January 29, 2001

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, TOP GUIDE/CORE PLATE REPAIR DESIGN CRITERIA (BWRVIP-50)," EPRI REPORT TR-108722, MAY, 1998 (TAC NO. MA1926)

Dear Mr. Terry:

The NRC staff has completed its review of the Electric Power Research Institute (EPRI) proprietary report TR-108722, "BWR Vessel and Internals Project, Top Guide/Core Plate Repair Design Criteria (BWRVIP-50)," dated May, 1998. Both proprietary and non-proprietary versions of the BWRVIP-50 report were submitted to the U. S. Nuclear Regulatory Commission for staff review by letter dated May 14, 1998. This report was supplemented by a letter dated December 6, 1999, which was in response to the NRC staff's requests for additional information (RAI), dated April 7, 1999 and May 24, 1999. The BWRVIP-50 report provides general design acceptance criteria for the permanent or temporary repair of the top guide or core plate. These guidelines are intended to maintain the structural integrity and system functionality of the top guide or core plate during normal operation and under postulated transient and design basis accident conditions. The BWRVIP provided the BWRVIP-50 report to support generic regulatory efforts related to the repair of BWR top guide and/or core plate.

The NRC staff has reviewed the BWRVIP-50 report, as well as its associated RAI response, and finds, in the enclosed safety evaluation (SE), that the BWRVIP-50 report, as modified and clarified in the SE, is acceptable for providing guidance for permanent repairs of the top guide and/or core plate. 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 the BWRVIP-50 report will provide an acceptable repair design criteria for the safety-related components addressed. However, as discussed in the attached SE (see Responses to RAI Items 5-1 and 11), licensees will need to submit requests for Code alternatives in accordance with 10 CFR 50.55a if the repair, either temporary or permanent, does not meet the Code or applicable regulatory requirements. The staff requests that the BWRVIP address this in a revision to the BWRVIP-50 report.

Also, pursuant to 10 CFR 50.55a(a)(3)(i), the staff finds that licensee installation of core support plate wedges to structurally replace the lateral resistance provided by the core support rim hold down bolts to be an acceptable alternative to Section XI of the ASME Boiler and Pressure Vessel Code that will provide an acceptable level of quality and safety for the subject safety-related components. Licensees who install the subject wedges are requested to inform the NRC staff by letter at least 90 days prior to installation, or, if they have already installed the wedges, within 90 days of the date of this letter.

The BWRVIP-50 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 an 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/

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, TOP GUIDE/CORE PLATE REPAIR DESIGN CRITERIA (BWRVIP-50)," EPRI REPORT TR-108722

1.0 INTRODUCTION

1.1 Background

By letter dated May 14, 1998, as supplemented by letter dated December 6, 1999, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted proprietary and non-proprietary versions of the Electric Power Research Institute (EPRI) Report TR-108722, "BWR Vessel and Internals Project, Top Guide/Core Plate Repair Design Criteria (BWRVIP-50)," dated May 1998, for NRC staff review. The supplemental information was in response to the staff's requests for additional information (RAI), dated April 7, 1999 and May 24, 1999.

The BWRVIP-50 report, as supplemented, provides general design acceptance criteria for the permanent or temporary repair of the top guide or core plate. These guidelines are intended to maintain the structural integrity and system functionality of the top guide or core plate during normal operation and under postulated transient and design basis accident conditions. BWRVIP provided the BWRVIP-50 report to support generic regulatory efforts related to the repair of the top guide or core plate.

1.2. Purpose

The staff reviewed the BWRVIP-50 report, as supplemented in response to the staff's RAI, to determine whether its proposed guidance 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-50 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-50 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-50 report.

ENCLOSURE

2.0 SUMMARY OF BWRVIP-50 REPORT

The BWRVIP-50 report addresses the following topics in the following order:

- Component Characteristics and Safety Functions - The top guide and core plate are described in detail with brief descriptions of each component's safety related function.

  Differences among the various models of BWRs (BWR/2 through BWR/5 and BWR/6) are identified. 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 addressed, including degradation of the top guide/core plate and addition of structural wedge-type components between the top guide/core plate and shroud.

- Design Objectives - The following design objectives are presented and briefly discussed: repair design life, safety design bases, safety analysis events, structural integrity, loose parts considerations, physical interfaces with other reactor internals, installation, load path alterations and existing structures.

- Codes and Standards - The design criteria of the top guide and core plate 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) 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, consideration of shroud repair or cracking, repair impact on existing internal components, radiation effects on repair design, analysis codes, thermal cycles, and corrosion allowance.

- System Evaluation - The following system evaluations are discussed: reactor coolant flow distribution and pressure drop, emergency operating procedure (EOP) calculations and power uprate

- Materials, Fabrication and Installation - The materials specifications are given along with the regulatory requirements pertaining to austenitic stainless steel alloys. Crevices and fabrication guidelines are also discussed. Pre-installation as-built inspection, installation cleanliness, ALARA considerations, and qualification of critical design parameters are presented

- Inspection and Testing - Inspection and testing of the reactor internal components are addressed in inspection access and pre- and post-installation inspection.

3.0 STAFF EVALUATION

The top guide and core plate are classified as safety related components in BWR/2 - BWR/6 plants. The structural integrity of the top guide is relied upon for assuring the correct position of the top of fuel assemblies to ensure control rod insertion. An additional safety function of the top guide is to provide lateral support for the fuel assemblies during seismic loads. The structural integrity of the core plate is relied upon for assuring the correct position of the bottom end of the fuel assemblies, fuel support castings, and top end of the control rod guide tubes to ensure control rod insertion. The core plate is also relied on for lateral support for the above components during seismic loads and vertical support for peripheral fuel assemblies.

3.1 BWRVIP Response to Staff's RAI

The staff's April 7, 1999 and May 24, 1999, RAI's, provided eleven open items. The BWRVIP, in its letter of December 6, 1999, addressed these items, which are discussed below.

RAI Item 1: Clarification is needed for the case of a licensee previously granted relief or an alternative to the regulations regarding inspection of reactor vessel welds based either on inaccessibility or the provisions of Generic Letter 98-05. Specifically, if during the course of the repairs, the subject welds are exposed sufficiently to allow for inservice inspection access, there needs to be an adequate technical justification as to why these welds should not be inspected in accordance with the ASME Code or Regulations.

BWRVIP Response to RAI Item 1: It is recognized by BWRVIP members that when access is gained to previously inaccessible components or welds, that as a minimum, a visual inspection should be performed. When this is done, inspection results will be provided to NRC as part of the routine reporting from BWRVIP.

However, visual examinations of reactor vessel welds are of limited benefit and temporary access to a weld in the reactor vessel during a repair will not generally allow for a meaningful volumetric examination. Volumetric examination of reactor vessel welds requires outage time, inspection tooling and qualified examination procedures and personnel. Having the time required to adequately plan and implement a vessel inspection during the course of an internal repair is a significant logistical problem. Also the amount of weld metal likely to be exposed by a repair is not likely to be sufficient to warrant the cost/effort to perform the examination. This is especially true of reactor vessel welds outside of the beltline region which are exposed to less fluence and thus do not suffer from significant reduction in fracture toughness. (The NRC evaluation of BWRVIP-05 confirms the limited benefit of a small area examined, especially for circumferential welds.) Also, if the repair was unanticipated prior to the outage in which it is performed, it is unlikely that the appropriately qualified personnel and inspection equipment would be available.

Finally, it is not appropriate that an approved relief request regarding inspection of one component (e.g. RPV) become unapproved due to work on another component (e.g. top guide), unless the repaired component's problem has direct bearing on the functionality of the component for which relief was granted.

Staff's Evaluation to BWRVIP Response to RAI Item 1: The staff finds that the BWRVIP's response adequately addressed this item.

RAI Item 2: What is the basis for the 0.003 inch corrosion allowance for stainless steel exposed to the primary coolant? This corrosion allowance is based on a 40 year reactor life, what are the consequences of using this allowance for the license renewal period?

BWRVIP Response to RAI Item 2: The 0.003 inch corrosion allowance for stainless steel is the value typically applied by GE for BWR internals repair and modification design. This design allowance is documented in a GE Internal Materials Handbook. Based on some prior study of GE and published test results, it is estimated that the actual BWR general stainless steel corrosion rate is approximately a factor of three lower than this design value. This value is therefore sufficiently conservative that it can also be applied for a 60 year life.

Additional information can be found in the article "Electrochemical Measurements of Corrosion Processes in a Boiling Water Nuclear Reactor," by Cowan and Kaznoff in the journal Corrosion, April, 1973. Measured corrosion rates for Type 304 SS exposed for 12 months to a hot water environment are reported as the equivalent of 6.6×10^{-6} inches per year. A conservative linear extrapolation of this value yields 0.0002 inches over a 30 year period. This is an order of magnitude lower than the 0.003 in value suggested in the Repair Design Criteria.

Staff's Evaluation to BWRVIP Response to RAI Item 2: The staff finds that BWRVIP's response adequately addressed this item.

RAI Item 3: Provide further details for the "CIB" condition for Alloy X-750. Provide the basis for the use of Alloy 750 in the "CIB" condition.

BWRVIP Response to RAI Item 3: Details of the CIB heat treatment condition for alloy X-750 and the basis for its use in the BWR environment are found in EPRI Document NP-7032, "Material Specification for Alloy X-750 for Use in LWR Internal Components (Revision 1)," which has previously been reviewed by NRC in conjunction with review of the Shroud Repair Design Criteria (BWRVIP-02).

Staff's Evaluation to BWRVIP Response to RAI Item 3: In the "BWRVIP Response to NRC Safety Evaluation on BWRVIP-16 and BWRVIP-19," dated December 6, 1999, the BWRVIP stated, "The rising load test as described in NP-7032 will be retained in order to provide verification by physical testing that the specified heat treatment was properly performed."

In this same response the BWRVIP also stated, "The allowable cobalt level for individual heats of alloy X-750 will be specified as 0.25 percent maximum. If this limit is exceeded, an alternative evaluation protocol that can be implemented by the licensee will be provided. The alternative criteria will be a maximum allowable weighted average cobalt level of 0.25 percent, taking into account the surface area of all newly installed components wetted by reactor coolant."

In order for the BWRVIP to be consistent, both of the requirements stated above should be included in the BWRVIP-50 report. With the inclusion of these two statements, the staff finds that BWRVIP's response adequately addressed this item.

RAI Item 4: Will any lubricants be used on the fasteners in the top guide/core plate repair? If so, what lubricants will be used? Provide additional analyses or evaluation on the effects of using a lubricant in terms of the torque-tension relationship and the effect of the lubricant on stress corrosion cracking (SCC) behavior.

BWRVIP Response to RAI Item 4: Lubricants are normally used for threaded fasteners or with a configuration where sliding of adjacent surfaces is required for installation or assembly of repairs. The torque-tension relationship is determined based on the expected range of friction coefficients for the selected lubricant. The lubricant selected would be one that is on the plant's approved list for the reactor coolant system. Typically, these lists reference, or are based on the "BWR Operators Manual For Materials and Processes," the latest version of which is NEDE-31735-A-P-98 Class III, December 1998. This proprietary document, issued by GE, is utilized to define the acceptability of numerous non-metallic items and substances for contact with reactor coolant and/or surfaces exposed to reactor coolant.

Staff's Evaluation to BWRVIP Response to RAI Item 4: The staff finds that BWRVIP's response adequately addressed this item.

RAI Item 5: Clarifications should be made to the BWRVIP-50 report so that individual licensees will make plant-specific submittals for the following instances:

RAI Item 5-1: Licensees with materials not covered by the scope of the ASME Code or in the original design Code of Record should submit a plant-specific alternative to the NRC for review and approval.

BWRVIP Response to RAI Item 5-1: 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 10CFR50.55a.

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 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 to the regulatory authorities for approval, on a plant specific basis.

<u>Staff's Evaluation to BWRVIP Response to RAI Item 5-1</u>: The staff finds the BWRVIP response to be acceptable. However, the staff requests that the third paragraph of the BWRVIP's response to Item 5-1, above, be included in the BWRVIP-50 report in Section 9.1.

<u>RAI Item 5-2</u>: Licensees should provide to the NRC the bases for the adequacy of their water chemistry control measures and neutron irradiation effects on materials used in the repair.

<u>BWRVIP Response to RAI Item 5-2</u>: BWRVIP members assure the adequacy of their water chemistry by implementing the 1996 EPRI Water Chemistry Guidelines. All members have committed to follow these guidelines and the recommended corrective measures. NRC has previously indicated that this practice is adequate.

It is the responsibility of the owner and the repair designer to understand and account for the environment in which a repair will be used. The BWRVIP guidelines specify that material be selected with regard to their suitability of their environment. BWRVIP believes this is adequate. Plant-specific submittals are not required except as noted in response to Item 5-1 above.

<u>Staff's Evaluation to BWRVIP Response to RAI Item 5-2</u>: The staff finds that BWRVIP's response adequately addressed this item.

<u>RAI Item 5-3</u>: Licensees should provide the bases for the minimization of the effects of crevice-induced SCC in their individual design and repairs for NRC review and approval.

<u>BWRVIP Response to RAI Item 5-3</u>: A typical approach to minimizing the effects of SCC would be to utilize materials that are highly resistant to IGSCC and to minimize the number of crevices adjacent to susceptible materials. This information would be included in the specification for the repair and would be provided for NRC review.

<u>Staff's Evaluation to BWRVIP Response to RAI Item 5-3</u>: The staff finds that BWRVIP's response adequately addressed this item.

<u>RAI Item 6</u>: With respect to Appendix A, "Specific Design Criteria and Requirements for Wedge-Type Repairs for Top Guide and Core Plate"

<u>RAI Item 6-1</u>: Will any lubricants be used on components described in Appendix A for "wedge-type" repairs? If so, give the basis for the type of lubricant with respect to inducing SCC susceptibility, including those components in contact with the wedge.

<u>BWRVIP Response to RAI Item 6-1</u>: See response to Item 4 (above).

<u>Staff's Evaluation to BWRVIP Response to RAI Item 6-1</u>: The staff finds that BWRVIP's response adequately addressed this item.

<u>RAI Item 6-2</u>: Will the design of the wedge region consider crevice conditions with respect to SCC?

<u>BWRVIP Response to RAI Item 6-2</u>: See the response to Item 5-3 above.

<u>Staff's Evaluation to BWRVIP Response to RAI Item 6-2</u>: The staff finds that BWRVIP's response adequately addressed this item.

RAI Item 6-3: Describe the provisions provided to facilitate inservice inspection (e.g., surface flatness and finish).

BWRVIP Response to RAI Item 6-3: Typically, there are no requirements on flatness or surface finish imposed by inservice inspection requirements for non-welded repairs (and the document does not permit welded repairs). The designs must consider the requirement to perform inservice inspection to verify the continuing functional and structural integrity of the repair but flatness or surface finish normally have no impact on inspectability.

Staff's Evaluation to BWRVIP Response to RAI Item 6-3: The staff finds that BWRVIP's response adequately addressed this item.

RAI Item 6-4: Individual utilities using wedge-type repairs should provide the NRC with information on the above and materials, lubricants and inspection procedures.

BWRVIP Response to RAI Item 6-4: The BWRVIP agrees that it is appropriate for utilities to provide the NRC with information regarding, ISI, materials and lubricants and crevice issues. The document will be revised to indicate this.

Staff's Evaluation to BWRVIP Response to RAI Item 6-4: The staff finds that BWRVIP's response adequately addressed this item.

RAI Item 7: Section 3.2, Safety Related Functions of Analyzed Components, refers to a General Electric document GENE-771-44-0482, "Justification of Allowable Displacements of the Core Plate and Top Guide - Shroud Repair," Rev. 2, November 16, 1994. Is this the same document as GENE-771-44-0894 Rev. 2, which has the same title and date?

BWRVIP Response to RAI Item 7: The GENE document numbers in 3.2.1 and 3.2.2 are in error. The GENE document numbers in Section 13 (Reference 8) and Appendix A Section 7 (Reference A-4) are also wrong. The correct reference is General Electric document GENE-771-44-0894, "Justification of Allowable Displacements of the Core Plate and Top Guide - Shroud Repair," Rev. 2, November 16, 1994.

Staff's Evaluation to BWRVIP Response to RAI Item 7: The staff finds that BWRVIP's response adequately addressed this item and requests that the BWRVIP-50 report be modified to address the BWRVIP response to item 7.

RAI Item 8: Section 8, System Evaluation, does not discuss potential leakage caused by a top guide or core plate repair. If the repair results in a new source of leakage or increases the leakage from known leakage points, this leakage should be evaluated as part of the repair.

BWRVIP Response to RAI Item 8: Section 8.1 requires that the effects of the repair on the reactor coolant flow distribution be minimized. This would include the assessment of the effects on leakage of any new shroud leakage paths, e.g., from any holes EDM'd in the shroud. The section will be clarified to state that evaluation of the coolant flow distribution includes consideration of leakage caused by the repair.

Staff's Evaluation to BWRVIP Response to RAI Item 8: With the inclusion of the above clarification in the BWRVIP-50 report, the staff finds that BWRVIP's response adequately addresses this item.

RAI Item 9: It is stated in Section 4.0 that wedge-type components will be used between the top guide/core plate and the shroud. The design criteria for the wedge-type repair state that wedges will be adjustable. Indicate whether or not these wedges, or other components that would be used in the repair, would be expected to slide against each other or other hard surfaces during differential thermal expansion or seismic loadings.

BWRVIP Response to RAI Item 9: The wedge components would be expected to slide against the supported top guide/core plate/shroud surfaces and would need to have features to accommodate the sliding while maintaining their functional and structural integrity.

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

RAI Item 10: Since the frictional coefficient between metallic surfaces often varies significantly from location to location and has a tendency to change over a period of time, discuss how the repair design criteria will assure that unanticipated frictional resistance between the repair parts will not result in excessive loading of the top guide/core plate structures.

BWRVIP Response to Item 10: The wedge designs would have to address differential motion between supported surfaces. Typically, this would be accomplished by minimizing the effects of changes in friction coefficients by providing small clearances or by minimizing the extent of the relative motion.

Staff's Evaluation to BWRVIP Response to RAI Item 10: Since it is difficult to maintain small clearances under operating conditions, the BWRVIP should be aware that it may be prudent to minimize the relative motion between supported surfaces in the event differential motion is completely inhibited due to friction or other unanticipated causes. The structural integrity of the affected subassemblies should be maintained and no loose parts generated. The staff finds that, with the preceding modification to the BWRVIP-50 report, the BWRVIP's response adequately addressed this item.

RAI Item 11: Distinguish between the repair criteria for temporary and permanent repairs.

BWRVIP Response to RAI Item 11: These Repair Design Criteria are intended to define the requirements for a permanent repair. Situations may occur where due to, for example, material availability, a licensee cannot meet all requirements for a permanent repair, but could install a repair that would be fully functional for a limited time or under limited operating conditions. This type of repair would be considered a temporary repair.

Staff's Evaluation to BWRVIP Response to RAI Item 11: The staff has a concern over the vagueness of the distinction between temporary and permanent repairs. Specifically, the staff requests that the BWRVIP revise its definition of temporary and permanent repairs to acknowledge that repairs may be in accordance with the ASME Code and meet the requirements of 10 CFR Part 50, under which circumstances no regulatory review or action would be needed. Conversely, if the repair, either temporary or permanent, does not meet the Code or the regulatory requirements, staff review and approval would be required. The staff requests that the BWRVIP address this in a revision to the BWRVIP-50 report.

3.2 Systems Evaluation

In Section 3.2, Safety Related Functions of Analyzed Components, reference was made to a General Electric (GE) report which discusses the allowable displacements of the top guide and core plate. This proprietary report has not been formally reviewed by the staff. However, the staff noted that the reference number to the GE document appeared to be incorrect. The BWRVIP has reviewed the staff's observation and agreed that the reference document number was incorrect. This should be corrected in Revision 1 of the BWRVIP-50 report.

Section 8, System Evaluation, provides guidance on evaluating reactor coolant flow distribution and pressure drop, emergency operating procedure calculations, and power uprate for proposed repairs to the top guide or core plate. The BWRVIP-50 guidance recommends that the effect of the proposed repair on the reactor coolant flow distribution and pressure drop through the reactor fuel and internals should be minimized. However, the guidance does not explicitly mention the evaluation of potential leakage sources caused by a repair. The BWRVIP has agreed to modify the section to clarify that evaluation of the coolant flow distribution includes consideration of leakage paths caused by the repair. The staff has reviewed the guidance provided in the BWRVIP-50 report and finds it consistent with other existing guidance. The staff has concluded that the guidance in BWRVIP-50 should be sufficient for permanent or temporary repairs of the top guide and core plate. Although no specific type of repair has been recommended, the staff finds that the guidance provided should be applicable to any potential repair that may be considered.

3.3 Structural and Design Evaluation

The top guide/core plate repair shall be designed to provide structural integrity for all specified loading conditions. Hydraulic loads acting on the top guide/core plate structures for normal, upset, emergency and faulted conditions including seismic loads shall be considered. The pressure differences used for these events shall be consistent with the current plant licensing basis documents.

Alterations to the original design load path(s) as a result of the repair shall be analyzed to demonstrate that the structural integrity of the affected component(s) and the reactor vessel meets the original design basis for those components.

The applied loads on the reactor internals generally 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, and thermal and pressure anchor displacements.

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.

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 a high steam flow rate through the vent system where the steam is condensed in a region near the vent exit that results in oscillation. The resulting hydrodynamic pressure oscillations may cause dynamic excitations of the structure and contained equipment.

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

SRV air clearing loads are induced by SRV actuation which produces a rapid compression of the air mass inside the SRV discharge pipes. The internal pressure drives the water out of the submerged SRV discharge device (rams head or quencher) and ejects a high-pressure air bubble into the suppression pool below the water surface, causing oscillating pressures on the suppression pool boundary. The oscillating pressures impart structural motions which may cause dynamic excitations of the structure and contained 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 loads are induced during a DBA by the continued injection of drywell air into the suppression pool, and the subsequent expansion of the air bubble which results in the rise of the suppression pool surface. Structures above the pool surface may experience loads. In addition to the initial impact loads, these structures may experience drag loads as water flows past them.

Annulus pressurization refers to the loading on the biological shield and the reactor vessel following a postulated pipe rupture. The break is assumed to be instantaneous guillotine rupture occurring at the vessel nozzle safe end to pipe weld. The mass and energy released during the postulated pipe rupture cause a short-term transient, asymmetric, differential pressure within the annular region between the biological shield wall and the RPV. In addition, there is a reaction to the jet stream release of the RPV inventory and the impact of the ruptured pipe against the pipe whip restraint, which is attached to the biological shield wall. Based on its review as discussed above, the staff finds that all loads applicable to the jet pumps have been considered and are therefore acceptable.

<u>Service Level Conditions</u>

The applicable service level conditions shall be in accordance with the individual plant SAR, or other plant commitments for RPV internals mechanical design. Where commitments exist to utilize the requirements of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," 3rd Edition, July 1981, or no plant-specific guidance exists, the following descriptions may apply. These descriptions of general load combinations by service level are based on the current regulatory guidance (provided in NCA-2142.4 of ASME Code Section III, and Appendix A of Section 3.9.3 of NUREG-0800).

Service Level A loads should include the combination of all sustained loads that are anticipated during normal plant/system operation. These include deadweight of all supported components, differential pressures, and thermal-hydraulic loads, including flow-induced vibration (FIV).

Service Level B loads include loads due to anticipated operational occurrences that have the potential to increase the loads acting on the reactor internal components above those experienced during normal operation. Typical events include normal operating loads plus system operating transients (SOTs). The SOTs shown on the applicable RPV thermal cycle diagram should be used to determine the applicable transient conditions. Also, the combination of normal loads plus OBE loads is considered an upset event.

Service Level C or emergency loads include the combination of all sustained normal operating loads in conjunction with loads from the design-basis pipe break (DBPB). The DBPB includes all postulated pipe breaks other than a LOCA, main steam line break (MSLB), or feedwater line break. These include postulated pipe breaks in Class I branch lines that result in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system.

Service Level D loads include the combination of all sustained loads in conjunction with several combinations of design basis events. These combinations include the DBPB, MSLB/feedwater line break, or LOCA and the SSE (where applicable, per the plant-specific design basis). All components of these loads should be considered.

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

Load Combinations

The load combinations used in the evaluation should be consistent with the requirements of the plant SAR or related licensing basis documentation. Typically, Section 3.9 of the SAR contains the necessary information on this subject including, for some plants, hydrodynamic loads (i.e., "new loads").

Load combinations used to analyze reactor internals vary, depending on the plant vintage. There are two major categories of plants: those with Mark II or Mark III containments, where hydrodynamic events cause vessel internals loads, and those with Mark I containments, where hydrodynamic effects in the torus do not cause significant loads on the vessel internals.

In the event that load combinations are not specified in the SAR, the set of load combinations shown in Tables 3 and 4 of the report are recommended for Mark I, II and III plants. The staff finds these load combinations reasonable and acceptable.

Allowable Stresses and Design Criteria

Allowable stresses under the above loading combinations will be consistent with the current plant SAR. Allowable stresses shall be based on ASME, Section III, Class 1 allowable stress intensity tables. (Allowable stress intensity tables are found in ASME Section II, Part D in the 1992 and later Code editions.)

The use of ASME Code (in 10 CFR 50.55a) editions and Addenda not yet specifically endorsed by the NRC will need to be evaluated by the NRC on a case-by-case basis. The functional evaluation criteria have also been specified. 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 such as a pipe or support structure can perform its 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. Appendix A of Section 3.9.3. of the Standard Review Plan (SRP) provides specific guidance on this subject for core support structures.

For the design of repair hardware, this requirement is satisfied by implementing the requirements of Section C (Position) and Table I of NUREG-0800. The evaluation of primary and secondary stress limits for the service level A & B is required by the ASME Code. The evaluation of primary plus secondary stresses for the faulted conditions is not required by the Code, but is recommended in these criteria as a means of verifying the functional capability of the passive components.

The repair will be designed to address the potential for flow-induced vibration. The vibratory stresses shall be shown to be less than the endurance limit of the repair and existing materials. Testing may be used as an alternative, or to supplement the vibration analysis. Flow during accident scenarios will be evaluated as well as the effects of increased core flow associated with power uprates. Thermal cycles based on actual plant operating data may be employed if technically justified. Using this thermal cycle information, repaired components will be evaluated for fatigue loading along with other design vibratory loads. All thermal-hydraulic and structural computer codes utilized in the design analysis shall be benchmarked.

4.0 CONCLUSION

The NRC staff has reviewed the BWRVIP-50 report as well as its associated RAI response. The staff finds that the BWRVIP-50 report, as modified and clarified to incorporate the staff's comments above, is acceptable for providing guidance for repairs of the top guide or core plate. This finding, based upon the information submitted in the subject report and RAI response, is consistent with NRC approved methodology. Therefore, the staff has concluded that licensee implementation of the guidelines in BWRVIP-50, as modified, will provide an acceptable repair design criteria of the safety-related components addressed in the BWRVIP-50 document. The modifications stated in the RAI and addressed above should be incorporated in Revision 1 of the BWRVIP-50 report.

Further, pursuant to 10 CFR 50.55a(a)(3)(i), the staff finds that licensee installation of core support plate wedges to structurally replace the lateral resistance provided by the core support rim hold down bolts to be an acceptable alternative to Section XI of the ASME Boiler and Pressure Vessel Code and will provide an acceptable level of quality and safety for the subject safety-related components. Licensees who install the subject wedges are requested to inform the NRC staff by letter at least 90 days prior to installation, or, if they have already installed the wedges, within 90 days of the date of this letter.

The BWRVIP-50 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, Top Guide/Core Plate Repair Design Criteria (BWRVIP-50),” EPRI Report TR-108722, dated May, 1998.

2. C. E. Carpenter, USNRC, to Carl Terry, BWRVIP, “Propriety Request for Additional Information - Review of BWR Vessel and Internals Project Report, Top Guide/Core Plate Repair Design Criteria (BWRVIP-50),” dated April 7, 1999.

3. C. E. Carpenter, USNRC, to Carl Terry, BWRVIP, “Propriety Request for Additional Information - Review of BWR Vessel and Internals Project Report, Top Guide/Core Plate Repair Design Criteria (BWRVIP-50),” dated May 24, 1999.

4. Carl Terry, BWRVIP, to USNRC, “BWRVIP Response to NRC Request for Additional Information on BWRVIP-50,” December 6, 1999.

5. Carl Terry, BWRVIP, to USNRC, “BWRVIP Response to NRC Safety Evaluation on BWRVIP-16 and BWRVIP-19,” December 6, 1999.