October 26, 2000

Carl Terry, BWRVIP Chairman
Niagara Mohawk Power Company
Post Office Box 63
Lycoming, NY 13093

SUBJECT: SAFETY EVALUATION OF THE "BWRVIP VESSEL AND INTERNALS PROJECT, STANDBY LIQUID CONTROL LINE REPAIR DESIGN CRITERIA (BWRVIP-53)," EPRI REPORT TR-108716, JULY 1998 (TAC NO. MA2328)

Dear Mr. Terry:

The NRC staff has completed its review of the Electric Power Research Institute (EPRI) proprietary report TR-108716, "BWR Vessel and Internals Project, Standby Liquid Control Line Repair Design Criteria (BWRVIP-53)," dated July 1998. Both proprietary and non-proprietary versions of the BWRVIP-53 report were submitted to the U. S. Nuclear Regulatory Commission (NRC) for staff review by letter dated July 2, 1998, and an expanded non-proprietary version of the BWRVIP-53 report was submitted by letter dated March 24, 2000. The BWRVIP-53 report was supplemented by a letter dated December 6, 1999, which was in response to the NRC staff's request for additional information (RAI), dated April 7, 1999. The BWRVIP-53 report provides general design acceptance criteria for the temporary and permanent repairs of the standby liquid control and core differential pressure (SLC and CDP) nozzles and internal lines. These guidelines are intended to maintain the structural integrity of the SLC and CDP nozzles and internal lines during normal operation and under postulated transient and design basis accident conditions. BWRVIP provided the BWRVIP-53 report to support generic regulatory efforts related to the repair of BWR standby liquid control lines.

The NRC staff has reviewed the BWRVIP-53 report, as well as its associated RAI response, and finds that the BWRVIP-53 report, as modified and clarified to incorporate the staff's comments in the enclosed safety evaluation (SE), is acceptable for providing guidance for permanent repairs of the cracked or leaking SLC and CDP nozzles and cracked or broken SCL and CDP lines inside the reactor vessel. Therefore, the staff concludes that licensee implementation of the guidelines in the BWRVIP-53 report, as modified, will provide an acceptable repair design criteria of the safety-related components addressed in the BWRVIP-53 document. The staff will consider the guidelines provided in the BWRVIP-53 report for temporary repairs as a part of its determination of an alternative for plant-specific temporary repairs pursuant to 10 CFR 50.55a(a)(3).

The BWRVIP-53 report is considered by the staff to be acceptable for licensee usage, as modified and approved by the staff, at any time during either the current operating term or during the extended license period. The modifications stated in the RAI and addressed in the enclosed SE should be incorporated in a revision of the BWRVIP-53 report.

Please contact C. E. (Gene) Carpenter, Jr., of my staff at (301) 415-2169 if you have any further questions regarding this subject.

Sincerely

/ra RHWessman f/

Jack R. Strosnider, Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: See next page

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U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

SAFETY EVALUATION OF THE "BWRVIP VESSEL AND INTERNALS PROJECT, STANDBY LIQUID CONTROL LINE REPAIR DESIGN CRITERIA (BWRVIP-53)," EPRI REPORT TR-108716

1.0 INTRODUCTION

1.1 Background

By letter dated July 2, 1998, as supplemented by letter dated December 6, 1999, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the Electric Power Research Institute (EPRI) proprietary Report TR-108716, "BWR Vessel and Internals Project, Standby Liquid Control Line Repair Design Criteria (BWRVIP-53)," dated July 1998, for U. S. Nuclear Regulatory Commission (NRC) staff review. The supplemental information was in response to the staff's request for additional information (RAI), dated April 7, 1999. Both proprietary and non-proprietary versions of the BWRVIP-53 report were submitted for NRC staff review by letter dated July 2, 1998, and an expanded non-proprietary version of the BWRVIP-53 report was submitted by letter dated March 24, 2000.

The BWRVIP-53 report provides general repair criteria for the temporary and permanent repair of the standby liquid control and core differential pressure (SLC and CDP) nozzles and internal lines. These guidelines are intended to maintain the structual integrity of the SLC and CDP nozzle and internal lines during normal operation and under postulated transient and design-basis accident conditions. The BWRVIP provided the BWRVIP-53 report to support generic regulatory efforts related to the repair of BWR SLC and CDP nozzles and internal lines.

1.2. Purpose

The staff reviewed the BWRVIP-53 report, as supplemented, to determine whether its proposed guidance, along with the response to the staff's RAI, will provide an acceptable repair design criteria of the subject safety-related reactor pressure vessel (RPV) internal components. The review assessed the design objectives, structural evaluation, system evaluation, materials, fabrication and installation considerations, as well as the required inspection and testing requirements.

1.3. Organization of this Report

Because the BWRVIP report is proprietary, this SE was written not to repeat information contained in the report. The staff does not discuss in any detail the provisions of the guidelines nor the parts of the guidelines it finds acceptable. A brief summary of the contents of the BWRVIP-53 report is given in Section 2 of this SE, with the evaluation presented in Section 3. The conclusions are summarized in Section 4. The presentation of the evaluation is structured according to the organization of the BWRVIP-53 report.

ENCLOSURE

2.0 SUMMARY OF BWRVIP-53 REPORT

The BWRVIP-53 report addresses the following topics in the following order:

- Component Description and Safety Function - The SLC and CDP nozzles and internal line configurations are described in detail with brief descriptions of each configuration's function and characteristics. Differences among the various models of BWRs (BWR/2, BWR/3-5 and BWR/6) are identified. The safety design bases for the SLC and CDP nozzles are to: (1) support the SLC system delivery of sodium pentaborate solution to the lower plenum whenever required, and (2) to provide a portion of the primary pressure boundary. The specific safety design features are subsequently addressed. An event analysis is also provided for various operational conditions to ensure the component safety functions are maintained.
- Scope of Repairs - The SLC and CDP nozzles and internal lines' susceptibility to intergranular stress corrosion cracking (IGSCC) is briefly discussed. Appendices A, B and C further address repair concepts of the subject safety-related components as well as alternate operational approaches in case of impairment of the core plate ΔP (differential pressure) signal.
- Design Objectives - The following design objectives are presented and briefly discussed: design life, safety design bases, safety analysis events, structural integrity, retained flaw(s), loose parts considerations, and physical interfaces with other reactor internals. Two features of component repair also considered in order to minimize in-vessel time for installations were vessel drain down (in order to support repair of the SLC and CDP nozzles and internal lines without draining the vessel) and repair accessibility.
- Design Criteria - The design criteria of the SLC and CDP nozzles and internal components are presented. In summary, all repair designs should meet the individual plant safety analysis report (SAR) as well as NRC and ASME Code established methodology for reactor pressure vessel (RPV) and internals mechanical design.
- Structural and Design Evaluation - Terms (e.g., hydraulic loads, fuel lift loads, etc.) associated with applied loads on the reactor vessel are briefly discussed. The various events and operational service level conditions are also considered to ensure the repairs do not inhibit safety and operational functions of the internal components. Other structural and design topics addressed are: load combinations, functional evaluation criteria, allowable stresses, flow induced vibration, repair impact on existing internal components, radiation effects on repair design, analysis codes, thermal cycles, and corrosion allowance.
- System Evaluation - The following system evaluations, performed in support of the SLC and CDP nozzles and internals repairs and modifications, are briefly discussed: SLC solution distribution, loss of the above core plate pressure reading, and power uprate.
- Materials, Fabrication and Installation - The materials specifications are given along with the regulatory requirements pertaining to austenitic stainless steel alloys. Welding and fabrication guidelines are also discussed. Installation considerations include indicating the as-built dimensional tolerance the repair can accommodate as well as controls to avoid in-vessel debris generation. Reducing radiation exposure using ALARA practices

and qualification of critical design parameters (e.g., preload in tensioned members, critical tolerances) was presented.

- Inspection and Testing - Inspection and testing of the reactor internal components are addressed in the following topics: inspection access, pre- and post-installation inspection, system hydrostatic test, flow test, and instrumentation checks.

3.0 STAFF EVALUATION

The BWRVIP-53 report provides the general design acceptance criteria for permanent and temporary repairs of the BWR standby liquid control and core differential pressure (SLC and CDP) nozzles and internal lines. While it does not present specific designs to effect repairs of the subject safety-related BWR internal components, it does present a methodology for BWR licensees to follow in designing repairs which maintain the structural integrity and system functionality of the SLC and CDP nozzles and internal lines during normal operation and under postulated transient and design basis accident conditions for the specified service life of the components.

The SLC system is designed to shut down a reactor from full power by injecting sodium pentaborate, a neutron absorber, into the reactor core. This is done when the normal method of controlling core reactivity with control rods cannot be accomplished. In most BWR/2 through BWR/6's, a line from the SLC and CDP nozzles in the vessel bottom head supplies liquid sodium pentaborate solution to a standpipe or sparger inside the RPV. The standpipe or sparger then distributes the liquid through holes to the coolant entering the core.

However, five plants (i.e., Hope Creek, Limerick Units 1 and 2, Nine Mile Point Unit 2, and Perry) are configured such that the sodium pentaborate solution is injected through the core spray piping. For these plants, the BWRVIP-53 report states that the repair criteria are not applicable to the repair of the standby liquid control hardware used for injection through the core spray piping. The repair criteria does, however, apply to the repair of the SLC and CDP nozzles and internal hardware in the lower plenum of these five plants.

The NRC staff has reviewed the BWRVIP-53 report and the associated response to the staff's RAI, and finds that the BWRVIP-53 report, as modified and clarified to incorporate the staff's comments, below, is acceptable for providing guidance for permanent repairs of the SLC and CDP nozzles and internal lines. Based on the following evaluation, the staff concludes that licensee implementation of the guidelines in the BWRVIP-53 report, as so modified, will provide an acceptable repair design criteria of the safety-related components addressed in the BWRVIP-53 report for licensee usage. The BWRVIP-53 report is considered by the staff to be acceptable for licensee usage, as modified and approved by the staff, at any time during either the current operating term or during the extended license period. The modifications stated in the RAI and addressed below should be incorporated in a revision of the BWRVIP-53 report. Inspections of the repaired components should be in accordance with the BWRVIP-27 guidance, as approved by the staff.

The staff will consider the guidelines provided in the BWRVIP-53 report for temporary repairs as a part of its determination of an alternative for plant-specific temporary repairs pursuant to 10 CFR 50.55a(a)(3).

3.1 BWRVIP Response to Staff's RAI

The staff's April 7, 1999, RAI, provided eleven open items. The BWRVIP, in its letter of April 30, 1999, addressed these items, which are discussed below.

RAI Item 1: In Section 3.2.1.2, "Alternative SLC and CDP Nozzles Safety Basis," the BWRVIP-53 report discusses roll expansion repairs of the SLC and CDP nozzles. As this repair was previously denied in the staff's review of the BWRVIP-17 report, the staff will not accept roll expansion as a permanent repair method.

BWRVIP Response to Item 1: It is recognized that the Staff will not accept roll expansion as a permanent repair. However, in some instances, roll expansion may be appropriate for a temporary repair. Section 3.2.1.2 will be modified to clarify that roll expansion can be considered only for a temporary repair.

Staff's Evaluation of BWRVIP Response to Item 1: As was previously stated in the staff's SE of EPRI's November 1996 proprietary report TR-106712, "Review of 'BWR Vessel and Internals Project, Roll/Expansion Repair of Control Rod Drive and In-core Instrument Penetrations in BWR Vessels (BWRVIP-17),'" dated March 13, 1998, the staff determined that roll expansion is not a structural repair but is only leak limiting. Further, the staff found that the corrective action intended by the ASME Code requirements, upon discovery of a flaw in a Class 1 pressure retaining boundary component, is to either repair the flaw or replace the flawed component in order to return it to a condition of Code compliance. An ASME Code-acceptable repair of a crack in a control rod drive (CRD) stub tube or in-core penetration would require a weld repair. Although the roll/expansion method may for some time period control the symptom of the flaw (leakage), it does not reestablish structural integrity by repairing or replacing the degraded item consistent with 10 CFR 50.55a. Therefore, the NRC staff determined that the BWRVIP-17 report did not provide a sufficient generic technical basis and criteria for performing a non-Code repair to an ASME Code component to warrant a generic alternative to the ASME Code. The same reasoning holds for roll expansion of the SLC and CDP nozzles.

However, as described above, roll expansion can control leakage for some short time. A licensee may utilize the BWRVIP-53 report as part of the technical basis for a plant-specific request for an alternative repair per 10 CFR 50.55a(a)(3) to utilize roll expansion to temporarily repair the SLC and CDP nozzles for no more than one (1) operating cycle. The request will be reviewed by the staff on a plant-specific basis prior to its implementation.

RAI Item 2: In Section 6, "Design Criteria," the staff requests that the paragraph be modified to state: "...use of ASME Code Editions and Addenda, ASME Code Cases, or ASTM standards or specifications not yet specifically approved by the regulatory authorities will need to be evaluated by the regulator on a plant-specific case by case basis."

BWRVIP Response to Item 2: (See combined response below)

RAI Item 3: In Section 6.2, "SLC and CDP Internal Component," the staff recommends that the first sentence in the fourth bullet be modified to state: "Materials shall meet the criteria

and methods in ASME Section II specifications, or in ASME Code Cases, ASTM specifications, or other material specifications that have been previously approved by the regulatory authorities, or else be submitted on a case by case basis to the regulatory authorities for approval. EPRI Document #84-MG-18, 'Nuclear Grade Stainless Steel Procurement, Manufacturing and Fabrication Guidelines,' and EPRI Document NP-7032, 'Material Specification for Alloy X-750 for Use in LWR Internal Components,' are examples of other, non-ASME, non-ASTM specifications that have been previously approved by the regulatory authorities for use in BWR reactor pressure vessel and internals repair applications."

BWRVIP Response to Item 3: (See combined response below)

RAI Item 4: Section 9.1, "Materials"

RAI Item 4-1: The staff recommends that Section 9.1.2 be modified to state: "Materials shall be manufactured in accordance with ASME Section III, Class 1 (Subsection NB-2000), and shall meet the criteria and methods in applicable ASME Section II specifications, or in ASME Code Cases, ASTM specifications, or other material specifications that have been previously approved for use by the regulatory authorities. EPRI Document #84-MG-18, 'Nuclear Grade Stainless Steel Procurement, Manufacturing and Fabrication Guidelines,' and EPRI Document NP-7032, 'Material Specification for Alloy X-750 for Use in LWR Internal Components,' are examples of other, non-ASME, non-ASTM specifications that have been previously approved by the regulatory authorities for use in BWR reactor pressure vessel and internals repair applications. Design stress intensity and allowable stress values shall be established for the limiting design conditions consistent with the methodology of ASME Section III, Appendix III."

BWRVIP Response to Item 4-1: (See combined response below)

RAI Item 4-2: The staff finds Section 9.1.5 to be generally acceptable; however, the staff encourages licensees to set limits on the amount of cobalt in Alloy X-750 materials in order to limit the levels of cobalt-60 induced radiation fields.

BWRVIP Response to Item 4-2: The allowable cobalt level for individual heats of alloy X-750 will be specified as 0.25% maximum. If this limit is exceeded, an alternative evaluation protocol which can be implemented by the licensee will be provided. The alternative criteria will be a maximum allowable weighted average cobalt level of 0.25%, taking into account the surface area of all newly installed components wetted by reactor coolant.

Staff's Evaluation of BWRVIP Response to Item 4-2: as was stated in the staff's final SE on the BWRVIP-16 and -19 reports, dated August 10, 2000, regarding the BWRVIP's response on allowable cobalt levels for individual heats of alloy X-750, this criterion does not meet the requirements of NP-7032 for cobalt limits. However, in a June 27, 2000, e-mail from K. Wolfe of EPRI to C. E. Carpenter of NRC, Mr. Wolfe states:

> "The older number [i.e., 0.015 percent, which is the current limit for cobalt found in NP-7032] was based on manufacturers estimates of the lowest cobalt that could be maintained in new melts of material. It was included in the [specification] for the

> purpose of overall cobalt reduction (i.e., it was not a metallurgical consideration) although its basis was not directly related to that end. The revised number is also included in the [specification] for purposes of overall cobalt reduction. It is based on realistic consideration of the cobalt level necessary to limit radiation from cobalt. Given the small amount of X-750 typically used in a repair and the fact that it is not typically used in an abrasive application (i.e., one in which a significant amount of material could find its way into the reactor water and migrate through the system), the revised, higher number is appropriate."

With this new information on cobalt limits, the staff finds this section acceptable.

RAI Item 4-3: Section 9.1.8 changes

The staff recommends deleting Section 9.1.8, by making the following changes:

RAI Item 4-3-1: Incorporate into Section 9.1.2: "Materials which are not covered by the scope of ASME Section III Material Requirements or the original Design Code of Record will be considered by the regulatory authorities as alternatives to the Code and will need to be submitted for review on a case-by-case basis if they have not been previously approved for use by the regulatory authorities. Thus, materials manufactured in accordance with unapproved ASME Code Cases, ASTM specifications or material specifications may also be used provided that they are approved on a case by case basis by the regulatory authorities, and provided that they have been demonstrated to be acceptable for use in BWR environments."

BWRVIP Response to Item 4-3-1: (See combined response below)

RAI Item 4-3-2: Move the Note in Section 9.1.8 as an added Note to the end of Section 9.1.6.

BWRVIP Response to Item 4-3-2: (See combined response below)

RAI Item 5: In Section 12, "Documentation," the staff recommends that the third bullet be amended to add a statement: "Use of unapproved ASME Code editions, ASME Code Cases, ASTM standards, or other materials, design, or fabrication related specifications will need to be submitted for approval by the appropriate regulatory authorities."

BWRVIP Response to Items 2, 3, 4-1, 4-3-1, 4-3-2 and 5: It is recognized by the BWRVIP members that repair and replacement designs for plants with internals which were designed and constructed in accordance with ASME Section III must meet the rules of ASME Section XI. Section XI requires that repairs or replacements meet the applicable requirements of ASME Section III and the Owner's Original Design Specification. This would include the applicable Code materials requirements. If the Code is not met, a relief request to allow a technical alternative to the Code pursuant to 10 CFR 50.55a must be requested.

Section XI rules for repair and replacement also applies to components that were not designed to Section III, but are classified by the Owner as "Welded Core Support

Structures" and are subject to inspection under Section XI Category B-N-1 from Table IWB-2500-1. These components are to be repaired or replaced in accordance with the Owner's original Design Specification and Construction Code. NRC allows later approved versions of Section III to be used. If this requirement is not met, approval of a technical alternative must be sought pursuant to 10 CFR 50.55a.

Repair and replacement designs for plants which were not designed and constructed in accordance with ASME Section III (and components not subject to Section XI) must meet the individual plant SAR and other plant commitments for RPV internals mechanical design, as stated in Section 6. In that instance, materials must meet the requirements of ASME Section III specifications, ASME Code Cases, ASTM specifications, or other material specifications that have been previously approved by the regulatory authorities. This would include material specifications/criteria submitted by the BWRVIP and approved by NRC. Otherwise, it is recognized that a repair or replacement design that uses a material not meeting these criteria must be submitted to the regulatory authorities for approval, on a plant specific basis.

Staff's Evaluation of BWRVIP Response to Items 2, 3, 4-1, 4-3-1, 4-3-2 and 5: The staff finds that, based on the above combined response, the BWRVIP adequately addressed these items.

RAI Item 6: Appendix C, "Repair Concepts For SLC and CDP Nozzles,"

RAI Item 6.1: Section C.2.1.2, "Japanese Owners Group In-core Repair," describes the major steps for this repair. The staff requests either a separate topical report or an additional appendix to this report to support approval of the Japanese welding repair methods.

RAI Item 6.2: Section C.2.3, "Non-Structural Thermal Spray Leakage Barriers," discusses a repair technique that ASME does not currently recognize or approve. As such, the staff requests that the use of this method as a repair technique be considered only on a case-by-case basis.

RAI Item 6.3: Section C.2.5, "Mechanical Seals." Pursuant to 10 CFR 50.55a(a)(3)(i), the staff has approved Combustion Engineering (CE) designed mechanical nozzle seal assemblies as an alternative repair technique for leaks in ASME Code Class 1 nozzles in the hot legs, pressurizers, and steam generators of domestic PWRs. However, if this repair technique is to be proposed for repairs of SLC and CDP line nozzles, additional detailed analyses would need to be submitted by BWRVIP.

BWRVIP Response to Item 6: The repair concepts discussed in Appendix C are intended to provide the repair designer with a number of potential repair approaches. They were not included in the report for the purpose of obtaining NRC acceptance or approval. The introduction to Appendix C will be clarified to indicate that the potential repair approaches have not necessarily been accepted by the NRC.

Staff's Evaluation of BWRVIP Response to Item 6: Appendix C should state that it is for "Information Only." Use of the repair methods and concepts described therein will need to be submitted for staff approval as an alternative to 10 CFR 50.55a on a case basis.

RAI Item 7: Section 1.1, Background, states that "there has only been one report of cracking in any BWR SLC and CDP nozzles or internals." This refers to the 1965 failure of a SLC sparger in an overseas BWR/1. Clarification to this statement should be made based on the findings at Big Rock Point (Report dated April 24, 1998). Big Rock Point found the discharge piping of the SLC line severed during decommissioning.

BWRVIP Response to Item 7: The discussion of previous failures was intended only as an introductory remark and does not have substantial bearing on the remainder of the document. However, if the April 24, 1998 report is provided, the BWRVIP will review this new information and determine if there is sufficient basis to identify this in the document as a second cracking incident.

Staff's Evaluation of BWRVIP Response to Item 7: The information on cracking of the SLC discharge piping at the Big Rock Point site is publicly available. BWRVIP is requested to identify this incident and evaluate its significance in a revision to the BWRVIP-53 report.

RAI Item 8: Section 3.2.2, SLC and CDP Internal Lines Safety Bases, states that the original design bases of the SLC and CDP internals included the distribution of sodium pentaborate to the lower plenum. However, it appears that BWRVIP-53 is concluding that this function is no longer required based on the BWRVIP-06, "BWR Vessel and Internals Project: Safety Assessment of BWR Reactor Internals," document. The purpose of the BWRVIP-06 document was to demonstrate that a safe shutdown could be accomplished if a failure in a BWR internal occurred, not to change the design bases of the components. Additionally, the staff does not believe that the discussion in the BWRVIP-06 document supports the statement in the BWRVIP-53 conclusion that distribution of sodium pentaborate to the lower plenum is no longer required. Provide further clarification as to why the design bases of the SLC and CDP internal lines should be changed and the review process of that change.

BWRVIP Response to Item 8: The reference to BWRVIP-06 was incorrect. The reference should have been to BWRVIP-27 (SLC Inspection and Flaw Evaluation Guidelines). Section 2.2 in BWRVIP-27 discusses boron mixing under various postulated accident conditions and power levels. The technical basis of the BWRVIP-27 discussion of mixing is documented in Reference 3 of BWRVIP-27 ("Summary of BWR Boron Remixing" GENE-A00-05652-03, Prepared for the BWR Owners' Group, February 1996, GE proprietary). This discussion concludes that there is no boron mixing concern associated with potentially large cracks in the ΔP/Standby Liquid Control (P/SLC) hardware internal to the reactor.

Staff's Evaluation of BWRVIP Response to Item 8: The staff finds that BWRVIP's response adequately addressed this item.

RAI Item 9: In Section 8, System Evaluation, clarifications should be made in the BWRVIP-53 report so that individual licensees will make plant-specific submittals for the following instances:

- Licensees whose repair design to the SLC and CDP internals eliminate the distribution function of the original plant design.

- Licensees whose repair design of the SLC and CDP internals eliminate the above core plate pressure sensing function of the original plant design.

BWRVIP Response to Item 9: The BWRVIP agrees with the comment and the document will be revised.

Staff's Evaluation of BWRVIP Response to Item 9: The staff finds that BWRVIP's response adequately addressed this item.

RAI Item 10: Appendix A.1, "Abandoning in Place," states that "...liquid control injection into the lower head, without the liquid control piping and sparger, is acceptable; a review of plant-specific analyses would be appropriate to document this." The staff disagrees with this statement. The staff does not believe that this has been shown to be acceptable. The BWRVIP-53 document should provide guidance on the plant-specific analysis that could be done to show acceptable results on a plant-specific basis.

BWRVIP Response to Item 10: The primary basis for "Abandoning in Place" as a repair option is the testing and analysis of boron mixing as referenced in the response to Item 8 above. Plant-specific analysis would be required to address other concerns such as potential vibration and loose parts from SLC and CDP internal components known to be cracked.

Staff's Evaluation of BWRVIP Response to Item 10: The staff finds that the BWRVIP-53 report should be modified to address the staff's RAI comments to Item 10.

RAI Item 11: Provide the frequency of periodic flushing of the SLC internal line which could be used to determine if the line is pinched or not.

BWRVIP Response to Item 11: For repair concepts which could potentially result in some internal lines in a 'pinched off' condition, a flushing test could be developed and applied at each refueling outage. This would verify that no hydraulically significant change has occurred since the outage when the repair was first implemented. It is anticipated that the frequency of flushing could be reduced over time based on experience with that specific repair.

Staff's Evaluation of BWRVIP Response to Item 11: The staff finds that the BWRVIP-53 report should be modified to address the staff's RAI comments to Item 11. General guidelines should be developed by the BWRVIP to address line blockage, as described above.

4.0 CONCLUSION

The NRC staff has reviewed the BWRVIP-53 report, as well as its associated RAI response, and finds that the BWRVIP-53 report, as modified and clarified to incorporate the staff's comments above, is acceptable for providing guidance for permanent repairs of the cracked or leaking SLC and CDP nozzles and cracked or broken SCL and CDP lines inside the reactor vessel. Therefore, the staff concludes that licensee implementation of the guidelines in the BWRVIP-53 report, as modified, will provide an acceptable repair design criteria of the safety-related components addressed in the BWRVIP-53 document. The staff will consider the guidelines provided in the BWRVIP-53 report for temporary repairs as a part of its

determination of an alternative for plant-specific temporary repairs pursuant to 10 CFR 50.55a(a)(3).

The BWRVIP-53 report is considered by the staff to be acceptable for licensee usage, as modified and approved by the staff, at any time during either the current operating term or during the extended license period. The modifications stated in the RAI and addressed above should be incorporated in a revision of the BWRVIP-53 report.

5.0 REFERENCES

1. Carl Terry, BWRVIP, to USNRC, “BWR Vessel and Internals Project, Standby Liquid Control Line Repair Design Criteria (BWRVIP-53),” EPRI Report TR-108716, dated July 1998.

2. C. E. Carpenter, USNRC, to Carl Terry, BWRVIP, “Propriety Request for Additional Information - Review of BWR Vessel and Internals Project Report, Standby Liquid Control Line Repair Design Criteria (BWRVIP-53),” dated April 7, 1999.

3. Carl Terry, BWRVIP, to USNRC, “BWRVIP Response to NRC Request for Additional Information on BWRVIP-53,” December 6, 1999.