November 2, 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, SHROUD SUPPORT AND VESSEL BRACKET REPAIR DESIGN CRITERIA (BWRVIP-52)," EPRI REPORT TR-108720, JUNE 1998 (TAC NO. MA2326)

Dear Mr. Terry:

The NRC staff has completed its review of the Electric Power Research Institute (EPRI) proprietary report TR-108720, "BWR Vessel and Internals Project, Shroud Support and Vessel Bracket Repair Design Criteria (BWRVIP-52)," dated June 1998. Both proprietary and non-proprietary versions of the BWRVIP-52 report were submitted to the U. S. Nuclear Regulatory Commission (NRC) for staff review by letter dated June 26, 1998, and an expanded non-proprietary version of the BWRVIP-52 report was submitted by letter dated May 19, 2000. The BWRVIP-52 report was supplemented by letter dated December 6, 1999, which was in response to the NRC staff's request for additional information (RAI), dated August 13, 1999. The BWRVIP-52 report provides general design acceptance criteria for the permanent and temporary repair of the shroud support structure and vessel internal attachments. These guidelines are intended to maintain the structural integrity and system functionality of the shroud support structure and vessel internal attachments during normal operation and under postulated transient and design basis accident conditions. The BWRVIP provided the BWRVIP-52 report to support generic regulatory efforts related to the repair of the BWR shroud support structure and vessel internal attachments.

The NRC staff has reviewed the BWRVIP-52 report, as well as its associated RAI response, and found, in the enclosed safety evaluation (SE), that the BWRVIP-52 report, as modified and clarified to incorporate the staff's comments in the SE, is acceptable for providing guidance for permanent repairs of the shroud support structure and vessel internal attachments. The staff concludes that licensee implementation of the guidelines in the BWRVIP-52 report, as modified, will provide an acceptable permanent repair design criteria of the safety-related components addressed in the BWRVIP-52 document. The BWRVIP-52 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-52 report. Inspections of the repaired components should be in accordance with the BWRVIP-38 guidance, as approved by the staff.

The staff has concluded that the guidelines provided in the BWRVIP-52 report should be useful to licensees in preparing alternatives pursuant to 10 CFR 50.55a(a)(3) for plant-specific temporary repairs. The staff will consider these guidelines in its review of such submittals.

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

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U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

SAFETY EVALUATION OF THE "BWRVIP VESSEL AND INTERNALS PROJECT, SHROUD SUPPORT AND VESSEL BRACKET REPAIR DESIGN CRITERIA (BWRVIP-52)," EPRI REPORT TR-108720

1.0 INTRODUCTION

1.1 Background

By letter dated June 26, 1998, as supplemented by letters dated December 6, 1999, and May 19, 2000, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the Electric Power Research Institute (EPRI) proprietary Report TR-108720, "BWR Vessel and Internals Project, Shroud Support and Vessel Bracket Repair Design Criteria (BWRVIP-52)," dated June 1998, and a non-proprietary version for NRC staff review. An expanded non-proprietary version of the BWRVIP-52 report was submitted by letter dated May 19, 2000. The BWRVIP-52 report was supplemented by letter dated December 6, 1999, which was in response to the NRC staff's request for additional information (RAI), dated August 13, 1999.

The BWRVIP-52 report provides general design acceptance criteria for the permanent and temporary repair of the shroud support structure and vessel pressure vessel (RPV) internal attachments. These guidelines are intended to maintain the structural integrity and system functionality of the shroud support structure and vessel internal attachments during normal operation and under postulated transient and design basis accident conditions. The BWRVIP provided the BWRVIP-52 report to support generic regulatory efforts related to the repair of the BWR shroud support structure and vessel internal attachments.

1.2. Purpose

The staff reviewed the BWRVIP-52 report, as supplemented, to determine whether its proposed revised 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 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-52 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-52 report.

ENCLOSURE

2.0 SUMMARY OF BWRVIP-52 REPORT

The BWRVIP-52 report addresses the following topics in the following order:

- Configuration and Safety Function - The shroud support structure and vessel internal attachments configurations are described in detail with brief descriptions of each configuration's function and characteristics. Differences among the various models of BWRs and plants are identified. The safety design bases for the shroud support structure and vessel internal attachments are given.

- Scope of Repairs - The scope of the proposed repairs is given, including mechanical repairs, welded repairs and flaw removal.

- 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, physical interfaces with other reactor internals, and design verification. Installation to minimize the critical path in-vessel time and shroud support structure leakage are considered.

- General Design Criteria - The design criteria of the shroud support structure and vessel internal attachments 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. RPV pressure boundary involvement in the repairs is also discussed.

- Structural and Design Evaluation - Terms (e.g., deadweight, hydraulic loads, fuel lift loads, etc.) associated with applied loads on the reactor internals are briefly discussed. 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 repair or cracking of other internals, flow induced vibration and hydraulic loads, repair impact on existing internal components, thermal and radiation effects on repair design, analysis codes, thermal cycles, and corrosion allowance.

- System Evaluation - The following system evaluations are discussed: leakage impact and acceptance criteria for normal operation and accident conditions, leakage evaluation methodology for accident conditions, impact to core flow distribution, 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 welding and fabrication guidelines are also discussed. Pre-installation as-built inspection, installation cleanliness, ALARA considerations, and qualification of critical design parameters are presented.

- Inspection - Inspection of the reactor internal components is addressed in the following topics: inspection access, pre- and post-installation inspection, inservice inspection requirements and vessel inspections.

3.0 STAFF EVALUATION

The BWRVIP-52 report provides the proposed general design acceptance criteria for permanent and temporary repairs of the shroud support structure and vessel internal attachments. 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 shroud support structure and vessel internal attachments during normal operation and under postulated transient and design basis accident conditions for the specified service life of the components.

The core shroud support structure consists of the shroud support plate and cylinder, including the design dependent gussets or legs. The vessel internal attachments include core spray bracket attachments, jet pump riser brace attachments, guide rod bracket attachments, steam dryer support bracket attachments, steam dryer holddown bracket attachments, feedwater sparger bracket attachments, and surveillance sample holder bracket attachments. According to the BWRVIP-52 report, the only safety-related attachments are the core spray piping bracket and jet pump riser brace attachments, along with the shroud support structure. The shroud support also supports the weight of the peripheral fuel bundles and provides lateral restraint during seismic or other dynamic events. The staff notes that the core shroud support structure forms part of the core coolant envelope which is required to provide two-thirds core cooling in BWR/3s through BWR/6s in the event of a recirculation line loss of coolant accident (LOCA).

The NRC staff has reviewed the BWRVIP-52 report and the associated response to the staff's RAI, and finds that the BWRVIP-52 report, as modified and clarified to incorporate the staff's comments, below, is acceptable for providing guidance for permanent repairs of the shroud support structure and vessel internal attachments. Based on the following evaluation, the staff concludes that licensee implementation of the guidelines in the BWRVIP-52 report, as so modified, will provide for licensee usage an acceptable design criteria for the permanent repair of the safety-related components addressed in the BWRVIP-52 report. The BWRVIP-52 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-52 report. Inspections of the repaired components should be in accordance with the BWRVIP-38 guidance, as approved by the staff.

The staff has concluded that the guidelines provided in the BWRVIP-52 report should be useful to licensees in preparing alternatives pursuant to 10 CFR 50.55a(a)(3) for plant-specific temporary repairs. The staff will consider these guidelines in its review of such submittals.

3.1 BWRVIP Response to Staff's RAI

The staff's August 13, 1999, RAI, provided one open item. The BWRVIP, in its letter of December 6, 1999, addressed this item, which is discussed below.

<u>Item 1</u>: In Section 9.3, Welding and Fabrication, it is stated that the susceptibility to intergranular attack of all welded XM-19 or Grade 300 series austenitic stainless steel materials shall be determined in accordance with either ASTM A 262, Practice A or E, and that the materials tested in accordance with the ASTM A 262, Practice A, requirements

must have no more than five percent of the total grain boundary length exhibiting ditching. ASTM A 262, Practice A, is designed as a screening test. If the results of Practice A test are not acceptable, Practice E test may be used for acceptance. The staff notes that the acceptance criteria specified in Section 9.3 for ASTM A 262, Practice A test, is different from that provided in the ASTM standards. In the ASTM standards, for A 262 Practice A test, the materials are considered not acceptable if its etched structures exhibit ditch structure. The ditch structure is defined as one or more grains completely surrounded by ditches. Provide a comprehensive discussion, including the supporting data, regarding how the acceptance criteria defined in Section 9.3 for ASTM A 262, Practice A, were established and the bases for its acceptability to replace that provided in ASTM Standards.

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

3.2 Systems Evaluation

The BWRVIP-52 report discussed the potential for leakage when evaluating repairs of the core shroud support structure. Leakage from the shroud support structure to the annulus could

affect the design performance of the jet pumps by adversely affecting the normal flow of water out of the jet pump diffusers. Additionally, post accident reflooding capability may be impacted by leakage through the shroud support structure. The staff notes that leakage is not a concern for vessel internal attachments.

The BWRVIP-52 report provided guidance on evaluating the impact of leakage through the shroud support structure during normal and accident conditions. The BWRVIP-52 report recommended that the acceptability of the leakage be based on plant specific analyses of the reduction in core flow for normal operations and the increase in peak clad temperature (PCT) for accident conditions. The BWRVIP-52 report recommended that the evaluation of the quantity of leakage through the shroud support structure be based on the system temperature, pressure and flow conditions that are consistent with the licensee's existing licensing basis LOCA analysis.

The total leakage should also account for other leakage sources such as a cracked shroud, core spray piping flaws, repair modifications, etc. The BWRVIP-52 report also provided recommendations on evaluating core flow distribution, emergency operating procedure calculations, and power uprates.

The staff has reviewed the guidance provided in the BWRVIP-52 report and finds it consistent with other existing guidance on the evaluation of potential leakage sources. The staff has concluded that the guidance in BWRVIP-52 should be sufficient for permanent repairs of the shroud support structure. 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 applied loads on the shroud support structure and vessel internal attachments generally are due to the following: deadweight, differential pressure, hydraulic phenomena, seismic inertia, seismic anchor displacements, fuel lift, LOCA, safety relief valve (SRV) opening, 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. The staff finds the consideration of these loads to be appropriate in the evaluation of the repairs of the shroud support and vessel internal attachments.

The fuel bundles in the reactor core are supported horizontally by the fuel support casting and core plate at their lower end, and by the top guide at their upper end. In the vertical direction, the core plate supports the weight of about two dozen fuel bundles in peripheral fuel supports. The remainder of the fuel is supported by a system composed of the fuel support casting, the control rod guide tube, the upper portion of the control rod drive housing, and the RPV bottom head. 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, peripheral fuel bundles could move relative to their supports, and, if sufficient to exceed take-up tolerances, could unseat. Upon reseating on the core support plate, impact forces may result. Fuel lift loads are applicable to the shroud support structure and not the vessel internal attachments.

During a seismic or other dynamic event, it is possible in some plants for the fuel to lift which creates a vertical load caused by the friction of the fuel against the inside of the top guide cell. This “fuel lift load” should be considered, if appropriate. 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 response from natural phenomena (safe shutdown earthquake - SSE), and from other events, such as the design-basis accident and SRV discharge.

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 and pressures during transient conditions are based on the information provided in the applicable vessel thermal cycle diagram. Displacements during normal operation, operating transients (including pump transients), and loss of coolant accident, need to be considered.

Depending upon the component under consideration, the reactor water level during normal, operating conditions may be above the highest elevation of the component. Therefore, during a seismic or other dynamic event, the component would vibrate in the submerged condition. This requires the addition of a hydrodynamic mass term for the analysis of dynamic loads.

<u>Handling loads</u> associated with the removal/ replacement, or misalignment of reactor internals may exist. For example, the guide rod bracket experiences handling loads in the event the shroud head or steam dryer are misaligned. Consideration shall be given to handling loads if they are determined to exist.

Service Level A loads 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 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 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.

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, UFSAR, and/or related licensing basis documentation. Typically, Section 3.9 of the FSAR or UFSAR contains the necessary information on this subject including, for some plants, hydrodynamic loads (i.e., "new loads") and/or AP loads.

For Mark II and III plants, the method for load combination was specified at the time that the loads caused by hydrodynamic events were defined and labeled “new loads.” 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 Service Levels A and B is required by the ASME Code. The evaluation of primary plus secondary stresses and displacements for the faulted condition is not required by the ASME Code, but is recommended in these criteria as a means of verifying the functional capability of any passive components. A set of load combinations, in lieu of plant-specific documentation, is shown in Table 7-2 of the report. The staff finds these recommended load combinations in conformance with accepted guidelines for combining loads, and therefore acceptable.

A summary of the recommended ASME Code design guidance is provided in Table 6-1 of the report. For the design of special hardware such as support clamps, tie rods, and configurations not addressed specifically in ASME Code Section III, the requirements of Subsection NG are recommended to be used. These requirements are also recommended for the design of shroud support structures, and core shrouds (including pads and attachments). These items are included since they potentially affect the subject repairs. Based on its review, the staff finds the recommended design guidance for repair of components reasonable and acceptable.

In some cases, other internal components (e.g., jet pumps) may be cracked or have been repaired. In such cases, the appropriate loads (including anchor displacements and dynamic response characteristics) which were calculated as part of the repair design or flaw evaluation analyses will be used in analyzing affected shroud support or vessel attachment repairs.

The repair will be designed to address the potential for FIV and hydraulic loads on the affected component, and to keep FIV to a minimum. As changes are made to vessel attachments that affect the natural frequency of any assembly, the contribution made by the FIV loads can significantly increase. The vibratory stresses shall be shown to be less than the endurance limit of the repair and affected component materials. Forces to be considered include: (1) pump vane passing frequency acting within the repaired or attached component, (2) turbulent fluid flow within a repaired or attached component, (3) cross flow over the repaired or attached component, especially in the annulus between the RPV and shroud, and, (4) leakage flow from a repaired or attached component. Testing may be used as an alternative, or to supplement

the vibration analysis. Flow during accident scenarios shall be considered, as well as the effects of increased core flow associated with power uprate.

Repaired components and the repair hardware will be evaluated for fatigue loading along with any other design vibratory loads. Thermal cycles based on actual plant operating data may be used, with appropriate justification, for the fatigue analysis

Based on the staff's review as discussed above, the staff finds the recommended applied loads, service level conditions and load combinations for the structural design evaluation of the shroud support and vessel bracket repair, in conformance with ASME Code, Section III requirements and therefore acceptable.

3.4 General Comment

In order to be consistent with other BWRVIP repair procedures, such as the BWRVIP-16 and BWRVIP-19 reports, the following requirements should be added or changed in Section 9.1, Materials, of the BWRVIP-52 report:

1. 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.
2. 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.
3. The following statement should be included in the BWRVIP-52 report in this Section. "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."

4.0 CONCLUSION

The NRC staff has reviewed the BWRVIP-52 report and the associated response to the staff's RAI, and finds that the BWRVIP-52 report, as modified and clarified to incorporate the staff's comments, above, is acceptable for providing guidance for permanent repairs of the shroud support structure and vessel internal attachments. 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-52 report, as modified, will provide licensees an acceptable design criteria for the permanent repair of the safety-related components addressed in the BWRVIP-52 report. The modifications stated in the RAI and addressed above should be incorporated in Revision 1 of the BWRVIP-52 report.

The BWRVIP-52 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-52 report. Inspections of the repaired components should be in accordance with the BWRVIP-38 guidance, as approved by the staff.

The staff has concluded that the guidelines provided in the BWRVIP-52 report should be useful to licensees in preparing alternatives pursuant to 10 CFR 50.55a(a)(3) for plant-specific temporary repairs. The staff will consider these guidelines in its review of such submittals.

5.0 REFERENCES

1. Carl Terry, BWRVIP, to USNRC, "BWR Vessel and Internals Project, Shroud Support and Vessel Bracket Repair Design Criteria (BWRVIP-52)," EPRI Report TR-108720, dated June 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-52," December 6, 1999.

4. C. E. Carpenter, USNRC, to Carl Terry, BWRVIP, "Final Safety Evaluation of the BWRVIP Vessel and Internals Project, Internal Core Spray Piping and Sparger Replacement Design Criteria (BWRVIP-16), and of the BWRVIP Vessel and Internals Project, Internal Core Spray Piping and Sparger Repair Design Criteria (BWRVIP-19), dated August 10, 2000.