P8b. BWRVIP-18 Revision 2 requires that these welds be reinspected frequently.

It is difficult to state generically how many of the welds can have through wall leaks without compromising safety consequences or functionality. Each plant maintains a number of plant-specific input assumptions for their LOCA analysis including assumptions for the delivery and distribution of the core spray system. For each plant, the allowable number of cracks depends on the total length of the through-wall cracking, which may be associated with a single weld or with more than one weld. For this reason, BWRVIP-18, Rev. 2 requires that a plant specific leakage

assessment that considers all identified cracks be performed using the methodology described in Section 5 of the Guideline. The calculated leakage is compared to the plant specific design margin to ensure that the leakage is not sufficient to challenge safety or functionality.

RAI 13

Provide information on establishing the timing of the first inspection that is to be performed on any repair or replaced weld in the core spray and sparger systems.

BWRVIP Response to RAI 13

Inspection of repaired component is addressed in the Internal Core Spray Piping and Sparger Replacement Design Criteria (BWRVIP-16) and the Internal Core Spray Piping and Sparger Repair Design Criteria (BWRVIP-19). These documents state:

> Inspections required for the entire repaired internal core spray piping and spargers assembly for the remaining life of the unit, shall be specified commensurate with design considerations and Code requirements applicable to the specific design. These inspections shall be consistent with the requirements and scope of BWRVIP-18 [9].

Definition of inspections for repaired components is left to the discretion of the repair designer but must be consistent with the methodology of BWRVIP-18, Rev. 2. This approach was taken by the BWRVIP for core spray as well as for most other repaired components because it is impossible for the BWRVIP to envision all manner of repairs that might be developed in the future. As such, the designer is in the best position to develop appropriate inspections that take into account relevant aspects of the design, materials utilized, estimated operational stresses, etc.

RAI 14

The BWRVIP is requested to confirm whether the following attributes were included in developing the technical bases for the proposed inspection criteria addressed in the BWRVIP-251 report, and the attributes are: (a) functionality of core spray and sparger systems; (b) consequence of failure consideration i.e., safe shutdown mode, and, (c) minimum number of flawed welds (with 70 percent or higher through wall cracks) that the core spray and or sparger can sustain during normal operation or during seismic and LOCA conditions.

BWRVIP Response to RAI 14

The functionality, consequence of failure and allowable number of flawed welds are implicitly addressed by the report. The basis for the inspection guidance is not probabilistically based. Rather, it is based on an inspection approach that assures that all welds are inspected frequently enough to ensure that the combined cracking in the core spray system is maintained low enough to ensure that the original function of the system is maintained.

RAI 15

The past inspections of the majority of the subject core spray line and sparger welds addressed in Appendix A to BWRVIP-251 indicate that the area of inspection coverage is less than 50 percent of the surface area of the welds. Please describe how this lack of coverage affects the overall functionality of the core spray system assuming the uninspected lengths of all original welds (both creviced and non-creviced) are flawed in the core spray line and sparger components.

BWRVIP Response to RAI 15

While the inspection coverage for some welds is less than 50 percent, the summary figures in Section 3.5 show that the coverage for many other welds is as high as 100-percent. This condition is typical of inspection results for many components in a plant. If the entire uninspected region of all welds was indeed flawed, a serious challenge to the functionality of the system might be predicted. However, the methodology adopted by the BWRVIP (as well as the ASME Code) does not require that the uninspected region of all components be considered flawed. Rather, it assumes that the inspected region provides a fair representation of the condition of the uninspected region. If the inspected region is flawed, the methodology conservatively requires that prescriptive assumptions be made regarding the existence of flaws in the uninspected region and that those flaws be considered in structural and leakage evaluations. If the inspected region is unflawed, the methodology does not require the assumption of flaws in the uninspected region.

RAI 16

On June 8, 2009, General Electric (GE)- Hitachi issued Safety Communication (SC) 09-01, "Annulus Pressurization Loads Evaluation," related to Annulus Pressurization (AP) loads, also referenced as "New Loads," and the corresponding stresses on the RPV, internals, and containment structures. SC 09-01 identifies that "...the AP loads used as input for design adequacy evaluations of NSSS [nuclear steam supply system] safety related components for "New Loads" plants might have resulted in non-conservative evaluations." The NRC also recently became aware of three other related GE SCs, namely SC 09-03, Revision 1 related to core shroud recirculation line break loads, SC 11-07 related to a new load combination, and SC 12-20 related to acoustic load errors, all of which were issued on June 10, 2013. With respect to the issues related to AP loads and these four SCs, the licensee is requested to provide the following information.

> The NRC is aware of some plant-specific re-evaluations of New Loads performed that increased the AP loads acting on the core spray piping and sparger components. The NRC staff requests that the BWRVIP address whether the AP loads and associated calculations included in BWRVIP-18, Revision 2, properly reflect the correct hydrodynamic loads in response to SC 09-01.

BWRVIP Response to RAI 16

The BWRVIP is aware of the numerous GEH Safety Communications and understands that they may have an effect on one or more of the BWRVIP Guidelines. The potential impact on BWRVIP-18 Revision 2 would be a revision of the flaw analysis method contained in Section 4. However, the inspection requirements, which are not based fundamentally on flaw tolerance, would not be impacted. As such, the BWRVIP proposes that no changes be made to BWRVIP-18 Revision 2 at this time. Note that the BWRVIP is currently evaluating the impact of the SCs on all of the BWRVIP Inspection Guidelines and will issue revised guidance where deemed necessary.