September 21, 2001

Carl Terry, BWRVIP Chairman
Niagara Mohawk Power Company
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SUBJECT: SAFETY EVALUATION OF THE "BWRVIP VESSEL AND INTERNALS PROJECT, LOWER PLENUM REPAIR DESIGN CRITERIA (BWRVIP-55)," EPRI REPORT TR-108719, SEPTEMBER 1998 (TAC NO. MA3673)

Dear Mr. Terry:

The NRC staff has completed its review of the Electric Power Research Institute (EPRI) proprietary report TR-108719, "BWR Vessel and Internals Project, Lower Plenum Repair Design Criteria (BWRVIP-55)," dated September 1998. Both proprietary and non-proprietary versions of the BWRVIP-55 report were submitted to the U. S. Nuclear Regulatory Commission (NRC) for staff review by letter dated September 22, 1998. This report was supplemented by a letter dated December 6, 1999, which was in response to the staff's request for additional information (RAI), dated August 13, 1999. The BWRVIP-55 report provides general design acceptance criteria for the temporary and permanent repairs of cracked or leaking internal components in the reactor vessel lower plenum area. These guidelines are intended to maintain the structural integrity of the internal components in the reactor vessel lower plenum area during normal operation and under postulated transient and design basis accident conditions. The BWRVIP provided the BWRVIP-55 report to support generic regulatory efforts related to the repair of BWR internal components in the reactor vessel lower plenum area.

The staff has reviewed the BWRVIP-55 report, as well as its associated RAI response, and found, in the enclosed safety evaluation (SE), that the BWRVIP-55 report is acceptable for providing guidance for permanent or temporary repairs of cracked or leaking internal components in the reactor vessel lower plenum area. This finding, based upon the information submitted by the above cited letters, is consistent with NRC approved methodology. The staff has concluded that licensee implementation of the guidelines in BWRVIP-55 will provide an acceptable repair design criteria of the safety-related components addressed in the BWRVIP-55 document.

The BWRVIP-55 report is considered by the staff to be applicable 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.

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/

William H. Bateman, Chief
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Enclosure: As stated

cc: BWRVIP Service List

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

Sincerely

Jack R. Strosnider, Director
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U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

SAFETY EVALUATION OF THE “BWRVIP VESSEL AND INTERNALS PROJECT, LOWER PLENUM REPAIR DESIGN CRITERIA (BWRVIP-55),” EPRI REPORT TR-108719

1.0 INTRODUCTION

1.1 Background

By letter dated September 22, 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-108719, “BWR Vessel and Internals Project, Lower Plenum Repair Design Criteria (BWRVIP-55),” dated September 1998 for NRC staff review. The supplemental information was in response to the staff’s request for additional information (RAI), dated August 13, 1999. The BWRVIP-55 report provides general repair criteria for the temporary and permanent repair of cracked or leaking internal components in the reactor vessel lower plenum area. These guidelines are intended to maintain the structual integrity of the internal components in the reactor vessel lower plenum area during normal operation and under postulated transient and design-basis accident conditions. The BWRVIP provided the BWRVIP-55 report to support generic regulatory efforts related to the repair of BWR internal components in the reactor vessel lower plenum area.

1.2. Purpose

The staff reviewed the BWRVIP-55 report, as supplemented, to determine whether its proposed guidance adequately addressed the open items in the staff’s RAI submittal, and if it will provide an acceptable repair design criteria of the subject safety-related reactor pressure vessel (RPV) internal components. The review assessed the design objectives and criteria, structural and design 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-55 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-55 report.

ENCLOSURE

2.0 SUMMARY OF BWRVIP-55 REPORT

The BWRVIP-55 report addresses the following topics in the following order:

- Component Configurations and Safety Functions - The internal components in the reactor vessel lower plenum area are described in detail with brief descriptions of each component's function and characteristics. The safety design bases for the internal components in the reactor vessel lower plenum area are given. An event analysis is also provided for various operational conditions to ensure the component safety functions are maintained.

- Scope of Repairs - The scope of the proposed repairs is given, which primarily addresses cracking and/or leaking in IGSCC susceptible stainless steel and nickel-chrome-iron alloy components in the reactor vessel lower plenum area.

- 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 internal components in the reactor vessel lower plenum area without draining the vessel) and repair accessibility.

- Design Criteria - The design criteria of the internal components in the reactor vessel lower plenum area are presented. In summary, all repair designs shall 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 - No system evaluation should be required in support of repairs to lower plenum components. Power uprate is briefly discussed for those units currently undergoing a power uprate program.

- Materials, Fabrication and Installation - The materials specifications are given along with the regulatory requirements pertaining to austenitic stainless steel alloys. The minimization of crevices and welding and fabrication guidelines are also discussed. Installation considerations included indicating the as-built dimensional tolerance the repair can accommodate as well as the minimization of 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, and scram tests.

3.0 STAFF EVALUATION

The BWRVIP-55 report is provided to assist BWR owners in designing repairs which maintain the structural integrity of the lower plenum components during normal operation and under postulated transient and design basis accident conditions for the remaining plant life or other service life as specified by the plant owner.

This document is applicable to General Electric BWR/2 through BWR/6 plants which plan to implement repairs to lower plenum items. The following lower plenum components are addressed in this document: Control Rod Drive Housing, Control Rod Drive Stub Tube, In-core Housing, In-core Guide Tube, In-core Stabilizer and BWR/2 Flow Baffle. The aligner pin for the Control Rod Guide Tube (CRGT) and Orifice Fuel Support (OFS) and the peripheral fuel support assembly are also included in this repair criteria, even though they are located on the upper surface of the core support plate. The shroud support legs, and standby liquid control and core delta pressure nozzles and internal lines are addressed in separate repair criteria and are not included in the scope of this report.

3.1 BWRVIP Response to Staff's RAI

The staff's August 13, 1999, RAI, provided six open items. The BWRVIP, in its letter of December 6, 1999, addressed these items, which are discussed below.

Item 1: Section 5.1 discusses the design life of a repair to include plant life extension beyond the current operating license. For expediency, the staff is reviewing BWRVIP submittals to current plant life. Repairs that will be part of the plant at the end of its current license may require plant specific evaluations as part of any license extension. These repairs should be part of a separate section or submittal specifically addressing plant life extension.

BWRVIP Response to Item 1:

> In accordance with a telephone discussion with the NRC lead project manager on Nov. 29, 1999, it is the understanding of the BWRVIP that the NRC considers the Repair documents as applicable to the renewal term. The BWRVIP is preparing a letter to the NRC staff to fully document this position.

Staff's Evaluation:

> The staff finds that BWRVIP's response adequately addressed this item.

Item 2: Section 9.1.3 states that austenitic stainless steel shall meet the requirements of EPRI document No. 84-MG-18, "Nuclear Grade Stainless Steel, Procurement, Manufacturing and Fabrication Guidelines." This document provides guidance for the use of Types 304 and 316 stainless steels in nuclear applications. Further, other stainless steels (SS), e.g., stabilized SS, could be appropriate for certain repair applications. Has the BWRVIP considered including these materials in the subject guidelines?

BWRVIP Response to Item 2:

> A variety of materials have been considered for use in BWR internals applications, such as stabilized types 347 and 321 and nickel-based alloys 625 and 718. The BWRVIP has no plans to address these materials at this time.

Staff's Evaluation:

> The staff finds that BWRVIP's response adequately addressed this item; however, austenitic stainless steel or any other materials shall meet the requirements of EPRI document No. 84-MG-18 or the requirements of other materials proven through testing, performance demonstrations, and field experience to be satisfactory for the application.

Item 3: Section 9.3.4 references ASTM A262 Practice A and E for determining the IGSCC susceptibility in austenitic stainless steels. Practice A provides a screening for IGSCC. Material found nonacceptable by Practice A may be tested according to Practice E. Practice E exposes the test material to an acidified environment followed by a bend test.

If the material exhibits intergranular cracking on the bent surface, the material is susceptible to IGSCC. The acceptance criteria are based on comparing the bend test surface with photographs in ASTM A262. Instead of the acceptance criteria in ASTM A262 Practice A, this document (Section 9.3.4) establishes an acceptance criterion defined as a maximum of 5 percent of the total grain boundary length exhibiting ditching. Discuss how this acceptance criterion was derived, how to determine the percent ditching, and how does this acceptance criterion compare with ASTM practice?

BWRVIP Response to Item 3:

> The criteria found in numerous BWRVIP Repair Criteria, permitting no more than five percent total grain boundary length ditching when evaluating in accordance with ASTM A 262-93a (hereinafter A 262) Practice A, is intended to supplement, not replace, the criteria of A 262. The effect is more, not less, stringent than the criteria of A 262.
>
> When Practice A is used as a screening test in conjunction with Practice E, step and dual etch structures are acceptable in accordance with Table 5 of A 262. Ditch etch structures are not acceptable.
>
> A "Dual Structure" is defined in A 262 paragraph 6.3.2 as "Some ditches at grain boundaries in addition to steps, but no single grain completely surrounded by ditches." A "Ditch Structure" is defined in A 262 paragraph 6.3.3 as "One or more grains completely surrounded by ditches." Based on these definitions, it should be noted that if no one grain is observed to be completely encircled by ditches, an etch structure can exhibit significant lengths of ditching at grain boundaries without being classified as an unacceptable "Ditch Structure" in accordance with A 262.

The evaluation of the etched surface is carried out by optical microscopy as specified in A 262 paragraphs 6.1, 6.2, and 6.4. If any one grain is encircled by ditching, it is evaluated as an unacceptable Ditch Structure. The judgement of the overall percentage of grain boundary ditching, even if no one grain is completely encircled, is also performed by the metallographer using optical microscopy.

The result of applying only the acceptance criteria and definitions of terms found in A 262 would be to allow a very large percentage of grain boundary ditching, as long as no one grain is completely surrounded by ditches. For this reason, the five percent total grain boundary ditching criteria has also been applied as a supplementary acceptance criteria, at least since BWR replacement recirculation pipe specifications were developed in the early 1980s. Using this criteria, more heats fail the screening of Practice A and require further evaluation by Practice E. This "raising of the bar" beyond that required by A 262 assures that marginal heats are effectively evaluated by Practice E.

Staff's Evaluation:

The staff finds that BWRVIP's response adequately addressed this item.

Item 4: Section 9.1.6 (c) states that heat treatments other than solution annealing may be used for stainless steel providing the heat treatment is demonstrated by appropriate qualification testing. Discuss what constitutes an appropriate qualification test and identify any guidance documents.

BWRVIP Response to Item 4:

There is no single way of assessing the suitability of a material that can be described ahead of time. But a general process can be described. A judgement of what constitutes appropriate qualification testing of alternative alloys and heat treatments must take into account the electrochemical and irradiation environment to which a particular material product form and condition will be subjected. Service experience must also be taken into account. Sources of data would include a variety of fractographic, metallographic, and microchemical techniques using accelerated environments, stresses, and/or strain rates, including ASTM A 262 Practices A and E, rising load testing in accordance with MIL-DTL24114F(SH), constant extension rate testing, slow strain rate testing, creviced bent beam specimens, swelling mandrel testing, u-bends, corrosion fatigue, and ASTM E 399 fracture mechanics specimens. EPRI TR-109668-SI-R1, "Materials Handbook for Nuclear Plant Pressure Boundary Applications," Revision 1, December 1998, provides a broad overview of the application of such techniques to the evaluation of suitability of materials. If a licensee chooses to utilize alternative alloys or heat treatments on the basis of such qualification testing, the licensee will inform the NRC.

Staff's Evaluation:

The staff finds that BWRVIP's response adequately addressed this item.

Item 5: Section 9.1.7 suggests that re-solution annealing is an acceptable technique for removing stresses. If localized solution annealing is an option for reducing stresses in a stainless steel weld, discuss the potential for sensitizing part of the cross section area in the temperature gradient between the solution annealed and non-annealed material.

BWRVIP Response to Item 5:

> In the context of Section 9.1.7, localized solution annealing would only be applied to nuclear grade types 304 and 316 and XM-19, materials which already exhibit excellent resistance to sensitization. The effect of local solution annealing would be to relieve stresses, anneal cold work, and re-solutionize carbides. This would be a concern in high stress concentration regions such as threads, and in the exposed surface of a weld heat affected zone which is unavoidably subjected to significant plastic strain as a result of the welding process. In the unlikely event that sensitization does occur in these materials in the temperature gradient between solution annealed and non-annealed material, it would be in a region remote from significant stress concentrations.

Staff's Evaluation:

> The staff finds that BWRVIP's response adequately addressed this item; however, for stress relaxation of weldments, only low carbon (like type 304L and 316L) or carbon stabilized (like type 347 and 321) stainless steels which exhibit excellent resistance to sensitization, can be subjected to re-solution annealing.

Item 6: Section 10.2 states that inspection required for repaired shroud support structures and/or vessel internal attachments shall be specified commensurate with design considerations and Code requirements applicable to the specific design. Provide a list of pre- and post-inspections for lower plenum component non-Code repairs. Identify the method and technique required and the method and technique provided in guidance for a baseline inspection. Discuss design considerations that would prompt an inspection after a repair.

BWRVIP Response to Item 6:

> As described in 10-2, the inspections must be specified commensurate with considerations of the specific repair design. A specific list of pre- and post-repair inspections cannot be made until a design has been proposed. The method and technique defined for the baseline inspection would also depend on the specifics of the design. However, in general, it would be performed to ensure the structural integrity of the attachment points and to confirm that the repair had been properly installed. Additional inspections may be appropriate after the repair is installed to ensure the continued integrity of the repair. An example with
>
> which the BWRVIP has some considerable experience is shroud repair. In this case, the hardware is inspected on an ongoing basis to ensure, for example, that the tie-rod hardware remains properly in place.

Staff's Evaluation:

The staff finds that BWRVIP's response adequately addressed this item.

3.2 Structural and Design Evaluation

This topical report provides the general design acceptance criteria for temporary and permanent repair of BWR components located in the lower plenum of the reactor. It is provided to assist BWR owners in designing repairs which maintain the structural integrity of the lower plenum components during normal operation and under postulated transient and design basis accident conditions for the remaining plant life or other service life as specified by the plant owner.

Repair of the lower plenum components may not be the only viable method for resolving cracking in the components. Due to variation in the material, fabrication, environment and as-found condition of the individual lower plenum components, repair is only one of several options that are available. The action to be taken for individual plants will be determined by the plant licensee.

Based on a review of the description of the loads as discussed in the BWRVIP-55 report, the staff finds that all applicable loads are included in the repair design.

Structural and Design Evaluation

The applied loads on the reactor internals consist of the following: deadweight, differential pressure, hydraulic loads, seismic inertia, seismic anchor displacements, fuel lift, LOCA phenomena, safety relief valve (SRV) opening, loads due to flow induced vibration, thermal and seismic anchor displacements, and control rod drive reactions. Each of these loads is discussed in detail in the topical report. A summary of how these loads are incorporated in the structural analysis and their effect on the lower plenum is provided in the following staff evaluation:

In general, hydrodynamic loads incurred due to SRV discharge, pool swell, condensation oscillation, and chugging are applicable to Mark II and III containment types. These loads are not significant for the vessel and internals in Mark I containment types where the torus and drywell are not dynamically coupled to a substantial degree. Also, the annulus pressurization loads may not be included in the licensing basis for Mark I containment plants.

The fuel lift loads are essentially due to hydrodynamic effects, frictional forces, and the relative motion between the components. The more severe loading combination that contributes to fuel lift loads includes responses from natural phenomena [safe shutdown earthquake (SSE)], and from other events, such as the LOCA and SRV discharge.

In the vertical direction, the bundle weight resists the upward forces caused by the flow of reactor coolant inside the vessel and any additional dynamic forces due to postulated accident loads which could potentially result in relative motion between the fuel bundle and its supports. Under postulated dynamic conditions, the bundle could move and, upon reseating in the fuel supports, produce impact forces. The staff finds the application of the fuel lift loads as discussed above reasonable and acceptable.

Seismic inertia consists of horizontal and vertical inertia forces acting on the entire component due to seismic excitation of the RPV. The locations where the seismic excitation is imparted to the component are at the connection to the RPV and the shroud. Generally, either the peak seismic accelerations, the acceleration spectra, or both, are available for some or all of the connection points. Alternatively, if the acceleration spectrum information is available, a modal analysis of the component can be conducted to obtain the seismic inertia stresses. Both the horizontal and vertical seismic excitation must be considered. The method of combination should be consistent with the plant Final Safety Analysis Report (FSAR).

The reactor water level during normal operating conditions is above the level of most of the reactor internals. Therefore, during a seismic event some components will be vibrating in the submerged condition. This requires the addition of a hydrodynamic mass term for the seismic analysis. The staff finds the application of the inertial loads in the seismic analysis as discussed above reasonable and acceptable.

Seismic anchor displacements are applied at the attachment points of the component to the RPV or core shroud. These displacements, if available, are obtained from the RPV internals seismic analysis report. If not available, additional analysis may be required. The vertical direction relative seismic anchor displacements are generally expected to be insignificant. However, if such displacements are determined to be significant, the corresponding stresses should be included along with the other displacement stresses.

Pool Swell, Condensation Oscillation and Chugging

Condensation oscillation (CO) loads are induced during an intermediate-break accident (IBA) and design basis LOCA (DBA) following vent air clearing and pool swell (PS). There is a period of high steam flow rate through the vent system where the steam is condensed in a region near the vent exit which results in oscillation. The resulting hydrodynamic pressure oscillations may cause dynamic excitations of the structure and equipment.

Main vent chugging (CHG) loads are induced during an SBA (small break accident), IBA and DBA when there is insufficient steam flow to maintain a steady steam jet at the vent exit. A random formation of steam bubbles, which alternatively form and collapse at the vent exit, produces hydrodynamic pressure oscillations on the pool boundary for Mark II pressure suppression containments and on the weir wall and pool boundary for Mark III containments. These pressure oscillations may cause dynamic excitations of the structure and the contained equipment.

Pool Swell (PS) loads are induced during DBA by the continued injection of drywell air into the suppression pool during the LOCA and the subsequent expansion of the air bubble which results in the rise of the suppressing pool surface. These pool swell excitations can result in dynamic loading of the reactor vessel and internals.

The anchor points of the reactor internals grow vertically and horizontally at different rates due to differences in the materials (low alloy steel for the vessel versus stainless steel for many of the internals). Also, these displacements are expected to vary during certain transients due to the differences in temperatures and pressures. The RPV temperatures during the transient conditions are based on the information provided in the vessel thermal cycle diagram. The displacements during normal operations, operating transients (typically enveloped by the loss of feedwater pumps transient but should be verified on a plant specific basis) and during a loss of coolant accident, need to be considered. Loads / stresses associated with steady state or transient conditions are also included along with the differential thermal displacement effects.

Based on a review of the description of the SRV actuation, annulus pressurization, pool swell and other loads as discussed above, the staff finds that all applicable loads are included in the repair design.

Load Combinations

The descriptions of general load combinations by service level are based on current regulatory guidance provided in NCA-2142.4 of ASME Section III and Appendix A of Section 3.9.3 of NUREG-800. Service levels A, B, C and D refer to the Normal Operating, Upset , Emergency and Faulted conditions, respectively.

Service Level C loads include the combination of all sustained normal operation loads in conjunction with loads from the design basis pipe break (DBPB).

Service Level D loads include the combination of all sustained loads in conjunction with several combinations of design basis events. All components of these loads should be considered.

For plants that use systems for injection (e.g., jet pumps for LPCI injection and core spray injection), the loads associated with the injection are treated as a faulted condition. This assumption is acceptable provided that the system functional requirements for delivery of coolant under long term DBA conditions are ensured.

The load combinations used in the evaluation should be consistent with the requirements of the plant FSAR or related licensing basis documentation. Load combinations used to analyze reactor internals vary, depending on the plant vintage. In the event that load combinations are not specified in the FSAR, the set of load combinations shown in Table 4 of the BWRVIP-55 report may be used for Mark I Plants and in Table 5 of the report for Mark II and III Plants.

To insure the integrity of preloaded mechanical repair joints, which are critical to meet functional requirements, it may be necessary to also evaluate the effect of secondary loadings with emergency and faulted load combinations.

In response to the staff's concern regarding footnote (2), Tables 4 and 5, relating to the combination of DBA and SSE loads, the BWRVIP has proposed to delete footnote (2) from the tables. Thus, plants which lack guidance in their FSAR regarding combination of DBA and SSE loads will use the load combinations recommended in Tables 4 and 5 to evaluate the structural adequacy of the repairs. In addition, clarification will be provided in a revised BWRVIP-55 report as to how and when Tables 4 and 5 should be used. The staff finds the recommended load combinations with the proposed modifications reasonable and acceptable.

Functional Evaluation Criteria

For some reactor internal components the code specified stress limitations alone may not be sufficient to verify that the function of the component is satisfied. Some faulted loading conditions may require a more detailed evaluation of the effects of displacement induced stresses (secondary) to ensure that the passive components (e.g., pipe support structure) can perform their intended function. The recognition of this concept requires more detailed evaluations to demonstrate that specific allowable stress limits are satisfied for each of the code prescribed service levels.

The design report for the repair shall identify the specific sections and subsections of the ASME Code utilized to designate allowable limits.

Flow Induced Vibration

The repair shall be designed to address the potential for vibrations, and to keep vibration to a minimum. The natural frequency of the repaired internal lower plenum component shall be determined. The vibratory stresses shall be shown to be less than the endurance limit of the repair materials. Forcing functions to be considered include the coolant flow and the vibratory forces transmitted via the end point attachments for the repair. Flow across the piping during accident scenarios shall be considered.

Repair Impact on Existing Internal Components

Changes in the lower plenum components which result from repairs shall be evaluated to demonstrate that the entire system (e.g., piping, anchors, and supports) satisfy design basis stress limits. This may require parametric evaluations (sensitivity studies) to determine the bounding configuration of postulated cracks. Increased stress on existing internal components, e.g., vessel penetrations, core support plate or shroud which are used to support the repair, is acceptable as long as the current plant licensing basis or code allowables are met. Potential impacts of a lower plenum repair on previously implemented repairs and current analytical justifications for continued operation with defects shall also be addressed.

All thermal-hydraulic and structural codes utilized in the design analysis shall be appropriately benchmarked. New or improved calculational methods may be utilized by the designer. For these techniques, appropriate benchmark information shall be provided to demonstrate that the method is conservative and bounding for the application.

The design and analysis of lower plenum component repairs shall consider the operating conditions and events specified in the original plant RPV and nozzle thermal cycle diagrams.

3.3 General Comments:

1. On the page after the title page under the heading, “Results,” the BWRVIP-55 report states, “the document provides general design acceptance criteria for the repair of SLC piping.” The staff recommends this be reworded to state, “the document provides general design acceptance criteria for the repair of lower plenum components.”

2. In order to be consistent with other BWRVIP repair procedures, such as the BWRVIP-16 and BWRVIP-19 reports, the following requirements in Section 9.1.2, Materials, of the BWRVIP-55 report should be modified to read: “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 II 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 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 on a case by case basis to the regulatory authorities for approval, on a plant specific basis."

3. The staff requests licensees to determine the weldability of all materials to be welded since some fasteners may be made of generally unweldable materials or require very special conditions to weld them, such as AISI 4140, 4340 (B7) low alloy materials or 410 (B6) type stainless steel alloys. Alternatively, BWRVIP could eliminate all welding on fasteners in this document.

4.0 CONCLUSION

The NRC staff has reviewed the BWRVIP-55 report as well as its associated RAI response and found that the BWRVIP-55 report, as modified and clarified to incorporate the staff's comments above, is acceptable for providing guidance for the temporary and permanent repairs of cracked or leaking internal components in the reactor vessel lower plenum area. This finding, based upon the information submitted in the subject report and RAI, is consistent with NRC approved methodology. Therefore, the staff has concluded that licensee implementation of the guidelines in the BWRVIP-55 report, as modified, will provide an acceptable repair design criteria of the safety-related components addressed in the BWRVIP-55 document. The modifications stated in the RAI and addressed above should be incorporated in Revision 1 of the BWRVIP-55 report. The BWRVIP-55 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.

5.0 REFERENCES

1. Carl Terry, BWRVIP, to USNRC, "BWR Vessel and Internals Project, Lower Plenum Repair Design Criteria (BWRVIP-55)," EPRI Report TR-108719, dated September, 1998.

2. C. E. Carpenter, USNRC, to Carl Terry, BWRVIP, "Propriety Request for Additional Information - Review of EPRI Topical Reports TR-108720, TR108719, and TR-108721," dated August 13, 1999.

3. Carl Terry, BWRVIP, to USNRC, "BWRVIP Response to NRC Request for Additional Information on BWRVIP-55," December 6, 1999.

4. Carl Terry, BWRVIP, to USNRC, "BWRVIP Response to NRC Safety Evaluation on BWRVIP-16 and BWRVIP-19," December 6, 1999.