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Subject: Submittal of Topical Report CE NPSD-683, Rev. 06, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," September 2000

The CE Owners Group herewith submits 12 copies of non-proprietary topical report CE NPSD-683 Rev. 06 for Nuclear Regulatory Commission (NRC) review and approval. Revision 06 supercedes, in their entirety, all previous versions of this topical report.

The enclosed report presents a methodology to enable CEOG utilities to relocate the Pressure-Temperature (P-T) limit curves and low temperature overpressure protection (LTOP) setpoint curves and values typically contained in Technical Specifications to a licensee-controlled document.

The format of CE NPSD-683, Rev. 06 conforms to the guidance of GL-96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." Rev. 06 has been completely revised and reformatted to reflect feedback obtained from discussions with the NRC staff, incorporate responses to staff RAIs issued on Rev. 05, and consolidate the methodology into a single generic approach.

Appendices A, B & C were provided in Rev. 05 as examples of implementation approaches and not for staff review and approval; these Appendices have been removed from Rev. 06. Likewise, Appendix D has been removed and essential elements of this appendix have been incorporated into the body of Rev. 06. Changes to plant specific TS or to NUREG-1432 (CE Standard TS) will be submitted under the existing TS change processes.

D047

OPPD remains the CEOG lead utility to reference CE NPSD-683, Rev. 06 and the PTLR methodology documented therein for implementation at the Fort Calhoun Station nuclear power plant.

The NRC should address technical questions related to CE NPSD-683 Rev. 06 to the Chairman of the CE Owners Group. Invoices for review fees should also be directed to the Chairman with a copy to:

Mr. Gordon Bischoff
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If you have any questions, please contact me.

Very truly yours,

R J Phelps

Ralph Phelps, Chairman
CE Owners Group

Enclosure: CE NPSD-683, Rev 06 (8 copies)

cc: J. Cushing, NRC w/4 copies
K. Holthaus, OPPD w/1 copy
J. Brown, APS w1 copy
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CEOG COMBUSTION ENGINEERING OWNERS GROUP

CE NPSD-683
Rev 06

Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications

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CE NPSD-683, Revision 06

Development of a RCS Pressure and Temperature Limits Report for Removal of P-T Limits and LTOP Requirements from the Technical Specifications

CEOG Task 1174, Final Report

September 2000

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ABSTRACT

An approach and methodology description is presented in this report [1] for utilities to relocate the Pressure-Temperature (P-T) limit curves, low temperature overpressure protection (LTOP) setpoint values and curves currently contained in the Technical Specifications (TSs) to a licensee-controlled document. The approach is based upon the guidance contained in Nuclear Regulatory Commission (NRC) Generic Letter (GL) 96-03. As part of the relocation, additional considerations were the Reactor Vessel (RV) surveillance program, including the capsule withdrawal schedule, and the calculation of Adjusted Reference Temperature (ART), including the determination of the neutron fluence and analysis of post-irradiation surveillance capsule measurements.

To substantiate relocation of the detailed information for affected Limiting Conditions for Operation (LCOs), a new licensee-controlled document called a Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) needs to be developed. This document is consistent with the recommendations of GL 96-03 and contains the detailed information needed to support the pertinent LCOs, which would remain in the TS.

Per GL 96-03, methodology descriptions for developing RCS P-T limits, establishing LTOP setpoints, calculating the ART, developing a RV Surveillance Program, and calculating Neutron Fluence to support the PTLR are provided in Sections 1-7.

No other methodologies beyond those currently used for CE NSSS designs are included herein, since a license amendment is required anytime a licensee changes methodology. A license amendment will also be required if the licensee chooses to change from one method to another method, even though both methods are described in this topical report.

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[1] This document is a product of a Combustion Engineering Owner's Group (CEOG) effort undertaken to create a generic PTLR document based on guidance presented in NRC GL 96-03. Revision 6 of CE NPSD-683 is a total revision and supercedes all previous revisions, i.e., Revisions 0 through 5.

PREFACE

Generic Letter (GL) 96-03, (Reference 3), was issued by the NRC to improve the maintenance of Technical Specifications (TSs) by allowing the relocation of certain requirements from the TSs into another licensee-controlled document called a Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR). This relocation simplifies the regulatory processing of frequently revised items such as RCS Pressure-Temperature (P-T) limits, Low Temperature Overpressure Protection (LTOP) setpoints, Reactor Vessel (RV) surveillance program post-irradiation test results, and neutron fluence calculation updates.

Once incorporated into the plant's Technical Specification, changes can be made to the PTLR per guidance outlined in GL 96-03. The GL recommends that a licensee submit a new administrative section that refers to the specific version of the methodology that has been approved by the NRC staff for generating P-T limit curves and LTOP system setpoints. The intent of this process is to allow licensees to relocate P-T Limits and LTOP setpoints from the Technical Specifications so that they can be changed using an NRC approved methodology without prior NRC approval.

BACKGROUND

In 1972, the Summer Addenda to the ASME Boiler and Pressure Vessel Code, Section III, incorporated Appendix G, "Protection Against Nonductile Failure" (Reference 9). This Appendix, although not mandatory, was issued to provide an acceptable design procedure for obtaining allowable loadings for ferritic pressure retaining materials in the Reactor Coolant Pressure Boundary (RCPB) components.

Shortly after publication of ASME Code Section III, Appendix G, a new Appendix to 10 CFR 50 entitled "Appendix G - Fracture Toughness Requirements" became effective on August 16, 1973. This Appendix imposed fracture toughness requirements on ferritic material of pressure-retaining components of the RCPB and mandated compliance with ASME Code Section III,

Appendix G. Compliance with 10 CFR 50 Appendix G was applicable to all light water nuclear power reactors both operating and under construction at that time. 10 CFR 50 Appendix G, was further revised in 1979, 1983 and 1995. (Note: In 1995, 10 CFR 50 redirected compliance to ASME Code Section XI, Appendix G.)

In addition to Appendix G, the RCPB must meet the requirements imposed by 10 CFR 50, Appendix A, General Design Criteria 14 and 31. These design criteria require that the RCPB be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture. The criteria also require that the RCPB be designed with sufficient margin to assure that when stressed under operating, maintenance, and testing loadings, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. Appropriate and conservative methods that protect the RCPB against nonductile failure have been developed by CE to comply with 10 CFR 50, Appendix G.

DESCRIPTION OF ACTIVITIES

GL 96-03 advised licensees that they may request a license amendment to relocate cycle dependent information, such as the P-T limit curves and LTOP system limits from their plant TSs to a PTLR or similar licensee-controlled document. Per GL 96-03, this topical report is divided into seven provisions to be addressed in the PTLR. Each provision addresses the recommended information to be provided including specific methodology descriptions in each. The seven provisions are:

1. Neutron Fluence Values
2. Reactor Vessel Surveillance Program
3. LTOP System Limits
4. Beltline Material ART
5. P-T Limits using limiting ART in the P-T Curve calculation

6 Minimum Temperature Requirements in the P-T curves

7 Application of Surveillance Data to ART calculations

These provisions are addressed in Sections 1-7, respectively, of this topical report in conformance with the matrix of GL 96-03.

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ACRONYMS AND ABBREVIATIONS

ADV	Atmospheric Dump Valve
AOO	Anticipated Operational Occurrence
ART	Adjusted Reference Temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
BNL	Brookhaven National Laboratory
BTP	Branch Technical Position
CE	CE Nuclear Power, LLC
CEOG	Combustion Engineering Owner's Group
CSEWG	Cross Section Evaluation Working Group
EFPY	Effective Full Power Years
ENDF	Evaluated Nuclear Data File
GL	Generic Letter
HAZ	Heat-Affected-Zone
HPSI	High Pressure Safety Injection
ISA	Instrument Society of America
LCO	Limiting Condition for Operation
LEFM	Linear Elastic Fracture Mechanics
LTOP	Low Temperature Overpressure Protection
NDE	Non-Destructive Examination
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ORNL	Oak Ridge National Laboratory
PCA	Pool Critical Assembly
PHTP	Pre-Service Hydrostatic Test Pressure
PORV	Power-Operated Relief Valve
P-T	Pressure-Temperature
PTLR	Pressure and Temperature Limits Report
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump

ACRONYMS AND ABBREVIATIONS

RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RSB	Reactor Systems Branch (of the NRC)
RV	Reactor Vessel
SDC	Shutdown Cooling
SG	Steam Generator
SIAS	Safety Injection Actuation Signal
SIT	Safety Injection Tank
TS	Technical Specification
WRC	Welding Research Council

TECHNICAL PREFACE

Reactor Coolant Pressure Boundary Operational Description

General

Currently 10 CFR 50, Appendix G imposes special fracture toughness requirements on the ferritic components of the RCPB. These fracture toughness requirements result in pressure restrictions which vary with RCS temperature. Determination of these restrictions requires that specific loading conditions be evaluated and the resulting P-T limits not be exceeded. The specific loading conditions, for which P-T limits are required, are as follows:

1. Normal operations (including anticipated operational occurrences as referenced in Section IV.A of 10CFR50, Appendix G) which include RV boltup, heatup and cooldown
2. Inservice hydrostatic pressure and leak tests when the core is not critical
3. Reactor core operation

A brief description of these conditions is provided below to highlight the typical process that must be followed to determine the physical loadings resulting from a particular mode of operation.

Normal Operation

Reactor Vessel Boltup

RV boltup loads are generated by stud tensioners when securing the closure head against the RV. Prior to tensioning of the studs to the required preload, the reactor coolant temperature and the volumetric average temperature of the closure head region must be at or above the minimum boltup temperature. Once the studs have been tensioned, the RCS is capable of being pressurized and heated. The heatup transient begins when a Reactor Coolant Pump

(RCP) is started or when Residual Heat Removal (RHR) system flow is altered to allow elevation of the RCS temperature.

Heatup

Heatup is the process of bringing the RCS from a COLD SHUTDOWN condition to a HOT SHUTDOWN condition. The increase in temperature from COLD SHUTDOWN to HOT SHUTDOWN is achieved by RCP heat input and any residual core heat.

During the heatup transient, the reactor coolant temperature is considered essentially the same throughout the RCS with the exception of the pressurizer. The pressurizer is used to maintain system pressure within the normal operating window which is between the minimum pressure associated with RCP operating limits and the maximum pressure meeting the RV material fracture toughness requirements. Also, the heatup rate must not exceed the rates specified by the P-T limits.

Cooldown

During cooldown the RCS is brought from a HOT SHUTDOWN condition to a COLD SHUTDOWN condition. Initially, coolant temperature reduction is achieved by removing heat through use of the SGs by dumping the steam directly to the condenser or to the atmosphere through the Atmospheric Dump Valves (ADVs). The fluid temperature is decreased from approximately 550°F to 300°F using this method. To complete the cooldown, the RHR System is utilized.

Typically, cooldown is initiated by securing one or more RCPs. Any remaining pumps provide coolant circulation through the RCS so that heat is transferred from the RCS to the secondary side of the SGs. The RCS cooldown rate is controlled by the steam flow rate on the secondary-side that is in turn controlled by the steam bypass control system or ADVs. The RCS pressure

is controlled with the pressurizer through use of heaters and spray. Once pressure and temperature have been reduced to within the design values of the RHR, the RHR can be utilized to control the cooldown rate and the remaining RCPs can be stopped. It is advisable to initiate RHR flow prior to stopping all RCPs to provide sufficient mixing and minimize the thermal shock to RCPB components.

The pressure during cooldown is maintained between the maximum pressure needed to meet the fracture toughness requirements for this condition and the minimum pressure mandated by RCP operating limits. The cooldown rate must not exceed the appropriate rates specified by the P-T limits.

Inservice Hydrostatic Pressure Test And Leak Tests

In order to perform a system leak test or hydrostatic pressure test, the system is brought to the HOT SHUTDOWN condition. The heatup or cooldown processes, described previously, would be followed to achieve a HOT SHUTDOWN condition.

The pressure tests are performed in accordance with the requirements given in ASME Code Section XI, Article IWA-5000. For the system leakage test, the test pressure must be at least the nominal operating pressure associated with 100% rated reactor power. In the case of the hydrostatic pressure test, the test pressure is based on the requirements of ASME Code Section XI, IWB-5222. It is CE's practice to recommend that the inservice hydrostatic test for CE NSSS designs be performed at a test pressure corresponding to 1.1 times the operating pressure with the reactor core not critical. The minimum temperature for the required pressure is determined by the fracture toughness requirements and guidance provided in 10 CFR 50, Appendix G.

Reactor Core Operation

The minimum temperature at which the core can be brought critical is controlled by core physics and safety analyses. This temperature is typically in excess of 500°F. The heatup process described previously is used to attain the required temperature. Also, this minimum temperature is much higher than the requirements imposed by 10 CFR 50 Appendix G which only address brittle fracture.

1.0 NEUTRON FLUENCE CALCULATIONAL METHODS

This section describes an outline of a general methodology for neutron fluence calculations. Due to the variety of dosimeter types which may be in use by any plant, and the plant specific nature of calculations for fluence, specific details of the methodology with regards to the dosimeter types used for the plant, methods qualification including analytical benchmark analyses to determine bias and uncertainty, and plant-specific methods and results (including uncertainties) shall be addressed in detail by the plant-specific PTLR fluence analysis section.

The methods and assumptions described in this report apply to the calculation of vessel fluence for core and vessel geometrical and material configurations typical of CE NSSS designed pressurized water reactors. This methodology meets the guidance of Draft Regulatory Guide 1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

The prediction of the vessel fluence is made by a calculation of the transport of neutrons from the core out to the vessel and cavity. The calculations consist of the following steps: (1) determination of the geometrical and material input data, (2) determination of the core neutron source, and (3) propagation of the neutron fluence from the core to the vessel and into the cavity. A qualification of the calculational procedure is described later.

The discrete ordinate method should be used for the calculation of pressure vessel fluence. The DOT-4 code was commonly used in the United States and has been recently replaced by the DORT (2-D) and TORT (3-D) transport codes.

1.1 Input Data

1.1.1 Materials and Geometry

Detailed material and geometrical input data are used to define the physical characteristics that determine the attenuation of the neutron flux from the core to the locations of interest on the

pressure vessel. These data include material compositions, regional temperatures, and geometry of the pressure vessel, core, and internals. The geometrical input data includes the dimensions and locations of the fuel assemblies, reactor internals (shroud, core support barrel, and thermal shield), the pressure vessel (including identification and location of all welds and plates) and cladding, and surveillance capsules. For cavity dosimetry, input data also includes the width of the reactor cavity and the material compositions of the support structure and concrete (biological) shielding, including water content, rebar and steel. The input data are based, to the extent possible, on documented and verified plant-specific as-built dimensions and materials. The isotopic compositions of important constituent nuclides within each region are based on as-built materials data. In the absence of plant-specific information, nominal compositions and design dimensions can be used; however, in this case conservative estimates of the variations in the compositions and dimensions should be made and accounted for in the determination of the fluence uncertainty. The determination of the concentrations of the two major sources of isotopes responsible for the fluence attenuation (e.g., iron and water) are emphasized. The water density is based on plant full power operating temperatures and pressures, as well as standard steam tables. The data input includes an accounting of axial and radial variations in water density caused by temperature differences in the core and inside the core barrel.

1.1.2 Cross-Sections

The calculational method to estimate vessel damage fluence uses neutron cross-sections over the energy range from ~0.1 MeV to ~15 MeV. Draft Regulatory Guide 1053 recommends the use of the latest version of the Evaluated Nuclear Data File (ENDF/B-VI). The ENDF/B-VI files were prepared under the direction of the Cross Section Evaluation Working Group (CSWEG) operated through the National Nuclear Data Center at Brookhaven National Laboratories (BNL). These data have been thoroughly reviewed, tested, and benchmarked. Cross-section sets based on earlier or equivalent nuclear data sets that have been thoroughly benchmarked for a specific application may be used for that application.

1.1.2.1 Multi-group Libraries

Since the discrete ordinates transport code used to determine the neutron fluence uses a multi-group approximation, the basic data contained within the ENDF files must be pre-processed into a multi-group structure. The development of a multi-group library considers the adequacy of the group structure, the energy dependence of the flux used to average the cross-sections over the individual groups, and the order of the Legendre expansion of the scattering cross-section. Sufficient details of the energy- and angular-dependence of the differential cross-sections (e.g., the minima in the iron total cross-section) should be included to preserve the accuracy in attenuation characteristics.

It should be noted that in many applications the earlier ENDF/B-IV version and the first three Mods of the ENDF/B-V iron cross-sections result in substantial underprediction of the vessel inner-wall and of the cavity fluence. Updated ENDF/B-V iron cross-section data have been demonstrated to provide a more accurate determination of the flux attenuation through iron and are strongly recommended. These new iron data are included in ENDF/B, version VI.

1.1.2.2 Constructing a Multi-group Library

The ENDF files (including ENDF/B-VI) were first processed into problem-independent, fine-multi-group, master library containing data for all required isotopes. This master library (e.g., VITAMIN-B6) was developed at Oak Ridge National Laboratory and includes a sufficiently large number of groups (199) such that differences between the shape of the assumed flux spectrum and the true flux have a negligible effect on the multi-group data. This library includes 62 energy groups above 1 MeV and 105 groups above 0.1 MeV. The library also contains 42 photon energy groups.

The master library is collapsed into a job (broad group) library over spectra that closely approximate the true spectra. The resulting library should contain ~47 neutron and ~20 photon groups. This reduction is accomplished with a one-dimensional calculation that includes the discrete regions of the core, vessel internals, by-pass and downcomer water, pressure vessel,

reactor cavity, shield, and support structures. This job library should include approximately 20 energy groups above ~0.1 MeV. The collapsing is performed over four different spectra typical of PWRs, i.e. the core, downcomer, concrete and vessel. Both master (VITAMIN-B6) and job libraries are available from Oak Ridge National Laboratory.

1.2 Core Neutron Source

The determination of the neutron source for the pressure vessel fluence calculations accounts for the temporal, spatial, and energy dependence together with the absolute source normalization.

The spatial dependence of the source is based on two-dimensional or three-dimensional depletion calculations that incorporate actual core operation or from measured data. The accuracy of the power distributions shall be demonstrated. The depletion calculations may be performed in three dimensions, so as to provide the source in both the radial and axial directions.

The core neutron source is determined by the power distribution (which varies significantly with fuel burnup), the power level, and the fuel management scheme. The detailed state-point dependence must be accounted for, but a cycle average power distribution inferred from the cycle incremental burnup distribution can also be used. The cycle average power distribution is updated each cycle to reflect changes in fuel management. For the extrapolation to the end of life fluence, a best estimate power distribution is used, which is consistent with the anticipated fuel management of future cycles.

The peripheral assemblies, which contribute the most to the vessel fluence, have strong radial power gradients, and these gradients are accounted for to avoid overprediction of the fluence. The pin-wise source distribution generated by the depletion calculation is used for best-estimate, and represents the absolute source distribution in the assembly. When the actual planar core rectangular geometry can not be modeled (e.g., in the case of (r-θ) discrete

ordinates calculations), the pin power distribution in (x-y) geometry is converted into a (r-θ) distribution as required by the (r-θ) transport code geometry.

The local source is determined as the product of the fission rate and the neutron yield. The energy dependence of the source (i.e., the spectrum) and the normalization of the source to the number of neutrons per megawatt account for the fact that changes in the isotopic fission fractions with fuel exposure (caused by Pu build-up) result in variations in the fission spectra, the number of neutrons produced per fission, and the energy released per fission. These effects increase the fast neutron source per megawatt of power for high-burnup assemblies. The variations in these physics parameters with fuel exposure may be obtained from standard lattice physics depletion calculations. This effect is particularly important for cycles that have adopted low-leakage fuel management schemes in which once-, twice-, or thrice-burned fuel is located in peripheral locations.

The horizontal core geometry is described using an (r,θ) representation of the nominal plane. A planar-octant representation is used for the octant-symmetric fuel-loading patterns typically used in CE NSSS plants. For evaluating dosimetry, the octant closest to the dosimeter capsules may be used. For determining the peak fluence, fuel-loading patterns that are not octant symmetric may be represented in octant geometry using the octant having the highest fluence. For evaluating dosimetry, the octant in which the dosimetry is located may be used. To accurately represent the important peripheral assembly geometry, a θ-mesh of at least 40 to 80 angular intervals is applied over the octant geometry. The (r,θ) representation should reproduce the true physical assembly area to within ~0.5% and the pin-wise source gradients to within ~10%. The assignment of the (x,y) pin-wise powers to the individual (r,θ) mesh intervals is made on a fractional area or equivalent basis.

The overall source normalization is performed with respect to the (r,θ) source so that differences between the core area in the (r,θ) representation and the true core area do not bias the fluence predictions.

1.3 Fluence Calculation

1.3.1 Transport Calculation

The transport of neutrons from the core to locations of interest in the pressure vessel is determined with the two-dimensional discrete ordinates transport program DORT in (r,θ) geometries.

An azimuthal (θ) mesh using at least 40 to 80 intervals over an octant in (r,θ) geometry in the horizontal plane provides an accurate representation of the spatial distribution of the material compositions and source described above. The radial mesh in the core region is about 1 interval per centimeter for peripheral assemblies, and coarser for assemblies more than two assembly pitches removed from the core-reflector interface. The Draft Regulatory Guide 1053 recommends that in excore regions, a spatial mesh that ensures the flux in any group changes by less than a factor of ~2 between adjacent intervals should be applied, and a radial mesh of at least ~3 intervals per inch in water and ~1.5 intervals per inch in steel should be used. Because of the relatively weak axial variation of the fluence, a coarse axial mesh of about 2 inches per mesh may be used in the axial (Z) geometry except near material and source interfaces, where flux gradients can be large. For the discrete ordinates transport code, an S_8 a fully symmetric angular quadrature is used as a minimum for determining the fluence at the vessel.

Past calculations were limited by computer storage and had to be performed in two or more "bootstrap" steps to avoid compromising the spatial mesh or quadrature (the number of groups used usually does not affect the storage limitations, only the execution time). In this approach, the problem volume was divided into overlapping regions. In a two-step bootstrap calculation, for example, a transport calculation was performed for the cylinder defined by $0 < r < R'$ with a fictitious vacuum-boundary condition applied at R'. From this initial calculation a boundary source is determined at the radius $R'' = R' - \Delta$ and was subsequently applied as the internal-boundary condition for a second transport calculation from R" to R (the true outer boundary of the problem). The adequacy of the overlap region had to be tested (e.g., by

decreasing the inner radius of the outer region) to ensure that the use of the fictitious boundary condition at R' had not unduly affected the boundary source at R" or the results at the vessel. Current workstations normally do not present this computer storage limitation, and the entire problem can now be solved as one fixed source problem.

A point-wise flux convergence criterion of < 0.001 should be used, and a sufficient number of iterations should be allowed within each group to ensure convergence. To avoid negative fluxes and improve convergence, a weighted difference model should be used. The adequacy of the spatial mesh and angular quadrature, as well as the convergence criterion, must be demonstrated by tightening the numerics until the resulting changes are negligible. In discrete ordinates codes, the spatial mesh and the angular quadrature should be refined simultaneously. In many cases, these evaluations can be adequately performed with a one-dimensional model.

Although the term "fluence calculation" is commonly used, one must recognize that the calculated quantity is a multi-group flux distribution, and that the fluence is obtained by integrating the flux over energy and over the duration of full power operation (in seconds).

The transport calculations may be performed in either the forward or adjoint modes. When several transport calculations are needed for a specific geometry, assembly importance factors may be pre-calculated by either performing calculations with a unit source (with the desired pin-wise source distribution) specified in the assembly of interest or by performing adjoint calculations. The adjoint fluxes are used to determine the fluence contribution at a specific (field) location from each source region, while the forward fluxes from the unit-source calculations determine the fluence at all locations in the problem. Once calculated, these factors contain the required information from the transport solution. By weighting the source distribution of interest by the assembly importance factors, the vessel (or capsule) relative fluence may be determined without additional transport calculations, assuming the in-vessel geometry, material, and in-assembly source distribution remain the same.

The use of forward solution is made on the basis of the number of configurations to be solved for the end of life fluence determination. The computational speed achieved with modern workstations may justify the exclusive use of forward solutions.

In performing calculations of surveillance capsule fluence (Regulatory Position 1.4), it should be noted that the capsule fluence is extremely sensitive to the representation of the capsule geometry and internal water region (if present), and the adequacy of the capsule representation and mesh must be demonstrated using sensitivity calculations (as described in Regulatory Position 1.4.1). The capsule fluence and spectra are sensitive to the radial location of the capsule and its proximity to material interfaces (e.g., at the vessel, thermal shield, and concrete shield in the cavity), and these should be represented accurately. The core shroud former plates can result in a 5-10% underprediction of the accelerated surveillance capsule dosimeter response and should be included in the model. (No significant effect is generally observed on the dosimeters located at the vessel inner-wall and in the cavity.)

1.3.2 Synthesis of the 3-D Fluence

Since 3-D calculations are not usually performed, the Regulatory Guide 1053 recommends that a 3-D fluence representation be constructed by synthesizing calculations of lower dimensions using the expression

$$\Phi(r, \theta, z) = \Phi(r, \theta) * L(r,z) \qquad \text{(Equation 1)}$$

where $\Phi(r, \theta)$ is the groupwise transport solution in (r,θ) geometry for a representative plane and $L(r,z)$ is a group-dependent axial shape factor. Two simple methods available for determining $L(r,z)$ are defined by the expressions

$$L(r, z) = P(z) \qquad \text{(Equation 2)}$$

where $P(z)$ is the peripheral-assembly axial power distribution, or

$$L(r, z) = \Phi(r, z) / \Phi(r) \qquad \text{(Equation 3)}$$

where $\Phi(r)$ and $\Phi(r, z)$ are one- and two-dimensional flux solutions, respectively, for a cylindrical representation of the geometry that preserves the important axial source and attenuation characteristics. The (r,z) plane should correspond to the azimuthal location of interest (e.g., peak vessel fluence or dosimetry locations). The source per unit height for both the (r, θ)- and (r)- models should be identical, and the true axial source density should be used in the (r,z) model.

Equation 2 is only applicable when (a) the axial source distribution for all important peripheral assemblies is approximately the same or is bounded by a conservative axial power shape and (b) the attenuation characteristics do not vary axially over the region of interest. Since the axial flux distribution tends to flatten as it propagates from the core to the pressure vessel, for typical axial power shapes, use of Equation 2 will tend to overpredict axial flux maxima and underpredict minima. This underprediction is nonconservative and can be large near the top and bottom reflectors, as well as when minima are strongly localized as occurs in some fluence-reduction schemes.

Equation 3 is applicable when the axial source distribution and attenuation characteristics vary radially but do not vary significantly in the azimuthal (θ) direction within a given annulus. For example, this approximation is not appropriate when strong axial fuel-enrichment variations are present only in selected peripheral assemblies.

In summary, an (r,θ)-geometry fluence calculation and a knowledge of the peripheral assembly axial power distribution are needed when using Equation 2. Use of this equation may result in fluence overpredictions near the midplane at relatively large distances from the core (e.g., in the cavity) and underpredictions at axial locations beyond the beltline that are at relatively large radial distances from the core. Conservatism may be included in the latter case by using the peak axial power for all elevations.

Both radial and axial fluence calculations are needed when using Equation 3; thus, it is generally more accurate in preserving the integral properties of the three-dimensional fluence. Both Equation 2 and Equation 3 assume separability between the axial and azimuthal fluence calculations, which is only approximately true.

1.3.3 Cavity Fluence Calculations

Accurate cavity fluence calculations are used to analyze dosimeters located in the reactor cavity. The calculation of the neutron transport in the cavity is made difficult by (a) the strong attenuation of the E > 1 MeV fluence through vessel and the resulting increased sensitivity to the iron inelastic-scattering cross-section and (b) the possibility of neutron streaming (i.e., strong directionally dependent) effects in the low-density materials (air and vessel insulation) in the cavity. Because of the increased sensitivity to the iron cross-sections, ENDF/B-VI cross-section data should be used for cavity fluence calculations. Properly benchmarked alternative cross-sections may also be used, however, for cavity applications, the benchmarking must include comparisons for operating reactor cavities or simulated cavity environments. Typically, the width of the cavity together with the close-to-beltline locations of the dosimetry capsules result in minimal cavity streaming effects, and an S8, angular quadrature is acceptable. However, when off-beltline locations are analyzed, the adequacy of the S8 quadrature to determine the streaming component must be demonstrated with higher-order Sn calculations.

The cavity fluence is sensitive to both the material and the local geometry (e.g., the presence of detector wells) of the concrete shield, and these should be represented as accurately as possible. Benchmark measurements involving simulated reactor cavities are recommended for methods evaluation. When both in vessel and cavity dosimetry measurements are available, an additional verification of the measurements and calculations may be made by comparing the vessel inner-wall fluence determined from (1) the absolute fluence calculation, (2) the extrapolation of the in-vessel measurements, and (3) the extrapolation of the cavity measurements.

1.4 Methodology Qualification and Uncertainty Estimates

Draft Regulatory Guide 1053 recommends that the neutron transport calculational methodology be qualified and that flux uncertainty estimates be determined. The neutron flux undergoes several decades of attenuation before reaching the vessel, and the calculation is sensitive to the material and geometrical representation of the core and vessel internals, the neutron source, and the numerical schemes used in its determination. The uncertainty estimates are used to determine the appropriate uncertainty allowance to be included in the application of the fluence estimate. While adherence to the guidelines described in the Draft Regulatory Guide will generally result in accurate fluence estimates, the overall methodology must be qualified in order to quantify uncertainties, identify any potential biases in the calculations, and provide confidence in the fluence calculations. In addition, while the methodology, including computer codes and data libraries used in the calculations, may have been found to be acceptable in previous applications, the qualification ensures that the licensee's implementation of the methodology is valid. The methods qualification consists of three parts: (1) the analytic uncertainty analysis, (2) the comparison with benchmarks and plant-specific data, and (3) the estimate of uncertainty in calculated fluence.

1.4.1 Analytic Uncertainty Analysis

The determination of the pressure vessel fluence is based on both calculations and measurements; the fluence prediction is made with a calculation, and the measurements are used to qualify the calculation. Because of the importance and the difficulty of these calculations, the method's qualification by comparison to measurements must be made to ensure a reliable and accurate vessel fluence determination. In this qualification, calculation-to-measurement comparisons are used to identify biases in the calculations and to provide reliable estimates of the fluence uncertainties. When the measurement data are of sufficient quality and quantity that they allow a reliable estimate of the calculational bias (i.e., they represent a statistically significant measurement data base), the comparisons to measurement may be used to (1) determine the effect of the various modeling approximations and any calculational bias and, if appropriate, (2) modify the calculations by applying a correction to account for bias or by

model adjustment or both. As an additional qualification, the sensitivity of the calculation to the important input and modeling parameters must be determined and combined with the uncertainties of the input and modeling parameters to provide an independent estimate of the overall calculational uncertainty.

An analytic uncertainty analysis must be performed to demonstrate the accuracy of the methodology. This analysis includes identification of the important sources of uncertainty. For typical fluence calculations, these sources include:

- Nuclear data (cross-sections and fission energy spectrum),
- Geometry (locations of components and deviations from the nominal dimensions),
- Isotopic composition of material (density and composition of coolant water, core barrel, thermal shield, pressure vessel with cladding, and concrete shield),
- Neutron sources (space and energy distribution, burnup dependence),
- Methods error (mesh density, angular expansion, convergence criteria, macroscopic group cross-sections, fluence perturbation by surveillance capsules, spatial synthesis, and cavity streaming).

Other uncertainties that are specific to a particular reactor or a particular calculational method should be considered. In typical applications, the fluence uncertainty is dominated by a few uncertainty components, such as the geometry, which are usually easily identified and substantially simplify the uncertainty analysis.

The sensitivity of the flux to the significant component uncertainties should be determined by a series of transport sensitivity calculations in which the calculational model input data and modeling assumptions are varied and the effect on the calculated flux is determined. (A typical sensitivity would be ~10-15% decrease in vessel >1 MeV fluence per centimeter increase in vessel inside radius.) Estimates of the expected uncertainties in these input parameters must be made and combined with the corresponding fluence sensitivities to determine the total calculated uncertainties.

1.4.2 Comparison with Benchmark and Plant-Specific Measurements

Calculational methods must be validated by comparison with measurements and calculational benchmarks. Three types of comparisons are required:

- operating reactor in-vessel or ex-vessel dosimetry measurements,
- pressure vessel simulator
- calculational benchmarks

The methods used to calculate the benchmarks must be consistent with those used to calculate fluence in the vessel. Calculated reaction rates at the dosimeter locations must agree with the measurements to within about 20% for in-vessel capsules and 30% for cavity dosimetry. If the observed deviations are larger, the methodology must be examined and refined to improve the agreement.

1.4.2.1 Operating Reactor Measurements

Comparisons of measurements and calculations should be performed for the specific reactor being analyzed or for reactors of similar physical and fuel management design. This plant-specific data can be compared to the benchmark analyses to validate that plant-specific calculations are within the tolerances expected by the benchmark uncertainty. A good estimate of the vessel attenuation can be obtained when both in-vessel and cavity dosimetry are available. These measurements should not be used to bias or adjust the fluence calculations unless a statistically significant number of measurements is available, the various dosimeter measurements are self-consistent, and a reliable estimate of the calculational bias can be determined. Similarly, plant-specific biases should not be used unless sufficient reliable measurement data are available. As capsule and cavity measurements become available, they should be incorporated into the operating reactor measurements data base and the calculational biases and uncertainties should be updated as necessary.

1.4.2.2 Pressure Vessel Simulator Measurements

A number of experimental benchmarks providing detector reaction rates in the peripheral fuel assemblies, within the vessel wall, and in the cavity are available for the purpose of methods calibration. These benchmark experiment were carried out by several laboratories, and dosimetry measurements using different techniques were compared to provide experimental results with well known and documented uncertainties. Examples include the Pool Critical Assembly (PCA), VENUS, and H.B. Robinson Unit 2 benchmarks.

1.4.2.3 Calculational Benchmarks

A calculational benchmark commissioned by the NRC and prepared by Brookhaven National Laboratory (Reference 8) provided a very detailed input description as well as the flux solution at several mesh points. An analysis of this benchmark, which addresses both standard out-in and low-leakage fuel management, provides a detailed test of the cross sections and various calculational options for transport calculation. The benchmark calculation results may be used for methods qualification. The calculation being used as the benchmark must be the actual original referenced benchmark calculation, and not just a second independent calculation of the benchmark.

1.4.3 Overall Bias and Uncertainty

An appropriate combination of the analytical uncertainty analysis and the results of the uncertainty analysis based on the comparisons to the benchmark results provide the bias and uncertainty to be applied to the predicted fluence.

2.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

This section addresses Provision 2 of Attachment 1 to GL 96-03 (Reference 3) on compliance with 10CFR50, Appendix H (Reference 17). Appendix H presents the requirements for RV material surveillance programs. The design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of ASTM E185 (Reference 18) that is current on the issue date of the ASME Code to which the CE NSSS RV was purchased (Circa 1966-73). For each capsule withdrawal, the test procedures and reporting must meet the requirements of ASTM E185-82 to the extent practicable for the configuration of specimens in the capsule.

ASTM E 185, "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels," provides for the monitoring and periodic evaluation of neutron radiation-induced changes in the mechanical properties of the vessel beltline materials. The ASTM standard provides procedures for the selection of materials, the design and quantity of test specimens, the design and placement in the RV of the test specimen compartments, and the means for measuring neutron fluence and irradiation temperature. These are aspects pertaining to the design of the program. ASTM E 185 also provides the guidelines for a schedule for the withdrawal of capsules for testing and a procedure for the pre- and post-irradiation testing of the surveillance program materials, neutron fluence monitors and temperature monitors.

The RV material surveillance program for the CE NSSS design was to meet or exceed the requirements of the version of ASTM E 185 in effect at the time that the vessel was purchased. For each vessel, base metal was selected from one of the beltline plates and used to fabricate test specimens for pre-irradiation testing and for inclusion in the surveillance capsule compartments. Similarly, a weldment was fabricated using portions of the beltline plates and the same welding process as used for one or more of the beltline welds; both weld metal and heat-affected-zone (HAZ) specimens were fabricated from the weldment for pre-irradiation testing and for inclusion in the surveillance capsule compartments. A section from the surveillance plate and weld was retained as archive material. Neutron flux and temperature monitors, and test specimens from the surveillance plate, weld and HAZ together with

specimens from a correlation monitor material were loaded into compartments and assembled into surveillance capsules. A minimum of six surveillance capsules were originally provided for each plant. Records were compiled that documented the source of the materials, including fabrication history, the location and orientation of test specimens in the original material, the design of the specimen compartments, and the location of individual specimens in the compartments for each capsule assembly.

The six surveillance wall capsules were installed in holders on the inside surface of the RV and within the region surrounded by the effective height of the active reactor core. The vessel wall location provides for irradiation of the surveillance materials under conditions closely approximating the neutron fluence rate, temperature, and variations thereof, over time of the RV that is being monitored[1].

The surveillance capsule withdrawal schedule was originally established following the requirements of the version of E 185 in effect at the time of vessel design/fabrication; the schedule may have been originally established based on the requirements of 10CFR50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." The schedule called for at least three capsules to be removed and tested during the design life of the RV. The remaining capsules are available to provide a higher frequency of testing if required or retained to provide supplemental information in the future. The surveillance capsule withdrawal schedule may be modified. If the surveillance capsule withdrawal schedule is located within the TSs, such proposed modifications will be submitted to the NRC with a technical justification for approval and require a license amendment (pursuant to 10 CFR 50.90). For those plants not having the surveillance capsule removal schedule located in the TSs, any proposed changes to the program or withdrawal schedule will be submitted to the NRC for review and approval. Proposed changes to the surveillance capsule withdrawal schedule that are not consistent with

[1] See plant-specific details for azimuthal location of the wall capsules and, if applicable, for additional surveillance or dosimetry capsule locations. In some cases, additional capsules were installed in holders attached to the core barrel for accelerated irradiation or in the upper plenum region away from the beltline where the fast neutron fluence is negligible. In other cases, replacement surveillance capsules have been installed in empty capsule holders to obtain additional vessel material or neutron fluence data. Examples of the latter are dosimetry capsules installed inside the vessel or in the annulus between the vessel and the biological shield.

the withdrawal criteria of the version of ASTM E-185 of record or with one of the later versions of E-185 amount to alternatives to the requirements of 10 CFR Part 50, Appendix H, and pursuant to 10 CFR 50.60(b), must be accompanied with an exemption request for their use. Such requests will be evaluated on their technical merits against the exemption acceptance criteria of 10 CFR 50.12.

Post-irradiation testing is presently performed on the specimens from the withdrawn capsule in accordance with the requirements of ASTM E 185-82 (or later versions, as specified in Appendix H) and 10CFR50, Appendix H. The test data and evaluation results are compiled and presented in a report to the NRC within one year of the date of capsule withdrawal, unless an extension is granted by the NRC. Application of the data for the PTLR are discussed in Sections 4.0 and 7.0.

The initial properties of the RV beltline plates and welds were established in parallel to the establishment of the RV surveillance program. For each of the beltline plates, Charpy impact tests and/or drop weight tests were performed to demonstrate compliance with applicable ASME Code and vessel specification requirements. The welding procedures used for beltline welds were qualified and the welding materials certified to applicable AWS, ASME Code and vessel specification requirements. Chemical analyses of the plates and weld deposits were obtained in accordance with the vessel specification. The data were processed to obtain values of the initial reference temperature and the copper and nickel content. [Note: The data that are available for a specific vessel will vary because of differences in the requirements for testing and certification.] For beltline plates and welds, the initial RT_{NDT} was determined in accordance with the ASME Code, Section III, NB-2331, for which drop weight tests and Charpy impact tests (complete transition curve) were performed. For the earlier CE NSSS RV designs for which test requirements were different, the initial RT_{NDT} was determined using BTP MTEB 5-2, "Fracture Toughness Requirements (for Older Plants)," or a generic value of initial RT_{NDT} was determined based on measurements for a specific set of materials. References (4), (5) and (6) are some pertinent CEOG sponsored efforts.

3.0 LOW TEMPERATURE OVERPRESSURE PROTECTION REQUIREMENTS

3.1 Introduction

3.1.1 Scope

This section addresses Provision 3 of Attachment 1 to GL 96-03 (Reference 3) that allows LTOP system limits, developed using NRC-approved methodologies and contained in TS, to be relocated to a plant-specific PTLR. The methods described are those utilized by CE in the analyses supporting LTOP to ensure adequate protection of the RCPB and, especially of the RV, against brittle fracture during heatup, cooldown, and shutdown operations. These methods must be followed by the participating CEOG utilities for CE NSSS designs in the calculation of the plant-specific LTOP limits in their original PTLRs and revisions thereto.

No other methodologies beyond those currently used for CE NSSS designs are included herein, since a license amendment is required anytime a licensee changes methodology. A license amendment will also be required if the licensee chooses to change from one method to another method, even though both methods are described in this topical report.

The relationship between LTOP setpoints and limitations, and RCS P-T limits is also discussed.

The two kinds of P-T limits that are used as a basis for the LTOP setpoints and limitations for CE NSSS designs are considered herein. These are Appendix G P-T limits and LTOP P-T limits as defined in Section 3.4.2. Both are based upon the NRC-approved methodology of Appendix G to Section XI of the 1995 Edition and addenda through the 1996 Addenda of the ASME Code (Reference 10) as currently specified in 10 CFR 50, Appendix G (Reference 1).

Appendix G P-T limits developed using ASME Code Case N-640 (Reference 11) can also serve as a basis for the LTOP setpoints and limitations. An exemption must be obtained to use the Code Case via 10 CFR 50.60 (b) pertaining to proposed alternatives to the discussed

requirements in Appendices G and H on fracture toughness. Section 3.4.2 provides specifics related to the use of the Code Case.

Additionally, two methods of calculating the LTOP enable temperatures are addressed; one, per BTP RSB 5-2 (Reference 12), the other as prescribed by Appendix G to Section XI of the 1995 Edition and Addenda through the 1996 Addenda of the ASME Code (Reference 10).

3.1.2 Background

Current requirements defined in Section III, Article NB-7000 of the ASME Boiler and Pressure Vessel Code provide for overpressure protection of the RCPB during power operation. Additional requirements are also given by 10 CFR 50, Appendix A, General Design Criteria 15 and 31. These criteria require that the RCS be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during normal operation including anticipated operational occurrences, and the RCPB be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, it behaves in a nonbrittle manner and the probability of rapid propagating fracture is very low.

The NRC has also provided guidance to ensure overpressure protection for anticipated operational occurrences at conditions other than power operation. This guidance, originally published in NUREG-75/087 (currently NUREG-0800), is provided in Standard Review Plan 5.2.2, "Overpressure Protection" (Reference 2), which includes BTP RSB 5-2 (Reference 12).

The primary concern of BTP RSB 5-2 pertains to operation at low temperatures, especially in a water-solid condition. The applicable operating limits in the low temperature region are based on an Appendix G evaluation which provides much lower allowable pressures than the design limit considered at normal operation (power operation) pressure and temperature. The consequences resulting from an overpressurization event at low temperatures are clearly threatening to the integrity of the RCPB. Therefore, the objective of BTP RSB 5-2 is to protect the Appendix G limits in order to meet the criteria established in 10 CFR 50, Appendix A.

The LTOP system is required to protect the P-T limits that constitute a basis for the LTOP setpoints and other limitations at the plant. In the plants using P-T limits generated via Appendix G to ASME Code Section XI (Reference 10) as a basis for LTOP, the plant's LTOP system is required to protect these Appendix G P-T limits. Conversely, if the plant has chosen to use LTOP P-T limits, as a basis for LTOP, the plant's LTOP system is required to protect these LTOP P-T limits.

LTOP is a combination of measures that ensure that the applicable P-T limits will not be exceeded during heatup, cooldown, and shutdown operations. The LTOP range is the operating condition when any RCS cold leg temperature is less than the applicable LTOP enable temperature. LTOP is accomplished through either alignment and operability of the LTOP relief valves for automatic protection or establishment and maintenance of an adequate vent size as defined in the TS to provide an alternate means of LTOP. The definitions for this alternate configuration are plant-specific and provided in the TS.

As a minimum, an LTOP system may include relief valves with a single setpoint that must be aligned below the enable temperature, and restrictions on RCS heatup and cooldown rates. Such a system would result when the P-T limits are not overly restrictive, the LTOP relief valves are of high capacity, and the relief valve setpoint allows for an acceptable operating window. Conversely, if the P-T limits are restrictive, the LTOP relief valves are small, and/or the operating window is challenged, the LTOP system may include a combination of valves, power-operated relief valves (PORV) with multiple setpoints, or with a variable setpoint controlled as a function of reactor coolant temperature. Other restrictions may be added to make the LTOP system adequate.

Historically, the LTOP-related limitations have been required to be included in the TS, along with the applicable Appendix G P-T limit figures. P-T limits, except those for the RV, do not typically change, as these apply to the RCPB components that are not subject to irradiation. P-T limits based upon the RV beltline do change with time due to irradiation. As a result, every time P-T limits changed due to irradiation, the TS had to be revised to incorporate these new P-

T limits, unless the changes were offset by simultaneous reductions in the pressure correction factors and/or instrumentation uncertainties. If the P-T limits became more limiting, the LTOP limits had to be re-evaluated with these new limits. The LTOP limits had to also be re-evaluated following plant modifications and changes in configuration, which may adversely impact the LTOP analyses. If LTOP limits become more restrictive, the TS were required to be revised.

GL 96-03 gives utilities the opportunity to avoid TS revisions due to changes in P-T limits by relocating the appropriate limits to plant-specific PTLRs. GL 96-03 also establishes the conditions under which LTOP system limits can be relocated from the TS to a plant-specific PTLR. Attachment 1 to GL 96-03 specifies the requirements for the methodology that must be provided in a methodology report, which is a prerequisite for a PTLR. According to Provision 3 of Attachment 1, the LTOP methodology must include a description of how the LTOP system limits are calculated applying system/thermal hydraulics and fracture mechanics and must reference SRP 5.2.2, ASME Code Case N-640, and ASME Code Appendix G, Section XI, as applied in accordance with 10 CFR 50.55.

GL 96-03 specifies that only the figures, values and parameters associated with the P-T limits and the LTOP setpoints can be controlled in the PTLR, rather than in the TS. Other LTOP-driven limitations, such as the limits on RCP starts, limitations on high pressure safety injection (HPSI) and/or charging pump operation, pressurizer level, etc., must remain and be controlled in the TS. These TS limits must be used in the future as analysis inputs to identify changes to the parameters that will be controlled in the PTLR. If a change to the LTOP-related TS is required to recapture operating margin that may disappear due to changes to the P-T limits, this change must first be implemented via a license amendment and only then can this change be credited in analyses.

The following sections describe the LTOP methodology for CE NSSS designs that has been used to develop and analyze LTOP systems and that must be adhered to in the plant-specific PTLRs and revisions thereto. Based on GL 96-03, following NRC approval of a plant-specific PTLR that has this topical report as its basis, future changes to LTOP-specific operating

restrictions, modifications to the approved LTOP analysis methods, and/or LTOP system redesigns will require NRC approval prior to implementation.

3.2 General Methodology

3.2.1 Description

The LTOP methodology for CE NSSS designs makes use of an iterative process in the determination of LTOP limitations, which balances the adequacy of the LTOP system with acceptable operating restrictions. The methodology is based upon a supposition that an adequate LTOP system can be designed in more than one way by varying assumptions and setpoints/parameters such that the resulting operational restrictions are optimized. As an example, keeping the existing relief valve setpoint but further restricting the RCS heatup and cooldown rates may be more beneficial than keeping the rates but reducing the setpoint, which, in turn, reduces the operating window. Restricting a heatup and/or cooldown rate results in an increase in the P-T limit allowable pressure at a given temperature, which allows an increase in the relief valve set pressure, which, in turn, expands the operating window between the relief valve set pressure and the RCP minimum operating pressure limit. Each utility decides on the optimal approach itself.

Since it protects the RCPB integrity, LTOP is a safety related function. Consequently, any analysis supporting of LTOP must be quality assured in accordance with a QA Program that complies with the 10 CFR 50 Appendix B requirements. NRC guidance on performing LTOP related analyses is documented in BTP RSB 5-2 (Reference 12). The LTOP methodology of this report for CE-NSSS designs is consistent with BTP RSB 5-12.

3.2.2 LTOP Evaluation Components

Analyses that support the determination of LTOP requirements generally fall into three major analytical areas:

1) Analysis of P-T limits for use as operating guidelines and as a basis for LTOP requirements. The methodology for P-T limits is detailed in Section 5.0.

2) Analyses of postulated overpressure events in the RCS, including energy addition (RCP start) and mass addition events. These analyses yield peak transient pressures which are compared with the P-T limits to identify LTOP-related limitations. The sources for the transients most often remain unchanged. However, changes in operational practices and plant configuration may cause changes in the applicable transients and/or temperatures.

3) LTOP evaluation, which compares the applicable P-T limits and peak transient pressures to identify the LTOP-related limitations.

The following sections describe the methods to be used in the various analyses that comprise the LTOP evaluations.

3.3 Transient Analysis Methodology

3.3.1 Limiting Event Determination

The determination of LTOP-driven restrictions is based upon the consideration of multiple requirements. Currently, 10 CFR 50 Appendix A requires that the initiating event be established considering any condition that may occur during normal operation, including anticipated operational occurrences (AOOs). AOOs are defined therein as those conditions of normal operation which are expected to occur one or more times in the life of the nuclear power unit.

According to BTP RSB 5-2, "All potential overpressurization events should be considered when establishing the worst-case event". Potential causes (sources) of RCS overpressurization at low temperatures in CE NSSS designs have been considered at the time LTOP systems were

being designed. Out of those causes, two types of events were determined to most challenge LTOP systems. They are:

(1) Energy addition to the RCS during an RCP start with the secondary SG inventory at a higher temperature than reactor coolant, and

(2) Mass addition to the RCS during operation of HPSI pumps and/or charging pumps that results from an inadvertent Safety Injection Actuation Signal (SIAS).

Presently, the RCP start continues to be the most limiting energy addition event. With respect to mass addition, the most limiting event is mass addition from a HPSI/charging pump combination with the highest flow rate, as allowed by the TS. The applicability of the most limiting mass addition event may extend over the entire LTOP range, or may be restricted to a certain temperature range, in accordance with the TS. If the applicability of the most limiting event is restricted, then other mass addition events, with a smaller number of operating pumps and/or flow rate restrictions (as allowed by the TS) become the limiting events at other temperatures.

An additional qualifier for the limiting events is pressurizer water level. This is one of the design bases for LTOP limitations. Each energy addition and mass addition event's definition must be supplemented by this parameter as "under water-solid conditions" or "with a pressurizer steam volume of...% (or *cu ft*)." The LTOP setpoints and limitations can be based on the transient analyses that assume a steam volume in the pressurizer only if a limit on pressurizer steam volume is in the TS. To take credit for a restriction for transient mitigation in the pressure transient analyses, this restriction must be in the TS. If there is no TS controlling the restriction (e.g., limitations on HPSI and charging pump operation or pressurizer level), then the restriction cannot be credited in the analysis or put in the PTLR. The analysis must also account for pressurizer level (volume) instrumentation uncertainty. The uncertainty must be determined using the guidance contained in Regulatory Guide 1.105 (Reference 20) and ISA Standard S67.04-1994 (Reference 13).

3.3.2 Approach and Major Assumptions

The limiting events must be analyzed for each pump combination (mass addition), with each applicable means for transient mitigation, and for the most limiting fluid conditions in the RCS and pressurizer. If an LTOP system is comprised of two or more PORV setpoints or a variable PORV setpoint, and water-solid conditions in the pressurizer may exist during PORV discharge for transient mitigation, the transient analyses must assume water-solid conditions and must be performed for each fixed setpoint or for a number of setpoints for a variable setpoint arrangement. In the latter case, the setpoints for the analyses can be selected based upon the existing, or a preliminary PORV setpoint vs. temperature curve, from the lowest setpoint at the minimum boltup temperature to the highest setpoint at the LTOP enable temperature. The setpoints may be selected at 50 – 200 psi increments, with the smaller values at the lower temperatures, in order to provide more peak pressure data points where the operating window is most restricted.

The purpose of analyzing a number of setpoints in the transient analysis is to develop a peak pressure vs. setpoint curve, which will be used in the follow-up LTOP evaluation to select the most optimal setpoints for a fixed or variable setpoint arrangement. The curve should envelope the range from the setpoint expected at the minimum boltup temperature to the setpoint expected at the LTOP enable temperature. The curve is not a mandatory requirement; it is a tool used to minimize the number of analyzed setpoints and, at the same time, provide a sufficient database for the final setpoint selection.

This topical report cannot mandate the number of the setpoints to be analyzed for the curve, nor can it require using certain increments, because these are plant-specific values, which depend on a number of plant-specific conditions. The selection of the setpoints for the analyses will be addressed in the plant-specific PTLRs.

Both the mass and energy addition transient analyses results are considered in combination in the LTOP evaluation to select the most limiting peak pressure at a given setpoint.

The transient analyses must assume the most limiting allowable operating conditions and systems configuration at the time of the postulated cause of the overpressure event, as stated in BTP RSB 5-2. Consequently, unacceptable peak pressures may result if only bounding analyses are performed, as these analyses typically assume the most limiting fluid conditions and plant configurations over the entire LTOP range. As an alternative, a transient analysis can be performed in a parametric manner for two or more initial reactor coolant temperatures, pressurizer water levels, RCS pressures, decay heat rates, etc. Such an approach yields lower peak pressures at the less limiting fluid conditions (where these apply), while producing the bounding peak pressure values at the most limiting conditions that would only be applicable in a narrower temperature range. This approach also benefits the LTOP evaluation, since a peak pressure database will be generated that can facilitate meeting the ultimate goal - protection of the P-T limits with minimum operational limitations, a sufficient operating window, and best possible heatup and cooldown rates.

Both energy and mass transient analyses will use the following major assumptions:

- *When relief valves mitigate the transient, only one valve must be used in the transient analysis.*

  This assumption meets the single failure criterion of BTP RSB 5-2. Past studies demonstrated that unavailability of one relief valve is the most limiting single failure with respect to the peak transient pressures. Relief valve discharge characteristics must be selected as indicated in Section 3.3.3.

- *Credit must not be taken for letdown, RCPB expansion, and heat absorption by the RCPB for transient mitigation.*

  This assumption places the entire burden for transient mitigation on the LTOP system.

- *A water-solid pressurizer must be assumed for bounding analyses.*

  This assumption expedites the transient response and maximizes pressurization rate. Additionally, assuming that the pressurizer inventory is at saturation conservatively reduces discharge flow rates through the PORVs and relief valves on the pressurizer, due to lower water subcooling at the valve inlet during discharge.

  Subcooled water conditions in the pressurizer may be assumed at the low reactor coolant temperatures when the pressurizer is filled but is not at saturation. This assumption will have to be properly justified in the plant-specific PTLRs. It will yield greater discharge flow rates as compared with those under saturated water conditions.

- *A pressurizer steam volume can be credited in transient mitigation.*

  A steam volume in the pressurizer can be assumed, if the TS contain a limitation on a maximum pressurizer water level (or a minimum steam volume) for the LTOP temperature region, or a portion thereof. As the energy addition and mass addition transient analysis methods differ, discussions on the application of the steam volume are provided in the appropriate sections of this report (see Sections 3.3.4 and 3.3.5).

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

  This input increases transient pressure.

- *Decay heat will be assumed as an additional input to maximize reactor coolant expansion.*

  This assumption increases the peak pressure and is the result of an assumed loss of SDC heat removal capability. The most conservative method for calculating decay heat rate must account for cooldown at the maximum rates allowed by the TS to reach the LTOP enable temperature or another temperature point in the LTOP region after reactor

shutdown. These decay heat rates must then apply to the transients occurring during both heatup and cooldown.

An acceptable alternate method is to determine decay heat rates separately for heatup and cooldown, recognizing the fact that the times after reactor shutdown to reach the same temperature during cooldown and heatup differ. Decay heat input may not have to be included at all during cooldown or isothermal conditions, if decay heat removal either by the secondary system or by SDC can be relied upon. When other than the most conservative methods are used to reduce or eliminate the decay heat contribution, a justification must be provided in the plant-specific PTLRs for NRC review and approval. That justification shall account for decay heat rate dependent issues such as outage length.

- *Operator action time is 10 minutes.*

  Generally, operator action for transient mitigation or termination can be credited 10 minutes after initiation of the event. If the licensee can demonstrate that it would take less than 10 minutes for the operator to recognize and mitigate (terminate) the transient, less time can be used. The NRC must approve the justification.

- *PORV setpoint for the analyses must be greater than the nominal setpoint to account for the actuation loop uncertainty and pressure accumulation due to finite PORV opening time.*

  This assumption recognizes that due to loop instrumentation uncertainties, the PORV may start its opening at a higher pressurizer pressure than the nominal setpoint (if the loop "reads" low). Additionally, it accounts for pressure accumulation above the opening pressure during the time delay between the signal initiation and the valve plug reaching the full flow position. See Section 3.3.3 for further discussion.

3.3.3 LTOP Relief Valves

3.3.3.1 General Description

Current CE NSSS designs incorporate LTOP relief capability during low temperature operation of the RCS. This is done in one of several ways. LTOP is provided by either two PORVs on top of the pressurizer, two dedicated relief valves on top of the pressurizer, relief valves in the SDC suction line, or a combination of the PORVs and SDC relief valves.

The PORVs and the relief valves on the pressurizer are the only LTOP relief valves with a setpoint that can be adjusted with relative ease. A change in the PORV LTOP setpoint or relief valve setpoint can be factored into the LTOP transient analyses if needed, as these setpoints are for LTOP only. The SDC relief valves, on the other hand, are spring loaded relief valves with a fixed setpoint, whose main function is to protect the SDC system. A setpoint change is not typically an option in the LTOP transient analyses involving these valves. The specifics of each type with respect to transient analyses are discussed below.

3.3.3.2 Power-Operated Relief Valves

The PORVs at CE NSSS designs are fast acting pilot operated valves, with stroke times of the order of milliseconds[3].

The PORVs may pass subcooled water, saturated water, and/or steam, depending on the pressurizer conditions during transient mitigation. PORV discharge characteristics for these fluids must be developed using appropriate correlations and a conservative back pressure, as applicable. Especially important is accounting for discharge flow reduction due to flashing at the valve outlet when the discharged water has a low degree of subcooling. The characteristics, in the form of curves, must relate valve discharge flow rate with either PORV inlet pressure or pressurizer pressure, cover the anticipated pressure range, and not be related

[3] A slower opening time is assumed in the analyses for consistency with the acceptance criteria during PORV testing.

to a setpoint. PORV inlet piping pressure drop must be taken into account in the curves in terms of pressurizer pressure. The curves must be used in the transient analyses. The correlations used for developing of the PORV discharge characteristics (curves) are plant-specific and must be provided in the plant-specific PTLRs for NRC review and approval.

PORV actuation loop instrumentation uncertainty and PORV opening time must be accounted for in the determination of a conservative value for the PORV opening pressure at the rated flow position. The addition of the uncertainty to the nominal setpoint determines pressure at the beginning of opening, whereas addition of pressure accumulation during the opening time determines the highest pressure at opening.

The actuation loop instrumentation uncertainty must be determined using guidance contained in Regulatory Guide 1.105 (Reference 20) and ISA Standard S67.04-1994 (Reference 13). For development of a PORV setpoint curve for a continuously variable setpoint program, a conservative adjustment for uncertainty must be applied to the entire curve. Alternatively, the curve can be divided into segments and an uncertainty for each segment must then be determined, based on the slope of each segment.

The PORV opening time must be consistent with the acceptance criteria during in-service testing of the subject PORV. The transient analyses must assume a conservative PORV opening characteristic, which can be simplified by the assumption that during the opening time period, the PORV remains closed and then opens instantaneously. Pressure accumulation during this time must be added to the opening pressure (which is the nominal setpoint corrected for uncertainty) to obtain the maximum pressure at the opening which must be used in the transient analyses. The pressure accumulation is a product of the transient pressurization rate and the valve opening time. Pressurization rate, in psi/sec, is determined for each applicable transient via an analysis that produces a pressure vs. time function for discharge from a water-solid pressurizer. The function must extend to include all anticipated PORV opening pressures, such that pressurization rate can be determined at the moment just prior to the opening pressure.

Should a setpoint change be contemplated, one or more new setpoints can be assumed and analyzed to provide a peak pressure vs. setpoint function for the LTOP evaluation. This function could be developed using results from the analyses of both mass addition and energy addition transients performed for a number of setpoints. The greatest peak pressure for a given setpoint will be used as a data point for the function. The curve would allow the determination of an optimal PORV setpoint that yields the peak pressure below the applicable P-T limit.

3.3.3.3 SDC Relief Valves

The SDC relief valves pass subcooled water, due to their location in the SDC system piping inside containment. The opening and discharge characteristic for these valves must be consistent with the ASME Code requirements for spring loaded safety relief valves and/or manufacturer's recommendations, whichever is more conservative. The ASME Code requires that these valves start opening at 3% accumulation above the set pressure and reach rated flow position at 10% accumulation. The pressure drop in the piping from the hot leg to the valve inlet and the elevation head from the valve to the pressurizer must be considered in the adjustment of the peak pressure at the valve inlet to the pressurizer. A setpoint change is not typically contemplated in LTOP transient analyses involving these valves, because of their function to also support SDC system operation.

3.3.3.4 Pressurizer Relief Valves

A pair of dedicated safety relief valves connected to the top of the pressurizer are the sole means for LTOP in one of the CEOG member plants. These valves may pass subcooled water, saturated water, and/or steam, depending on the pressurizer conditions during transient mitigation. The opening and discharge characteristic for these valves must be consistent with the ASME Code requirements for spring loaded safety relief valves and/or manufacturer's recommendations, whichever is more conservative. The ASME Code requires that these valves start opening at 3% accumulation above the set pressure and reach rated flow position at 10% accumulation. The pressure drop in the inlet piping must be considered for its effect on

the peak transient pressure. Similar to the SDC relief valves, these valves are spring loaded safety relief valves with a fixed setpoint. However, unlike the SDC relief valves, the pressurizer relief valves have a dedicated use, and a setpoint change can be considered in the LTOP analyses involving these valves.

3.3.4 Energy Addition Event

An energy addition event can take place when the RCS is cooled via SDC, while the SGs remain at a higher temperature. A temperature difference between the secondary side of the SG and reactor coolant will transfer heat in the SG tubes to the reactor coolant, thus raising coolant temperature and pressure. With a water-solid pressurizer, pressure quickly reaches the relief valve opening pressure, the valve then opens and starts to discharge.

The temperature difference (Δt) represents the maximum initial limit allowed by the TS for the first idle RCP start, which consists of a highest secondary temperature and lowest reactor coolant cold leg temperature. The uncertainties in temperature measurements need to be considered in establishing a conservative analytical value for Δt.

If the relief valve is a PORV and its capacity at the opening exceeds the flow rate equivalent to the resulting coolant expansion, the transient will be mitigated at the opening pressure and the valve may reclose at the reseat pressure only to open again as pressure rises to repeat the cycle. This valve cycling continues until the cause of the transient is eliminated. The peak pressure in this case will be the maximum opening pressure.

If PORV capacity at the opening is less than the transient input, pressure rises until equilibrium is reached, at which point discharge matches input. That equilibrium pressure will be the peak pressure in the transient.

In the case of a SDC relief valve or a pressurizer relief valve, the peak pressure at the inlet, which will also be the equilibrium pressure, will either be maintained below 10% accumulation, if

valve capacity exceeds the input, or above 10% accumulation if a higher inlet pressure is needed to mitigate the transient.

In the case with a steam volume in the pressurizer, the maximum pressure can be reached either prior to the valve opening, or after the opening during steam discharge, or after the opening but during water discharge.

The analysis of this event under water-solid conditions typically uses an analytical model that uses equations for calculating heat transfer in the heated portions of the SG tubes from the secondary SG inventory to the reactor coolant. For a reverse temperature gradient to occur, the reactor coolant has been circulated through, and cooled down by, the SDC system, whereas the SGs remain at the SDC initiation temperature. A description of the analytical model will be included in the plant-specific PTLRs. A number of conservative assumptions are used in the analyses of this event to maximize peak pressures, in addition to those described in Section 3.3.2. These include: 1) additional heat input from the RCP, 2) fluid properties and heat transfer coefficients determined at the highest reactor coolant temperature, and 3) instantaneous RCP start.

The analysis of the energy addition transient for water-solid conditions must consider the entire LTOP temperature range, even though water-solid operations may procedurally be limited. The highest temperature in the range must be assumed to obtain a conservative peak pressure. The analysis could consider several narrower temperature spans, even if the relief valve setpoint remains unchanged, to obtain a less limiting peak pressure at these temperatures. If the LTOP system includes two or more relief valve setpoints, the analysis must be performed either for each setpoint, or for a number of setpoints sufficient to generate a peak pressure vs. setpoint function.

A steam volume in the pressurizer can be credited in an analysis that determines the conditions under which the LTOP relief valve would not be challenged in the energy addition event. The analysis must be based on the existing CE method for CE NSSS designs that assumes that reactor coolant expansion during the transient is absorbed by the steam, which is compressed

in a reversible adiabatic process. A maximum potential secondary-to-primary temperature difference must be assumed. The method assumes that at the end, the entire system reaches an equilibrium temperature, which depends on the initial conditions. The peak pressure must be determined at that temperature. No time factor is involved. When crediting a steam volume, the analysis must determine at least one combination of the initial pressurizer pressure and pressurizer level for each relief valve setpoint that would yield a peak pressure below the relief valve setpoint. As the objective of such an analysis is to prevent reaching the relief valve setpoint during RCP starts, this event (RCP start with steam bubble and no relief valve actuation) cannot be considered as a design basis overpressure event. Transient mitigation by the pressurizer steam volume must not be the only means for LTOP in any temperature range below the LTOP enable temperature. Two LTOP relief valves must always remain operable and capable of mitigating the overpressure transient within the LTOP region even when credit is taken for a steam bubble.

A steam volume in the pressurizer can also be assumed in non-bounding analyses of the energy addition transient, consistent with the assumption in Section 3.3.2. With a steam volume in the pressurizer, pressurization rate prior to valve opening can be significantly reduced as compared with that in the water-solid pressurizer. This results in a reduction in the valve opening pressure, due to a smaller pressure accumulation during opening time. Such an analysis determines the peak transient pressure with relief valve mitigation and can serve as a design basis for the LTOP system, whenever a pressurizer level limitation (or steam volume) is included in the TS.

3.3.5 Mass Addition Event

A mass addition event can take place whenever a HPSI and/or charging pump is aligned to the RCS. An inadvertent SIAS is assumed to initiate mass injection to the RCS from all the aligned pumps. The relief valve behavior in a mass addition event is similar to that described for an energy addition event (Section 3.3.4). As a different number of HPSI pumps and/or charging pumps may be operable in a particular temperature region, each pump combination represents an analytical case and should be analyzed, rather than postulating the worst possible

combination over the entire LTOP temperature range. Mass addition is assumed to take place at the cold leg centerline and adjustments can be made to the pressurizer. The HPSI pump inputs must be maximized by addition of a conservative margin of 3-10% of the nominal values. The charging pump input must be the maximum flow measured at the plant. For both pumps maximized performance is typically based on inservice test acceptance criteria.

The combined delivery of all operating pumps for a case is developed in the form of a delivery curve representing flow to the cold legs as a function of pressurizer pressure.
For analysis of this event on CE NSSS designs, CE uses a method of equilibrium pressures. The method consists of a superposition of the relief valve discharge curve on the mass addition curve, both in terms of flow rate as a function of pressurizer pressure. The mass addition curve includes not only pump flow rates, but also energy inputs from decay heat, pressurizer heaters, and RCP (if operating) converted into equivalent flow rate. These additional flow rates are determined by calculating reactor coolant temperature rise over an assumed period of time (10 minutes or as justified) resulting from these energy additions, which, in turn, determines reactor coolant expansion. The expansion is converted into the equivalent flow rate. The pump delivery curve is shifted to the right by this additional flow rate value, which effectively increases the equilibrium pressure. The equilibrium pressure is determined at the intersection of the two curves. It signifies the pressure at which the mass input matches the relief valve discharge flow rate. The equilibrium pressure is determined for liquid input and discharge.

The equilibrium pressure is then compared with the maximum pressure at the valve opening (see Section 3.3.3) to identify the peak transient pressure.

The equilibrium pressure is the greatest peak pressure that could be reached during this transient if it is higher than the maximum pressure at the opening. No time factor nor operator action are involved. As a result, this equilibrium pressure applies to both water-solid and steam volume initial conditions in the pressurizer.

A pressurizer steam volume is only credited in establishing pressurization rate prior to relief valve opening, which is then used in the calculation of the pressure accumulation. The latter is

added to the nominal setpoint to determine the maximum opening pressure (see Section 3.3.3). Depending on the assumed PORV opening time, a significant reduction in the maximum opening pressure on liquid can be realized, as pressurization rate on steam is much lower than on water. Note that because crediting the steam bubble is only used to improve the peak pressure at opening (and does not change the equilibrium pressure), steam bubble credit benefits those events where the peak pressure at opening (on water) exceeds the equilibrium pressure.

Pressurization rate, in psi/sec, is determined for each HPSI/charging pump combination considered, based on an analysis that produces a pressure vs. time function for a defined initial pressurizer steam volume. Mass input from the pumps into the RCS determines the decrease in the pressurizer steam volume each time increment. Then steam volume compression is calculated assuming a reversible adiabatic process. The resultant pressure rise is calculated assuming that steam behaves as an ideal gas.

The pressure vs. time function must extend to include all potential relief valve opening pressures, such that pressurization rate can be determined over the last second or two just prior to the opening pressure.

The requirements for the alignment of the Safety Injection Tanks (SIT) to the RCS while in the LTOP temperature range must be evaluated to ensure that the SITs are either at an operating pressure below the LTOP setpoint or securely isolated and thus do not constitute an additional mass addition source.

3.4 LTOP Evaluation Methodology

3.4.1 Criteria for Adequate LTOP System

An adequate LTOP system ensures that the applicable P-T limits are protected from being exceeded during postulated overpressure events with a minimal impact on plant operating flexibility. 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 temperature range, these pressures are compared with the applicable P-T limits. As each LTOP limitation is temperature related, for it to be valid, the applicable P-T limit pressure value must be demonstrated to be above the applicable controlling pressure at a given temperature. A controlling pressure is the most limiting (greatest) transient pressure of all events postulated for the subject temperature range.

The LTOP methodology for CE NSSS designs is consistent with BTP RSB 5-2 (Reference 12). The primary concern of BTP RSB 5-2 is that during startup and shutdown conditions at low temperature, especially in water-solid conditions, the RCS pressure might exceed the P-T limits established for protection against brittle fracture of the RV. Accordingly, BTP RSB 5-2 states, in part, that LTOP transient analyses determine the greatest system pressure that may challenge the P-T limits. No consideration is given in BTP RSB 5-2 to the lowest transient pressure that might occur at the re-closure of the LTOP relief valve following discharge to mitigate the pressure transient. Consistent with BTP RSB 5-2, the methodology for CE NSSS designs does not include the minimum transient pressure considerations.

3.4.1.1 Affect of Minimum Transient RCS Pressure on RCP Shaft Seal Integrity

For LTOP methodology for CE NSSS designs it is not necessary to consider the consequential effects of minimum transient pressure on RCP shaft seals. This is in part because BTP RSB 5-2 is concerned with overpressure transients that may challenge the P-T limits and no consideration is given to the lowest transient pressure that might occur at the re-closure of the LTOP relief valve following discharge. In the event of any pressure loss event, as a consequence of actuation of the plant's LTOP system or otherwise, the seal design in use at CE NSSS is not susceptible to catastrophic failure due to operation at low system pressure[4]. The robust design and the extensive operating experience[5] of the shaft seals in use allow operation at low system pressure for a reasonable period of time without resulting in excessive

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[4] Low system pressure is considered to be less than 100 psig.

[5] Query of the INPO event database did not identify any events of RCP seal damage at CE NSSS units as a result of low transient pressure.

coolant leakage or catastrophic failure. For example, the RCP seals will not be adversely affected if operated without operator action for an hour at very low system pressure. It is expected that the seals will operate for a considerably longer period of time at low pressure without excessive leakage or catastrophic failure.

All domestic CE NSSS currently operate with Byron-Jackson (now Flowserve, formerly BW/IP) or KSB manufactured reactor coolant pumps. The Byron Jackson pumps use a 4-stage shaft seal cartridge. The KSB pumps use a 3-stage shaft seal cartridge. The replaceable seal cartridges in use in these plants are from one of three manufacturers. These are the original manufacturers plus Sulzer Pumps (formerly Bingham-Willamette).

Each of these seal designs is of equivalent hydrodynamic design and specification. The design specifies multiple stages of either two or three rotating stages plus a final vapor stage. Each stage, including the vapor stage, is capable of sustaining full system differential pressure. Which means that any one stage alone will sustain RCS boundary integrity in the extremely unlikely event of multiple stage failure. Figure 3.1 provides a cut-away view of one representative seal stage configuration.

The hydrodynamic design of each rotating seal stage within the cartridge consist of a rotating and a static face. The rotating face, a circular ring of hard faced material, concentric to the shaft, is attached to and rotates with the shaft. The static face, also of a hard faced material, is attached to the pump. On each stage a large dynamic force provided by mechanical springs constantly forces the rotating face of the stage against the stationary face, which create the seal. Normal operating RCS pressure provides a hydraulic pressure which acts against the constant dynamic pressure of each stage. This hydraulic pressure acts to balance the pressure on the stage faces to minimize heat and wear. A small amount of controlled leakage cools the face of these stages during rotation. The controlled leakage is directed to the CVCS volume control tank. During normal operation, the pressure drop between the RCS and the VCT is broken down in equal increments across the stages of the seal cartridge.

The seal cartridge manufacturers specify normal minimum operating pressure limits for the seals which are based upon consideration of wear rates, since, as RCS hydraulic pressure is reduced the mechanical force against the seal faces increases, thus increasing heat and wear. The recommended operating pressure limit is not based on failure limitations. The typical minimum operating pressure is 150 to 250 psig at the RCP suction. This minimum operating pressure is factored into the RCP operating limits maintained in the plant operating procedures.

Extended operation of these RCP seals at a pressure below the recommended minimum limit may result in earlier seal cartridge replacement due to accelerated wear. Low pressure operation will not promote a failure mechanism other than accelerated seal face wear. For the duration of the LTOP transient, the RCP seal stages may experience some premature wear, but such operation will not result in a complete seal failure. Even in the unlikely event a single stage were to fail, the remaining stages would prevent a loss of RCP seal pressure retention function and thereby prevent excessive leakage. Since an LTOP transient would be of limited duration, the minimum pressure associated with relief valve re-closure does not pose a challenge to seal integrity.

The above description of RCP shaft seal performance is assured by maintaining a subcooled condition in the seal stages during any potential transient. In a transient situation, the hydraulic conditions in the seal cartridge will be maintained in a subcooled condition through the combination of normal seal cooling and the reseat pressure of the LTOP relief valve. Reseat pressure will limit the potential minimum pressure in the seal stages. The reseat pressure of the LTOP relief valve is specific to the plant's LTOP relief protection and the valve design.

3.4.2 Applicable P-T Limits

CE utilizes two kinds of P-T limits: 1) Appendix G P-T limits and 2) LTOP P-T limits. The Appendix G P-T limits are used at each plant as operating restrictions and are developed via the NRC-approved methodology of Appendix G to Section XI of the 1995 ASME Code Edition and Addenda through the 1996 Addenda (Reference 10) as currently specified in 10 CFR 50, Appendix G (Reference 1). Currently, these limits are found in the TS and operating

procedures and are also used as a basis for establishing the LTOP relief valve setpoints and other limitations in most of the CEOG member plants.

Those CEOG member plants, for which the use of the Appendix G P-T limits as a basis for LTOP would adversely impact operating flexibility, may choose an alternate methodology for generating the P-T limits to be utilized as a basis for the LTOP setpoints. This methodology is also contained in Appendix G to Section XI of the 1995 ASME Code Edition and addenda through the 1996 Addenda (Reference 10), where it is described as the one applicable to the plants with LTOP systems. It effectively increases the Appendix G limits by 10%, which allows for higher LTOP setpoints and is operationally less restrictive. CE uses the term "LTOP P-T limits" to distinguish them from the Appendix G P-T limits. For the CEOG member plants choosing this alternate methodology, the LTOP P-T limits are used only as a basis for the LTOP setpoints and other LTOP limitations, whereas the Appendix G P-T limits in the existing TS and operating procedures continue providing operating restrictions.
If the applicant utility has been approved to use ASME Code Case N-640 (Reference 11) via exemption granted under 10 CFR 50.60 paragraph (b) pertaining to proposed alternatives to the described requirements in Appendix G and H on fracture toughness, then the Appendix G P-T limits based on the Code Case cannot be adjusted up by 10% to obtain LTOP P-T Limits as described above; they shall be used as both the operating restrictions and the basis for the LTOP setpoints and limits.

The P-T limits that are protected by LTOP are mostly those for the RV beltline (and flange, as applicable) and apply to RCS heatup, cooldown, and isothermal conditions. The P-T limits at the beltline are adjusted to the pressurizer using pressure correction factors. A pressure correction factor is a pressure differential between the reference location in the RV beltline and the pressurizer pressure instrument tap. It includes, in part, static head between the RV beltline and pressurizer reference locations and a flow induced pressure drop between the RV inlet nozzle (or the beltline reference location, if available) and the surge nozzle in the hot leg. The flow induced pressure drop depends on the RV flow rate, which is a function of the number of operating RCPs. The maximum number of RCPs allowed by the TS to operate within a temperature range must be accounted for in determining the pressure drop.

For the existing TS, the P-T limits in terms of pressurizer pressure may or may not include pressure and temperature indication instrumentation uncertainties. As a basis for the LTOP evaluation, these adjusted P-T limits should not include pressure indication uncertainties, but may include temperature indication uncertainty. Pressure instrumentation uncertainty is accounted for in the determination of the PORV opening pressure, as described in Section 3.3.3.2. If temperature indication uncertainty is not part of the P-T limits, it must be included in the LTOP evaluation that determines LTOP-driven limitations such as the enable temperature, heatup and cooldown rate limitations, reference temperatures for LTOP setpoints, and all cases where temperature-related operating restrictions are applied. The temperature instrumentation uncertainty must be determined using guidance contained in Regulatory Guide 1.105 (Reference 20) and ISA Standard S67.04-1994 (Reference 13). The plant-specific PTLR must address this issue. The P-T limits are developed and applied down to the RCS temperature associated with the calculated minimum boltup temperature.

For the LTOP systems that use large capacity (over 1500 gpm) relief valves connected to the pressurizer, an adjustment must be made to account for the pressure differential between the RV and the pressurizer due to flow induced losses in the surge line. That pressure differential can either be included in the pressure correction factors for the P-T limits or be added to the peak transient pressures. As this pressure differential is not present when the relief valve is closed (i. e., most of the time) using it for the adjustment of the P-T limits would unnecessarily restrict them at other times.

Independent of which P-T limits are used as a basis for LTOP setpoints, the criterion for not exceeding these limits during postulated pressure transients remains valid.

3.4.3 LTOP Enable Temperatures

The LTOP system must be aligned and capable of mitigating any postulated overpressure event between the RV minimum boltup temperature and the LTOP enable temperature. Exceptions to this requirement would be if the RCS were incapable of being pressurized, such

as when the reactor vessel head is removed or an adequate vent area is established, as specified in the TS. The enable temperatures must be determined by the guidance provided in BTP RSB 5-2 (Reference 12), or Appendix G to Section XI of the 1995 ASME Code Edition and addenda through 1996 Addenda (Reference 10). Per BTP RSB 5-2, the LTOP enable temperature is defined as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) which is controlling in the Appendix G limit calculation. The LTOP enable temperature must account for the temperature gradient between the reactor coolant and metal at the controlling location. This is accomplished by performing a heat transfer analysis of the specific transient on the RV (i.e., a finite element thermal analysis of the metal wall is performed). The results from this analysis yield the temperature differential between the metal temperature and the reactor coolant. This information is used in the determination of the LTOP enable temperature for each transient. The overall LTOP enable temperature is developed from these individual results.

In accordance with ASME Code (Reference 10) guidance, the LTOP enable temperature is at the greater of 200°F or the reactor coolant inlet temperature corresponding to a RV metal temperature less than RT_{NDT} + 50°F. The RV metal temperature is the temperature at 1/4t at the beltline location.

A single LTOP enable temperature value is typically determined for cooldown based upon isothermal conditions. With respect to heatup, however, LTOP enable temperature is a function of heatup rate. The selected LTOP enable temperature for heatup must be that for the highest applicable heatup rate within the LTOP region. The resulting enable temperatures are then corrected for instrumentation uncertainty, as applicable. A single value, equal to the greater of the two, may be used, if desired. Use of two values, one for heatup and another for cooldown, is also acceptable.

3.4.4 LTOP-Related Limiting Conditions for Operation

As the RV gets irradiated with time, the Appendix G limits become more restrictive and additional limitations may be placed on operation of the plant. These operational restrictions must be placed into TS, in accordance with BTP RSB 5-2 and GL 96-03.

Typical restrictions that are placed on plant operations are listed below. These restrictions are in addition to P-T limits and relief valve setpoints and are always included in TS. This list is not intended to be complete or be applicable to every plant but is provided as an overview of possible restrictions.

1. RCS heatup and cooldown rates are restricted to rates lower than the RCS design rates.

2. HPSI flow is restricted by locking out power to the pumps or closing header isolation valves and locking out power to the valves while in the LTOP region.

3. Charging pump operation is limited by locking out power to the pumps and either closing an appropriate valve, or using another means that will result in at least two actions/failures that would be required to start a pump.

4. The number of operating RCPs is limited.

5. Water solid operation is restricted to a temperature region.

6. Limitations on start of the first RCP are specified that may include the secondary-to-primary temperature differential, pressurizer level, and/or initial pressure.

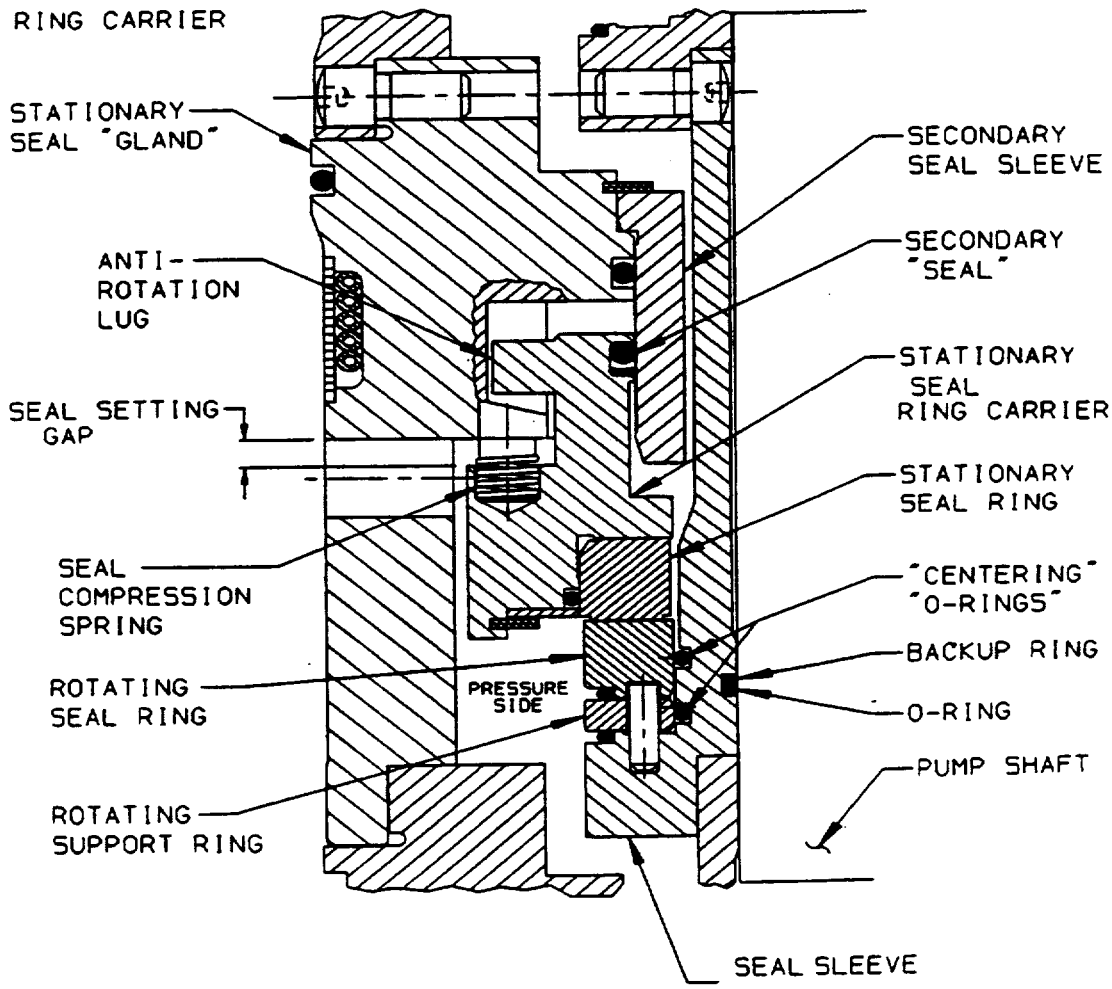
7. Closing of SIT isolation valves.

In accordance with GL 96-03, only the P-T limits and LTOP setpoints may be relocated into the plant-specific PTLR.

The TS must be modified to include the approved version (i.e., "-A") of this topical report in the Administrative Controls Section.

Figure 3.1

Single Stage of a Typical CE NSSS RCP Seal Design



4.0 METHOD FOR CALCULATING BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURE (ART)

This section addresses provision 4 of Attachment 1 to GL 96-03 (Reference 3) for the calculation of the ART. The ART is determined in accordance with Regulatory Guide 1.99, Revision 2 (Reference 16), "Radiation Embrittlement of Reactor Vessel Materials". The ART is determined as follows:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

"Initial RT_{NDT}" is the reference temperature for the beltline plate or weld material as described in Section 2.0. ΔRT_{NDT} is the shift in reference temperature calculated using a chemistry factor (from Table 1 or 2, as applicable, of Regulatory Guide 1.99, Revision 02 based on the copper and nickel content) and a neutron fluence factor (using the neutron fluence at the vessel depth of interest). The margin is the root mean squared value using the uncertainty in the initial RT_{NDT}, σ_i, and the uncertainty in the reference temperature shift, σ_Δ. The uncertainty in the initial RT_{NDT}, σ_i, for a measured value of RT_{NDT} is based on the precision of the test method; the uncertainty for a generic value is the standard deviation of the data used to obtain the generic value[5]. The reference temperature shift uncertainty, σ_Δ, for base material (e.g., plates) is 17°F and for welds is 28°F.

When credible surveillance data, as defined by Regulatory Guide 1.99, Revision 2, are available, the chemistry factor may be modified and the uncertainty in the shift in reference temperature may be reduced in accordance with Position 2.1. The process is as described in the Regulatory Guide and is discussed further in Section 7.0[6].

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[5] When using the generic value for welds made using Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes, initial RT_{NDT} = -56°F, and σ_i =17°F.

[6] Upon issuance of a new revision of Regulatory Guide 1.99, the ART calculation methodology will be evaluated and, if applicable, the new methodology will be cited in subsequent revisions of the PTLR.

5.0 APPLICATION OF FRACTURE MECHANICS IN CONSTRUCTING P-T CURVES

This section addresses Provision 5 of Attachment 1 to GL 96-03 (Reference 3), on calculation of pressure and temperature limit curves. It describes the analytical techniques and methodology for developing P-T limits for the beltline region of the reactor vessel that are utilized in the generation of composite RCS operating limits. The methodology presented in the following sections is directly applicable to plant heatup, cooldown and inservice hydrostatic tests.

5.1 General Discussion

A brief description of the procedures practiced by CE to develop brittle fracture limits for the CE NSSS design is given below for the required components of the RCPB. These procedures are applicable to all CE NSSSs and have been applied to nuclear power plants since the incorporation of Appendix G to 10 CFR 50 in 1973. These procedures utilize analytical techniques that are based partially on LEFM and include appropriately conservative design loadings for the ferritic components of the RCPB to preclude brittle fracture.

Currently, the ferritic components of the RCPB specifically addressed by Appendix G to Section XI of the ASME Code (Reference 10) are delineated as follows:

1. Vessels
2. Piping, Pumps and Valves
3. Bolting

Of these components, the vessel is the only one for which a LEFM analysis is specifically required by 10 CFR 50 Appendix G. The test and acceptance standards to which the other components are designed are considered to be adequate to protect against nonductile failure.

Over the course of the CE NSSS fabrication history, the following RV regions were considered, but not necessarily specifically evaluated, in the analysis to establish brittle fracture limits for a CE plant:

1a. Beltline

1b. Vessel Wall Transition

1c. Bottom Head Juncture

1d. Core Stabilizer Lugs

1e. Flange Region

1f. Inlet Nozzle

1g. Outlet Nozzle

As part of RV design and fabrication of the CE NSSS, a P-T Limit evaluation in accordance with ASME Section III, Appendix G requirements (Reference 9) was performed for some or all of these regions, as appropriate. The results of those evaluations were used to establish an initial set of P-T limits for plant operation. Of these regions, the beltline is the only portion of the vessel that is exposed to sufficient neutron flux over time to significantly alter the fracture toughness properties of the RV material, eventually becoming the most limiting component of the P-T Limit evaluation. The P-T Limit evaluations for the remaining regions of the vessel, once established, do not change significantly due to exposure to neutron flux unless they are updated through regulation or more recent advances in the technology. In either case, the recommended CE practice is to continually consider these non-beltline locations, as necessary, in the updating of P-T Limits throughout plant life, as they are currently part of the plant design basis. However, for the purposes of this topical report, the P-T Limit methodology discussion will be limited to the beltline region only. The P-T Limits for the beltline will be combined with the non-beltline regions, as appropriate, into a set of composite curves for specific modes of operation and will be found in the plant specific PTLR submittal. The lower bound of these composite curves defines the plant P-T Limit for a plant at that mode of operation.

10 CFR Part 50, Appendix G defines 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 radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

For the CE NSSS design, the "beltline" refers to the region of the RV that immediately surrounds the reactor core and is exposed to the highest levels of fast neutron fluence. Typically, the beltline is restricted to the large cylindrical shell section of the RV below the vessel wall transition. However, for some plant designs, the beltline region may also include the vessel wall transition. In either case, the material with the highest adjusted RT_{NDT} value typically falls within the cylindrical shell region below the vessel wall transition.

The beltline location has been analyzed utilizing the principles of LEFM described by Appendix G to Section XI of the ASME Code. This analysis considered plant heatup, plant cooldown and an isothermal leak test. A brief description of the general criteria follows.

5.2 Regulatory Requirements

10 CFR Part 50, Appendix G outlines the pressure and temperature requirements for the reactor pressure vessel for the normal operating and hydrostatic pressure and leak tests conditions. Table 1 of 10 CFR Part 50, Appendix G, provides specific guidance on P-T requirements for critical and non-critical core conditions. The guidance is centered on P-T limits developed using the fracture toughness methods of ASME Section XI, Appendix G. Table 1 also sets criteria to establish the minimum temperature requirements for the RV. Composite P-T limit curves are normally generated by calculating the most conservative P-T limit points established by using the methods of ASME Section XI, Appendix G, and the methods for the minimum temperature requirements.

In accordance with the ASME Code Section XI, Appendix G (Reference 10), requirements, the general equation to be satisfied for any assumed rate of temperature change during

(1) Service Level A and B *(Normal and Upset Loads, respectively)* operation *(heatup and cooldown)* is

$2K_{IM} + K_{IT} < K_{IR}$ (Reference 10) and

(2) Hydrostatic and Test conditions when core is not critical is:

$1.5K_{IM} + K_{IT} < K_{IR}$ (Reference 10)

where,

K_{IM} = Allowable pressure stress intensity factor, Ksi $\sqrt{in}$

K_{IT} = Thermal stress intensity factor, Ksi $\sqrt{in}$

K_{IR} = Reference stress intensity factor, Ksi $\sqrt{in}$

The minimum temperature requirements for the RV of a PWR, as required by Table 1 to 10 CFR Part 50, Appendix G, are as follows:

- For pressure testing conditions of the RCS, when the RCS pressure is less than or equal to 20% of the preservice hydrostatic test pressure (PHTP), and the reactor core is not critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} limiting material in the closure flange region stressed by bolt preload.

- For pressure testing conditions of the RCS, when the RCS pressure is greater than 20% of the PHTP and the reactor core is not critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region plus 90 °F.

- For normal operations, when the RCS pressure is less than or equal to 20% of the PHTP and the reactor core is not critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload.

- For normal operations, when the RCS pressure is greater than 20% of the PHTP and the reactor core is not critical, the minimum temperature requirement for the RV must be at

least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload plus 120 °F.

- For normal operations, when the RCS pressure is less than or equal to 20% of the PHTP and the reactor core is critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload plus 40 °F, or the minimum permissible temperature for the inservice hydrostatic pressure test, whichever is larger.

- For normal operations, when the RCS pressure is greater than 20% of the PHTP and the reactor core is critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload plus 160 °F, or the minimum permissible temperature for the inservice hydrostatic pressure test, whichever is larger.

5.3 Reference Stress Intensity Factor

At each of the postulated flaw locations, the Mode I stress intensity factor, K_I, produced by each of the specified loads, is calculated and the summation of the K_I values is compared to a reference stress intensity factor, K_{IR}. The result is a relationship of pressure versus temperature for reactor vessel operating limits that preclude brittle fracture. K_{IR} is currently defined as K_{IA}, the lower bound of crack arrest critical K_I values, measured as a function of temperature. Another material stress intensity factor, K_{IC}, is based on the lower bound of static initiation critical K_I values measured as a function of temperature. Both K_{IA} and K_{IC} are obtained from a reference fracture toughness curve for reactor pressure vessel low alloy steels as defined in Appendix G and Appendix A to Section XI of the ASME Code. These governing curves are defined by the following expressions:

$$K_{IA} = 26.78 + 1.223e^{[0.0145(T-RT_{NDT}+160)]}$$ Reference 10(a), *(See Note 1 below)*

$$K_{IC} = 33.20 + 20.734e^{[0.0200(T-RT_{NDT})]}$$ Reference 10(b), *(See Note 2 below)*

where,

K_{IA} = Crack arrest reference stress intensity factor, Ksi $\sqrt{in}$

K_{IC} = Crack initiation reference stress intensity factor, Ksi $\sqrt{in}$

T = temperature at the postulated crack tip, °F

RT_{NDT} = adjusted reference nil ductility temperature (also called ART) at the postulated crack tip, °F

For any instant during the postulated heatup or cooldown, K_{IA} or K_{IC} is calculated using the metal temperature at the tip of the flaw, as well as the value of ART at that flaw location.

Note: (1) ASME Section XI, Appendix G, Article G-2110 specifies K_{IA} with a coefficient of 1.233 versus 1.223 in Figure G-2110-1. In the calculation of K_{IA}, a slightly lower (0.8%) and more conservative value is obtained using a constant of 1.223, instead of 1.233, which would give a higher allowable limit. However, a value of 1.223 is consistent with Welding Research Council Bulletin 175 (Reference 14), and NRC Standard Review Plan 5.3.2 (Reference 15).

(2) The use of K_{IC} as the basis for establishing the reference fracture toughness limit, K_{IR}, value for the vessel is currently outlined in ASME Code Case N-640. Use of the K_{IC} fracture toughness limit will yield less limiting Appendix G P-T limits as compared to the use of K_{IA}, the current fracture toughness limit. However, the use of this Code Case for the applicant plant must be approved by the NRC via an exemption granted under 10 CFR 50.60 paragraph (b) pertaining to proposed alternatives to the described requirements in Appendix G and H on fracture toughness and is restricted as follows:

- If a licensees wishes to use K_{IC} as the basis for establishing the K_{IR} value for the vessel, then the licensee shall limit the maximum pressure in the vessel to 100% of the pressure allowed by the P-T limit curves as the basis for establishing the setpoints for the Low Temperature Overpressure Protection (LTOP) system.

5.4 Pressure-Temperature Limit Generation Methods

5.4.1 General Description of P-T Limits Generation

This section describes the methods used by CE to develop the allowable pressure-temperature relationships for normal plant heatup, cooldown and hydrostatic test conditions. For the beltline region, CE utilizes a finite element based influence coefficient superposition method to calculate heatup and cooldown limits.

The following sections describe the calculation of pressure-temperature limits. First, a general process description is presented to outline the flow of calculating the minimum P-allowable. Next, a section describing the calculation of allowable pressure is provided. Subsequent sections describe further details on the beltline CE method.

5.4.1.1 Process Description

P-T limits are generated via the following process to calculate P-Allowable and is based on a general method utilizing Linear Elastic Fracture Mechanics procedures to calculate the thermal stress intensity factor, K_{IT}, at the ¼ T and ¾ T crack tip locations. Once K_{IT} is determined, the general equation found in Appendix G, ASME Section XI is used to relate the size of a flaw with the allowable loading that precludes crack initiation, thus generating an allowable pressure. This relation is based upon a component stress analysis of the anticipated thermal loading and upon experimental measurements of the beltline material fracture toughness properties, as prescribed in Appendix G to Section XI of the ASME Code (Reference 10(a)).

The general process to generate PT Limits is as follows:

a) Determine the limiting ART for the postulated crack tip locations taking into account any radiation embrittlement.

b) Perform a thermal analysis of a set of constant rate heatup and cooldown transients on a particular vessel geometry to obtain through-wall temperatures.

c) Calculate the thermal stress intensity factor, K_{IT}, at the postulated crack tips for each time point in each transient.

d) Calculate the material reference stress intensity factor, K_{IR}, at the postulated crack tips for each time point in each transient.

e) Calculate the transient P-Allowable by subtracting the thermal stress intensity factor, K_{IT}, from the material reference stress intensity factor, K_{IR}, via the Appendix G requirement and solving for the allowed pressure loading for each point in the transient.

f) Calculate the Isothermal P-Allowable from the material reference stress intensity factor, K_{IR}, via the Appendix G requirement and solving for the allowed pressure loading *(For the Isothermal condition, the thermal stress intensity factor, K_{IT}, is assumed to be zero).*

g) Determine minimum P-Allowable as the minimum of the Heatup/Cooldown transient P-Allowable and the Isothermal P-Allowable at the postulated crack tips. *(These results are tabularized and plotted as the Heatup/Cooldown PT Limits for a particular vessel).*

The following sections provide additional detail as to some of the specifics outlined in the general procedure above. In addition, the analysis of heatup and cooldown transients are described and discussed.

5.4.2 Calculation of Allowable Pressure

The Appendix G equation relating K_{IM}, K_{IT}, and K_{IR} is rearranged as shown below to solve for the allowable pressure stress intensity factor, K_{IM}, as a function of time with the calculated K_{IR} and K_{IT} values. As shown in the following equation, the thermal stress intensity is subtracted

from the available K_{IR} to determine the allowable pressure stress intensity factor and consequently the allowable pressure:

(1) For Service Level A and B operation:

$$K_{IM} = \frac{K_{IR} - K_{IT}}{2}$$

and

(2) For Hydrostatic and Test Conditions when the core is not critical and tests are performed at isothermal conditions (i.e. thermal stress intensity factor, $K_{IT,} = 0$)

$$K_{IM} = \frac{K_{IR}}{1.5}$$

where,

K_{IM}	=	Allowable pressure stress intensity factor as a function of coolant temperature, Ksi $\sqrt{in}$
K_{IR}	=	Reference stress intensity factor as a function of coolant temperature, Ksi $\sqrt{in}$
K_{IT}	=	Thermal stress intensity factor as a function of coolant temperature, Ksi $\sqrt{in}$

The allowable pressure is then derived from the calculated allowable pressure stress intensity factor, K_{IM}, shown above. The generation of K_{IM} is discussed in the following sections.

5.4.3 CE NSSS P-T Curve Method for the Beltline Region

In the development of operational limits, CE analyzes the RV beltline region considering the predicted effects of neutron fluence over a specific time period. The beltline region is the only location that receives sufficient neutron fluence to substantially alter the toughness properties of the RV material. Over time, CE considers the beltline region to be controlling, that is, the most limiting with respect to allowable pressure at any specific temperature. (The allowable pressure

is based on the highest Adjusted Reference Temperature, ART, for the beltline materials.) This philosophy is consistent with the guidance given in Standard Review Plan 5.3.2, Pressure-Temperature Limits (Reference 15). P-T limits for the beltline are generated in conjunction with the shift prediction methods of Regulatory Guide 1.99 Revision 2 (Reference 16), to account for the reduction in fracture toughness due to neutron irradiation.

In the CE NSSS PT Curve Methodology, the RV beltline region is analyzed assuming a semi-elliptical surface flaw oriented in the axial direction with a depth of one quarter of the RV beltline thickness and an aspect ratio of one to six. This postulated flaw is analyzed at both the inside diameter location (referred to as the 1/4t location) and the outside diameter location (referred to as the 3/4t location) to assure the most limiting condition is achieved. The above flaw geometry and orientation is the maximum postulated defect size (reference flaw) described in Appendix G to Section XI of the ASME Code (Reference 10(a)).

5.4.4 Thermal Analysis Methodology of the Beltline Region

The first step in the generation of P-T limits is a detailed thermal analysis of the RV beltline wall to calculate the Mode I thermal stress intensity factor, K_{IT}. One dimensional, three noded, isoparametric finite elements suitable for one-dimensional axisymmetric radial conduction-convection heat transfer are used. The vessel wall is divided into elements and an accurate distribution of temperature as a function of radial location and transient time is calculated. Convective boundary conditions on the inside wall of the vessel and an insulation boundary on the outside wall of the vessel are used in the analysis. Variation of material properties through the vessel wall is permitted thus allowing for the change in material thermal properties between the cladding and the base metal.

In general, the temperature distribution through the RV wall is governed by the partial differential equation,

$$\frac{1}{r}\frac{\partial}{\partial r}\left(kr\frac{\partial T}{\partial r}\right) = \rho\, C_p \frac{\partial T}{\partial t} \qquad \text{(Reference 19, p. 37, Equation (2-6b))}$$

subject to the following boundary conditions at the inside and outside wall surface locations which are derived from Reference 19, p. 35, Equations (2-16a) and (2-16b):

$$\text{At } r=r_i \qquad -K\frac{\partial T}{\partial r} = h(T-T_c)$$

$$\text{At } r=r_o \qquad \frac{\partial T}{\partial r} = 0$$

where,

ρ	=	density, lb/ft^3
C	=	specific heat, Btu/lb-°F
K	=	thermal conductivity, Btu/hr-ft-°F
T(r,t)	=	vessel wall temperature, °F
r	=	vessel radius, ft
t	=	time, hr
h	=	convective heat transfer coefficient, Btu/hr-ft^2-°F
T_c	=	RCS coolant temperature, °F
r_i, r_o	=	inside and outside radii of vessel, ft

The above expression is solved numerically using a finite element model to determine wall temperature as a function of radius, time, and thermal rate.

5.4.4.1 Analysis of HeatUp Transient

During a heatup transient, the thermal bending stress is compressive at the RV inside wall and is tensile at the RV outside wall. Internal pressure creates a tensile stress at the inside wall as well as the outside wall locations. Consequently, the outside wall location has the larger total stress when compared to the inside wall. However, neutron embrittlement, shift in material RT_{NDT}, and reduction in fracture toughness are greater at the inside location than at the outside. Therefore, results from both the inside and outside flaw locations must be compared to assure that the most limiting condition is recognized.

It is interesting to note that a sign change occurs in the thermal stress through the RV beltline wall. Assuming a reference flaw at the 1/4t location, the thermal stress tends to alleviate the pressure stress indicating that the isothermal steady state condition would represent the limiting P-T limit. However, the isothermal condition may not always provide the limiting P-T limit for the 1/4t location during a heatup transient. This is due to the difference between the base metal temperature and the RCS fluid temperature at the inside wall. For a given heatup rate (non-isothermal), the differential temperature through the clad and film increases as a function of thermal rate, resulting in a crack tip temperature which is lower than the RCS fluid temperature. Therefore, to ensure the accurate representation of the 1/4t P-T limit during heatup, both the isothermal and heatup rate dependent P-T limits are calculated to ensure the limiting condition is recognized. These limits account for clad and film differential temperatures and for the gradual buildup of wall differential temperatures with time.

To develop minimum P-T limits for the heatup transient, the isothermal conditions at 1/4t and 3/4t, 1/4t heatup, and 3/4t heatup P-T limits are compared for a given thermal transient.

The most restrictive P-T limits are then combined over the complete temperature interval resulting in a minimum PT curve for the RV beltline for the heatup event.

5.4.4.2 Analysis of Cooldown Transient

During cooldown, membrane and thermal bending stresses act together in tension at the RV inside wall. This results in the pressure stress intensity factor, K_{IM}, and the thermal stress intensity factor, K_{IT}, acting in unison creating a high stress intensity. At the RV outside wall, the tensile pressure stress and the compressive thermal stress act in opposition, resulting in a lower total stress than at the inside wall location. Also, neutron embrittlement, the shift in RT_{NDT}, and the reduction in fracture toughness are less severe at the outside wall compared to the inside wall location. Consequently, the inside flaw location is limiting for the cooldown event.

To develop a minimum P-T limit for the cooldown event, the isothermal P-T limit must be calculated. The isothermal P-T limit is then compared to the P-T limit associated with cooling rate, and the more restrictive allowable P-T limit is chosen, resulting in a minimum P-T limit curve for the RV beltline.

5.4.4.3 Calculation of Thermal Stress Intensity Factors, K_{IT}, and Allowable Pressure

ASME Section XI Appendix G (Reference 10) recognizes the limitations of the original method provided for calculating K_{IT} because of the assumed temperature profile. Since a detailed heat transfer analysis results in time varying temperature profiles (and consequently varying thermal stresses), an alternate method for calculating K_{IT} is employed as suggested by Article G-2214.3 of the ASME Boiler and Pressure Vessel Code Section XI, Appendix G (Reference 10).

In this alternate method, thermal stress intensity factors are determined from the calculated temperature profile through the beltline wall using thermal influence coefficients specifically generated for this purpose. The method employed uses a polynomial fit of the temperature profile and superposition using influence coefficients to calculate K_{IT}. The influence coefficients are dependent upon the geometrical parameters associated with the postulated defect, the geometry of the reactor vessel beltline region, and the assumed unit loading. These influence coefficients were calculated using a 2-dimensional finite element model of the reactor vessel. The influence coefficients were corrected for three-dimensional effects using ASME Section XI Appendix A procedures (Reference 10(b)).

The CE methodology calculates the K_{IT} and K_{IM} at any time point in a transient using influence coefficients generated by applying unit loads on a finite element model of the reactor vessel beltline region. The influence coefficients are calculations of stress intensity factors at the 1/4t and 3/4t crack depth location under the following unit loads:

a. for $K_{IM\text{-}P}$, pressure load of 1 ksi, ksi-$\sqrt{in}$/ksi

b. for $K_{IT\text{-}L}$, linear through-wall gradient with peak temperature of 1°F, ksi-$\sqrt{in}$/°F

c. for $K_{IT\text{-}Q}$, quadratic through-wall gradient with peak temperature of 1°F, ksi-$\sqrt{in}$/°F

d. for K_{IT-C}, cubic through-wall gradient with peak temperature of 1°F, ksi-$\sqrt{in}$/°F

Each stress intensity factor is calculated using a standard quarter point element formulation at the respective crack tips. Since all calculations performed are linear, superposition is then used to scale and combine these influence coefficients as necessary to determine the stress intensity factor for a given temperature profile.

In the case of K_{IT}, the CE method calculates through-wall temperature profiles which are then fit to the third order polynomial below:

$$T(x) = C_o + C_L (1-x/_h) + C_Q (1-x/_h)2 + C_C (1-x/_h)3$$

where,

T(x)	=	Temperature at radial location x from inside wall surface, °F
C_o, C_L, C_Q, C_C	=	Coefficients in polynomial fit, °F
x	=	Distance through beltline wall, in
h	=	Beltline wall thickness, in

The coefficients of this polynomial are then combined through the following equation to calculate K_{IT} at the 1/4t and 3/4t locations.

$$K_{IT-Total} = C_L * K_{IT-L} + C_Q * K_{IT-Q} + C_C * K_{IT-C}$$

To calculate the allowable pressure, P-Allowable, the CE method uses the resultant $K_{IT-Total}$ from above in conjunction with Equation (1) from G-2215, ASME Appendix G as described in Section 5.4.2 as follows.

for Normal Level A and B loads

$$P-Allowable=\frac{K_{IR}-K_{IT-Total}}{2*K_{IM-P}}$$

for Hydrostatic and Test Conditions, where Isothermal conditions result in $K_{IT-Total}$ =

$$P-Allowable=\frac{K_{IR}}{1.5*K_{IM-P}}$$

Note: The CE NSSS P-T curve method for the beltline region yields less conservative results relative to the methods presented in ASME Section XI, Appendix G. (i.e. Higher Allowable Pressures). The use of this method by the applicant plant must be approved by the NRC via an exemption granted under 10 CFR 50.60 paragraph (b) pertaining to proposed alternatives to the described requirements in Appendix G and H on fracture toughness.

5.4.5 Application of Output

The P-T limits developed using the method described above account for the temperature differential between the RV base metal and the reactor coolant bulk fluid temperature only. However, uncertainties for instrumentation error, elevation, and flow induced differential pressure differences between the RV beltline and pressurizer are not accounted for and must be included by the plant when final P-T limits are developed.

5.5 Typical Pressure-Temperature Limits

This section presents a calculation of example P-T limits for the RV beltline. The example P-T limits were developed using the methods described in Section 5.4 in conjunction with the minimum temperature requirements of 10CFR50, Appendix G and the following information. Note that uncertainties for instrumentation error, elevation, and flow induced differential pressure differences between the RV beltline and pressurizer are not accounted for in this example and will be included in the composite curves defined in the plant specific PTLR submittal.

Note: The bracketed information included below is provided for illustration purposes only and is not intended to be representative of all CE NSSS RVs.

Reactor Vessel Data

Design Pressure = [2500] psi

Operating Pressure = [2250] psi

Design Temperature = [650] °F

Vessel I.R. to Wetted Surface = [87.227] in.

Cladding Thickness = [5/16] in.

Beltline Thickness = [8.625] in.

Material

Cladding - [Type 304 Stainless Steel]

Beltline - [SA-533 Grade B Class 1]

Beltline ART

Flaw Location	ART (°F)
1/4 T	[191.0]
3/4 T	[137.0]

Initial RT_{NDT}

Flange Region = [+70]°F

Piping, Pumps and Valves = [+56]° F

5.5.1 Beltline Limit Curves

The beltline P-T limits calculated for heatup and cooldown are depicted in Figures 5.1 through 5.3 and have been developed utilizing the CE NSSS methodology described in Section 5.4. These figures provide the operating limits for the beltline region in terms of an allowable pressure over the operating temperature range for various linear rates of temperature change.

5.5.2 Hydrostatic Test Limits and Core Critical Limits

The beltline P-T limits calculated for hydrostatic test limits is depicted in Figure 5.3 and has been developed utilizing the CE NSSS methodology described in Section 5.4. As discussed in that section, the procedure to calculate hydrostatic test limits is the same as when calculating normal operation heatup and cooldown with the exception that the thermal stress intensity factor is neglected (i.e. $K_{IT} = 0$) since the hydrostatic leak test is performed at isothermal conditions.

The purpose of the hydrostatic test limit is to establish the minimum temperature required at the corresponding hydrostatic test pressure. It is CE's practice to recommend that the inservice hydrostatic test for CE NSSS designs be performed at a test pressure corresponding to 1.1 times the operating pressure with the reactor core not critical. Under these conditions, 10CFR50, Appendix G requires that the minimum temperature for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region plus 90 °F. However, this temperature is usually superceded by the beltline hydrostatic test limits at the recommended test pressure. Hence, it is only necessary to show the beltline inservice hydrostatic test limits in the vicinity of this pressure.

For this example, the recommended inservice hydrostatic test pressure is calculated as follows:

Inservice hydrostatic test pressure =

= (1.1 x Operating Pressure)

= (1.1 x [2250] psi) = [2475] psi

To obtain the minimum temperature for hydrostatic testing, the temperature corresponding to the inservice hydrostatic pressure of [2475] psi is linearly interpolated from the hydrostatic P-T limit in Figure 5.3. The value of [313]°F was obtained.

To define minimum temperature criteria for core critical operation, Appendix G to 10 CFR Part 50 specifies Pressure-Temperature limits based on two criteria. These criteria are that the reactor vessel must be at a temperature equal to or greater than the minimum temperature required for the inservice hydrostatic test and at least 40 °F higher than the minimum pressure-temperature curve for normal operation heatup or cooldown.

In addition, 10CFR50, Appendix G also requires that, for normal operations, when the RCS pressure is greater than 20% of the PHTP and the reactor core is critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload plus 160 °F, or the minimum permissible temperature for the inservice hydrostatic pressure test, whichever is larger. However, in most cases the outcome for this requirement is that the minimum permissible temperature for the inservice hydrostatic pressure test is more conservative.

A review of Figure 5.3 shows that the specified beltline heatup P-T limit is the more restrictive above [313]°F. Consequently, the core critical limits have been established as a combination of [313]°F and the specified beltline heatup P-T limit from ASME Appendix G plus 40°F.

Note: The core critical limits established utilizing this criterion are based solely upon fracture mechanics considerations. These limits do not consider core reactivity safety analyses that can control the temperature at which the core can be brought critical. This is considered in the plant specific PTLR submittal.

5.5.3 Minimum Boltup Temperature

The minimum boltup temperature, from Reference (1), shall be at least the initial RT_{NDT} temperature for the material in the stressed (flange) region plus any effects of irradiation and is applicable for pressures less than twenty percent of pre-service hydrostatic pressure. Since there is no significant irradiation effect in the flange region for CE NSSS's, the minimum boltup temperature is equal to the initial RT_{NDT} value of the flange region:

$$\text{Minimum Boltup Temperature} = \text{Flange Initial } RT_{NDT} = [70]°F$$

5.5.4 Flange Limits

Per 10CFR50 Appendix G, the temperature of the closure flange regions must exceed the initial RT_{NDT} of the material by at least 120°F for normal operation and by 90°F for inservice hydrostatic test and leak testing when the pressure exceeds twenty percent of pre-service hydrostatic test pressure.

Therefore, for normal operation the flange limit is:

$$\begin{aligned} \text{Flange}_{\text{Normal Op}} &= \text{Initial } RT_{NDT} + 120°F \\ &= [70]°F + 120°F = [190]°F \end{aligned}$$

and, for hydrostatic test conditions

$$\begin{aligned} \text{Flange}_{\text{Hydro}} &= \text{Initial } RT_{NDT} + 90°F \\ &= [70]°F + 90°F = [160]°F \end{aligned}$$

In practice, CE uses the flange limit for normal operation in the development of P-T limits for hydrostatic test conditions since it is always the more conservative.

5.5.5 Minimum Pressure

The minimum pressure serves as a regulatory breakpoint in the development of P-T limits and is defined by 10CFR50, Appendix G as twenty percent of pre-service hydrostatic test pressure. For CE NSSS plants, the Preservice Hydrostatic Test pressure is defined as 1.25 times Design Pressure. The function of minimum pressure in the development of P-T limits is to provide a transition between the various temperature only based P-T limits, such as minimum bolt up and the lowest service temperature or flange limits.

For this example, the minimum pressure is calculated as follows:

Minimum Pressure =

(1.25 x Design Pressure) x 0.20 =

(1.25 x [2500] psi) x 0.20 = [625] psi

5.5.6 Lowest Service Temperature

CE NSSS designs specified carbon steel materials for the fabrication of RCS piping, pumps and valves requiring the need for impact tests as outlined in Section III, Division 1, NB-2300. Per ASME Appendix G, G-3000, the tests and acceptance standards of Section III, Division 1 are considered adequate to prevent nonductile failure under loadings with defect sizes encountered under normal, upset, and testing conditions.

The lowest service temperature is the minimum required operating temperature for piping, pumps and valves in the RCS in order to exceed twenty percent of pre-service hydrostatic test pressure. Per NB-2332,

Lowest Service Temperature =

Initial RT_{NDT} of piping, pumps and valves + 100°F

For this example, the initial RT_{NDT} for these components was established as [56]°F. Hence,

Lowest Service Temperature =

[56]°F + 100°F = 156°F

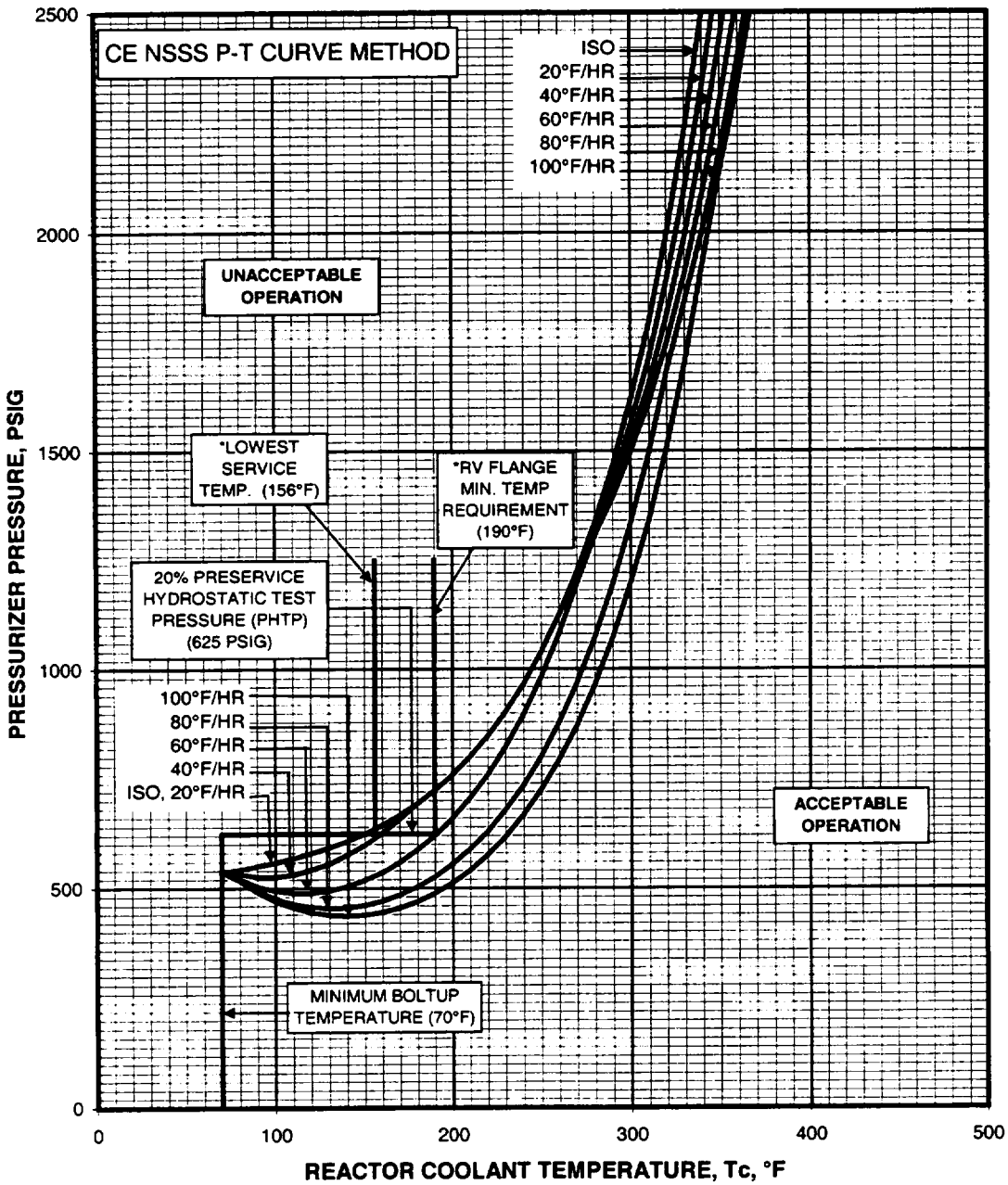
5.5.7 Summary

This section presented an example of the methodologies and practices utilized in the development of RCS P-T limits for the CE NSSS design. Typical P-T limits were developed for heatup, cooldown, hydrostatic test and core critical conditions and are presented in Figures 5.1 through 5.3, respectively. Figures 5.1 and 5.2 show results of a family of arbitrary rates selected for heatup and cooldown P-T limits for the beltline. The P-T limit curves for inservice hydrostatic test conditions and core critical operation are shown in Figure 5.3. The minimum temperature for inservice hydrostatic pressure test, [313]°F was established based on a hydrostatic test pressure of [2475] psi.

In these figures, the minimum boltup temperature was established as [70]°F, the lowest service temperature as [156]°F, the flange minimum temperature requirement as [190]°F for normal operating conditions and the minimum pressure was established as [625] psi. Together, these figures form the basis for further refinement in the plant specific PTLR to complete the development of P-T limits for a plant.

FIGURE 5.1

APPENDIX G P-T LIMITS
HEATUP (ISO/20/40/60/80/100°F)

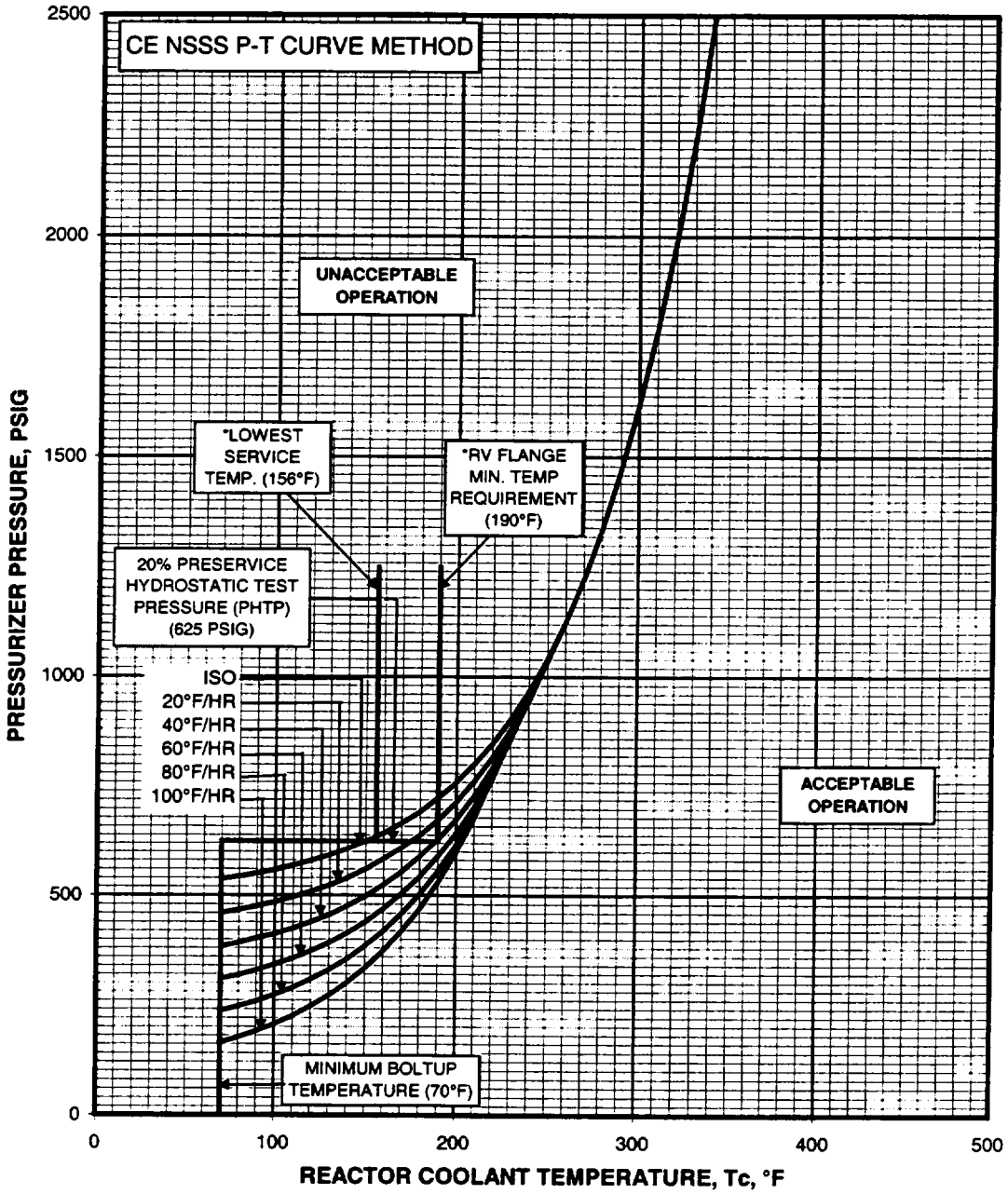


* The more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP should be used in the development of plant P-T limits.

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.2

APPENDIX G P-T LIMITS
COOLDOWN (ISO/20/40/60/80/100°F)

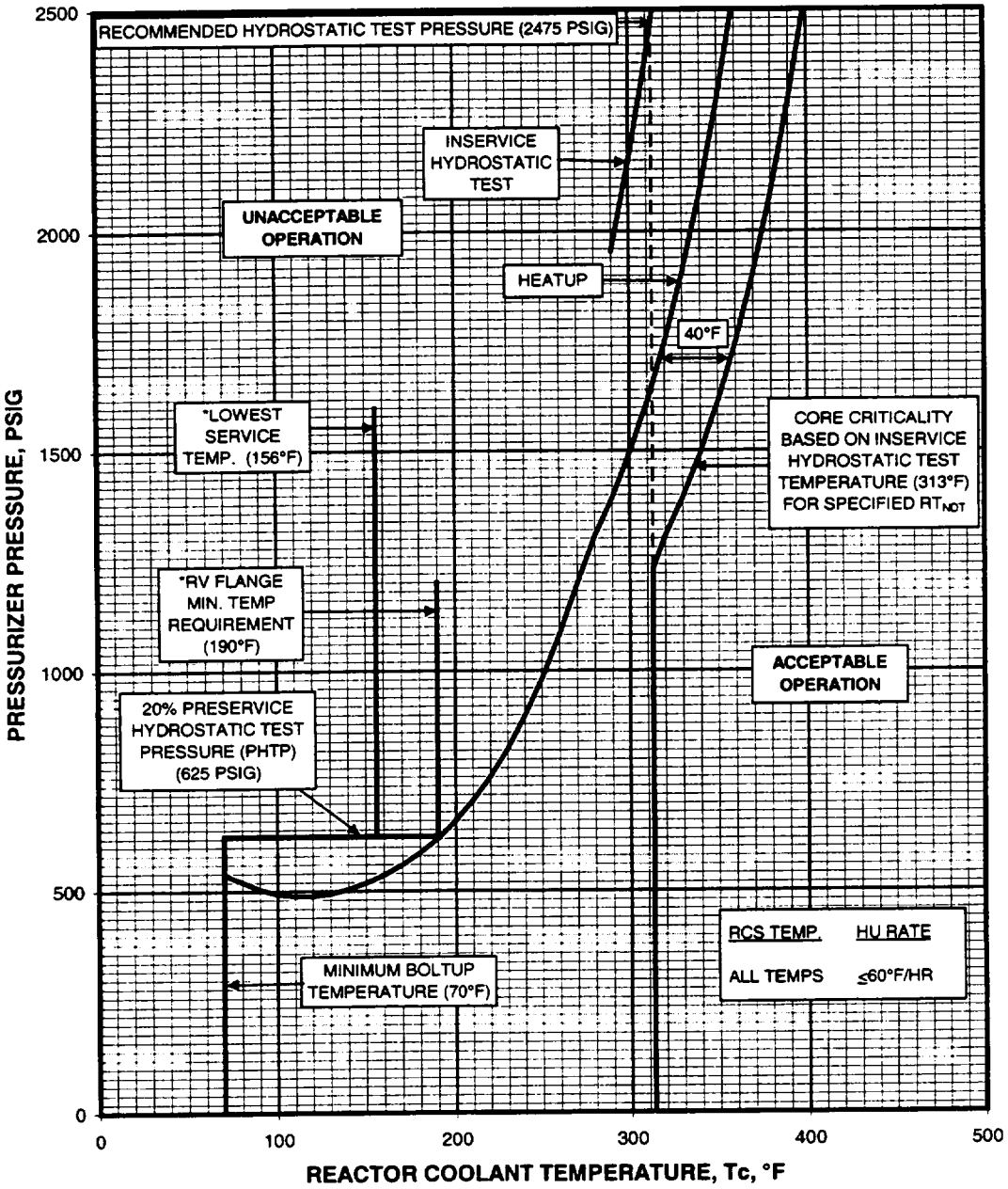


* The more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP should be used in the development of plant P-T limits.

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.3

TYPICAL REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR TECHNICAL SPECIFICATIONS HEATUP



* The more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP should be used in the development of plant P-T limits.

ART
1/4t = 191.0°F
3/4t = 137.0°F

6.0 METHOD FOR ADDRESSING 10 CFR 50 MINIMUM TEMPERATURE REQUIREMENTS IN THE P-T CURVES

6.1 Minimum Temperature Requirements in 10 CFR Part 50, Appendix G

The minimum temperature requirements for the RV of a PWR, as required by Table 1 to 10 CFR Part 50, Appendix G, are as follows:

- For pressure testing conditions of the RCS, when the RCS pressure is less than or equal to 20% of the preservice hydrostatic test pressure (PHTP), and the reactor core is not critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload.

- For pressure testing conditions of the RCS, when the pressure is greater than 20% of the PHTP and the reactor core is not critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region plus 90 °F.

- For normal operations, when the RCS pressure is less than or equal to 20% of the PHTP and the reactor core is not critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload.

- For normal operations, when the RCS pressure is greater than 20% of the PHTP and the reactor core is not critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload plus 120 °F.

- For normal operations, when the RCS pressure is less than or equal to 20% of the PHTP and the reactor core is critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by

bolt preload plus 40 °F, or the minimum permissible temperature for the inservice hydrostatic pressure test, whichever is larger.

- For normal operations, when the RCS pressure is greater than 20% of the PHTP and the reactor core is critical, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload plus 160 °F, or the minimum permissible temperature for the inservice hydrostatic pressure test, whichever is larger.

Note that the core critical limits established utilizing this criterion are based solely upon fracture mechanics considerations. These limits do not consider core reactivity safety analyses that can control the temperature at which the core can be brought critical.

6.2 Minimum Boltup Temperature

The minimum boltup temperature is established based on ASME Code Section XI, Subparagraph G-2222.c (Reference 10(a)). The recommendation is as follows:
"... when the flange and adjacent shell region are stressed by the full intended bolt preload and by pressure not exceeding 20% of the pre-operational system hydrostatic test pressure, minimum metal temperature in the stressed region should be at least the initial RT_{NDT} temperature for the material in the stressed region plus any effects of irradiation at the stressed regions."

6.3 Lowest Service Temperature

The lowest service temperature is defined by the ASME Code Section III, NB-2332 as "the minimum temperature of the fluid retained by the component or, alternatively, the calculated volumetric average metal temperature expected during normal operation, whenever pressure exceeds 20% of the pre-operational system hydrostatic test pressure." This requirement is applicable to piping, pumps, and valves and is intended to protect these components from brittle fracture.

The lowest service temperature is established based on the limiting RT_{NDT} for ferritic low alloy steel piping, pump, and valve materials in the RCPB. The lowest service temperature is the highest RT_{NDT} for those materials plus 100°F.

6.4 Minimum Permissible Temperature for Performing Hydrostatic Test

The minimum temperature for the inservice hydrostatic test pressure can be established conservatively by determining the test pressure corresponding to 1.1 times normal operating pressure and locating the corresponding temperature on the isothermal P-T curve. Hydrostatic testing of the RV after achieving core criticality is not allowed.

Note: In the development of Pressure-Temperature Limits, the intent of these requirements is to utilize the more conservative of either the Lowest Service Temperature or the other minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP.

7.0 APPLICATION OF SURVEILLANCE CAPSULE DATA TO THE CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

This section addresses Provision 7 of Attachment 1 to GL 96-03 (Reference 3) on application of surveillance capsule data.

Data from the RV surveillance program are used for two related purposes. The original purpose was to provide a system to monitor the radiation-induced changes to the toughness properties and provide assurance that the vessel materials are not behaving in an anomalous manner. The second purpose is to provide plant specific data for RV integrity analysis. Irradiation of materials in the surveillance capsules exposes specimens which are representative of the RV beltline in an irradiation environment nearly identical to the environment for the vessel. The post-irradiation analysis of the surveillance capsule contents provides measurements of the neutron fluence and of the changes in toughness properties of the surveillance plate and weld materials. These data can be used to refine both calculations of the vessel fluence and predictions of the ART for the beltline materials.

When data are available from two or more capsules (potentially from other plants), an evaluation may be performed to determine whether the data are credible as defined in Regulatory Guide 1.99, Revision 2. The data are deemed credible if:

1) One or more of the surveillance materials is controlling for that RV with respect to the ART,

2) The Charpy data scatter does not cause ambiguity in the determination of 30 ft-lb shift,

3) The measured shifts are within σ_Δ of the shift predicted using Position 2.1 (2 σ_Δ if the fluence range is large),

4) The capsule irradiation temperature is comparable to that of the RV, and

5) The correlation monitor material data, if available, are within the scatter band of the known data for that material.

The credible data can then be applied following Position 2.1 of the Guide to calculate a new chemistry factor for that material and to reduce the standard deviation for shift by half. If the revised chemistry factor and reduced standard deviation from application of Position 2.1 result in a higher value of ART than from that calculated using Position 1.1, the revised values must be incorporated into the PTLR methodology. If the Position 2.1 values result in a lower value of ART, either the Position 2.1 values will be incorporated or the original PTLR methodology will be retained.

When the plant-specific surveillance capsule data are credible in all respects except for the match of the surveillance material heat number to the controlling RV material heat number and there are data for the controlling material heat number available from another plant, the plant-specific PTLR may utilize surveillance data from that other plant as the basis for the ART prediction methodology. If such data are employed, the source of the data must be identified, the correspondence of the material heat numbers must be confirmed, and the basis for the manner in which the data are applied must be provided. The basis could be a previously generated safety evaluation report which would be referenced or a newly generated evaluation in which the licensee's surveillance data and the sister plant surveillance data are assessed with respect to the credibility criteria of Regulatory Guide 1.99, Revision 2 and, in addition, with respect to irradiation environment factors (e.g., neutron spectrum and irradiation temperature). Some recent CEOG sponsored efforts which are applicable to this discussion are CEOG Task 621 (Reference 7) which addresses methodology for the application of sister plant data and CEOG Task 904 which addresses methodology for the application of both plant-specific and sister plant data to refine ART calculations (Reference 21). Additionally, the use of this sister plant data must be reviewed and approved by the NRC if the licensee has not been approved to use integrated surveillance data or the sister plant data can be used directly if the NRC has

already determined that the licensee complies with the requirements for Integrated Surveillance Programs per Section II.3.C to 10 CFR Part 50, Appendix H.

8.0 REFERENCES

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