March 16, 2001

Mr. Richard Bernier, Chairman
CE Owners Group
Mail Stop 7868
Arizona Public Service Company
Palo Verde Nuclear Generating Station
P.O. Box 52034
Phoenix, Arizona 85072-2034

SUBJECT: SAFETY EVALUATION OF TOPICAL REPORT CE NPSD-683, REVISION 6, "DEVELOPMENT OF A RCS PRESSURE AND TEMPERATURE LIMITS REPORT [PTLR] FOR THE REMOVAL OF P-T LIMITS AND LTOP REQUIREMENTS FROM THE TECHNICAL SPECIFICATIONS" (TAC NO. MA9561)

Dear Mr. Bernier:

On September 29, 2000, the Combustion Engineering Owners Group (CEOG) submitted Topical Report CE NPSD-683, Revision 6, for staff review. The topical report (TR) provides the generic methodology to allow CEOG member utilities to remove the pressure-temperature (P-T) limits and the low temperature overpressure (LTOP) limits from the technical specifications and to place them in a PTLR or similar owner-controlled document. The TR was supplemented by information provided in the CEOG's letters of November 16 and 30, 2000.

The staff has found that CE NPSD-683, Revision 6, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," is acceptable for referencing in licensing applications for CE designed pressurized water reactors to the extent specified and under the limitations delineated in the report and in the associated NRC safety evaluation (SE). The SE defines the basis for acceptance of the report. Licensees requesting a license amendment to relocate the P-T limits and LTOP system limits will need to include in their plant-specific submittals appropriate responses to the information requests identified in Section 5.0 of the SE.

We do not intend to repeat our review of the matters described in the subject report, and found acceptable, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. Our acceptance applies only to matters approved in the report.

In accordance with procedures established in NUREG-0390, the NRC requests that the CEOG publish an accepted version, within 3 months of receipt of this letter. The accepted version shall incorporate (1) this letter and the enclosed SE between the title page and the abstract, and (2) an "-A" (designating "accepted") following the report identification symbol.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, the CEOG and/or the applicants referencing the TR will be expected to revise and resubmit their respective documentation, or submit justification for the continued applicability of the TR without revision of their respective documentation.

Sincerely,

/RA by Stephen Dembek for/

Stuart A. Richards, Director
Project Directorate IV and Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 692

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO TOPICAL REPORT CE NPSD-683, REVISION 6,

"DEVELOPMENT OF A RCS PRESSURE AND TEMPERATURE LIMITS

REPORT FOR THE REMOVAL OF P-T LIMITS AND LTOP REQUIREMENTS

FROM THE TECHNICAL SPECIFICATIONS"

PROJECT NO. 692

1.0 INTRODUCTION

On January 31, 1996, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 96-03 (Reference 1) to holders of operating licenses or construction permits for nuclear power reactors. In the GL, the NRC informed these licensees of their right to request a license amendment to relocate the pressure-temperature (P-T) limit curves and the low temperature overpressure protection (LTOP) system limits for their facilities from their plant-specific technical specifications (TS) to a P-T limits report (PTLR) or similar owner-controlled document.[(1)]

On September 29, 2000 (Reference 2), the Combustion Engineering Owners Group (CEOG) submitted CEOG Topical Report CE NPSD-683, Revision 6 (Reference 3), "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits from the Technical Specifications," for review by the NRC. On October 30, 2000 (Reference 4), the NRC issued a request for additional information (RAI) with regard to the P-T limit and LTOP limit methods stated in Topical Report CE NPSD-683, Revision 6. On November 16 and 30, 2000 (References 5 and 6), the CEOG supplemented the contents of Topical Report CE NPSD-683, Revision 6, with its responses to the staff's RAI. Topical Report CE NPSD-683, Revision 6, as modified by the contents of the CEOG's submittal of November 16 and 30, 2000, provides the CEOG's most current methodology for generating the P-T limit curves and LTOP limits that are designed to protect ferritic materials in the reactor pressure vessels (RPVs) and reactor coolant pressure boundaries (RCPBs) against fracture during normal plant operations (including operations during heatups and cooldowns of the reactor and during anticipated operational occurrences), and during leak-rate or hydrostatic-pressure testing conditions.

- Section 1.1 summarizes how the applicable regulations of 10 CFR Part 50 relate to the generation of plant-specific P-T limits and LTOP system limits.

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(1) GL 96-03 was issued as part of the NRC's process for reducing unnecessary regulatory burden on NRC stakeholders. This process is listed in NUREG-1614, Vol. 2, Part 1, "U.S. Nuclear Regulatory Commission Strategic Plan," as one of the NRC's performance goals for ensuring nuclear reactor safety.

- Section 1.2 discusses the NRC's criteria and position in GL 96-03 for allowing removal of P-T limit curves and LTOP limits from the TS.
- Section 1.3 provides an overview of the methods of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for generating P-T limit curves and LTOP limits.
- Section 1.4 discusses a number of exemptions that have previously been granted by the staff to allow use of alternative P-T limit/LTOP limit generation methods.
- Section 2.0 provides the staff's evaluation of Topical Report CE NPSD-683, Revision 6.
- Section 3.0 discusses what the plant-specific process is for submitting license amendment requests for relocating the P-T limits and LTOP limits from the TS into a PTLR or similar owner-controlled document.
- Section 4.0 provides the overall conclusions regarding the acceptability of Topical Report CE NPSD-683, Revision 6.
- Section 5.0 provides a concise list of supplemental information that licensees will need to include as part of their plant-specific license amendment submittals.
- Section 6.0 provides a list of applicable references used in the staff's evaluation.

1.1 *Code of Federal Regulations* Requirements for Generating Pressure-Temperature (P-T) Limit Curves and for Reactor Vessel Material Surveillance Programs

The NRC has established requirements in Section 50.60 and in Appendices G and H to Part 50 of Title 10, *Code of Federal Regulations* (10 CFR 50.60 and Appendices G and H to Part 50, respectively [References 7 , 8 and 9]), to protect the integrity of the RPV and RCPB in nuclear power plants. Clause (a) to 10 CFR 50.60 requires that commercial nuclear light-water reactor facilities must meet the fracture toughness requirements specified in Appendix G to Part 50 and the reactor vessel material surveillance program requirements specified in Appendix H to Part 50. Clause (b) to 10 CFR 50.60 allows licensees to use alternatives to the requirements of Appendices G and H to Part 50 if an exemption is granted by the Commission under the exemption provisions and criteria of 10 CFR 50.12 (Reference 10).

Holders of licenses for operation of nuclear power generation facilities are required by Section IV.A.2. of Appendix G to Part 50 to establish and implement these P-T limit curves at their respective nuclear plants. Criterion 2 of Paragraph (c)(2)(ii) of 10 CFR 50.36 (Reference 11), requires licensees to establish a limiting condition for operation (LCO) in their plant-specific TS for operating restrictions needed to preclude unanalyzed accidents and transients. These operating conditions include P-T limits and LTOP limits. Licensees typically incorporate these P-T limit curves and the LTOP system limits into the LCO for the reactor coolant system, and use them as one of the bases for protecting the RPV and RCPB against fracture during normal

plant operations (including operations during heatups and cooldowns of the reactor and during anticipated operational occurrences) and during pressure testing conditions.

In this case, Section IV.A.2 of Appendix G to Part 50 establishes the following criteria for generating plant-specific P-T limits:

- The P-T limits for an operating plant must be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (Appendix G to the Code) (Reference 12) were applied; and

- The minimum permissible temperature for the RPV, as summarized in Table 1 of Appendix G to Part 50, must be met for all conditions.

These criteria require that the P-T limit curves be generated from the most conservative combinations of the P-T data points from P-T limit calculations and the minimum temperature requirements listed in Appendix G to Part 50. The staff currently endorses editions of Appendix G to the Code through the 1995 Edition of Section XI, inclusive of the Summer and Winter 1996 Addenda.

Appendix H to Part 50, "Reactor Vessel Material Surveillance Program Requirements," provides the staff's requirements for monitoring the degree of irradiation induced embrittlement for the materials in the beltline region of nuclear RPVs. The appendix requires licensees to establish surveillance programs for the RPV beltline materials when the peak end-of-design life neutron fluence for the RPV is projected to exceed 1×10^{17} n/cm^2 (E > 1 MeV[2]). Appendix H to Part 50 also requires that the surveillance program be designed to conform with the RPV material surveillance program design and withdrawal criteria of the Edition of American Society for Testing and Materials (ASTM) Standard Practice E185 (Reference 13) that is in effect on the date of the ASME Code to which the plant's RPV was purchased. The appendix allows licensees to use later editions of ASTM Standard Practice E185 inclusive of the 1982 edition of the procedure. The data obtained from fracture toughness tests of test specimen removed in accordance with the surveillance program are directly applied to the methods for generating the P-T limits.

1.2 GL 96-03 Position for Submitting PTLR License Amendment Requests

In GL 96-03, the NRC advised the addressees of the opportunity to request a license amendment to relocate the P-T limit curves and LTOP limits from their plant-specific TS to an owner-controlled PTLR or similar document, and informed the addressees of the process to be followed for submittals requesting relocation of the P-T limits and LTOP limits to a PTLR.

As stated in GL 96-03, license amendments are generally required at the end of the effective period for P-T limits curves or when surveillance specimens are withdrawn and tested. Each

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(2) Mega-electron volt, a unit of energy equivalent to 1.60×10^{-13} Joules (a unit of energy in the SI System of weights and measures) or 1.18×10^{-13} foot-pounds-force (ft-lbf, a unit of energy in the English System of weights and measures).

time the P-T curves are revised, the LTOP system must be reevaluated to ensure that its functional requirements can still be met. Processing amendment requests for TS changes using an accepted methodology places an unnecessary burden on licensees and NRC resources alike. Therefore, an alternative approach for controlling these limits, similar to that of core operating limits, was proposed during the development of Standard Technical Specifications (STS) and adopted into the STS thereafter. This approach relocates the P-T curves and LTOP setpoint curves or values to a PTLR or a similar document, and references that document in the affected LCOs and Bases.

According to the GL, the methodology used to determine the P-T and LTOP system limit parameters must comply with the specific requirements of Appendices G and H to Part 50, be documented in an NRC-approved topical report or in a plant-specific submittal, and be incorporated by reference into the TS. According to the GL, updates of the P-T limits and/or LTOP limits that are implemented in accordance with the approved methodology will not need to be submitted for staff review pursuant to the 10 CFR 50.90 license amendment process. However, any subsequent changes in the approved methodology will require staff review and approval pursuant to the 10 CFR 50.90 license amendment process; 10 CFR 50.59 does not apply.

1.3 Methodology of Appendix G to ASME Code, Section XI, for Generating P-T Limits

The methodology of Appendix G to the Code postulates the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum applied stress. For materials in the beltline and upper and lower head regions of the RPV, the flaw is postulated to propagate to a maximum depth that is equal to one-fourth of the RPV wall thickness and a maximum length equal to 1.5 times the RPV wall thickness. For the case of evaluating RPV nozzles, the surface flaw is postulated to propagate parallel to the axis of the nozzle's corner radius. The basic parameter in Appendix G to the Code for calculating P-T limit curves is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. The methodology requires that licensees determine the lower bound crack arrest critical stress intensity factors (K_{Ia} factors), which vary as a function of temperature, from the reactor coolant system (RCS) operating temperatures, and from the adjusted nil-ductility reference temperature (RT_{NDT}) for the limiting material in the RPV. Thus, the critical locations in the RPV beltline and head regions are the 1/4-thickness (1/4T) and 3/4-thickness (3/4T) locations, which correspond to the points of the crack tips if the flaws are postulated to initiate and grow from the inside and outside surfaces of the RPV, respectively. Regulatory Guide (RG) 1.99, Revision 2 (Reference 14), provides an acceptable method of calculating RT_{NDT} values for ferritic RPV materials; the methods of RG 1.99, Revision 2, include methods for adjusting the RT_{NDT} values of materials in the beltline region of the RPV, where the effects of neutron irradiation may induce an increased level of embrittlement in the materials.

The methodology of Appendix G to the Code requires that P-T curves must be calculated to satisfy the following equation:

$$K_{Ia} \geq SF * K_{Im} + K_{It} \qquad (1)$$

where K_{Ia} is defined as the lower bound crack arrest critical stress intensity factor (as discussed previously), K_{Im} represents the stress intensity at the crack tip arising from primary membrane

stress, SF represents an additional safety factor to be imposed on K_{Im}, and K_{It} represents the stress intensity at the crack tip arising from the thermal gradient across the RPV shell wall. In this case, the methodology dictates that SF on K_{Im} be set at 2.0 during normal plant operations (including heatups, cooldowns, and transient operating conditions), and at 1.5 when leak-rate or hydrostatic-pressure tests are performed on the RCS. For areas of the RPV near nozzles, flanges, or other geometric discontinuities, the methodology states that the P-T calculation equation must be modified to account for stress intensities arising from primary bending stresses (including a safety factor of 2.0 imposed on these stresses), and for secondary membrane and bending stresses. In this case, the methodology of Appendix G to the Code states that the methodology in Appendix 5 to Welding Resource Council Bulletin WRC-175 (Reference 15) may be used to analyze the inside corner flaw of a nozzle joined to a cylindrical shell and to approximate the stress intensities arising from the internal pressure stress (membrane stress). The methodology of the 1995 Edition of Appendix G to the Code treats thermal stresses as secondary stresses, and allows them to be determined from either the appropriate equations in Paragraph G-2214.3 of the appendix or from the plant-specific thermal stress gradient determinations for plant heatups and cooldowns.

1.4 Exemptions to the Requirements of 10 CFR Part 50, Appendix G

As stated in Section 1.1, clause (b) to 10 CFR 50.60 allows licensees to use alternatives to the requirements of Appendix G to Part 50 if an exemption is granted by the Commission pursuant to the provisions and exemption acceptance criteria of 10 CFR 50.12. The staff has previously granted permission, through the exemption request process, to apply the methods in a number of ASME Code Cases to the methodology for plant-specific P-T limit calculations.

1.4.1 Code Case N-588

The current methods of Appendix G to the Code mandate consideration of an axial flaw in full penetration RPV welds, and thus, for circumferential welds, dictate that the flaw be oriented transverse to the axis of the weld. ASME Code Case N-588 (Reference 16) allows applicants seeking an exemption to evaluate circumferential RPV shell and head welds by postulating a circumferential flaw in the weld in lieu of the axially-oriented flaw typically assumed by the methods of analysis in Appendix G to the Code. Postulation of an axial flaw in a circumferential weld is unrealistic because the length of the flaw would extend well beyond the width of the circumferential weld and into the adjoining base metal material. Industry experience with the repair of flaw indications found in welds during preservice inspection, and data taken from destructive examination of actual vessel welds, confirms that any remaining flaws are small, laminar in nature, and do not transverse the weld bead orientation. Therefore, any potential defects introduced during the fabrication process, and not detected during subsequent nondestructive examinations, would only be expected to be oriented in the direction of weld fabrication. For circumferential RPV welds, the methods of the Code Case therefore postulate the presence of a flaw that is oriented in a direction parallel to the axis of the weld (i.e., in a circumferential orientation).

In an analysis provided to the ASME Code's Working Group on Operating Plant Criteria (WGOPC) (in which Code Case N-588 was developed), the effect of postulating axially or circumferentially oriented flaws for a circumferential weld was evaluated. The WGOPC determined that the acceptable pressure (as a function of temperature) for a postulated axial

flaw using a safety factor of 2 on stress intensities arising from primary membrane stresses was equivalent to that for a circumferentially oriented flaw using a safety factor of 4.18 on stress intensities arising from primary membrane stresses. Appendix G to the Code only requires that a safety factor of 2 be placed on the contribution of the pressure load (i.e., on stress intensities arising from primary membrane stresses) for the case of an axially-oriented flaw in an axial weld, shell plate, or forging. Consequently, the staff determined that the postulation of an axially-oriented flaw on a circumferential RPV weld adds a level of conservatism in the P-T limits that goes beyond the margins of safety required by Appendix G to the Code, and thus required by Appendix G to Part 50. By postulating a circumferentially-oriented flaw on a circumferential weld and using the appropriate correction factor, the safety margin of 2 is maintained for the primary membrane intensity calculations for circumferential welds. Based on this reason, the staff determined that methods of the Code Case for reducing the applied stress intensities for primary membrane stresses were acceptable.(3)

Application of Code Case N-588 will only matter if the Code Case is applied for the case where a circumferential weld is the most limiting material in the beltline region of the RPV. Since application of the Code Case methods allows licensees to reduce the stress intensities attributed to the circumferential weld, the net effect of the Code Case would allow an applicant to use the next most limiting base metal or axial weld material in the RPV as the basis for evaluating the vessel and generating the P-T limit curves, if a circumferential weld is the most limiting material in the RPV.

1.4.2 Code Case N-640

Code Case N-640 (Reference 17) permits application of the lower bound static crack initiation critical stress intensity factor equation (K_{Ic} equation) as the basis for establishing the curves in lieu of using the lower bound crack arrest critical stress intensity factor equation (i.e., the K_{Ia} equation, which is based on conditions needed to arrest a propagating crack, and which is the method invoked by Appendix G to Section XI of the ASME Code). Use of the K_{Ic} equation in determining the lower bound fracture toughness in the development of the P-T operating limits curve is more technically correct than the use of the K_{Ia} equation since the rate of loading during a heatup or cooldown is slow, and since crack initiation, which is more representative of a static condition than a dynamic condition, is principally at issue. The K_{Ic} equation appropriately implements the use of the static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The staff has required use of the initial conservatism of the K_{Ia} equation since 1974, when the equation was codified. This initial conservatism was considered to be necessary due to a limited knowledge of RPV material properties at the time. Since 1974, a significant amount of additional materials property data has been collected about RPV fabrication materials, and has provided the staff with a better

(3) The Code Case accomplishes this by reducing the M_m factors for circumferential welds that are used for calculations of the stress intensities attributed to primary membrane stresses (K_{Im}) and primary bending stresses (K_{Ib}). For RPVs with wall thicknesses in the range of 4.0-12.0 inches, the Code Case applies an M_m factor of 0.443 for circumferential welds and 0.926 for axial flaws. This reduction in the M_m factor for circumferential flaws is realistic since the postulated circumferential flaw in the vessel will propagate if a stress is applied in a direction normal to the axis of the flaw (i.e., by application of an axially oriented stress that results in Mode I crack propagation of the circumferential flaw). Such tensile stresses in the RPVs are typically about half the magnitudes of the corresponding membrane stresses.

understanding of how the RPV materials will behave in service. For this reason, the staff has concluded that this additional information is sufficient to permit the case of lower bound static crack initiation critical stress intensity factor K_{Ic} equation as an acceptable method for calculating P-T limits. In addition, P-T curves based on the K_{Ic} equation will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operations. Thus, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served.

1.4.3 Code Case N-514

ASME Code Case N-514 (Reference 18), recommends that the LTOP systems be effective at RCS inlet temperatures less than 200°F or at RCS inlet temperatures corresponding to a RPV metal temperature less than the limiting RT_{NDT} value + 50°F, whichever is greater. The Code Case further recommends that the LTOP systems limit the maximum pressure for the RPV to 110% of the pressure determined to satisfy Paragraph G-2215 of Appendix G to the Code. This recommendation is actually a relaxation of 10 percent in the limits used in the LTOP analysis. The methods of Code Case N-514 have been incorporated into Paragraph G-2215 of the 1995 edition of Appendix G to the Code. The staff will only grant exemptions to use the methods of Code Case N-640 (refer to Section 1.4.2 of this SE) if the LTOP system relief valve is set to lift at a pressure equivalent to 100 percent of the pressure determined to satisfy Paragraph G-2215 of the 1995 Edition of Appendix G to the Code.

2.0 EVALUATION

2.1 Fluence Methods

Technical Element 1 (Criterion 1) of the Table in Attachment 1 to GL 96-03[(4)] states that the methodology shall describe how the neutron fluence is calculated. To satisfy Criterion 1, the Table states that the methodology for generating P-T limit curves should (1) describe the methods for determining the neutron fluence values used in the generation of P-T limits, and (2) reference the reports and documents that contain these methods. The description of the neutron fluence transport calculational methods should include applicable computer codes, formulas, approximations, and cross sections used in the neutron fluence value calculations.

Section 1.0 of CE NPSD-683, Revision 6, describes the methodology for calculating neutron fluence values for the materials in the RPV. The CEOG states that the discussion of the proposed neutron fluence methodology meets Criterion 1 of the Table in Attachment 1 to GL 96-03, and Draft Regulatory Guide DG-1053. This is based on a benchmarked discrete ordinates transport method that will be validated with plant-specific dosimetry measurements. The CEOG's discussion of the methodology for calculating RPV neutron fluence values parallels the recommended guidelines of Draft Regulatory Guide DG-1053, but does not demonstrate how the benchmarking of neutron fluence will be performed. While the discussion does describe what is to be included in the benchmarking, it does not demonstrate the

(4) The Table, entitled "Requirements for Methodology and PTLR," is provided on pages 4 and 5 of Attachment 1 to GL 96-03.

benchmarking nor discuss the attributes of the data base upon which the benchmarking will be based. The methodology defers the details for benchmarking the neutron fluence to the plant-specific applications. In order to satisfy Criterion 1 of the Table in Attachment 1 to GL 96-03, licensees seeking to use CE NPSD-683, Revision 6, as the basis for relocating their P-T and LTOP limits to a PTLR will need to include the following information in the plant-specific license amendment requests:

- describe the methodology used to calculate the neutron fluence values for the reactor vessel materials, including a description of whether or not the methodology is consistent with the recommended guidance of Draft Regulatory Guide DG-1053, a description of the computer codes used to calculate the neutron fluence values, and a description of how the computer codes for calculating the neutron fluence values were benchmarked; and

- provide the values of neutron fluence used for the adjusted reference temperature calculations, including the values of neutron fluence for the inner surface (ID), 1/4T and 3/4T locations of the RPV.

2.2 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

Technical Element 2 (Criterion 2) of the Table in Attachment 1 to GL 96-03 states that the reactor vessel material surveillance program must comply with the requirements of Appendix H to Part 50, and that the reactor vessel material irradiation surveillance specimen removal schedule must be provided, along with a discussion of how the specimen examinations will be used to update the PTLR curves. To satisfy Criterion 2, the Table states that the methodology for generating P-T limit curves should briefly describe how these surveillances are to be implemented, preferably by discussing how design and implementation of the material surveillance program for a given facility will be sufficient to comply with the program design, surveillance capsule withdrawal schedule, specimen testing, and reporting requirements of Appendix H to Part 50.

Section 2.0 of CE NPSD-683, Revision 6, discusses how the RPV material surveillance programs for CE nuclear steam supply system (NSSS) designed reactors were designed to meet the RPV material surveillance program requirements of Appendix H to Part 50. Appendix H to Part 50 requires that the RPV material surveillance programs for light water nuclear reactors must conform with the RPV material surveillance program criteria (i.e., criteria for installation, design, withdrawal, and testing of RPV surveillance capsule specimens, and recording of fracture testing data) specified in the edition of ASTM Standard Practice E185 that is in effect on the date of the ASME Code to which the plant's RPV was purchased. The rule also allows licensees to use and apply the criteria and methods in later editions of ASTM Standard Practice E185 inclusive of the 1982 edition of the procedure. In Section 2.0 of the topical report, the CEOG discusses how the material surveillance programs for CE NSSS plants were designed and how these surveillance programs are sufficient to meet the design, withdrawal schedule, program implementation, surveillance capsule specimen testing, and reporting requirements of the version of ASTM Standard Practice E185 that was in effect at the time the RPV for the plant was purchased.

The CEOG emphasizes the need to comply with the following key regulatory criteria when licensees consider a change to their RPV material surveillance program withdrawal schedules:

- If the surveillance capsule withdrawal schedule is located within the TS, any proposed changes to the withdrawal schedule must be submitted as a license amendment request pursuant to the requirements of 10 CFR 50.90.

- If the surveillance capsule withdrawal schedule is not located within the TS, any proposed changes to the withdrawal schedule must be submitted to the NRC for review and approval pursuant to the requirements of Paragraph III.B.3. of Appendix H to Part 50.

- Proposed changes to the surveillance capsule withdrawal schedules that are not consistent with the withdrawal criteria of the version of ASTM Standard Practice E-185 of record, or with one of later versions of the standard practice endorsed in Appendix H to Part 50, must be accompanied with a request for an exemption for their use.

The CEOG's discussion of the CE NSSS RPV material surveillance programs and the criteria for changing surveillance capsule withdrawal schedules conform to the current NRC regulatory requirements in Appendix H to Part 50, and are therefore acceptable to the staff. In order to satisfy Criterion 2 of the Table in Attachment 1 to GL 96-03, licensees seeking to use CE NPSD-683, Revision 6, as the basis for relocating their P-T and LTOP limits to a PTLR must include the following information in the PTLRs that are submitted as part of their plant-specific license amendment requests:

- either provide the surveillance capsule withdrawal schedule in the proposed PTLR for the amendment, or reference in the PTLR by title and number the documents in which the schedule is located; and

- reference the surveillance capsule reports by title and number if the RT_{NDT} values are calculated using RPV surveillance capsule data.

Approval of a license amendment request to relocate the P-T limits and LTOP limits to a PTLR does not relieve a licensee of the requirement to submit any proposed changes to the reactor vessel material surveillance program to the NRC for review and approval. Consistent with findings given in Atomic Safety and Licensing Board's Memorandum and Order CLI-96-13 (Reference 19), and summarized in NRC Administrative Letter 97-04 (Reference 20), proposed changes to a material surveillance capsule program withdrawal schedule must be submitted to the staff for review and approval, and may require a license amendment. Pursuant to 10 CFR 50.90, changes to material surveillance program withdrawal schedules will require a license amendment, and an opportunity for public hearing (1) whenever the change to the withdrawal schedule involves a withdrawal schedule that is located within the plant-specific Technical Specifications, or (2) whenever the proposed withdrawal schedule is such that it no longer complies with the withdrawal schedule criteria stated in the ASTM Standard Practice E-185 within the facility's licensing basis. Proposed changes to a facility's withdrawal schedule which do not involve either of these conditions may be granted by the staff without need for a license amendment, but still need to be submitted for review and approval. For any other changes to a

facility's reactor vessel material surveillance program, licensees may need to fulfill appropriate exemption requirements as specified in clause (b) to 10 CFR 50.60 or review and approval requirements in Appendix H to Part 50 to obtain NRC staff approval.

2.3 LTOP Methodology

Technical Element 3 (Criterion 3) of the Table in Attachment 1 to GL 96-03 states that the LTOP system limits developed using NRC-approved methodologies may be included in the PTLR. To satisfy Criterion 3, the Table states that the methodology for generating P-T limit curves should describe how the LTOP limits will be calculated by applying system/thermal hydraulics and fracture mechanics.

Chapter 3.0 of CE NPSD-683, Revision 6, provides the CEOG's methodology for establishing the plant-specific LTOP limits. The methodology is proposed to allow, consistent with the recommendations in Criterion 3 of the Table, relocation of the LTOP system limits from the TS to a PTLR or similar licensee-controlled report. The proposed methodology states, and the staff emphasizes, that only the figures, values, and parameters associated with the P-T limits and LTOP setpoints may be relocated and controlled in the PTLR. Other LTOP driven limitations, such as the limits on reactor coolant pump (RCP) starts, RCP and decay heat removal pump operation, injection sources, pressurizer level, and other LTOP operational parameters must remain in and be controlled by the TS. In addition, a plant-specific NRC review of the implementation of the LTOP methodology would be required for any plant-specific proposal to change from one LTOP analysis method to another (e.g., a change in methodology to credit pressurizer steam volume for the first time), even if both methods are described and addressed in the topical report. The following subsections provide the staff's evaluation of the LTOP methodology proposed in CE NPSD-683, Revision 6.

2.3.1 <u>LTOP Enable Temperature</u>

The LTOP enable temperature is the reactor coolant inlet temperature below which the LTOP system is required to be aligned to the RCS and be capable of mitigating any postulated low temperature overpressure event. The proposed methodology provides two methods for calculating the LTOP enable temperature. The first method follows the guidance in Branch Technical Position (BTP) RSB 5-2. The second method follows the guidance of Paragraph G-2215 of the 1995 edition of Appendix G to the Code and corresponding addenda through 1996 Addenda.

According to the first method, which follows BTP RSB 5-2, the LTOP enable temperature is calculated as limiting RT_{NDT} + 90°F + $U_{instrument}$ + delta-T. In this formula, limiting RT_{NDT} refers to the highest RT_{NDT} for the beltline region, as determined from the RT_{NDT} evaluations for the beltline weld and base metal materials at the 1/4T and 3/4T locations of the RPV shell; $U_{instrument}$ refers to instrument error as determined using the guidance in RG 1.105 and ISA Standard S67.04-1994; and delta-T refers to the temperature difference between the reactor coolant and the RPV shell at the controlling location used for RT_{NDT}.

According to the second method, which follows Paragraph G-2215 of the 1995 edition of Appendix G to the Code and corresponding addenda through 1996 Addenda, the LTOP enable temperature is the greater of 200°F or limiting RT_{NDT} + 50°F + $U_{instrument}$ + delta-T. In this

formula, RT_{NDT} refers to the highest RT_{NDT} for the beltline region, as determined from the RT_{NDT} evaluations for the beltline weld and base metal materials at the 1/4T and 3/4T locations of the RPV shell; $U_{instrument}$ refers to instrument error as determined using the guidance in RG 1.105 and ISA Standard S67.04-1994; and delta-T refers to the temperature difference between the reactor coolant and the RPV shell at the controlling location used for RT_{NDT}.

For both methods, enable temperatures are allowed to be calculated separately for heatups and cooldowns. For cooldowns, the enable temperature must be based on the isothermal conditions (i.e., 0°F/hour cooldown rate) because this condition results in a bounding (i.e., highest) calculated enable temperature for all cooldown rates. For heatups, the enable temperature must be based on the highest heatup rate allowed within the LTOP region of interest because this condition results in a bounding (i.e., highest) calculated enable temperature for all heatup rates. In addition, the proposed methodology allows a licensee to use a single enable temperature if desired. When using only one, the greater of the two enable temperatures described above (i.e., the greater of the heatup and cooldown enable temperatures) must be used. This results in a conservative enable temperature for both heatup and cooldown operations.

The staff has reviewed the proposed methods for calculating the LTOP enable temperature and finds that the methods are either consistent with BTP RSB 5-2 or Paragraph G-2215 of the 1995 edition of Appendix G to the Code and corresponding addenda through 1996 Addenda, both of which have been accepted by the staff for calculating LTOP enable temperatures. In addition, the proposed methodology accounts for the temperature instrumentation uncertainty. Accounting for instrumentation uncertainty is necessary to ensure that the LTOP system is not enabled at temperatures less conservative than is required to protect the reactor vessel. Based on the above, the staff finds the proposed methodology acceptable.

2.3.2 Applicable P-T Limits for LTOP Analysis

Overpressure mass addition and energy addition transients are postulated and analyzed for low temperature conditions to demonstrate that the features provided for LTOP adequately protect the RCPB against brittle failure. The acceptance criteria used for these analyses are based on the P-T limits established for the reactor vessel beltline in accordance with the requirements of 10 CFR 50, Appendix G. Two methods for calculating the acceptance criteria for these analyses are provided in the proposed methodology. The first method utilizes the actual P-T limit values generated by Appendix G to the Code for acceptance criteria. These acceptance criteria are referred to as "Appendix G P-T limits". The second method utilizes 110 percent of the values of the "Appendix G P-T limits" for acceptance criteria. These acceptance criteria are referred to as "LTOP P-T limits."

Before being used in LTOP analyses, both types of P-T limits discussed above must be adjusted to the pressurizer using pressure correction factors to account for the static head between the reactor vessel beltline and the pressurizer reference locations, and the flow induced pressure drop between the reactor vessel inlet nozzle (or beltline) and the pressurizer surge nozzle in the hot leg. The maximum number of RCPs and shutdown cooling (SDC) pumps allowed to operate by TS must be accounted for when determining the flow induced pressure drop. In addition, for plants that have large capacity (over 1500 gpm) relief valves attached to the pressurizer, an adjustment must be made to account for the pressure

differential between the reactor vessel and the pressurizer. These pressure differentials result from flow-induced pressure losses in the surge line. The pressure differential resulting from the large capacity relief valves may either be included as part of the pressure correction factors used in calculating the P-T limits or be added to the peak transient pressure determined by the LTOP analysis.

Appendix G P-T limits and LTOP P-T limits are provided as P-T curves for various heatup and cooldown rates. The most conservative curve (i.e., the curve with the lowest pressure limit at a given temperature) must be used for LTOP transient analyses. P-T limits associated with certain heatup and/or cooldown rates may, however, be eliminated for certain temperature bands within the LTOP region if the applicable plant TS prohibits cooldowns or heatups at these rates. Using this technique, a licensee may propose to include TS restrictions to prohibit operations with certain heatup or cooldown rates and thereby increase the P-T limits for the LTOP analyses.

The staff has reviewed the above methods for establishing the acceptance criteria for LTOP transient analyses and finds them acceptable because they are consistent with either BTP RSB 5-2 or Paragraph G-2215 of the 1995 edition of Appendix G to the Code and corresponding addenda through the 1996 Addenda (as related to LTOP systems), both of which have been accepted by the staff.

A third method discussed in the proposed methodology utilizes ASME Code Case N-640 to calculate the acceptance criteria for LTOP analyses. Review of this method was not conducted for this safety evaluation because a plant-specific exemption request must be provided for its use. Plant-specific reviews for implementation of this third method will be conducted at the time of submittal of plant-specific exemption requests. It should be noted, however, that a plant may not apply both the ASME Code Case N-640 in combination with the "LTOP P-T limits" defined above. If a plant wishes to utilize Code Case N-640, it must use the "Appendix G, P-T limits" as the acceptance criteria for the LTOP transient analyses.

2.3.3 LTOP Transient Analysis Methodology

According to BTP RSB 5-2, "All potential overpressure events should be considered when establishing the worst-case event." Consideration of potential overpressure events has identified two limiting event types for LTOP analyses: an energy addition type and a mass addition type. Both analysis types must be performed for the entire LTOP range (i.e., from the LTOP enable temperature down to the reactor coolant inlet temperature corresponding to the boltup temperature) in order to demonstrate the adequacy of the LTOP system. After the most limiting peak pressures from both the energy addition and mass addition transient analyses have been identified and linked to specific reactor coolant temperatures ranges, these pressures are compared with the applicable P-T limits. The peak transient pressures from both types of analyses must be shown to be below the applicable P-T limits at the corresponding temperatures in order to demonstrate the adequacy of the LTOP system.

2.3.3.1 Analysis Approach and Assumptions

The proposed methodology utilizes the following assumptions for both energy addition and mass addition LTOP transient analyses:

1. Transient analyses must assume the most limiting operating conditions and system configurations allowed by TS at the time of the postulated cause of the overpressure event. Note that separate analyses may be performed for different temperature bands within the LTOP region.

2. The most limiting single failure must be assumed when analyzing LTOP events. Typically, when two relief valves are used for LTOP, the most limiting single failure is the failure of one of the relief valves. However, for redundancy, either relief valve must be capable of mitigating the LTOP events.

3. Credit must not be taken for letdown, RCPB expansion, or heat absorption by the RCPB for transient mitigation.

4. A water solid pressurizer must be assumed unless a limit on the maximum pressurizer water level (or a minimum steam volume) is included in the LTOP TS for the LTOP temperature region of interest. If a pressurizer level restriction is included in the TS, a steam volume may be assumed to exist. However, the steam volume may only be used in calculations of the pressurization rate for valve accumulation. For transient mitigation following valve lift, the analysis must assume a water solid RCS. The amount of steam volume assumed to initially exist for valve accumulation calculations must be less than the nominal limit in the TS by the amount that corresponds to the pressurizer level uncertainty as determined by using the guidance in RG 1.105 and ISA Standard S67.04-1994. In addition, when TS allow (or don't prevent) the presence of a gas other than steam in the steam volume (e.g., nitrogen), the analysis must be performed in a manner that bounds the resulting pressurization rate with that gas in the steam volume.

5. Heat input from pressurizer heaters' full capacity must be assumed.

6. Decay heat must be assumed as an additional input to maximize reactor coolant expansion. Decay heat must be calculated based on a cooldown at the maximum rate allowed by the TS from the point when the reactor is shut down to the point when the temperature of interest for the analysis is reached (i.e., LTOP enable temperature when one decay heat rate value is used or highest temperature in the temperature band for which the analysis will apply when different decay heat rate values are used for the different temperature bands). The calculated decay heat rate value may be used in transient analyses for temperatures below the temperature of interest used in calculating the decay heat rate. These decay heat rates must be used in the analyses for both heatup and cooldown operations.

7. Power operated relief valve (PORV) setpoints for the analyses must be greater than the nominal setpoint to account for the actuation loop uncertainty as determined using the guidance in RG 1.105 and ISA Standard S67.04-1994 and pressure accumulation due to finite PORV opening time.

These assumptions have been reviewed by the staff and found acceptable because: (1) they are needed to ensure that the analyses are performed in a manner that bounds actual plant operation, and (2) they are conservative with respect to peak transient pressure consideration.

In addition to the above, plants that use operator action for transient mitigation or termination must provide, in their plant-specific PTLR methodology submittal, a justification for the operator action time used in the analyses.

2.3.3.2 Energy Addition Analyses

For the energy addition case it is postulated that one RCP is started with the secondary side inventory of all steam generators at a higher temperature than the reactor coolant. For this case, energy is transferred from the secondary side of the steam generators to the reactor coolant causing the reactor coolant to heat up, expand, and pressurize the RCS.

The analytical model used for the energy addition event was not provided as part of the proposed methodology. Instead, the methodology states that the analytical model will be provided in the plant-specific PTLR methodology submittals. Therefore, the staff was not able to evaluate the adequacy of the energy addition analysis methodology. The staff will review each plant's energy addition analytical model and methodology on a plant-specific basis when submitted as part of a plant's PTLR methodology and provide its evaluation of the energy addition analysis methodology in the NRC's SE for the plant-specific license amendment.

2.3.3.3 Mass Addition Analyses

For the mass addition case it is postulated that an inadvertent safety injection actuation signal initiates injection from all high pressure safety injection (HPSI) pumps and charging pumps that are allowed, by TS, to be aligned to the RCS. While the safety injection tanks can be another source of injection for the mass addition event, injection from these tanks is excluded by placing restrictions in the applicable TS to ensure that the tanks are made incapable of causing an LTOP event. For the mass addition case, the mass addition results in pressurization of the RCS. The injection rate for the mass addition analyses is assumed to be the maximum possible combined flow rate from the HPSI and charging pumps allowed to be aligned to the RCS. The maximum flow rate may be determined by: (1) adding 10 percent to the design flow rate, (2) pump flow testing from the inservice testing program, or (3) referring to assumptions in the plant's safety analyses if these analyses establish the maximum delivery rates. When relying on testing, as is the case for the inservice testing program, measurement instrument uncertainty, as determined using the guidance in RG 1.105 and ISA Standard S67.04-1994, must be accounted for in order to ensure that the values used in the analyses bound actual plant operation.

In some cases, TS may contain different restrictions on injection capability for different temperature bands within the LTOP region. For these cases, different mass addition cases may be analyzed for the different temperature bands. A technique of dividing the LTOP region into smaller bands and including TS restrictions in certain bands can be used to obtain lower analysis peak pressures in the bands with the additional TS restrictions. For all cases, analyses assumptions must be consistent with the TS that are applicable in the temperature band of interest to the particular analysis.

To determine the magnitude of the pressurization that results from a mass addition event that is mitigated by a PORV, CE uses a method of equilibrium pressures. For this method, a mass addition curve is first generated. This curve includes the combined flow rates from all safety

injection and charging pumps allowed to be aligned to the RCS. The curve also includes equivalent flow rates calculated as a result of coolant expansion which is due to energy input from decay heat, pressurizer heaters, and RCPs allowed by TS to operate in the temperature range of interest. The equivalent flow rate resulting from the energy input is accounted for by shifting the combined mass addition curve for the injection pumps to the right (i.e., in the increasing flow rate direction) by the amount resulting from the energy input. The mass addition curve is developed in terms of flow rate into the RCS cold legs as a function of pressurizer pressure. The magnitude of the pressurization is determined by superposition of the mass addition curve on the relief valve discharge curve, both of which must be in terms of flow rate as a function of pressurizer pressure. The equilibrium pressure is taken as the pressure at the intersection of the two curves assuming liquid input and discharge. The equilibrium pressure is the pressure at which the mass addition rate matches the relief valve discharge flow rate. In addition to the equilibrium pressure, a maximum pressure at opening is also calculated for the PORV. This maximum pressure at opening is calculated as the sum of the opening pressure setpoint for the valve, pressure instrumentation uncertainty as determined using the guidance in RG 1.105 and ISA Standard S67.04-1994, and valve accumulation. Valve accumulation is calculated as the pressurization rate just prior to valve opening multiplied by the time it takes the valve to reach its full open position. The pressurization rate is the rate calculated by the mass addition transient corresponding to the temperature band of interest. For a water solid system, the pressurization rate is based on the rate of mass addition into a water solid system and a constant RCS volume. As stated earlier, for accumulation purposes, calculation of the pressurization rate may credit an initial steam volume in the pressurizer consistent with TS restrictions. In this case, the pressurization rate is calculated based on steam volume compression that is reversible and adiabatic and assuming the steam volume behaves as an ideal gas. The valve opening time must be consistent with the acceptance criteria for inservice testing of the subject valve. For the transient, the valve is assumed to stay closed until the maximum pressure at opening is reached. The peak transient pressure for a mass addition event mitigated by a PORV is then taken as the greater of the equilibrium pressure and the maximum pressure at opening.

Plants that rely on PORVs for LTOP transient mitigation must provide relief valve discharge curves for their PORVs as part of their plant-specific submittals for approval of their PTLR methodology. The PORV discharge curves must: (1) be developed using appropriate correlations, (2) be developed using a conservative back pressure, (3) account for discharge flow reduction due to flashing at the valve outlet when the discharged water has a low degree of subcooling, (4) relate the valve discharge flow rate with either valve inlet pressure or pressurizer pressure, (5) cover the anticipated range of pressures, (6) account for the inlet piping pressure drop, and (7) not be related to a pressure setpoint.

For events mitigated by a spring loaded SDC relief valve or a spring loaded pressurizer relief valve, analyses must assume that these valves will start to open at 3 percent accumulation above the set pressure. At 3 percent accumulation the valves are assumed to open to 30 percent of rated flow. Full rated flow is assumed to be reached at 10 percent accumulation. Between 3 and 10 percent accumulation, it is assumed that the discharge flow rate changes linearly with inlet pressure. These assumptions are used unless the valve manufacturer's recommendations are more conservative with respect to peak transient pressure. If the manufacturer's recommendations would result in a higher peak transient pressure, then the manufacturer's recommendations are used in the analyses. A set pressure tolerance is

normally required to be applied to the valve set pressure (i.e., to the pressure where the valve begins to open) for conservatism. However, the proposed methodology, which does not apply a set pressure tolerance, is acceptable since it assumes the valve inlet pressure accumulates 3 percent above the set pressure before any discharge occurs.

For the SDC relief valves, the pressure drop in the piping from the hot leg to the valve inlet must be considered for its effect on the peak transient pressure. In addition, the elevation head from the valve to the pressurizer must also be considered in the analysis. For the pressurizer relief valves, the pressure drop in the inlet piping must be considered for its effect on the peak transient pressure. For both types of valves, if the full rated flow at 10 percent accumulation exceeds the mass input from the mass addition transient, the peak pressure at the inlet will be maintained below 10 percent accumulation. If the full rated flow at 10 percent accumulation is less than the mass input from the mass addition transient, the peak pressure at the inlet will be above 10 percent accumulation. When relying on SDC relief valves or pressurizer relief valves, the peak transient pressure is calculated as the pressure, above 3 percent accumulation, at which the valve discharge rate equals mass input rate.

For a plant that relies on spring loaded SDC relief valves or pressurizer relief valves, if the peak transient pressure is above the 10 percent accumulation pressure, the plant must submit its valve discharge curves as part of their plant-specific submittals for approval of their PTLR methodology.

As stated earlier, for plants with large capacity (over 1500 gpm) relief valves attached to the pressurizer, an adjustment must be made to account for the pressure differential between the reactor vessel and the pressurizer due to flow induced pressure losses in the surge line. This pressure difference may either be included in the pressure correction factors for the P-T limits or be added to the peak transient pressure.

The staff has reviewed the methods proposed for performing mass addition analyses and finds that, when performed in the manner discussed above, the analyses will bound actual plant operation. In addition, the methods described are consistent with BTP RSB 5-2. Based on the above, the staff finds the proposed methods for analyzing the mass addition events acceptable.

2.4 Methodology for Calculating Adjusted Reference Temperatures for Reactor Vessel Materials

Technical Element 4 (Criterion 4) of the Table in Attachment 1 to GL 96-03 states that the adjusted reference temperature (abbreviated as ART in the Table and as RT_{NDT} in this SE) for each reactor beltline material shall be calculated in accordance with RG 1.99, Revision 2. To satisfy Criterion 4, the Table states that the methodology for generating P-T limit curves should briefly describe the method for calculating the RT_{NDT} values for the RV beltline materials consistent with the methodology of RG 1.99, Revision 2.

Section 3.0 of CE NPSD-683, Revision 6, discusses how the RT_{NDT} values for the RPV beltline materials will be calculated in a manner that is consistent with the methods of calculation stated in RG 1.99, Revision 2. This satisfies Criterion 4 of the Table in Attachment 1 to GL 96-03, and is therefore acceptable to the staff. In order to satisfy Criterion 4 of the Table in Attachment 1 to GL 96-03, licensees seeking to use CE NPSD-683, Revision 6, as the basis for relocating

their P-T and LTOP limits to a PTLR must include the following information in the PTLRs that are submitted as part of their plant-specific license amendment requests:

- identify the limiting materials and corresponding RT_{NDT} values for both the quarter-thickness (1/4T) and three-quarter-thickness (3/4T) locations of the RV shell; and

- for pressurized water reactor (PWR) design facilities, identify the limiting RT_{PTS} value for RV as calculated in accordance with the methods and criteria of 10 CFR 50.61.

2.5 Methodology for Generating P-T Limit Curves

Technical Element 5 (Criterion 5) of the Table in Attachment 1 to GL 96-03 states that the limiting RT_{NDT} value shall be incorporated into the calculation of the P-T limit curves in accordance with NUREG-0800, Standard Review Plan (SRP) Section 5.3.2, "Pressure Temperature Limits." To satisfy Criterion 5, the Table states that the methodology for generating P-T limit curves should describe how the application of fracture mechanics is used to construct the P-T limit curves consistent with the methods of Appendix G to the Code and SRP Section 5.3.2.

Section 5.0 of the topical report provides a detailed discussion of how the application of fracture mechanics is used in the construction of P-T limit curves. The discussion addresses the following topics:

- a general overview;
- regulatory requirements for generating P-T limit curves;
- methods for calculating limiting reference stress intensity factor (K_{iR}) values; and
- CE NSSS method for generating P-T limit curves, including the methods for calculating the stress intensities resulting from thermal and membrane stresses, and the maximum allowable pressures for the curves

The CEOG's discussion of the CE NSSS method for generating P-T limit curves is consistent with and satisfies Criterion 5 of the Table to GL 96-03. The CE NSSS method is designed to be consistent with requirements of Section VI.A.2 of Appendix G to Part 50, as exempted pursuant to the exemption request provision of 10 CFR 50.60(b). The sections that follow provide the staff's assessment of the individual topics discussed in Section 5.0 of CE NPSD-683, Revision 6.

2.5.1 General Overview

In Section 5.1 of the topical report, the CEOG states that the following ferritic components of the RCPB are addressed by Appendix G to the Code: (1) vessels, (2) piping, pumps and valves, and (3) bolting materials. The CEOG identifies that of these materials, the RPV is the only component for which a linear elastic fracture mechanics (LEFM) evaluation is necessary. The CEOG identifies that the test and acceptance standards to which the other components are designed are adequate to protect against potential non-ductile (brittle) failures. The CEOG states that over the CE NSSS fabrication history, the following RPV regions were considered,

but not necessarily specifically evaluated, in the analysis for establishing the brittle fracture limits for a CE-designed plant: (1) beltline, (2) vessel wall transition, (3) bottom head juncture, (4) core stabilizer lugs, (5) flange region, (6) inlet nozzle, and (7) outlet nozzle. Of these regions, the CEOG identifies that the beltline region is the only region of the RPV that will be exposed to a neutron flux that is high enough to result in radiation-induced embrittlement. The CEOG therefore identifies that the applicability of the CE NSSS method for generating P-T limit curves, as discussed in Chapter 5.0 of the report, will be limited to evaluations of the base metal and weld materials within the beltline region of RPVs.

The CEOG defines the *beltline* as the region of the RPV that "immediately surrounds the reactor core and is exposed to the highest levels of fast neutron fluence." In contrast, the NRC defines (in Appendix G to Part 50) the *beltline* as the "region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron irradiation damage to be considered in the selection of the most limiting material with regard to radiation damage." Appendix H to Part 50 defines that the threshold for irradiation damage occurs when the accumulated neutron fluence for the ferritic material is greater than $1.0x10^{17}$ n/cm^2 (E > 1 MeV). The CEOG definition tends to limit the RPV region being analyzed only to that immediately surrounding the active core.

In the RAI dated October 30, 2000, the staff informed the CEOG that the definition in CE NPSD-683, Revision 6, for the beltline region should conform to the definition stated in Section II of Appendix G to Part 50. In its submittal of November 16, 2000, the CEOG stated that the report will be modified to make the definition of the beltline region of the vessel consistent with the definition in Appendix G to Part 50, but qualified this statement by adding the definition does not include discontinuities, such as nozzles or ledges. CE NSSS designed reactors typically do not have these type of discontinuities within the beltline regions of the RPVs. However, to be consistent with definition of the *beltline* in Appendix G to Part 50, licensees requesting to use the CE NSSS method as the basis for generating their P-T limit curves need to ensure that the ferritic RPV materials that have accumulated neutron fluences in excess of $1.0x10^{17}$ n/cm^2 will be assessed according to Section 4.0 of CE NPSD-683, Revision 6, regardless of whether the materials are located within the region immediately surrounding the active core.

2.5.2 Regulatory Requirements for Generating P-T Limit Curves

In Section 5.2 of the topical report, the CEOG summarizes the regulatory requirements that need to be met when generating the P-T limit curves that will be included as part of the PTLR for a CEOG member PWR-designed nuclear power plant. The CEOG’s guidance for licensees requesting approval of a license amendment for a PTLR is centered on generating P-T limits in accordance with methods of Appendix G to the Code, as modified by certain exemptions to the methodology that will be requested by the licensees requesting approval of the PTLR.

In the section, the CEOG indicates that the P-T limits for CEOG member plants will satisfy the following equations:

$$2K_{Im} + K_{It} < K_{IR}, \quad \text{for Level A and B (normal and upset) loading conditions} \qquad (2)$$

$$1.5K_{Im} + K_{It} < K_{IR}, \quad \text{for hydrostatic/leak-rate testing conditions, core not critical} \qquad (3)$$

In these equations, K_{Im} is the stress intensity in the vessel arising from the primary membrane stress, K_{It} is the stress intensity arising from the thermal gradient across the RPV shell wall, and K_{IR} is the reference stress intensity factor for the limiting material in the RPV. This is consistent with Appendix G to Part 50, and with the methodology of Appendix G to the Code. In this section, the CEOG also provides a summary of the minimum temperature requirements for PWR designed facilities that are stated in Table 1 of Appendix G to Part 50. These guidelines are therefore acceptable to the staff.

2.5.3 Method for Calculating Reference Stress Integrity Factor Values

Section 5.3 of the topical report indicates that the K_{IR} values for the beltline materials will be determined in accordance with one of the following methods:

- with the methods of Article G-2110 of Appendix G to the Code for calculating K_{Ia} (the lower bound crack arrest critical stress intensity factor, K_{IA} in the topical report), as defined by the following expression:

$$K_{Ia} = 26.78 + 1.223 * e^{[\,0.0145 * (\,T - RT_{NDT} + 160)\,]} \qquad (4)$$

- with the methods of ASME Code Case N-640 for calculating K_{Ic} (the lower bound static crack initiation critical stress intensity factor, listed as K_{IC} in the topical report), as defined by the following expression:

$$K_{Ic} = 33.20 + 20.73 * e^{[\,0.0200 * (\,T - RT_{NDT}\,)\,]} \qquad (5)$$

Licensees seeking to use CE NPSD-683, Revision 6, as the basis for relocating their P-T and LTOP limits to a PTLR must identify which method (i.e., K_{Ic} or K_{Ia}) will be used to calculate the reference intensity factor (K_{IR}). Use of K_{Ic} will generate P-T curves that are less conservative than would be generated using K_{Ia}, which is the reference stress intensity factor used in the methods of Appendix G to the Code. Therefore, pursuant to the provisions of 10 CFR 50.60(b), any license amendment request for a PTLR that seeks to use Code Case N-640 and K_{Ic} as part of the bases for generating the P-T limit curves must be accompanied with an appropriate exemption request to deviate from complying with Section IV.A.2.b of Appendix G to Part 50. The staff will approve an exemption request to use Code Case N-640 and K_{Ic} as the bases for generating the P-T limit curves only if a licensee indicates that it will limit the maximum pressure in the vessel to 100 percent of the pressure satisfying Paragraph G-2215 of the 1996 Edition of Appendix G to the Code for establishing LTOP limit setpoints. These conditions are consistent with Note (2) on page 5-6 of CE NPSD-683, Revision 6.

2.5.4 P-T Limit Curve Generation Methods

Section 5.4 of the topical report provides the CEOG's CE NSSS methodology for generating the P-T limits that will be incorporated into a plant-specific PTLR. The methods include methods for calculating both K_{Im} and K_{It} values. The CEOG's methodologies for calculating these values are both based on finite element modeling (FEM) methods. In the letter dated October 30, 2000 (Reference 4), the NRC requested that the CEOG provide a description of the finite element modeling methods for calculating K_{Im} values. In its letter of November 16, 2000 (Reference 5), the CEOG stated that the FEM methods for calculating K_{Im} values were

provided to the NRC in proprietary Evaluation No. 063-PENG-ER-096, Revision 00 (Reference 21), which was submitted by the New York Power Authority as part of its exemption request to use an alternative method for calculating the K_{It} factors used in the generation of P-T limits for the Indian Point Unit 3 nuclear plant (Reference 22).

The current methodology of Appendix G to the Code endorsed by the NRC incorporates the most recent LEFM solutions for determining K_{Im} and K_{It} factors. These solutions are based on stress influence coefficients from FEM analyses for inside surface flaws performed at Oak Ridge National Laboratories (ORNL) and work published by Electric Power Research Institute (EPRI) for outside surface flaws. The FEM models considered uniform, linear, quadratic, and cubic thermal stress profiles respectively. The staff considers the current methodology in the 1995 edition of Appendix G to the Code provides a better method of estimating K_{Im} and K_{It} factors than does the methodology in the 1989 edition of Appendix G to the Code without reducing safety margins associated with these K_I values.

On the surface, the CE methodology appears to differ from the current Appendix G methodology in its K_{It} estimation. In the CE NSSS methodology, the K_{It} is calculated using thermal influence coefficients developed from 2-dimensional (2-D) FEM models with linear, quadratic, and cubic vessel temperature profiles. These thermal influence coefficients are then corrected for the 3-D elliptical crack geometry using the procedures of Appendix A to Section XI of the ASME Code (Reference 23). Theoretically, using CE's thermal influence coefficients is equivalent to using the stress influence coefficients of the current Appendix G methodology. The K_{It} estimation starts from a vessel temperature profile to an intermediate stress profile, then to a final K_{It} value. Mr. J. A. Keeney and Mr. T. L. Dickson of ORNL have demonstrated (Reference 24) that the influence coefficients developed by ORNL using 3-D FEM models agree with the influence coefficients using a shape-factor (Q-factor) approach to account for the 3-D crack geometry. This Q-factor approach is similar to the Section XI Appendix A approach used in the CE methodology. Thus, the alternative methodology in CE Evaluation No. 063-PENG-ER-096, Revision 00, for calculating K_{It} factors is similar to that in the most recent edition of Appendix G to the Code endorsed by the NRC. The staff approved New York Power Authority's request to use the CEOG's alternative methodology for calculating K_{It} factors on April 10, 1998 (Reference 25).

The CE NSSS methodology does not invoke the methods in the 1995 edition of Appendix G to the Code for calculating K_{Im} factors, and instead applies FEM methods for estimating the K_{Im} factors for the RPV shell. Upon a second review of CE Evaluation No. 063-PENG-ER-096, Revision 00, the staff has determined that the evaluation contains sufficient information to assess the method for calculating K_{Im} factors. Except for loading inputs, the staff has determined that the K_{Im} calculation methods apply FEM modeling that is similar to that used for the determination of the K_{It} factors. The staff has also determined that there is only a slight non-conservative difference between the P-T limits generated from the 1989 edition of Appendix G to the Code and those generated from CE NSSS methodology as documented in Evaluation No. 063-PENG-ER-096, Revision 00. The staff considers this difference to be reasonable and should be consistent with the expected improvements in P-T generation methods that have been incorporated into the 1995 edition of Appendix G to the Code. The staff therefore concludes that the CE NSSS methodology for generating P-T limits is equivalent to the current methodology in the 1995 edition of Appendix G to the Code, and is acceptable for P-T limit applications. However, since the staff cannot determine whether the CE NSSS

method for generating P-T limits will be as conservative as those which would be generated using the methods of the 1995 edition of Appendix G to the Code, licensees seeking to use the CE NSSS method as the basis for generating the P-T limits for their facilities will need to apply for an exemption against requirements of Section IV.A.2. of Appendix G to Part 50 to apply the CE NSSS methods to their P-T curves. This is consistent with the "note" on page 5-15 of CE NPSD-683, Revision 6. Exemption requests to apply the CE NSSS to the generation of P-T limit curves should be submitted pursuant to the provisions of 10 CFR 50.60(b) and will be evaluated on a case-by-case basis against the exemption request acceptance criteria of 10 CFR 50.12.

In order to satisfy Criterion 5 of the Table in Attachment 1 to GL 96-03, licensees seeking to use CE NPSD-683, Revision 6, as the basis for relocating their P-T and LTOP limits to a PTLR will also need to include in their PTLRs the P-T curves for heatup, cooldown, criticality, and hydrostatic and leak-rate tests of their reactors.

2.6 Minimum Temperature Requirements

Technical Element 6 (Criterion 6) of the Table in Attachment 1 to GL 96-03 states that the "minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves." To satisfy Criterion 6, the Table states that the methodology for generating P-T limit curves should describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to the P-T limit curves.

In Section 6.0 of the topical report, the CEOG discusses how the minimum temperature requirements of Appendix G to Part 50 will be applied to the calculations of P-T limit curves. In Section 6.1, the CEOG lists the minimum temperature requirements for CE NSSS designed RPVs during normal operating conditions and during hydrostatic or leak rate pressure testing conditions. These minimum temperature requirements are based on whether the RCS operating temperature is above or below 20 percent of the preservice hydrostatic test pressure (PHTP) for the RCPB. The minimum temperature requirements listed in Section 6.1 for PWR-designed RPVs are consistent with the minimum temperature requirements listed in Table 1 of Appendix G to Part 50, and are therefore acceptable to the staff.

In Section 6.2 of the topical report, the CEOG states that the minimum boltup temperature for CE NSSS vessel will be established in accordance with Subparagraph G-2222.c of Appendix G to the ASME Code. The recommendations of Subparagraph G-2222.c are consistent with the minimum temperature requirements of Table 1 to Appendix G to Part 50 for normal operating conditions and pressure test conditions at pressures less than or equal to 20 percent of the PHTP, and are therefore acceptable to the staff.

In Section 6.3 of the topical report, the CEOG states ASME Code Section III, Article NB-2000, Subparagraph NB-2332 will be used to establish the lowest service temperature requirement (LSTR) for the ferritic materials used to fabricate the RCPB piping, pumps and valves. For normal operating conditions above 20 percent of the PHTP, Subparagraph NB-2332 requires the lowest service temperature to be established at a value equal to the highest RT_{NDT} value for these materials plus 100°F. This is acceptable to the staff.

Appendix G to Part 50, in part, requires that the P-T limits for PWR-designed RPVs must meet the following minimum temperature requirements as listed in Table 1 of the rule:

- for leak rate and hydrostatic testing conditions at operating pressures less than or equal to 20 percent of the PHTP, the minimum temperature operating requirement must be set to at least the limiting RT_{NDT} value for the closure flange region.

- for leak rate and hydrostatic testing conditions at operating pressures greater than 20 percent of the PHTP, the minimum temperature operating requirement must be set to at least the limiting RT_{NDT} value for the closure flange region plus 90°F.

- for normal operating conditions[(5)], with the reactor core not critical, at operating pressures less than or equal to 20 percent of the PHTP, the minimum temperature operating requirement must be set to at least the limiting RT_{NDT} value for the closure flange region.

- for normal operating conditions, with the reactor core not critical, at operating pressures greater than 20 percent of the PHTP, the minimum temperature operating requirement must be set to at least the limiting RT_{NDT} value for the closure flange region plus 120°F.

- for normal operating conditions, with the reactor core critical, at operating pressures less than or equal to 20 percent of the PHTP, the minimum temperature operating requirement must be set to the larger of the minimum permissible temperature for performing the inservice hydrostatic test or the limiting RT_{NDT} value for the closure flange region plus 40°F.

- for normal operating conditions, with the reactor core critical, at operating pressures greater than 20 percent of the PHTP, the minimum temperature operating requirement must be set to the larger of the minimum permissible temperature for performing the inservice hydrostatic test or the limiting RT_{NDT} value for the closure flange region plus 160°F.

In order to satisfy Criterion 6 of the Table in Attachment 1 to GL 96-03, licensees seeking to use CE NPSD-683, Revision 6, as the basis for relocating their P-T and LTOP limits to a PTLR will need to demonstrate how the P-T curves for pressure testing conditions and normal operations with the core critical and not-critical will be in compliance with the appropriate minimum temperature requirements as given in Table 1 to Appendix G to Part 50. For normal operating conditions of the RCS with the reactor core not critical and operating pressures greater than 20 percent of the PHTP, the LSTR for piping, pumps, and valves in the RCPB may not substitute as an alternative for meeting the corresponding minimum temperature

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(5) Including operating conditions during heatups and cooldowns of the RCS, and anticipated operational occurrences.

requirement for the RPV in Appendix G to Part 50 if the LSTR value is less than the corresponding minimum temperature requirement value.(6)

2.7 Methodology for Applying Reactor Vessel Material Surveillance Program Data Into Adjusted Reference Temperature and P-T Limit Curve Calculations

Technical Element 7 (Criterion 7) of the Table in Attachment 1 to GL 96-03 states that "licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature" (measure ΔRT_{NDT} value) to "the predicted increase in reference temperature" (predicted ΔRT_{NDT} value). To satisfy Criterion 7, the Table states that the methodology for generating P-T limit curves should: (1) describe how the data from multiple surveillance capsules will be used in the RT_{NDT} calculations, and (2) describe the procedure for evaluating the data if the measured ΔRT_{NDT} value exceeds the predicted ΔRT_{NDT} value.

Section 7.0 of the topical report addresses how the application of material surveillance data will be applied to the adjusted reference temperature (RT_{NDT}) calculations. The discussion summarizes the criteria that should be used for determining whether the surveillance data are credible consistent with Position 2.1 of RG 1.99, Revision 2. Although the CEOG's summary of the credibility criteria is consistent with RG 1.99, Revision 2, the credibility criteria cited by the CEOG are only an abridged version of those stated in the RG; for more accurate and detailed descriptions, the credibility discussion and criteria, as stated on page 1.99-2 of the RG, should be used as the basis for assessing the credibility of surveillance data used in the RT_{NDT} assessments. Section 7.0 of the report also provides an acceptable discussion of the criteria for using surveillance data that are obtained from Charpy impact testing of surveillance capsule specimens that have been irradiated at another facility (integrated data).

Section 7.0 of the report, however, does not address how the Charpy Impact surveillance data will be applied if the data falls outside of the $2\sigma_\Delta$ scatterband for the predicted mean ΔRT_{NDT} trend curve.(7) The staff expects that licensees who apply to use this PTLR methodology will address the credibility of Charpy impact data in such data sets in order to ensure that an approximately conservative evaluation of RPV material properties is used in the RPV integrity evaluations.

In order to satisfy Criterion 7 of the Table in Attachment 1 to GL 96-03, licensees seeking to use CE NPSD-683, Revision 6, as the basis for relocating their P-T and LTOP limits to a PTLR

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(6) That is, for normal operating conditions above above 20% of the PHTP, the LSTR may only substitute as an alternative basis for meeting the corresponding minimum temperature requirement in Appendix G to Part 50 if:

$$\text{Limiting } RT_{NDT\text{-piping, pumps, \& valves}} + 100^\circ F \geq \text{Limiting } RT_{NDT\text{-RPV Flange}} + 120^\circ F \quad \text{(when the core is not critical)} \qquad (6)$$

$$\text{Limiting } RT_{NDT\text{-piping, pumps, \& valves}} + 100^\circ F \geq \text{Limiting } RT_{NDT\text{-RPV Flange}} + 160^\circ F \quad \text{(when the core is critical)} \qquad (7)$$

(7) That is, how the Charpy impact surveillance data will be evaluated and applied to the PTS and P-T limit assessments if the measured ΔRT_{NDT} values for the surveillance capsule specimens, as determined from the Charpy impact tests, exceed by $2\sigma_\Delta$ the mean ΔRT_{NDT} value that is predicted through application of the methods of analysis in Position 2.1 of RG 1.99, Revision 2.

will need to: (1) include in their PTLRs the supplemental surveillance data and calculations of the chemistry factors if surveillance data are used for the calculations of the adjusted reference temperatures; and (2) provide the evaluation of whether the surveillance data are credible in accordance with the credibility criteria of RG 1.99, Revision 2. In addition, if licensees seek to use surveillance data from supplement plant sources, licensees must: (1) identify the source(s) of the data; and (2) either identify by title and number the safety evaluation report that approved the use of the supplemental data, along with a justification of why the data is applicable, or compare the licensee's (applicant's) data with the data from the supplemental plant(s) for both the radiation environments (i.e., neutron spectrums and irradiation temperatures) and the surveillance test results, and submit the data to the NRC for review and approval. Pursuant to Section III.C of Appendix H to Part 50, use of integrated surveillance data from an alternate facility, if not previously approved by the NRC, need to be submitted to NRC for review and approval. The staff will evaluate the submittal of integrated surveillance data in accordance with the evaluation criteria of Section III.C.1.a-e. of Appendix H to Part 50.

3.0 GL 96-03 PROCESS CRITERIA FOR SUBMITTING PTLR LICENSE AMENDMENT REQUESTS

In Attachment 1 to the GL 96-03, the staff stated that requests for relocation of the P-T limits would require the following three actions for staff review: (1) the licensee must base its P-T limit curves and LTOP limits on a previously approved methodology for reference in the technical specifications, (2) the licensee must develop a report such as a PTLR to contain the figures, values, parameters, and explanations relative to establishing these limits, and (3) the licensee must modify the applicable sections of the technical specifications accordingly.

The first two of the three requirements for relocating the P-T curves and LTOP system limits are an NRC-approved methodology and the associated reporting requirements in the PTLR. The PTLR will consist of the explanations, figures, values and parameters derived from the calculations. Because the PTLR will be provided to the NRC upon issuance after each fluence period or effective full power years (EFPYs) and after approval of the methodology, a licensee should provide its PTLR when the methodology is submitted, so that questions regarding the content and format of the PTLR may be addressed prior to its formal completion. In Attachment 1 to the GL, the staff also provided a Table (i.e., the Table stated on pages 4 and 5 of Attachment 1 to the GL) containing seven key technical elements that would need to be addressed both in the technical methodologies and the PTLRs if approval were to be considered by the staff.

The third requirement for relocating the P-T curves and LTOP system limits is the modification of the plant TS. To modify the plant TS, three separate actions are necessary in the following TS subsections: (1) "Definitions" – add the definition of a named formal report (i.e., PTLR or a similar document) that would contain the explanations, figures, values and parameters derived in accordance with an NRC-approved methodology, and consistent with all of the design assumptions and stress limits for cyclic operation; (2) "LCOs" – add the references to the PTLR noting that the P-T limits shall be maintained within the limits specified in the PTLR; and (3) "Administrative Controls" – add a reporting requirement to submit the PTLR to the NRC, when it is issued, for each reactor vessel fluence period. In Attachment 2 to the GL, the staff provided a model plant-specific safety evaluation (SE) for this purpose. In Attachments 3a through 3d, the staff provided STS sections, LCOs, Actions, Surveillance Requirements, and

Reporting Requirements which are affected as a result. It should be noted that the final amended Administrative Controls page(s) must refer to both the approved methodology and the NRC's safety evaluation that will be issued in approval of the plant-specific PTLR license amendment request.

4.0 CONCLUSION

CE NPSD-683, Revision 6, as written by the CEOG and as supplemented by the CEOG's letters of November 16 and 30, 2000, provides a methodology that may be used by licensees as the basis for establishing the P-T limits and LTOP system limits for PWR-designed light water reactors. While the contents of the report are technically acceptable, the report leaves the description of certain key methodology details up to the licensee applying for a license amendment to relocate the P-T limits and LTOP system limits into a PTLR. These items have been identified in Section 2.0, and are collectively re-stated in Section 5.0. Licensees requesting a license amendment to relocate the P-T limits and LTOP system limits into a PTLR or similar owner-controlled document will therefore need to address in their plant-specific submittals the information requested in Section 5.0.

5.0 LIST OF INFORMATION TO BE INCLUDED IN PLANT-SPECIFIC PTLR LICENSE AMENDMENT SUBMITTALS IN ORDER TO MEET GL 96-03 TABLE CRITERIA

Information needed to satisfy Criterion 1 of the Table in Attachment 1 to GL 96-03, which deals with the topic of neutron fluence calculational methods – Licensees will need to:

(1) describe the methodology used to calculate the neutron fluence values for the reactor vessel materials, including a description of whether or not the methodology is consistent with the guidance of Draft Regulatory Guide DG-1053, a description of the computer codes used to calculate the neutron fluence values, and a description of how the computer codes for calculating the neutron fluence values were benchmarked; and

(2) provide the values of neutron fluence used for the adjusted reference temperature (RT_{NDT}) calculations, including the values of neutron fluence for the inner surface (ID), 1/4T and 3/4T locations of the RPV.

Information needed to satisfy Criterion 2 of the Table in Attachment 1 to GL 96-03, which deals with the topic of reactor vessel material surveillance program designs and withdrawal schedules – Licensees will need to:

(3) either provide the surveillance capsule withdrawal schedule in the proposed PTLR for the amendment or reference in the PTLR by title and number the documents in which the withdrawal schedule is located; and

(4) reference the surveillance capsule reports by title and number if the RT_{NDT} values are calculated using RPV surveillance capsule data

Information needed to satisfy Criterion 3 of the Table in Attachment 1 to GL 96-03, which deals with the topic of describing the methodologies that will be used to establish the LTOP system limits – Licensees will need to:

(5) provide a description of the analytical method used in the energy addition transient analysis;

(6) provide a description of the analytical method used in the mass addition transient analysis, if different from that in Section 3.3.5 of the topical report;

(7) provide a description of the method for selection of relief valve setpoints;

(8) provide a justification for use of subcooled water conditions or a steam volume in the pressurizer;

(9) provide a justification for a less conservative method for determination of decay heat contribution if the method used is less conservative than the "most conservative method" described in the topical report;

(10) provide justification for operator action time used in transient mitigation or termination;

(11) provide correlations used for developing PORV discharge characteristics;

(12) provide spring relief valve discharge characteristics if different from those described in the topical report or if the peak transient pressure is above the set pressure of the valve plus 10 percent;

(13) provide a description of how the reactor coolant temperature instrumentation uncertainty was accounted for;

(14) provide a justification for the mass and energy addition transient mitigation which credit presence of nitrogen in the pressurizer; and

(15) identify and explain any other deviation from the methodology included in Section 3.0 of the topical report.

Information needed to satisfy Criterion 4 of the Table in Attachment 1 to GL 96-03, which deals with the topic of describing the methodologies that will be used to calculate the adjusted reference temperature values for the RPV materials – Licensees will need to:

(16) identify the limiting materials and corresponding RT_{NDT} values for both the quarter-thickness (1/4T) and three-quarter-thickness (3/4T) locations of the RPV shell; and

(17) for pressurized-water-reactor (PWR) design facilities, identify the limiting RT_{PTS} value for RPV as calculated in accordance with the methods and criteria of 10 CFR 50.61.

<u>Information needed to satisfy Criterion 5 of the Table in Attachment 1 to GL 96-03</u>, which deals with the topic of describing the methodologies used to generate plant specific P-T limit curves – Licensees will need to:

(18) ensure that the ferritic RPV materials that have accumulated neutron fluences in excess of $1.0x10^{17}$ n/cm^2 (E > 1MeV) will be assessed according to Section 4.0 of the CE Topical Report CE NPSD-683, Revision 6, regardless of whether the materials are located within the region immediately surrounding the active core;

(19) identify which method (i.e., K_{IC} or K_{IA}) will be used to calculate the reference intensity factor (K_{IR}) values for the RPV as a function of temperature;

(20) *(applicable only if Code Case N-640 and K_{IC} are used as the basis for calculating the K_{IR} values)* submit an exemption request [pursuant to alternative program provisions of 10 CFR 50.60(b)] to use the methods of Code Case N-640 and apply them to the P-T limit calculations. Note that the staff will approve an exemption request to use Code Case N-640 and K_{IC} as the bases for generating the P-T limit curves only if a licensee indicates that it will limit the maximum pressure in the vessel to 100 percent of the pressure satisfying Paragraph G-2215 of the 1996 Edition of Appendix G to the Code for establishing LTOP limit setpoints. This condition is consistent with Note (2) on page 5-6 of CE NPSD-683, Revision 6;

(21) *(applicable only if the CE NSSS methods for calculating K_{im} and K_{lt} factors, as stated in Section 5.4 of CE NPSD-683, Revision 6, are being used as the basis for generating the P-T limits for their facilities)* apply for an exemption against requirements of Section IV.A.2. of Appendix G to Part 50 to apply the CE NSSS methods to their P-T curves. This is consistent with the *"note"* on page 5-15 of CE NPSD-683, Revision 6. Exemption requests to apply the CE NSSS to the generation of P-T limit curves should be submitted pursuant to the provision of 10 CFR 50.60(b) and will be evaluated on a case-by-case basis against the exemption request acceptance criteria of 10 CFR 50.12; and

(22) include in their PTLRs the P-T curves for heatup, cooldown, criticality, and hydrostatic and leak tests of their reactors.

<u>Information needed to satisfy Criterion 6 of the Table in Attachment 1 to GL 96-03</u>, which deals with the topic of describing how the P-T limit curves for normal operations and pressure testing conditions will satisfy the appropriate minimum temperature requirements, as stated in Table 1 of Appendix G to Part 50 – Licensees will need to:

(23) demonstrate how the P-T curves for pressure testing conditions and normal operations with the core critical and not-critical will be in compliance with the appropriate minimum temperature requirements as given in Table 1 to Appendix G to Part 50.

<u>Information needed to satisfy Criterion 7 of the Table in Attachment 1 to GL 96-03</u>, which deals with the topic of how the plant-specific RPV material surveillance data will be evaluated and applied to the adjusted reference temperature calculations – Licensees will need to:

(24) include in their PTLRs the supplemental surveillance data and calculations of the chemistry factors if surveillance data are used for the calculations of the adjusted reference temperatures;

(25) provide the evaluation of whether the surveillance data are credible in accordance with the credibility criteria of RG 1.99, Revision 2;

(26) In addition, if licensees seek to use surveillance data from supplemental plant sources, licensees must:

(a) identify the source(s) of the data; and

(b) either identify by title and number the safety evaluation report that approved the use of the supplemental data, along with a justification of why the data is applicable; or compare the licensee's (applicant's) data with the data from the supplemental plant(s) for both the radiation environments (i.e., neutron spectrums and irradiation temperatures) and the surveillance test results, and pursuant to Section III.C of Appendix H to Part 50, submit the proposed integrated surveillance program and evaluation of the data to the NRC for review and approval.

6.0 REFERENCES

1. Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996.

2. Letter CEOG-00-286 from R. Phelps, Chairman - Combustion Engineering Owners Group, to the U.S. Nuclear Regulatory Commission Document Control Desk, "Submittal of Topical Report CE NPSD-683, Revision 6, `Development of a RCS Pressure Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications . . . ,' " September 29, 2000.

3. Combustion Engineering Owners Group Topical Report CE NPSD-683, Revision 6, "Development of a RCS Pressure Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," September 2000.

4. Letter from J. Cushing, Project Manager for the Combustion Engineering Owners Group, U.S. Nuclear Regulatory Commission, to R. Bernier, Chairman - Combustion Engineering Owners Group, "Request for Additional Information Regarding CE NPSD-683, Revision 6, `Development of a RCS Pressure Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications,'" October 30, 2000.

5. Letter CEOG-00-326 from R. Bernier, Chairman - Combustion Engineering Owners Group, to the U.S. Nuclear Regulatory Commission Document Control Desk, "Response to Information Request Concerning CEOG Topical Report CE NPSD-683, Rev. 06," November 16, 2000.

6. Letter CEOG-00-340 from R. Bernier, Chairman - Combustion Engineering Owners Group, to the U.S. Nuclear Regulatory Commission Document Control Desk, "Clarification of Response to Information Request Concerning CEOG Topical Report CE NPSD-683, Rev. 06," November 30, 2000.

7. Section 50.60 to Part 50 of Title 10, *Code of Federal Regulations,* "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operations."

8. Appendix G to Part 50 of Title 10, *Code of Federal Regulations,* "Fracture Toughness Requirements."

9. Appendix H to Part 50 of Title 10, *Code of Federal Regulations,* "Reactor Vessel Material Surveillance Program Requirements."

10. Section 50.12 to Part 50 of Title 10, *Code of Federal Regulations,* "Specific Exemptions."

11. Section 50.36 to Part 50 of Title 10, *Code of Federal Regulations,* "Technical Specifications."

12. Appendix G to Section XI of the American Society for Mechanical Engineers Boiler and Pressure Vessel Code, Division 1, "Fracture Toughness Criteria for Protection Against Failure."

13. ASTM Designation E185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

14. Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

15. Welding Resource Council Bulletin WRC-175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," August 1972.

16. ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," Section XI, Division 1, December 12, 1997.

17. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," Section XI, Division 1, February 28, 1999.

18. ASME Code Case N-514, "Low Temperature Overpressure Protection," Section XI, Division 1, February 12, 1992.

19. Atomic Safety and Licensing Board Memorandum and Order CLI-96-13, "In the Matter of the Cleveland Electric Illuminating Company," Docket No. 50-440-OLA-3, December 6, 1996.

20. NRC Administrative Letter 97-04, "NRC Staff Approval for Changes to 10 CFR Part 50, Appendix H, Reactor Vessel Surveillance Specimen Withdrawal Schedules," September 30, 1997.

21. CE/ABB Proprietary Evaluation 063-PENG-ER-096, Revision 00, "Technical Paper Comparing ABB/CE PT Curve to ASME Section III, Appendix G," January 22, 1998.

22. Letter from J. Knubel, Senior Vice President, New York Power Authority, to the U.S. Nuclear Regulatory Commission, Document Control Desk, "Indian Point 3 Nuclear Power Plant, ... Proposed Exemption From Requirements of 10 CFR 50.60 to Utilize Alternate Methodology to Determine K_{IT}," January 28, 1998.

23. Appendix A to Section XI of the American Society for Mechanical Engineers Boiler and Pressure Vessel Code, Division 1, "Analysis of Flaws."

24. Oak Ridge National Laboratory Letter ORNL/NRC/LTR-93/33, December 1993.

25. Letter from G. F. Wunder, Project Manager, U.S. Nuclear Regulatory Commission, to J. Knubel, Senior Vice President, New York Power Authority, "Exemption from the Requirements of 10 CFR 50.60, `Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation,' to Allow for Use of Alternative Methodology for Construction of Pressure Temperature Limit Curves - Indian Point Nuclear Generating Unit No. 3 (TAC No. M99928)," April 10, 1998.

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