October 18, 2001

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

SUBJECT: ACCEPTANCE FOR REFERENCING OF EPRI PROPRIETARY REPORT TR-113596, "BWR VESSEL AND INTERNALS PROJECT, BWR REACTOR PRESSURE VESSEL INSPECTION AND FLAW EVALUATION GUIDELINES (BWRVIP-74)" AND APPENDIX A, "DEMONSTRATION OF COMPLIANCE WITH THE TECHNICAL INFORMATION REQUIREMENTS OF THE LICENSE RENEWAL RULE (10 CFR 54.21)"

Dear Mr. Terry:

By letter dated September 21, 1999, as supplemented by letter dated March 7, 2000, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted both proprietary and non-proprietary versions of the report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)," EPRI Report TR-113596, for staff review and approval. By letter dated March 7, 2000, the BWRVIP provided an expanded non-proprietary version of the subject report for use in license renewal reviews.

The BWRVIP-74 report provides generic guidelines intended to present the appropriate inspection and flaw evaluation recommendations to assure safety function integrity of the reactor pressure vessel (RPV) components during both the current operating term and the license renewal period. The components addressed include the vessel shell, top and bottom heads, closure flanges and studs, support skirts, nozzles, safe ends, penetrations, internal and external attachments, in-core monitor housings, control rod drive (CRD) stub tubes, and pressure boundary portions of CRD housings.

As documented in the attached license renewal (LR) final safety evaluation report (FSER), the NRC staff has completed its review of the BWRVIP-74 report. As indicated in the LR FSER, the staff finds the BWRVIP-74 report acceptable for licensees participating in the BWRVIP to reference in a LR application to the extent specified and under the limitations delineated in the LR FSER. In order for licensees participating in the BWRVIP to rely on the report, they shall commit to the accepted aging management programs (AMPs) defined therein, and complete the action items described in the LR FSER. By referencing the BWRVIP-74 report and the AMPs in it, and completing the action items, an applicant will provide sufficient information for the staff to make a finding that there is reasonable assurance that the applicant will adequately manage the effects of aging so that the intended functions of the reactor vessel components covered by the scope of the report will be maintained consistent with the current licensing basis during the period of extended operation.

The staff does not intend to repeat its review of the matters described in the report, and found acceptable in the FSER when the report appears as a reference in license renewal applications, except to ensure that the material presented applies to the specified plant.

In accordance with the procedures established in NUREG-0390, “Topical Report Review Status,” the staff requests that the BWRVIP publish the accepted version of the BWRVIP-74 report within 90-days after receiving this letter. In addition, the published version shall incorporate this letter and the FSER between the title page and the abstract.

To identify the version of the report that was accepted by the staff, the BWRVIP shall include “A” following the topical report number (e.g., BWRVIP-74-A).

Sincerely,

/***RA***/

Christopher I. Grimes, Branch Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 704

Enclosure: Final Safety Evaluation Report

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FINAL LICENSE RENEWAL SAFETY EVALUATION REPORT

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FINAL SAFETY EVALUATION OF EPRI PROPRIETARY REPORT TR-113596

"BWR VESSEL AND INTERNALS PROJECT, BWR REACTOR PRESSURE VESSEL

INSPECTION AND FLAW EVALUATION GUIDELINES (BWRVIP-74)"

AND COMPLIANCE WITH THE LICENSE RENEWAL RULE (10 CFR PART 54)

1.0 INTRODUCTION

1.1 Background

By letter dated September 21, 1999, as supplemented by letter dated March 7, 2000, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted both proprietary and non-proprietary versions of the Electric Power Research Institute (EPRI) report TR-113596, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)," dated September 1999, for staff review. The March 7, 2000, letter provided an expanded non-proprietary version of the EPRI TR-113596 report for use in license renewal reviews.

The BWRVIP-74 report provides generic guidelines intended to present the appropriate inspection and flaw evaluation recommendations to assure safety function integrity of the reactor pressure vessel (RPV) components during both the current operating term and the license renewal period. The components addressed include: vessel shell, top and bottom heads, closure flanges and studs, support skirts, nozzles, safe ends, penetrations, internal and external attachments, in-core monitor housings, control rod drive (CRD) stub tubes, and pressure boundary portions of CRD housings.

Appendix A, "BWR Reactor Pressure Vessel Components Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)," describes the intended function of the RPV components including the vessel shell, top and bottom heads, closure flanges and studs, support skirts, nozzles, safe ends, penetrations, internal and external attachments, in-core monitor housings, CRD stub tubes, and pressure boundary portions of CRD housings.

1.2 Purpose

The staff has reviewed the BWRVIP-74 report, including Appendix A and Appendix B, "Bounding [Upper-Shelf Energy] USE Analysis for Plant Licensing Renewal (Up to 60 Years / 54 Effective Full Power Years [EFPY])," to determine whether its guidance will provide acceptable levels of quality for inspection of the RPV components within the scope of the report during the period of extended operation. The staff also considered compliance with the License Renewal (LR) Rule in order to allow applicants for renewal the option of incorporating the BWRVIP-74 guidelines by reference in a plant-specific integrated plant assessment (IPA) and associated time-limited aging analyses (TLAA).

Enclosure

Section 54.21 of the LR Rule requires, in part, that each LR application contain an IPA and an evaluation of TLAA. The IPA shall identify and list those structures and components subject to an aging management review (AMR) and demonstrate that the effects of aging will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. In addition, 10 CFR 54.22 requires that each LR application include any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation as part of the LR application.

If a LR applicant participating in the BWRVIP confirms that the BWRVIP-74 report applies to it and that the results of the Appendices A and B, IPA and TLAA evaluations are in effect at its plant, then no further review by the NRC staff of the issues described in the documents is necessary, except as specifically identified by the staff. With this exception, the LR applicant may rely on the BWRVIP-74 report for the demonstration required by Section 54.21(a)(3) with respect to the components and structures within the scope of the report. Under these circumstances, the NRC staff intends to rely on the evaluation in this LR final safety evaluation report (FSER) to make the findings required by 10 CFR 54.29 with respect to a particular application, except as necessary to ensure that the BWRVIP-74 report's conclusions apply to the specified plant.

It should be noted that the staff has previously accepted for referencing in a LR application the following BWRVIP reports: BWRVIP-18, -26, -25, -27, -38, -41, -42, -47, -48, and BWRVIP-49. Unless otherwise stated below, the findings of these reports remain valid.

This LR safety evaluation (SE) specifically reviews the applicability of EPRI Report TR-105697, "BWR RPV Shell Weld Inspection Recommendations (BWRVIP-05)," September 1995, for applicant referencing. The applicability of the BWRVIP-01, BWRVIP-05 and BWRVIP-63 reports for LR will be evaluated in the staff's review of the BWRVIP-76 report. Unless otherwise stated in the staff's BWRVIP-76 LR FSER, the staff's findings for the BWRVIP-76 report will be applicable to the BWRVIP-74 report and will not obviate the following conclusions.

1.3 Organization of this Report

Because the BWRVIP-74 report is proprietary, this SE was written so as not to unnecessarily repeat information contained in the propriety portions of the report. The staff does not discuss in any detail the proprietary provisions of the guidelines nor the parts of the guidelines it finds acceptable. A brief summary of the contents of the BWRVIP-74 report is given in Section 2.0 of this SE, with the NRC staff's evaluation presented in Section 3.0. The conclusions are summarized in Section 4.0. The presentation of the evaluation is structured according to the organization of the BWRVIP-74 report.

2.0 SUMMARY OF THE BWRVIP-74 REPORT

The BWRVIP-74 report and its Appendices contain a generic evaluation of the management of the effects of aging of the subject safety-related RPV components such that their intended functions will be maintained consistent with the CLB for the period of extended operation. This evaluation applies to BWR LR applicants who have committed to implementing the BWRVIP-74 report and want to incorporate the report and the Appendices by reference into a plant-specific IPA and associated TLAA.

The BWRVIP-74 report contains Appendices A and B. Appendix A describes how the report provides the necessary information to comply with the technical information requirements pursuant to Paragraph 54.21 (a) and (c), and 54.22, and the NRC's findings under 54.29 (a) of the LR Rule, 10 CFR Part 54. Appendix B provides an equivalent margins analysis to establish the minimum Charpy Upper-Shelf-Energy (USE) limits for BWR/2-6 vessel beltline materials for 54 EFPY of operation.

2.2 Identification of Structures and Components Subject to an Aging Management Review

10 CFR 54.21(a)(1) requires that an IPA identify and list those structures and components within the scope of LR that are subject to an AMR. Structures and components subject to an aging management review are those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are also referred to as "passive" and "long-lived" structures and components, respectively.

All the components considered to make up the vessel or that are attached to the vessel are passive and long-lived and are subject to AMR.

2.3 Effects of Aging

BWR Reactor Pressure Vessel Industry Report, "BWR RPV License Renewal Industry Report, Revision 1," and the resolution to the NRC's questions on the Industry report are used to identify the aging mechanisms for the vessel components. Aging mechanisms are the cause of the aging effects. The BWRVIP-74 report used NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," dated October 1996, to correlate the aging effects and their associated aging mechanisms. Using these reports, the BWRVIP-74 report concluded that neutron embrittlement, fatigue crack initiation and growth, and stress corrosion crack (SCC) initiation and growth are the only aging effects that require aging management review for RPV components.

Table 3-1 in the report identifies the age-related degradation mechanism for each BWR reactor vessel component. A discussion of the potential locations degraded by these mechanisms is presented for each of the vessel components in subsections 3. through 3.3 of the report.

Aging mechanisms determined not to be relevant to the RPV are irradiation-assisted stress-corrosion cracking (IASCC), SCC of low alloy steel, general corrosion, flow-accelerated corrosion (FAC) and thermal embrittlement.

2.4 Aging Management Programs

10 CFR 54.21(a)(3) requires, for each component identified, that the applicant demonstrate that the effects of aging will be adequately managed so that the intended function will be maintained consistent with the CLB for the period of extended operation.

The effects of neutron embrittlement leading to reduced material toughness, cyclic loading on fatigue, and SCC, are degradation mechanisms for RPV components that require aging management for LR. These aging effects will be managed by on-going comprehensive inspection programs that are summarized in Section 4 of the report.

The inspection program includes examinations required by the ASME Code and augmented examinations imposed by the NRC or as recommended in the various BWRVIP reports.

2.5 Time-Limited Aging Analyses (TLAA)

10 CFR 54.21(1)(c) requires that each LR application contain an evaluation of TLAA as defined in 10 CFR 54.3, and that the applicant shall demonstrate:

(i) the analyses remain valid for the period of extended operation,

(ii) the analyses have been projected to the end of the period of extended operation, or

(iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

TLAA are those licensee calculations and analyses that:

(1) involve the RPV components,

(2) consider the effects of aging,

(3) involve time-limited assumptions defined by the current operating term,

(4) were determined to be relevant by the licensee in making a safety determination,

(5) involve conclusions or provide the basis for conclusions related to the capability of the RPV components to perform their intended function, and

(6) are contained or incorporated by reference in the CLB.

If a plant-specific analysis identified by an applicant meets all six criteria above, the analysis will be considered a TLAA for LR and evaluated by the applicant.

The TLAAs identified in the report are pressure-temperature curves, equivalent margins analysis for RPV materials with Charpy USE less than 50 ft-lb, fatigue, and a material evaluation to provide the technical basis supporting the elimination of RPV circumferential welds from inservice inspection.

For corrosion allowance, which was calculated based on a corrosion rate for unclad main steam nozzles, updated analyses have established that the original corrosion allowance is generically valid for the LR term. Therefore, the report concludes that there is no need for an associated TLAA.

3.0 STAFF EVALUATION

The staff has reviewed the BWRVIP-74 report and its Appendices A and B to determine if it demonstrates that the effects of aging on the RPV components intended functions will be maintained consistent with the CLB for the period of extended operation in accordance with 10 CFR 54.21(a)(3). This is the last step in the IPA described in 10 CFR 54.21(a).

Besides the IPA, 10 CFR Part 54 requires an evaluation of TLAA in accordance with 10 CFR 54.21(c). The staff reviewed the BWRVIP-74 report and its appendices to determine if the TLAA covered by the report was evaluated for LR in accordance with 10 CFR 54.21(c)(1).

The LR applicant is to verify that its plant is bounded by the BWRVIP-74 report. Further, the LR applicant is to commit to programs described as necessary in the BWRVIP-74 report to manage the effects of aging on the functionality of the RPV components during the period of extended operation. LR applicants will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the AMP within the BWRVIP-74 report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel components or other information presented in the report, such as materials of construction, will have to be identified by the LR applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1). This is Applicant Action Item 1.

3.1 Structures and Components Subject to Aging Management Review

The staff agrees that RPV components covered by the BWRVIP-74 report are subject to AMR because they perform their intended functions without moving parts or without a change in configuration or properties. The staff concludes that BWR LR applicants shall identify the RPV components as subject to AMR to meet the applicable requirements of 10 CFR 54.21(a)(1). 10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation. Those LR applicants referencing the BWRVIP-74 report for the RPV components shall identify applicable RPV components in the FSAR supplement along with shall provide a summary description of the applicable BWRVIP-74 programs and activities. This is Applicant Action Item 2.

10 CFR 54.22 requires that each LR application include any technical specification changes (and the justification for the changes) or additions necessary to manage the effects of aging during the period of extended operation as part of the LR application. In its Appendix A to the BWRVIP-74 report, the BWRVIP stated that technical specification changes resulting from neutron embrittlement will be made at the appropriate time prior to the end of the current license. Those LR applicants referencing the BWRVIP-74 report for the RPV components shall ensure that the inspection strategy described in the BWRVIP-74 report does not conflict or result in any changes to their technical specifications. If technical specification changes do result, then the applicant should ensure that those changes are included in its LR application. This is Applicant Action Item 3.

The staff is concerned that leakage around the reactor vessel seal rings could accumulate in the vessel flange leak detection (VFLD) lines, causing an increase in the concentration of contaminants and cause cracking in the VFLD line. The BWRVIP-74 report does not identify this component as within the scope of the report. However, since the VFLD line is attached to the RPV and provides a pressure boundary function, LR applicants should identify an AMP for the VFLD line. This is Renewal Applicant Action Item 4.

3.2 Intended Functions

Appendix A describes the intended function of the pressure vessel structure and supports, vessel nozzles and safe-ends, vessel internal attachments, instrumentation, in-core housing, CRD stub tube and CRD housing penetrations. The function of the pressure vessel structure is to (1) form a pressure boundary to contain reactor coolant/moderator and leakage of radioactive materials into the drywell and (2) provide structural support for the reactor core and internals. The intended function of the attachment welds is to provide structural support for the reactor core and internals. The intended function of the nozzles and safe-ends, instrumentation penetrations, in-core housing, CRD stub tube and CRD housing penetrations is to form a pressure boundary to contain reactor coolant/moderator and prevent leakage of radioactive materials into the drywell. The staff agrees that the intended safety functions of the RPV components are as stated.

3.3 Effects of Aging

BWR reactor pressure vessel industry report, "BWR RPV License Renewal Industry Report, Revision 1," and resolution to the NRC's questions on the industry report are used to identify the aging mechanisms for the vessel components. Aging mechanisms are the cause of the aging effects. The BWRVIP-74 report used NUREG-1557 to correlate the aging effects and their associated aging mechanisms. Using these reports, the BWRVIP-74 report concluded that neutron embrittlement, fatigue crack initiation and growth, and SCC initiation and growth are the only aging effects that require AMR for RPV components.

Aging mechanisms determined to be not relevant to the RPV are IASCC, SCC of low alloy steel, general corrosion, FAC and thermal embrittlement. The report indicates that IASCC is associated with stainless steel components which have been subject to a fluence greater than 5×10^{20} n/cm^2 (E>1MeV). The report indicates that none of the RPV components reach this fluence level and therefore IASCC is not relevant. In the staff's final safety evaluation of the BWRVIP-14 report, which is contained in a letter to Carl Terry dated December 3, 1999, the staff concluded that unirradiated material properties were applicable for a neutron fluence $< 5 \times 10^{20}$ n/cm^2 (E>1MeV). Therefore, IASCC is an aging mechanism that will not require aging management for RPV components.

Low alloy steels have a high resistance to SCC. The NRC accepts this conclusion based on information discussed in the BWRVIP-60 report. The staff's final safety evaluation on this report was dated July 8, 1999.

For BWR RPV components, much of the low alloy steel surfaces are clad to essentially eliminate general corrosion. The low alloy steel unclad surfaces have also exhibited excellent resistance to general corrosion. Therefore, the report indicates general corrosion is not a significant aging effect for RPV components. The staff agrees with this conclusion.

The report indicates that, for BWR vessels, the coolant flow rates are low (<5 ft/sec.) and the oxygen levels in the steam lines are relatively high which reduces the potential for FAC. Experience with all BWR materials, including unclad feedwater nozzle regions and carbon steel components, has been very good. Therefore, FAC is not significant for RPV components. The staff agrees with this conclusion.

Thermal embrittlement results in the degradation of the fracture toughness of cast austenitic stainless steel (CASS) components through the thermal aging process. CASS components are used for internal components; however, the vessel itself does not use any. The only duplex austenitic material found is the weld metal used in vessel cladding and some of the attachment welds. By design, these alloys, typically 308L, have ferrite levels that range from 8 to 15 percent. The weld microstructure is much finer than that associated with castings and has never been identified to have the potential for toughness loss. Additionally, the material is non-structural. These overall conclusions are consistent with other BWRVIP documents. Therefore, this mechanism is not relevant to the vessel and its associated components.

The staff agrees that neutron embrittlement, fatigue crack initiation and growth, and SCC initiation and growth are the only aging effects that require AMR for the reactor vessel and its attachments.

3.4 Aging Management Programs

Appendix A to BWRVIP-74 report indicates four specific criteria were used in developing the inspection strategy: (1) safety significance, (2) detectability of failure or cracking, (3) field cracking history, and (4) prior inspections. The staff has determined that there are 10 elements which should be addressed by the applicant in AMPs. These are (1) scope of the program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience. The report does not include a discussion of all of these 10 elements. The LR applicant shall describe how each plant-specific AMP addresses these elements. This is Renewal Applicant Action Item 5.

The effects of neutron embrittlement leading to reduced material toughness, cyclic loading on fatigue, and SCC are degradation mechanisms for RPV components that require aging management for LR. These aging effects will be managed by on-going comprehensive inspection programs that are summarized in Section 4 of the report. The inspection program includes examinations required by the ASME Code and augmented examinations imposed by the NRC or as recommended in the various BWRVIP reports.

Sections 4.3 and 4.4 of the BWRVIP-74 report identify augmented examinations imposed by the NRC. NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,” implemented augmented inservice examination on feedwater nozzle inner radius areas based on thermal fatigue concerns. The NUREG provides recommendations for an increased frequency of ultrasonic (UT) testing beyond ASME Section XI requirements and recommended performing liquid penetrant examinations periodically from every second to every ninth refueling outage, depending on the sparger design. In a letter from T. Essig to T. J. Rauch, dated June 5, 1998, the NRC endorsed the BWR Owners’ Group’s position as an acceptable alternative to NUREG-0619 and allowed improved automated UT in lieu of the liquid penetrant testing (PT) and reduced the frequency of the UT examination from the original NUREG requirements.

Generic Letter (GL) 88-01 implemented augmented examination requirements for stainless steel piping welds for 4 inches and larger piping carrying reactor water at temperatures above 200 ^{o}F during power operation. The staff, in a letter dated September 15, 2000, to Carl Terry of the BWRVIP, approved with open items the BWRVIP-75 report which allows for modification of inspection scope in the GL 88-01 program.

Sections 4.5 and 4.6 of the BWRVIP-74 report identifies BWRVIP reports that provide guidelines for augmented examination of RPV components. BWRVIP-05 recommends inservice inspection (ISI) on 100-percent of the RPV longitudinal shell welds, and eliminates the required ASME Section XI inspection of the circumferential shell welds. BWRVIP-03 provides the definition for high resolution visual examination, designated EVT-1. BWRVIP-41 recommends that EVT-1 examination be performed on 25-percent of the jet pump riser brace-to-vessel attachment welds at a frequency of every four refueling outages. BWRVIP-48 recommends that EVT-1 examination of all core spray piping bracket-to-vessel attachment welds be performed at a frequency of every four refueling outages. Where furnace-sensitized material of Inconel Alloy 182 weld material was used, BWRVIP-48 recommends EVT-1 be applied for feedwater sparger bracket-to-vessel and steam dryer support bracket-to-vessel attachment welds; however, no increase in the examination frequency beyond those of ASME Section XI are recommended. BWRVIP-38 recommends UT of the shroud support-to-vessel attachment weld be performed every 10 years or, alternatively, either EVT-1 or eddy current examination performed every six years. BWRVIP-27 recommends that UT be used in addition to the surface examination required by the ASME Code, but only for those safe-ends which are stainless steel or were fabricated by boring out a solid forging, thereby leaving a substantial amount of cold worked material on the inside surface of the nozzle. BWRVIP-47 and BWRVIP-49 evaluated the CRD nozzle welds and the in-core monitor housing (ICMH) nozzle welds, respectively, but imposed no additional examination requirements beyond those of the ASME Section XI.

The staff finds that inspection by itself is not sufficient to manage cracking. Cracking can be managed by a program that includes inspection and water chemistry. BWRVIP-29 describes a water chemistry program that contains monitoring and control guidelines for BWR reactor water that is acceptable to the staff. BWRVIP-29 is not discussed in the BWRVIP-74 report. Therefore, in addition to the previously discussed BWRVIP reports, LR applications shall contain water chemistry programs based on monitoring and control guidelines for reactor water chemistry that are contained in BWRVIP-29. This is Renewal Applicant Action Item 6.

The BWRVIP-74 report indicates concerns for neutron embrittlement addressed in 10 CFR 50, Appendix G and by 10 CFR 50, Appendix H material surveillance program. Appendix G requires pressure-temperature (P-T) limits and establishes limits on Charpy USE. P-T limits and Charpy USE limits are discussed in the TLAA section of this FSER. The Appendix H material surveillance program is dependent upon the neutron fluence of the reactor vessel. Since the neutron fluence of the reactor vessel increases during the LR period, the program could be impacted by extending the license for an additional 20 years. The BWRVIP has developed an integrated surveillance program (ISP) to replace their plant-specific in-vessel surveillance programs (BWRVIP-78). The ISP proposed by the BWRVIP is being reviewed by the staff. LR applicants shall provide a material surveillance program, either an ISP or plant-specific in-vessel surveillance program, applicable to the LR term. This is Renewal Applicant Action Item 7

For the reasons set forth above, the staff concludes that the inspection strategy and evaluation methodologies discussed in the BWRVIP-74 report, as revised to include Applicant Action

Items, is acceptable. Implementation of the above inspection program provides reasonable assurance that the aging effects will be adequately managed such that the intended functions of the subject safety-related RPV components will be maintained consistent with the CLB in the extended operating period.

3.5 Time Limited Aging Analyses (TLAA)

10 CFR 54.21(1)(c) requires that each LR application contain an evaluation of TLAA as defined in 10 CFR 54.3, and that the applicant shall demonstrate that :

(i) the analyses remain valid for the period of extended operation,

(ii) the analyses have been projected to the end of the period of extended operation, or

(iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The TLAAs identified in the report are Pressure Temperature curves, Equivalent Margins Analysis for RPV materials with Charpy USE less than 50 ft-lb, fatigue, and a material evaluation to provide the technical basis supporting the elimination of RPV circumferential welds from inservice inspection.

<u>Fatigue</u>

Section 3.2 of the BWRVIP-74 report describes the fatigue evaluation of reactor pressure vessel (RPV) components. The fatigue evaluation of BWR RPV components was based on design curves for RPV and nozzle low alloy steel materials given in ASME Section III. The report identifies several RPV locations where fatigue is a potentially significant age-degradation mechanism. These locations are:

- closure studs
- penetrations
- safe-ends
- nozzles
- vessel support skirt
- vessel external attachments

Section A.4 of the report addresses the TLAAs for fatigue. The report indicates that the fatigue usage for the 60-year LR life can be obtained by multiplying the current 40-year usage by 1.5. Since operating plants typically monitor the number of occurrences of design transients, the LR applicant should verify that the number of cycles assumed in the original fatigue design is conservative to assure that the estimated fatigue usage for 60 years of plant operation is not underestimated. This is part of Renewal Applicant Action Item 8.

Section A.4.4 of the report provides alternative actions for cases where the estimated fatigue usage is projected to exceed 1.0. The alternatives are shown in Figure A-3. Step 5d allows for the use of ASME Section XI Appendix L procedures to qualify the component. The staff has identified technical concerns regarding the use of the Appendix L flaw tolerance procedure. As a consequence, the staff has not endorsed this procedure on a generic basis for use in LR evaluations. The use of this alternative will require case-by-case staff review and approval. This is part of Renewal Applicant Action Item 8.

Section 3.2 of the report indicates that the specific case of environmental fatigue was considered. The report further indicates that the overall conservatism in the ASME Code fatigue evaluation process assures that even in the presence of environmental conditions, sufficient safety factors are maintained. The report references the staff assessment for the current license period contained in SECY-95-245. In SECY-95-245, the staff indicated that its assessment could not be extrapolated beyond the current facility design life (40 years). The staff recommended that a sample of components with high fatigue usage factors be evaluated for any period of extended operation.

By letter dated February 9, 1998, EPRI submitted two technical reports dealing with the fatigue issue, EPRI Reports TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant," and TR-105759, "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Evaluations." These reports were part of an industry attempt to resolve GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life." As recommended in SECY 95-245, EPRI analyzed components with high usage factors, using environmental fatigue data. The staff has open technical concerns regarding the EPRI reports. The staff technical concerns were transmitted to the Nuclear Energy Institute (NEI) by letter dated November 2, 1998. NEI responded to the staff concerns in a letter dated April 8, 1999. The staff submitted its assessment of the response in an August 6, 1999, letter to NEI. As indicated in the staff letter, the NEI response did not resolve all staff technical concerns regarding the EPRI reports.

Although the August 6, 1999, letter identified staff concerns regarding the EPRI procedure and its application to PWRs, the technical concerns regarding the application of the Argonne National Laboratory (ANL) statistical correlations and strain threshold values are also relevant to BWRs. EPRI Report TR-107943, "Environmental Fatigue Evaluations of Representative BWR Components," addressed a BWR-6 plant and EPRI Report TR-110356, "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a Boiling Water Reactor Plant," used plant transient data from a newer vintage BWR-4 plant. In a July 28, 2000, letter to Southern Company, the staff requested that the applicant provide additional information regarding the use of the EPRI LR fatigue studies to resolve the environmental fatigue issue at the Hatch Plant.

The staff's concerns regarding environmental fatigue for BWRs, including the reactor vessel nozzles, have not been resolved by the referenced EPRI topicals and the BWRVIP-74 report for the LR period. Therefore, a LR applicant shall address environmental fatigue for the components listed in the BWRVIP-74 report for the LR period. This is part of Renewal Applicant Action Item 8.

<u>Pressure-Temperature (P-T) Limits</u>

The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; GL 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and Standard Review Plan

(SRP) Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in Charpy USE resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their RPV data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit curves. Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 thickness (1/4T) of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the ¼-thickness (¼T) and ¾-thickness (¾-T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor (CF) was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

Appendix A to the BWRVIP-74 report identifies the regulatory requirements for P-T Limits. The report indicates that a set of P-T curves should be developed for the heatup and cooldown operating conditions in the plant at a given EFPY in the LR period. This is Renewal Applicant Action Item 9.

Charpy (USE)

Section IV.A.1a. of Appendix G to 10 CFR Part 50 requires, in part, that RPV beltline materials shall have Charpy USE in the transverse direction for base metal and along the weld for weld material of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

By letter dated April 30, 1993, the Boiling Water Reactor Owner's Group (BWROG) submitted a topical report entitled "10 CFR 50 Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," to document that BWR RPVs could meet the margins of safety against fracture equivalent to those required by Appendix G of the ASME Code for Charpy USE values less than 50 ft-lb. In a letter dated December 8, 1993, the staff concluded that the topical report demonstrated that the materials evaluated had the margins of safety against fracture equivalent to Appendix G of the ASME Code, in accordance with Appendix G of 10 CFR Part 50. In this report, the BWROG derived through statistical analysis the initial USE values for materials that originally did not have documented Charpy USE values. Using these statistically derived Charpy USE values, the BWROG predicted the end-of life (40 years of operation) USE values in accordance with RG 1.99, Revision 2 (RG 1.99, Rev. 2). According to this RG, the decrease in USE is dependent upon the amount of copper in the material and the neutron fluence predicted for the material. The BWROG analysis determined that the minimum allowable Charpy USE in the transverse direction for base metal and along the weld for weld metal was 35 ft-lb.

Appendix B in the BWRVIP-74 report provides a bounding Charpy USE for BWR plants for 54 EFPY. The BWRVIP-74 analysis utilized an unirradiated Charpy USE in the longitudinal direction of 91 ft-lb for BWR/3-6 plates and 70.5 ft-lb for non-Linde 80 submerged arc welds. The value for the plates is the lowest value from the database and is less than the lower 95/95 confidence value. The value for the non-Linde 80 submerged arc welds is the value corresponding to the lower 95/95 confidence value. Since these values are statistically determined with at least 95/95 confidence, these values may be used in the evaluation of Charpy USE.

The analysis in the BWRVIP-74 report determined the reduction in the unirradiated Charpy USE resulting from neutron radiation using the methodology in RG 1.99, Rev. 2. Using this methodology and using a correction factor of 65-percent for conversion of the longitudinal properties to transverse properties, the lowest irradiated Charpy USE at 54 EFPY for all BWR/3-6 plates is projected to be 45 ft-lb. The correction factor for specimen orientation in plates is based on NRC Branch Technical position MTEB 5-2. Using the RG methodology the lowest irradiated Charpy USE at 54 EFPY for BWR non-Linde 80 submerged arc welds is projected to be 43 ft-lb. The BWRVIP-74 report indicates that the percent reduction in Charpy USE for the limiting BWR/3-6 plates and BWR non-Linde 80 submerged arc welds are 23.5 percent and 39 percent, respectively. To demonstrate that the beltline materials meet the criteria specified in the report, the applicant shall demonstrate that the percent reduction in Charpy USE for their beltline materials are less than those specified for the limiting BWR/3-6 plates and the non-Linde 80 submerged arc welds and that the percent reduction in Charpy USE for their surveillance weld and plate are less than or equal to the values projected using the methodology in RG 1.99, Revision 2. This is Renewal Applicant Action Item 10.

Circumferential RPV Weld Inspection

The BWRVIP-05 report recommends inservice inspection on 100-percent of the RPV longitudinal shell welds, and eliminates the required ASME Section XI inspection of the circumferential shell welds. This technical alternative is discussed in the staff's FSER of the BWRVIP-05 report, which is contained in a July 28, 1998, letter to Carl Terry, BWRVIP Chairman. In this letter, the staff concludes that, since the failure frequency for circumferential welds in BWR plants is significantly below the criteria specified in RG 1.154 and the core damage frequency (CDF) of any BWR plant, and that continued inspection would result in a negligible decrease in an already acceptably low value, elimination of the ISI for RPV circumferential welds is justified. The staff's letter indicated that BWR licensees may request relief from ISI requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds by demonstrating: (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the evaluation, and (2) they have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the report. The letter also indicated that the requirements for inspection of circumferential RPV welds during an additional 20-year LR period will be reassessed, on a plant-specific basis, as part of any BWR LR application.

Section A.4.5 of report BWRVIP - 74 indicates that Appendix E of the NRC staff's FSER conservatively evaluated BWR RPV's to 64 EFPY, which is 10 EFPY greater than what is realistically expected for the end of the LR period. Therefore, the staff's analysis provides a technical basis for relief from the current ISI requirements of the ASME Section XI for volumetric examination of the circumferential welds as they may apply for the LR period. To obtain relief, the BWRVIP report indicates each licensee will have to demonstrate that (1) at the end of the renewal period, the circumferential welds will satisfy the limiting conditional failure frequency for circumferential welds in the Appendix E of the staff's FSER, and (2) that they have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the staff's FSER. This is Renewal Applicant Action Item 11.

As an alternative to satisfying the limiting conditional failure frequency for circumferential welds in Appendix E of the staff's July 28, 1998, FSER, the BWRVIP proposes either:

(1) Perform a one-time inspection of the circumferential welds in accordance with the requirements of ASME Section XI within two refueling outages of the start of the renewal period. If the results from the inspection meet ASME Code acceptance criteria, no further examination or analyses are required for the remainder of the renewal period.

or,

(2) Perform plant-specific analysis to assess the probability of vessel failure at the end of the renewal period. The analysis should be consistent with the analytical approach in the NRC FSER including any subsequent revisions, etc., and should be based on the chemistry of the limiting weld and the predicted neutron fluence at the end of the LR period. The calculated probability of failure should be less than or equal to that stated in Appendix E of the staff's FSER. The plant-specific analysis should be submitted to the NRC for inspection relief.

The second alternative is acceptable to the NRC; but the first alternative is not acceptable. The staff needs to know that the plant-specific probability of vessel failure is below NRC's 64 EFPY value before it could consider authorizing a one-time inspection in lieu of ASME Section XI inspection requirements.

Axially Oriented RPV Welds

The BWRVIP-74 report does not indicate the impact of neutron embrittlement on BWR axially oriented RPV welds. However, in its July 28, 1998, letter to Carl Terry, the staff identified a concern about the failure frequency of axially oriented welds in BWR RPVs. In a response to this concern, the BWRVIP provided evaluations of axial weld failure frequency in letters dated December 15, 1998 and November 12, 1999. The staff's evaluation of these analyses is contained in a March 7, 2000, letter to Carl Terry. The FSER enclosed in that letter states that the RPV failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is below 5×10^{-6} per reactor year, given the assumptions on flaw density, distribution and location, as described in the FSER. Since the results apply only for the initial 40-year license period of BWR plants, LR applicants shall provide plant-specific information applicable to 60 years of operation.

The BWRVIP identified Clinton and Pilgrim as the reactor vessels with the highest mean RT_{NDT} in the BWR fleet. The staff confirmed this conclusion by comparing the information contained in the BWRVIP analysis and the information contained in the reactor vessel integrity database (RVID) for all BWR RPV axial welds. The staff performed analyses of the Clinton and Pilgrim plants. The results from the staff calculations are provided in Table 1. The staff calculations used the basic input information for Pilgrim, with three different assumptions for the initial RT_{NDT}. The calculations of the actual Pilgrim condition used the docketed initial RT_{NDT} of -48 °F and a mean RT_{NDT} of 68 °F. A second calculation, listed as "Mod 1" in Table 1, is consistent with the BWRVIP calculations, with an initial RT_{NDT} of 0 °F and a mean RT_{NDT} of 116 °F. A third calculation, with an initial RT_{NDT} of -2 °F and a mean RT_{NDT} of 114 °F, was performed to identify the mean RT_{NDT} value required to provide a result which closely matches the RPV failure frequency of 5×10^{-6} per reactor-year.

Table 1: Comparison of Results from Staff and BWRVIP

Plant	Initial RT_{NDT} (°F)	Mean RT_{NDT} (°F)	Vessel Failure Freq.	
			Staff	BWRVIP
Clinton	-30	91	2.73 E -6	1.52 E -6
Pilgrim	-48	68	2.24 E -7	-------
Mod 1 *	0	116	5.51 E -6	1.55 E -6
Mod 2 **	-2	114	5.02 E -6	-------

* A variant of Pilgrim input data, with initial RT_{NDT} = 0 °F.
** A variant of Pilgrim input data, with initial RT_{NDT} = -2 °F

As indicated in the March 7, 2000, letter, an applicant shall monitor the axial beltline weld embrittlement. One acceptable method is to determine the mean RT_{NDT} of the limiting axial beltline weld at the end of the extended period of operation is less than the values specified in Table 1. This is Renewal Applicant Action Item 12.

Neutron Fluence of the RPV

The Charpy USE, P-T limit, circumferential weld and axial weld RPV integrity evaluations are all dependent upon the neutron fluence. The neutron fluence shall be calculated prior to the extended operating term using an approved methodology or the neutron fluence methodology shall be submitted for staff review. This is Renewal Applicant Action Item 13.

Analytical Evaluation of Indications

Subsection IWB-3600 to Section XI of the ASME Code states that flaw indications that exceed the size of allowable indications defined in IWB-3500 may be evaluated by analytical procedures, such as described in Appendix A of Section XI of the ASME Code in order to calculate its growth until the next inspection or the end of service lifetime of the component. Therefore, components that have indications that have been previously analytically evaluated in accordance with IWB-3600 until the end of the 40-year service period shall be re-evaluated for the 60-year service period corresponding to the LR term. This is Renewal Applicant Action Item 14.

4.0 CONCLUSIONS

The staff has reviewed the BWRVIP-74 report and the associated Appendices A and B, as supplemented. On the basis of its review, as set forth above, the staff concludes that, LR applications referencing the BWRVIP-74 report and completing the applicant action items, will have provided an acceptable demonstration that aging effects applicable to the reactor vessel components within the scope of the report will be adequately managed so that there is reasonable assurance that the RPV components will perform their intended functions in accordance with the CLB during the period of extended operation. The staff also concludes that, upon completion of the LR applicant action items, applications referencing the BWRVIP-74 report will have provided an acceptable evaluation of TLAA for the RPV components for the BWRVIP member utilities for the period of extended operation.

Any BWRVIP member utility may reference this report in a LR application to satisfy the requirements of (1) 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the RPV components within the scope of this topical report will be adequately managed and (2) 10 CFR 54.21(c)(1) for demonstrating the appropriate findings regarding evaluation of TLAA for the RPV components for the period of extended operation. The staff also concludes that, upon completion of the LR applicant action items set forth in Section 4.1 below, referencing the BWRVIP-74 report and its Appendices in a LR application and summarizing in an final safety analysis report (FSAR) supplement the aging management programs and the TLAA evaluations contained in this report, will provide the staff with sufficient information to make the necessary findings required by Sections 54.29(a)(1) and (a)(2) for components within the scope of this report.

As an alternative to satisfying the limiting conditional failure frequency for circumferential welds in Appendix E of the staff's FSER in a July 28, 1998, letter to Carl Terry, BWRVIP Chairman, the BWRVIP proposes either to allow applicants to (1) perform a one-time examination in lieu of ASME Section XI requirements or (2) perform plant-specific analysis to predict the probability of vessel failure at the end of the renewal period. The second alternative is acceptable to the NRC; but the first alternative is not acceptable. The staff needs to know that the plant-specific probability of vessel failure is below NRC's 64 EFPY value before it could consider authorizing a one-time inspection in lieu of ASME Section XI inspection requirements.

4.1 Renewal Applicant Action Items

The following are LR applicant action items to be addressed in the plant-specific LR application when incorporating the BWRVIP-74 report and the associated Appendices A and B in a LR application:

(1) The LR applicant is to verify that the BWRVIP-74 report is applicable to its plant. Further, the LR applicant is to commit to programs described as necessary in the BWRVIP-74 report to manage the effects of aging on the functionality of the RPV components during the period of extended operation. LR applicants will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the AMP within the BWRVIP-74 report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel components or other information presented in the report, such as materials of construction, will have to be identified by the LR applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

(2) 10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation. Those LR applicants referencing the BWRVIP-74 report for the RPV components shall ensure that the programs and activities specified as necessary in the BWRVIP-74 report are summarily described in the FSAR supplement.

(3) 10 CFR 54.22 requires that each LR application include any technical specification changes (and the justification for the changes) or additions necessary to manage the effects of aging during the period of extended operation as part of the LR application. In its Appendix A to the BWRVIP-74 report, the BWRVIP stated that technical specification changes resulting from neutron embrittlement will be made at the appropriate time prior to the end of the current license. Those LR applicants referencing the BWRVIP-74 report for the RPV components shall ensure that the inspection strategy described in the BWRVIP-74 report does not conflict or result in any changes to their technical specifications. If technical specification changes do result, then the applicant should ensure that those changes are included in its LR application.

(4) The staff is concerned that leakage around the reactor vessel seal rings could accumulate in the VFLD lines, cause an increase in the concentration of contaminants and cause cracking in the VFLD line. The BWRVIP-74 report does not identify this component as within the scope of the report. However, since the VFLD line is attached

to the RPV and provides a pressure boundary function, LR applicants should identify an AMP for the VFLD line.

(5) LR applicants shall describe how each plant-specific aging management program addresses the following elements: (1) scope of the program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience.

(6) The staff believes inspection by itself is not sufficient to manage cracking. Cracking can be managed by a program that includes inspection and water chemistry. BWRVIP-29 describes a water chemistry program that contains monitoring and control guidelines for BWR water that is acceptable to the staff. BWRVIP-29 is not discussed in the BWRVIP-74 report. Therefore, in addition to the previously discussed BWRVIP reports, LR applications shall contain water chemistry programs based on monitoring and control guidelines for reactor water chemistry that are contained in BWRVIP-29.

(7) LR applicants shall identify their vessel surveillance program, which is either an ISP or plant-specific in-vessel surveillance program, applicable to the LR term.

(8) LR applicants should verify that the number of cycles assumed in the original fatigue design is conservative to assure that the estimated fatigue usage for 60 years of plant operation is not underestimated. The use of alternative actions for cases where the estimated fatigue usage is projected to exceed 1.0 will require case-by-case staff review and approval. Further, a LR applicant must address environmental fatigue for the components listed in the BWRVIP-74 report for the LR period.

(9) Appendix A to the BWRVIP-74 report indicates that a set of P-T curves should be developed for the heatup and cooldown operating conditions in the plant at a given EFPY in the LR period.

(10) To demonstrate that the beltline materials meet the Charpy USE criteria specified in Appendix B or the report, the applicant shall demonstrate that the percent reduction in Charpy USE for their beltline materials are less than those specified for the limiting BWR/3-6 plates and the non-Linde 80 submerged arc welds and that the percent reduction in Charpy USE for their surveillance weld and plate are less than or equal to the values projected using the methodology in RG 1.99, Revision 2.

(11) To obtain relief from the inservice inspection of the circumferential welds during the LR period, the BWRVIP report indicates each licensee will have to demonstrate that (1) at the end of the renewal period, the circumferential welds will satisfy the limiting conditional failure frequency for circumferential welds in the Appendix E of the staff's July 28, 1998, FSER, and (2) that they have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the staff's FSER.

(12) As indicated in the staff's March 7, 2000, letter to Carl Terry, a LR applicant shall monitor axial beltline weld embrittlement. One acceptable method is to determine the

mean RT_{NDT} of the limiting axial beltline weld at the end of the extended period of operation is less than the values specified in Table 1 of this FSER.

(13) The Charpy USE, P-T limit, circumferential weld and axial weld RPV integrity evaluations are all dependent upon the neutron fluence. The applicant may perform neutron fluence calculations using a staff approved methodology or may submit the methodology for staff review. If the applicant performs the neutron fluence calculation using a methodology previously approved by the staff, the applicant should identify the NRC letter that approved the methodology.

(14) Components that have indications that have been previously analytically evaluated in accordance with sub-section IWB-3600 of Section XI to the ASME Code until the end of the 40-year service period shall be re-evaluated for the 60 year service period corresponding to the LR term.

5.0 REFERENCES

1. Letter from Carl Terry, BWRVIP, EPRI proprietary report TR-113596, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)," dated September 29, 1999.

2. "BWR RPV License Renewal Industry Report, Revision 1," EPRI Report TR-103836, July 1994.

3. Letter to Carl Terry, BWRVIP Chairman, Subject: Final Safety Evaluation of Proprietary Report TR 105873, "BWR Vessel Internals Project Evaluation of Crack Growth in BWR Stainless Steel Internals (BWRVIP-14)" (TAC NO. M94975).

4. NUREG-1557, Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal, October 1996.

5. "BWR Vessel Internals Project, Evaluation of Stress Corrosion Cracking Growth Rate in Low Alloy Steel Vessel Materials in the BWR Environment (BWRVIP-60)," EPRI Report TR-108709, March 1999.

6. Letter to Carl Terry, BWRVIP, dated July 8, 1999, FSER on BWRVIP-60.

7. NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," U. S. NRC, November 1980.

8. Letter from T. Essig to T. J. Rauch, dated June 5, 1998, Subject: "BWROG-Safety Evaluation of Proposed Alternative to BWR Feedwater Nozzle Inspection," (TAC NO.94090).

9. NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," January 25, 1988.

10. “Technical Report on Materials selection and Processing Guidelines for BWR Coolant Boundary Piping - Final Report,” NUREG-0313, Rev. 2, U. S. NRC.

11. Letter to Carl Terry, BWRVIP, SE approving modification to GL 88-01 (BWRVIP-75).

12. “BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05),” EPRI Report TR-105697, September 1995.

13. “BWR Vessel and Internals Project, RPV Internals Examination Guidelines (BWRVIP-03),” dated November 10, 1995.

14. “BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41),” EPRI Report TR-108728, October 1997.

15. “BWR Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Evaluation Guidelines (BWRVIP-48),” EPRI Report TR-10874, February 1998.

16. “BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38),” EPRI Report TR-108823, September 1997.

17. “BWR Vessel and Internals Project, BWR Standby Liquid Control System/Core Plate delta P Inspection and Flaw Evaluation Guidelines (BWRVIP-27),” EPRI Report TR-107286, April 1997.

18. “BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47),” EPRI Report TR-108727, December 1997.

19. “EPRI Water Chemistry Guidelines - 1996".

20. Letter dated April 30, 1993, “10 CFR 50 Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy BWR/2 Through BWR/6 Vessels,” BWROG.

21. Letter dated December 8, 1993, FSER on Ref. 20, U. S. NRC.

22. Letter to Carl Terry, BWRVIP Chairman dated July 28, 1998, “Final FSER of the BWR Vessels and Internals Project BWRVIP Report.”

23. Letter to Carl Terry, BWRVIP Chairman dated March 7, 2000, “Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report.”

24. “BWR Vessel and Internals Project, Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18),” EPRI Report TR-106740, July 1996.

25. “BWR Vessel and Internals Project, BWR Top Guide Inspection and Flaw Evaluation Guidelines (BWRVIP-26),” EPRI Report TR-107285, December 1996.

26. “BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25),” EPRI Report TR-107284, December 1996.

27. “BWR Vessel and Internals Project, BWR LPCI Inspection and Flaw Evaluation Guidelines (BWRVIP-42),” EPRI Report TR-108726, December 1997.

28. “BWR Vessel and Internals Project, Instrument Penetration Inspection and Flaw Evaluation Guidelines,” EPRI Report TR-108695, March 1998.