October 28, 2000

Carl Terry, BWRVIP Chairman
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SUBJECT: SAFETY EVALUATION OF THE "BWRVIP VESSEL AND INTERNALS PROJECT, JET PUMP REPAIR DESIGN CRITERIA (BWRVIP-51)," EPRI REPORT TR-108718, MAY 1998 (TAC NO. MA1927)

Dear Mr. Terry:

The NRC staff has completed its review of the Electric Power Research Institute (EPRI) proprietary report TR-108718, "BWR Vessel and Internals Project, Jet Pump Repair Design Criteria (BWRVIP-51)," dated May 1998. Both proprietary and non-proprietary versions of the BWRVIP-51 report were submitted to the U. S. Nuclear Regulatory Commission for staff review by letter dated May 14, 1998, and an expanded non-proprietary version of the BWRVIP-51 report was submitted by letter dated March 7, 2000. The BWRVIP-51 report was supplemented by a letter dated December 6, 1999, which was in response to the NRC staff's request for additional information (RAI), dated April 8, 1999. The BWRVIP-51 report provides general design acceptance criteria for the permanent and temporary repairs of jet pump assemblies. These guidelines are intended to maintain the structural integrity and system functionality of the jet pump assemblies during normal operation and under postulated transient and design basis accident conditions. BWRVIP provided the BWRVIP-51 report to support generic regulatory efforts related to the repair of BWR jet pumps.

The NRC staff has reviewed the BWRVIP-51 report, as well as its associated RAI response, and finds, in the enclosed safety evaluation (SE), that the BWRVIP-51 report, as modified and clarified to incorporate the staff's comments in the SE, is acceptable for providing guidance for permanent repairs of the jet pump assemblies. The staff concludes that licensee implementation of the guidelines in the BWRVIP-51 report, as modified, will provide an acceptable permanent repair design criteria of the safety-related components addressed in the BWRVIP-51 report. The staff will consider the guidelines provided in the BWRVIP-51 report for temporary repairs as a part of its determination of an alternative for plant-specific temporary repairs pursuant to 10 CFR 50.55a(a)(3).

The BWRVIP-51 report is considered by the staff to be acceptable for licensee usage, as modified and approved by the staff, at any time during either the current operating term or during the extended license period. The modifications stated in the RAI and addressed above should be incorporated in a revision of the BWRVIP-51 report.

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

Sincerely

Jack R. Strosnider, Director
Division of Engineering
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Enclosure: As stated

cc: See next page

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SAFETY EVALUATION OF THE "BWRVIP VESSEL AND INTERNALS PROJECT, JET PUMP REPAIR DESIGN CRITERIA (BWRVIP-51)," EPRI REPORT TR-108718

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 the Electric Power Research Institute (EPRI) proprietary Report TR-108718, "BWR Vessel and Internals Project, Jet Pump Repair Design Criteria (BWRVIP-51)," dated May 1998, for NRC staff review. Both proprietary and non-proprietary versions of the BWRVIP-51 report were submitted to the U. S. Nuclear Regulatory Commission for staff review by letter dated May 14, 1998, and an expanded non-proprietary version of the BWRVIP-51 report was submitted by letter dated March 7, 2000. The supplemental information was in response to the staff's request for additional information (RAI), dated April 8, 1999.

The BWRVIP-51 report provides general design acceptance criteria for the permanent and temporary repairs of jet pump assemblies. These guidelines are intended to maintain the structural integrity and system functionality of the jet pump assemblies during normal operation and under postulated transient and design basis accident conditions. The BWRVIP provided the BWRVIP-51 report to support generic regulatory efforts related to the repair of jet pumps.

1.2. Purpose

The staff reviewed the BWRVIP-51 report, as supplemented, to determine whether its proposed guidance, along with the response to the staff's RAI, will provide an acceptable repair design criteria of the subject safety-related reactor pressure vessel (RPV) internal components. The review assessed the design objectives, structural evaluation, system evaluation, materials, fabrication and installation considerations, as well as the required inspection and testing requirements.

1.3. Organization of this Report

Because the BWRVIP-51 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-51 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-51 report.

ENCLOSURE

2.0 SUMMARY OF BWRVIP-51 REPORT

The BWRVIP-51 report addresses the following topics in the following order:

- Configuration and Safety Function - The jet pump configurations are described in detail with brief descriptions of each configuration's function and characteristics. Differences among the various models of BWRs (BWR/3 through BWR/6) are identified. BWR/2 plants do not contain jet pumps. The safety design bases for the jet pump are: For post-accident core reflooding, the jet pump assembly assures reflooding to no less than 2/3 core height. An additional safety function of the jet pump assembly at some plants is to provide a flow path for LPCI flow into the core. 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 are given.
- 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, installation, jet pump performance and leakage, and design verification.
- Design Criteria - The design criteria of the jet pumps are presented. In summary, all repair designs should meet the individual plant safety analysis report (SAR) as well as NRC and ASME Code established methodology for reactor pressure vessel (RPV) and internals mechanical design.
- Structural and Design Evaluation - Terms (e.g., hydraulic loads, fuel lift loads, etc.) associated with applied loads on the reactor vessel are briefly discussed. The various events and operational service level conditions are also considered to ensure the repairs do not inhibit safety and operational functions of the internal components. Other structural and design topics addressed are: load combinations, allowable stresses, consideration of shroud repair or cracking, flow induced vibration, impact on existing internal components, radiation effects on repair design, analysis codes, thermal cycles, and corrosion allowance and restrainer bracket adjusting screw gap evaluation.
- 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, internal jet pump pressure drop, flow distribution impact, 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. The effect of water chemistry and neutron flux on any materials used in the repair as well as crevices are discussed. 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 jet pumps are addressed in the following topics: inspection access, pre- and post-installation inspection and testing, which includes monitoring by plant operators.

3.0 STAFF EVALUATION

The BWRVIP-51 report provides the general design acceptance criteria for permanent and temporary repairs of jet pump assemblies. 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 jet pump assemblies during normal operation and under postulated transient and design basis accident conditions for the specified service life of the components.

Jet pump assemblies are classified as safety related components in BWR/3 - BWR/6 plants; however, they are not part of the BWR/2 plant design. The structural integrity of the jet pump assembly is relied upon for assuring the ability to reflood the core, up to two-thirds core height, following a design basis accident (DBA) loss of coolant accident (LOCA). An additional safety function of the jet pump assemblies is to provide a flow path for low pressure coolant injection (LPCI) flow into the core. This safety function is applicable to all BWR/3s and BWR/4s except Hope Creek and Limerick Units 1 and 2. For these three plants and the BWR/5s and BWR/6s, LPCI injection occurs through the LPCI coupling which provides a flow path between the reactor vessel nozzle thermal sleeve and the core shroud.

The NRC staff has reviewed the BWRVIP-51 report and the associated response to the staff's RAI, and finds that the BWRVIP-51 report, as modified and clarified to incorporate the staff's comments, below, is acceptable for providing guidance for permanent repairs of the jet pump assemblies. Based on the following evaluation, the staff concludes that licensee implementation of the guidelines in the BWRVIP-51 report, as so modified, will provide an acceptable repair design criteria of the safety-related components addressed in the BWRVIP-51 report for licensee usage. The BWRVIP-51 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-51 report. Inspections of the repaired components should be in accordance with the BWRVIP-41 guidance, as approved by the staff.

The staff will consider the guidelines provided in the BWRVIP-51 report for temporary repairs as a part of its determination of an alternative for plant-specific temporary repairs pursuant to 10 CFR 50.55a(a)(3).

3.1 BWRVIP Response to Staff's RAI

The staff's April 8, 1999, RAI provided six open items. The BWRVIP, in its letter of December 6, 1999, addressed these items, which are discussed below.

RAI Item 1: The information related to Information Notice (IN) 93-101 and GE RICSIL No. 065 should be added to the discussion included under Section 4.0. Specifically, on September 13, 1993, a BWR/6 experienced a scram and upon restart, discovered jet pump flow differential pressure anomalies. The cause of the anomalies was that a jet pump holddown

beam had cracked and failed and the assembly had lifted several inches. This failure prompted the NRC staff to issue IN 93-101, "Jet Pump Hold-Down Beam Failures," on December 17, 1993. The General Electric Co. (GE) issued RICSIL No. 065, "Jet Pump Beam Cracking, " on December 3, 1993, which called for the replacement of all jet pump holddown beams from the group of beams described in SIL No. 330 with more than eight years of service.

BWRVIP Response to Item 1: Details of specific failure mechanisms have typically been included in the BWRVIP Inspection and Flaw Evaluation Guidelines rather than in the Repair Design Criteria. Since RICSIL 065 is discussed in the Jet Pump I&E Guideline (BWRVIP-41), the BWRVIP proposes that it not be discussed in the Jet Pump Repair Design Criteria.

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

RAI Item 2: Existing EPRI guidelines recommend that threaded fasteners not be lock welded. In Section 5.6, the BWRVIP-51 report recommends lock welding [fasteners] or [fitting them] with lock welded retainers. Justify this recommendation.

BWRVIP Response to Item 2: Longstanding industry experience has shown that both mechanical and welded locking devices are reliable, effective means of preventing loose parts. The only consideration here is the reliability of the field installation equipment.

Staff's Evaluation of BWRVIP Response to Item 2: The staff requires 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. The staff requests that the BWRVIP-51 report be modified to require licensees to determine the weldability of the materials to be welded.

RAI Item 3: What is the basis for the 0.003 inch corrosion allowance for stainless steel exposed to the primary coolant in Section 7.11? In as 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 Item 3: 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 of BWRVIP Response to Item 3: The staff finds that BWRVIP's response adequately addressed this item.

RAI Item 4: Clarifications should be made to the BWRVIP-51 report, so that individual licensees will make plant-specific submittals for the following instances:

RAI Item 4-1: 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 Item 4-1: 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 Ill and the Owner's Original Design Specification. This would include the applicable Code materials requirements. If the Code is not met, a relief request to allow a technical alternative to the Code pursuant to 10 CFR 50.55a must be requested.

Section XI rules for repair and replacement also applies to components that were not designed to Section III but are classified by the Owner as "Welded Core Support Structures" and are subject to inspection under Section XI Category B-N-1 from Table IWB-2500-1. These components are to be repaired or replaced in accordance with the Owner's original Design Specification and Construction Code. NRC allows later approved versions of Section III to be used. If this requirement is not met, approval of a technical alternative must be sought pursuant to 10 CFR 50.55a.

Repair and replacement designs for plants which were not designed and constructed in accordance with ASME Section III (and components not subject to Section Xl) 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.

Staff's Evaluation of BWRVIP Response to Item 4-1: The staff finds that, with the inclusion of the third paragraph of the BWRVIP's response to Item 4-1, above, into Section 9.1.2 of the BWRVIP -51 report, this response is acceptable.

RAI Item 4-2: 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.

BWRVIP Response to Item 4-2: 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 4-1 above.

Staff's Evaluation BWRVIP Response to Item 4-2: The staff finds that BWRVIP's response adequately addressed this item.

RAI Item 4-3: 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.

BWRVIP Response to Item 4-3: 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.

Staff's Evaluation of BWRVIP Response to Item 4-3: The staff finds that BWRVIP's response adequately addressed this item.

RAI Item 5: 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 Item 5: Details of the CIB heat treatment condition for alloy X-750 and the basis for its use in the BWR environment are found in ERPI Document NP-7032, "Material Specification for Alloy X-750 for Use in LWR Internal Components (Revision 1)," which has previously been reviewed by NRC as part of the review of the Shroud Repair Design Criteria (BWRVIP-02).

Staff's Evaluation BWRVIP Response to Item 5: 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 shall be included in the BWRVIP-51 report. With the inclusion of these two statements, the staff finds that BWRVIP's response adequately addressed this item.

RAI Item 6: Clarification is needed for the case of a licensee which has been 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 Item 6: 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 BWRVIP Response to Item 6: The staff has stated many times in the past that meaningful inspections should be performed if conditions change and accessibility becomes possible. Therefore, the BWRVIP-51 report should be modified to include language stating that licensees should plan on inspecting welds made accessible during the course of repairs performed on the subject safety-related internal components, or develop a technical justification as to why these welds should not be inspected during this accessibility period.

3.2 Systems Evaluation

Jet pump assemblies are classified as safety related components in BWR/3 - BWR/6 plants. Jet pump assemblies are not part of the BWR/2 plants design. The structural integrity of the jet pump assembly is relied upon for assuring the ability to reflood the core, up to two-thirds core height, following a design basis accident (DBA) loss of coolant accident (LOCA). An additional safety function of the jet pump assemblies is to provide a flow path for low pressure coolant injection (LPCI) flow into the core. This safety function is applicable to all BWR/3s and BWR/4s except Hope Creek and Limerick Units 1 and 2. For these three plants and the BWR/5s and BWR/6s, LPCI injection occurs through the LPCI coupling which provides a flow path between the reactor vessel nozzle thermal sleeve and the core shroud.

The BWRVIP-51 report discussed the importance of maintaining the amount of leakage from a jet pump repair to a minimum. Leakage from the jet pump flow path 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, repairs should not impact the design performance of the jet pumps by obstructing the flow of water in and around the jet pumps.

The BWRVIP-51 report provided guidance on evaluating the impact of leakage through the jet pumps during normal and accident conditions. The BWRVIP-51 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 for BWR/3 and BWR/4s. The BWRVIP-51 report recommended that the evaluation of the quantity of leakage through the jet pump assemblies 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-51 report also provided recommendations on evaluating internal jet pump pressure drop, flow distribution, emergency operating procedure calculations, and power uprates. The staff has reviewed the guidance provided in the BWRVIP-51 report and finds it consistent with other existing guidance on the evaluation of potential leakage sources. The staff has concluded that the guidance in the BWRVIP-51 report should be sufficient for permanent or temporary repairs of the jet pump assemblies. 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 loads and load combinations used in analyzing a repair design will be consistent with the current licensing basis of the plant. In the event that adequate information is not contained in the licensing documents, the loads and load combinations presented in Sections 7.1 and 7.3 are proposed to be used. The staff has reviewed these loads and load combinations and a general discussion and evaluation of these loads with specific application to the jet pumps is as follows.

The applied loads on the shroud support structure and vessel internal attachments 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 accidents (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 that results in oscillation. The resulting hydrodynamic pressure oscillations may cause dynamic excitations of the structure and contained equipment.

Main vent chugging (CHG) loads are induced during a small break accident (SBA), 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, produce hydrodynamic pressure oscillations on the pool boundary for Mark II pressure suppression containment, and on the weir wall and pool boundary for Mark III containment. 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 as postulated pipe rupture causes 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 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.

3.3.1 Service Level Conditions

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, 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 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.

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.

3.3.2 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") and/or AP loads. In the event that adequate definition of load combinations is not contained in the plant licensing basis documentation, the following load combinations may be used.

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 containment, where hydrodynamic events cause vessel internals loads, and those with Mark I containment, 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 recommend for Mark I, II and III plants. The staff finds these load combinations reasonable and acceptable.

3.3.3 Allowable Stresses and Design Criteria

A summary of the recommended ASME design guidance is as follows.

Repair of the existing jet pump piping components will conform to the ASME Section III, Subsection NC requirements. Subsections NB or ND can also be used with the appropriate Code allowables. Special repair hardware such as supports, clamps and tie rods will conform to Subsection NG requirements. The evaluation of the connections to the core shroud and revised loads will also be performed in accordance with Subsection NG requirements. The evaluation of connections to the reactor pressure vessel and nozzles including pads will be in accordance with the Subsection NB requirements. These, and recommended ASME design guidance for material properties, non-destructive examinations, materials, fabrication, welding procedures and personnel qualifications are summarized in Table 2.0 of the report. The staff finds these recommended design guidelines reasonable and acceptable.

The repair will be designed to address the potential for <u>flow induced vibration</u>. 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. <u>Thermal cycles</u> 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-51 report, as modified by the associated RAI response, and finds that the BWRVIP-51 report, as modified and clarified to incorporate the

staff's comments above, is acceptable for providing guidance for permanent repairs of the jet pump assemblies. Therefore, the staff concludes that licensee implementation of the guidelines in the BWRVIP-51 report, as modified, will provide an acceptable repair design criteria of the safety-related components addressed in the BWRVIP-51 report. The staff will consider the guidelines provided in the BWRVIP-51 report for temporary repairs as a part of its determination of an alternative for plant-specific temporary repairs pursuant to 10 CFR 50.55a(a)(3).

The BWRVIP-51 report is considered by the staff to be acceptable for licensee usage, as modified and approved by the staff, at any time during either the current operating term or during the extended license period. The modifications stated in the RAI and addressed above should be incorporated in a revision of the BWRVIP-51 report.

5.0 REFERENCES

1. Carl Terry, BWRVIP, to USNRC, "BWR Vessel and Internals Project, Jet Pump Repair Design Criteria (BWRVIP-51)," EPRI Report TR-108718, 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, Jet Pump Repair Design Criteria (BWRVIP-51)," dated April 8, 1999.
3. Carl Terry, BWRVIP, to USNRC, "BWRVIP Response to NRC Request for Additional Information on BWRVIP-51," December 6, 1999.
4. Carl Terry, BWRVIP, to USNRC, "BWRVIP Response to NRC Safety Evaluation on BWRVIP-16 and BWRVIP-19," December 6, 1999.